NUREG/CP-0168
Vol. 1

Transactions of the Twenty-Seventh Water Reactor Safety Information Meeting

To Be Held at
Bethesda Marriott Hotel
Bethesda, Maryland
October 25–27, 1999

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

Proceedings prepared by
Brookhaven National Laboratory

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Date Published: October 1999

Compiled by: Susan Monteleone, Meeting Coordinator

S. Nesmith, NRC Project Manager

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

PREFACE

This report contains summaries of papers on reactor safety research to be presented at the 27th Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel in Bethesda, Maryland, October 25-27, 1999. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, U.S. NRC. 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 are also included. The summaries have been compiled in one report to provide a basis for meaningful discussion of information exchanged during the course of the meeting, and are given in order of their presentation in each session.

An asterisk [*] in place of a page number in the Contents indicates summary not submitted in time for inclusion in this report.

A summary of the agenda is printed on the inside of the back cover. Blank note pages are also provided.

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27TH WATER REACTOR SAFETY INFORMATION MEETING
OCTOBER 25-27, 1999

Monday, October 25, 1999

SESSION 1: RISK-INFORMED REGULATION
(APPROACHES, METHODS, AND EXPERIENCE)
Chaired by: J. Murphy (NRC), S. Floyd (NEI)
Rapporteur: T. King
Objective: To show that RES results have been or will be useful in increasing/improving the use of PRA in regulatory decision-making.

SESSION 2: INTEGRITY OF PRIMARY COOLANT PRESSURE BOUNDARY - PART I
Chaired by: M. Mayfield (NRC)
Rapporteur: J. Craig
Objective: To describe one element of the work supporting the risk-informing of Part 50 and 50.61 (PTS Rule) in particular.

Tuesday, October 26, 1999

SESSION 3: ENHANCING REGULATORY EFFECTIVENESS
(GENERIC ISSUES, OPERATING EXPERIENCE, AND REQUIREMENTS)
Chaired by: S. Rubin (NRC), S. Rosen (STP, ANS Stds Steering Committee)
Rapporteur: C. Rossi
Objective: To illustrate some of the diverse ongoing and completed activities within the Office of Nuclear Regulatory Research and industry in support of enhancing regulatory effectiveness.

SESSION 4: INTEGRITY OF PRIMARY COOLANT PRESSURE BOUNDARY - PART II

Chaired by: E. Hackett (NRC), R. Dyles (Inservice Engineering)

Rapporteur: J. Craig

Objective: To describe the results of research responding to issues that have emerged from licensing actions or operating experience and to understand the response of nuclear plant primary system components to transients, active degradation and/or repairs.

SESSION 5: HIGH BURNUP AND MIXED OXIDE FUEL (REGULATORY CRITERIA AND METHODS)

Chaired by: R. Meyer (NRC), R. Yang (EPRI)

Rapporteur: C. Rossi

Objective: To discuss new research results, which will form the basis for regulatory criteria and evaluation models for high-burnup fuel, MOX fuel, and various cladding alloys.

SESSION 6: FUEL BURNUP CREDIT

Chaired by: R. Lee (NRC), S. Turner (Holtec Int'l.)

Rapporteur: C. Rossi

Objective: To provide (a) an overview of the regulatory and industry perspectives in the use of burnup credit, (b) an international perspective on how burnup credit is taken into account in criticality studies, and (c) experimental determinations of burnup credit of spent nuclear fuel.

Wednesday, October 27, 1999

SESSION 7: FISSION PRODUCT SOURCE TERMS AND APPLICATIONS

Chaired by: C. Tinkler

Rapporteur: C. Rossi

Objective: The objective of this session is to address the application of improved accident fission product source terms for operating reactors to improve safety and reduce regulatory burden and to describe the ongoing experimental research to better characterize fission product source terms.

SESSION 8: RESEARCH SUPPORTING NEW TECHNOLOGY

Chaired by: F. Coffman
Rapporteur: J. Craig
Objective: To explore safety implications related to new technologies by (a) describing current initiatives and results, (b) looking for common cautions and lessons, and (c) improving the general cognizance about such technologies.

System Reliability Studies

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This paper summaries the results of studies that were performed to determine the reliability of eight nuclear power reactor safety systems over a several-year operational period. The risk-important systems include High-Pressure Coolant Injection (HPCI), High-Pressure Core Spray (HPCS), Reactor Core Isolation Cooling System (RCIC), Isolation Condenser (IC), Auxiliary Feedwater (AFW), Emergency Diesel Generator Power System (EDG), Westinghouse Reactor Protection System (W RPS), and General Electric Reactor Protection System (GE RPS).

The objectives of each study were to: (1) estimate the system unreliability based on operating experience and to compare these estimates with the assumptions, models, and data used in probabilistic risk assessments and individual plant examinations (IPEs); and (b) review the operational data from an engineering perspective to determine trends and patterns in the data and provide insights into the failures and failure mechanisms associated with the operation of the system. Each study analyzed operating experience data contained in the Licensee Event Reports (LERs). The Sequence Coding and Search System database was used to identify LERs for review and classification, and the full text of each identified LER was reviewed from a risk and reliability perspective. Operational data from the Nuclear Plant Reliability Data System was used to supplement LER data for the reactor protection systems.

The results of each study are summarized in the table below, where a downward-sloping arrow indicates a decreasing trend and a horizontal arrow indicates no evidence of a trend.

Study	Mean Unreliability	Unplanned Demand Trend	Failure Rate Trend	Unreliability Trend	Consistency with PRA/IPEs	Unreliability vs Plant Age Trend
HPCI (1987-1993)	0.06	↘	↘	⇨	General agreement--few plants lower than operating experience	None
EDG--RG1.108 (1987-1993)	0.04	↘	↘	⇨	General agreement--fail-to-run higher in PRAs	None
IC (1987-1993)	0.02	⇨	⇨	⇨	General agreement--nature of failures differ	None
RCIC (1987-1993) Short (<15 min) Long (>15 min)	 0.04 0.2	↘	⇨	↘	General agreement--restart different in PRAs	None
HPCS (1987-1993)	0.08	↘	⇨	⇨	Fail-to-run contribution higher in operating experience	None
AFW (1987-1995)	3E-5	↘	⇨	⇨	Fail-to-run and suction contributions higher in operating experience	None
W RPS (1984-1995)	2E-5	↘	N/A	N/A	General agreement--reactor trip breaker contribution different	N/A
GE RPS (1984-1995)	6E-6	↘	N/A	N/A	One order of magnitude lower than IPEs	N/A

Development of Risk-Based Performance Indicators

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The NRC is moving toward more risk-informed performance-based oversight and regulation. The EDO tasking memo of August 25, 1998, detailed the work to be done in this area, which included the Performance Assessment Workshop conducted in September 1998. The development of risk-based performance indicators (RBPIs) is part of the long-term activities to improve the risk focus of NRC assessments of plant safety performance. The following describes the program plan activities which will continue through calendar year 2001. The paper and presentation will focus on the results of the development process including candidate risk-based performance indicators and the results of the supporting analyses.

There are several attributes that RBPIs should have. First, they should be related to plant risk by accident sequence logic so that a direct relationship exists between the indicator and safety performance expectations. This will make them meaningful and easily understood. Second, the data for producing the RBPIs should be readily available, clear, and objective. Third, the RBPIs should be easily measured or capable of being calculated from the data, and they should be able to be independently validated.

In addition to these attributes, the RBPIs should have the following technical characteristics: (1) they should be directly related to risk parameters such as frequency, reliability, availability, and probability; (2) they should cover a broad sample of the systems, structures, and components that are important contributors to risk; (3) they should be capable of accounting for differences in plant design and operation; (4) they should account for data density and variability when identifying generic and plant-specific performance changes; and (5) their thresholds for determining deviations from expected norms and establishing limits of acceptable performance should be able to distinguish between normal fluctuations and genuine trends of poor performance.

The development of RBPIs will be done in the following phases: (1) formation of an interoffice coordination group and the awarding of a technical support contract; (2) development of a candidate set of indicators; (3) trial application of the candidate indicators, including NRC and public comment; (4) development of final indicators based on public, ACRS, staff, and Commission review of the trial application; (5) development of an automated process for producing RBPIs; and (6) start of periodic publication of RBPIs.

Risk Perspectives Regarding Low Power Shutdown Operations

Mary Drouin, Erasmia Lois, (U.S. Nuclear Regulatory Commission)
Allen Camp, Timothy Wheeler, Donnie Whitehead, (Sandia National Laboratories)
Louis Chu, John Lehner (Brookhaven National Laboratory)

Past and recent nuclear power plant operational experience indicates that the risk associated with low power and shutdown (LPSD) operations could be significant. Therefore, it is important that the NRC considers the risk from LPSD conditions in its decision-making processes.

During the last few years the NRC has undertaken significant efforts for developing guidance on using risk information in its decision-making process (risk-informed regulation). Specifically, the NRC has published Regulatory Guide (RG) 1.174 which provides guidance for the use of risk information in regulatory decision-making regarding licensee requests to change their licensing basis. The guidelines of RG 1.174 are applicable to both full power and low power and shutdown operations.

The NRC initiated in early 1999 a program with the objective to provide (or develop, as necessary) an understanding of the risk associated with LPSD operations sufficient to support risk-informed decision-making and in particular to support the use of RG 1.174. This paper summarizes preliminary perspectives regarding how LPSD risk can be factored into the regulatory decision-making process, the strengths and weaknesses of existing methods and tools, and the need for any methods development.

Applications for Risk-Informed Regulation

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US Nuclear Regulatory Commission

In the past several years, the NRC staff's work to make its reactor regulatory requirements and processes more "risk-informed" has made several significant steps forward. This work has resulted in:

- Development and application of guidance on the use of risk information, in concert with traditional engineering information, for changes to the licensing bases of commercial nuclear power plants;
- Preliminary revision of the reactor oversight process (which includes inspection, enforcement, and assessment) to make greater use of objective performance indicators and to focus staff and licensee resources on the most safety-significant issues and plant activities; and
- Initiation of two programs to modify the NRC's fundamental reactor regulations, contained in 10CFR50, to provide:
    - risk-informed scope requirements and safety definitions, and
    - risk-informed modifications to technical regulations.

The paper will focus on recent work in the Office of Nuclear Regulatory Research for this last program, that is, providing modifications to technical regulations in 10CFR50. The paper will discuss: the staff's plans to define the process by which regulations are identified for change, and to develop the technical bases for making needed changes; and initial results, including the process and evaluation criteria to be used to identify "targeted" regulations, and the identification of specific regulations initially appearing to merit revision.

PRESSURIZED THERMAL SHOCK -- A PROGRAM TO REVISIT THE TECHNICAL BASIS

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In the late 1970's, a challenge to the integrity of embrittled reactor pressure vessels was identified that involved a rapid cooldown of the pressure vessel wall accompanied by either sustained high system pressure or a subsequent repressurization of the system. This challenge was termed Pressurized Thermal Shock (PTS). Working from the state of the art at that time for probabilistic risk assessment, thermal hydraulic analysis, vessel fracture analysis, and material embrittlement estimation methods, the NRC staff developed one of the first performance-based, risk-informed regulations to assure the safe operation of the pressure vessel. The pressurized thermal shock rule, 10 CFR 50.61, established an embrittlement screening criterion above which licensees would be required to demonstrate that their pressure vessels could be operated safely. The screening criterion was derived using both conservative deterministic analyses and risk concepts that established an acceptable probability of vessel failure that implicitly considered the conservative nature of the deterministic analyses. The NRC subsequently developed regulatory guidance, Regulatory Guide 1.154, on the format and content of analyses that could be used to demonstrate the continued safe operation of pressure vessels that would exceed the PTS screening criterion.

In late 1989 and early 1990, the staff and the licensee for the now decommissioned Yankee Rowe plant entered into an intensive evaluation of the pressure vessel for the Yankee Rowe plant. The staff had identified a high level of embrittlement for the pressure vessel and both the licensee and staff turned to Regulatory Guide 1.154 as a basis for evaluating the integrity of the pressure vessel. During the course of that evaluation, the staff and industry identified a number of shortcomings and limitations in the regulatory guide methodology; chief among them being the technical basis for the fabrication flaw distributions used in the probabilistic fracture mechanics analyses. The Yankee Rowe evaluation, as well as the earlier evaluations that had formed the basis for the rule and regulatory guide, demonstrated that the way flaws were modeled, using 1970's non-destructive examination data and the resulting Marshall flaw distribution, dominated the uncertainty in the calculated probability of vessel failure. Other variables were also shown to be important, such as the embrittlement estimation methods, the fracture toughness curves, and the thermal hydraulics calculations.

In the intervening years, the staff and the industry have worked both separately and jointly to improve the data bases for the flaw-related variables and for the other key variables. Recent analyses that combined these advances to evaluate the potential impact on acceptable levels of pressure vessel embrittlement have shown that the significant levels of conservatism in the screening criterion can be reduced.

The NRC has initiated a program to revisit the technical bases for the PTS rule and to, potentially, propose a revision to that rule that would significantly reduce the unnecessary level of conservatism in the rule. This effort has been undertaken as a full-participatory activity where active participation by the public and industry in evaluating the changes in the relevant technologies has been sought.

This paper will describe the issues and the progress made in the areas of probabilistic fracture analyses, thermal hydraulics, and probabilistic risk analyses, and the plans for the future work.

Developing a Generic Flaw Distribution for Reactor Pressure Vessels

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ABSTRACT

The U.S. Nuclear Regulatory (NRC) is re-evaluating the guidance and criteria in the code of federal regulations (CFR) as it relates to reactor vessel integrity, specifically pressurized thermal shock (PTS). Recent ultrasonic examination (UT) of considerable vessel material at Pacific Northwest National Laboratory (PNNL) and industry experience with Yankee Rowe have provided the NRC with a better understanding of PTS issues. The reevaluation of PTS will consider a risk-informed approach to the PTS rule and also provide important benefits for licensees considering license renewal.

Pressurized thermal shock transients can lead to reactor vessel failure. These transients have occurred at operating reactors but to date they have not resulted in vessel failure. To properly determine the potential or probability for vessel failure from a PTS event an accurate estimate of fabrication flaws is necessary. The characteristics of the fabrication flaw are inputs to fracture mechanics structural calculations which will determine the probability of vessel failure during a PTS event. Also the results will indicate the sizes and locations of flaws that are most likely to cause failures. This information is also an integral input to the overall pressure vessel safety program.

In order to obtain an accurate estimate of fabrication flaws to address PTS events for all classes of reactors a generic flaw distribution must be developed. An expert elicitation process will be used in conjunction with empirical data from PNNL RPV studies and modeling (RR PRODIGAL Code) in developing generic flaw distributions.

This paper will demonstrate the important relationship between reactor vessel integrity and flaw distributions in reactor pressure vessel material; present the PNNL work to date on developing flaw density and distributions for domestic RPVs; and describe the expert elicitation process that is being planned to verify that a generalized flaw distribution can be properly developed and then to assist in developing a generalized flaw distribution

Reactor Vessel Lower Head Failure Experiments and Analyses

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ABSTRACT

The lower head of the reactor pressure vessel can be subjected to significant thermal and pressure loads in the event of a postulated core meltdown accident; consequently, the possibility exists that the lower head will fail releasing large quantities of core material into the containment. The objectives of this NRC sponsored program are to characterize the mode, timing, and size of lower head failure (LHF). The program consisted of coordinated experimental and analytical efforts. The present paper provides a concise summary of the experimental and the modeling results of the LHF program.

A Phenomena Identification and Ranking Table (PIRT) was prepared, and a scaling analysis was performed at the start of the program to guide the design and conduct of the experiment program. The experiment program consisted of eight experiments. Each test vessel was a 1/5-scale model of a typical PWR lower head made from prototypic materials (SA533B1 steel). By using geometrical scaling and prototypical material, membrane stress and material behavior were preserved in the experiments. Each experiment was pressurized and heated from the inside until failure occurred.

The test matrix addressed issues of heating patterns, lower head penetrations, lower head weldments, and reactor coolant system (RCS) pressure. Of the eight experiments, seven were performed at a RCS pressure of 10 MPa (nominal membrane stress of 75 MPa) and one experiment was performed at 5 MPa (nominal membrane stress of 37.5 MPa). One replicate test was performed. The key experimental observations are:

- Failure size was typically smaller than the heated region and localized heating led to localized failure;
- Failure typically initiated near locations of high membrane stress (due to variations in wall thickness or temperature or both) and the membrane stress at failure initiation was typically 40% to 60% of yield. Therefore, failure was due to creep;
- Penetrations can lead to premature local weld failure;
- Change of membrane stress (pressure) can result in significant changes in the characteristics and size of local failure; and
- The experiments were highly repeatable and, therefore, amenable to modeling.

The results of the experiments were analyzed using both FEM numerical simulation and Larson-Miller correlation-based system code models. An exhaustive review and analysis of existing material database were made, and selected property measurements were made to anchor the diverse material database. A power-law secondary creep constitutive model was implemented in the ABAQUS code. The FEM analyses show good agreement with experimental observations including time to failure and failure location. Vessel failure time calculated from the system code was shown to be in good agreement with experiments. One important lesson-learned in the analysis program was the need to develop accurate constitutive model for the specific material used in the experiment.

Reactor Pressure Vessel Embrittlement: The Road Ahead

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The last decade has seen remarkable progress in developing a mechanistic understanding of irradiation embrittlement of reactor pressure vessel steels. This has been exploited in formulating robust, physically based and statistically calibrated predictive models of Charpy V-notch indexed transition temperature shifts. These semi-empirical models account for key embrittlement variables, and variable interactions, including the effects of copper, nickel, phosphorous, fluence, flux and temperature. Models of nanoscale precipitates rich in copper, manganese and nickel are quantitatively consistent with experimental observations of the complex interplay between these elements and other embrittlement variables. The models also rationalize other effects, such as those associated with post weld heat treatment and many aspects of the interactive flux and temperature dependence of embrittlement. Models have been extended to treat post irradiation annealing and re-embrittlement based on tracking the fate of key alloy constituents and defects. Finally, revolutionary advances have been made in directly measuring fracture toughness using a relatively small number of relatively small specimens based on the master curve (MC) concept.

Thus it is appropriate to reflect the significance of potential technical surprises as well as new opportunities for relaxing embrittlement margins. This presentation will attempt to provide explicit assessments of a number of such inter-related issues and opportunities, including:

Quantification of the effect of thermal-mechanical processing history on the effective copper (and other elements) content of steels including the role of alloy composition and microstructure.

Better quantification of the effects of product form and secondary elements, like phosphorous and manganese.

Characterizing the variables that result in the large scatter and significant uncertainties in the so-called matrix defect contribution to embrittlement, which has a weak or negligible copper dependence, but can be significant, particularly at high fluence.

Characterizing the effects of very long irradiation times and/or very low flux.

Assessing the potential for late blooming phases, that may result in a rapid acceleration of embrittlement at high fluence even in low copper steels.

Evaluating through vessel thickness property gradients, including effects of material variability and the effective attenuation of dose.

Establishing a firm micromechanical underpinning for the empirical MC, including effects of irradiation on MC shifts and shape and their relations to initial properties, Charpy shifts and irradiation microstructures and hardening.

The use of surrogate materials and the corresponding implications of corresponding uncertainties to actual vessel embrittlement predictions.

Ongoing and future research efforts to address these issues are described, including the potential contributions from micro-micromechanics, non-intrusive biopsy techniques, and post-service vessel inspections.

OVERVIEW OF IRRADIATION EFFECTS ON FRACTURE TOUGHNESS AND CRACK-ARREST TOUGHNESS OF RPV STEELS*

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SUMMARY

The safety of commercial light-water reactors (LWRs) is highly dependent on the structural integrity of the reactor pressure vessel (RPV). The degrading effects of neutron irradiation on carbon and low-alloy pressure vessel steels have been recognized and investigated since the early 1950s. In those steels at LWR operating temperatures (~288°C), radiation damage is produced when neutrons of sufficient energy displace atoms from their lattice sites. The defects formed in the steel as a result of those displacements typically cause hardening and a decrease in toughness. Tensile behavior exhibits an increase in yield strength, a decrease in the ultimate to yield strength ratio, and a loss of ductility as measured by specimen elongation. The decrease in toughness is most commonly represented by an increase in the ductile-brittle transition temperature and a decrease of the upper-shelf energy as measured by the Charpy V-notch (CVN) impact test. The synergistic effects of neutron fluence, flux, and spectrum, the irradiation temperature, and the chemical composition and microstructure of the steel must be understood to allow for reductions in uncertainties associated with the development of predictive models of embrittlement. The CVN toughness, however, is a qualitative measure which must be correlated with the fracture toughness and crack-arrest toughness properties, K_{Ic} and K_{Ia}, necessary for structural integrity evaluations.

During the 1960s, it was well recognized that the effects of irradiation could degrade the materials, but definitive effects on fracture properties, especially in thick sections, were not available. Then, the field of fracture mechanics was in the early stages and even the amount of data on other material properties under LWR conditions was deficient. When the Heavy-Section Steel Technology (HSST) Program was initiated in 1967, the U.S. Atomic Energy Commission had, in fact, already sponsored two irradiation effects projects, and the HSST Program assumed managerial responsibility for them and for the formulation of plans for extensions of those projects. The results from those early programs were important in that they showed irradiation-induced degradation of fracture toughness, a strong temperature dependence of postirradiation fracture toughness, a need for larger specimens, and that the K_{Ic} temperature shift was about the same as the CVN 41-J shift. At about the same time, in late 1968, Potapovs and Hawthorne had reported that some residual elements, particularly copper, increased irradiation sensitivity. Subsequently, Title 10, Part 50 of the *Code of Federal Regulations* (10CFR50) evolved to include requirements for fracture toughness of RPVs. Those requirements included surveillance testing with CVN specimens and required fracture toughness specimens if the surveillance materials were predicted to exhibit marginal properties. Furthermore, 10CFR50 requires prediction of radiation effects using *Regulatory Guide 1.99* (Rev. 2). Additionally, screening criteria are specified for

*Research sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, under Interagency Agreement DOE 1886-N695-3W with the U.S. Department of Energy under Contract No. DE-AC05-96OR22464 with Lockheed Martin Energy Research Corp.

toughness transition temperatures which, if attained by the surveillance tests or by prediction, require plant-specific analyses to demonstrate adequate protection against pressurized thermal shock (PTS). As part of those requirements, 10CFR50 refers to the ASME *Boiler and Pressure Vessel Code,* Sects. III and XI, for fracture toughness and ASTM E 185 for surveillance testing and analysis as well as application of the test results. Moreover, *Regulatory Guide 1.154* incorporates estimates of the variability in fracture toughness and crack-arrest toughness for the critical RPV material in PTS analyses. Thus, it is important to recognize that the developments of fracture mechanics and knowledge of irradiation effects in RPV steels have occurred concurrently; moreover, the rate of developments in fracture mechanics have been significantly compelled by radiation effects research on RPV steels.

In 1972, the HSST Program began a series of irradiation experiments in response to the need for information regarding effects of neutron irradiation on the mechanical properties, particularly fracture toughness, of RPV steels. [In 1989, the HSST Program irradiation effects task was organized into a separate HSSI Program.] The eight completed projects include irradiation effects on (1) dynamic fracture toughness; (2) and (3) ductile tearing resistance; (4) state-of-the-art welds; (5) and (6) temperature shift and shape of K_{Ic} and K_{Ia} curves; (7) stainless steel cladding; (8) commercial low upper-shelf welds; and (9) thermal annealing. Two ongoing projects include effects of irradiation on the shape of the fracture toughness master curve for highly embrittled steel, and the effects of irradiation, post-irradiation thermal treatment, and reirradiation on the propensity for temper embrittlement in heat-affected zones. Concurrently, a number of other research programs on radiation effects in RPV steels were also in progress within the United States and in other countries. As the developments in fracture mechanics have led from the linear-elastic to the elastic-plastic regimes, the specimen size requirements for measuring fracture toughness has significantly decreased. At ORNL, the HSSI Program has irradiated and tested a large number of fracture toughness specimens up to and including 100-mm in thickness (4T). Such experiments have provided results leading to understanding of specimen size effects and, in fact, have contributed significantly to the development of ASTM standard test methods which allow the use of relatively small specimens (e.g., 0.5T and precracked CVN) to establish the material fracture toughness. The results have also provided important data regarding variability of fracture toughness in RPV steels which allow for statistically-based analysis. Such results have been obtained both for cleavage initiation in the ductile-brittle transition region and for ductile tearing resistance. Similar experiments have provided key results on the effects of irradiation on crack-arrest toughness and have demonstrated a decrease in the temperature difference between the K_{Ic} and K_{Ia} curves with irradiation embrittlement. Statistical analysis of data from around the world have indicated that the assumption of equivalent irradiation-induced CVN 41-J and fracture toughness transition temperature shifts may not be appropriate for all RPV steels. The experimental programs have provided key results regarding the fracture behavior of RPV steels under conditions of irradiation, thermal annealing, and reirradiation, to include effects of copper and nickel content, relationships between Charpy impact toughness and fracture toughness/crack-arrest toughness, specimen size effects, and statistical variability. Relative to PTS analysis, the results are directly applicable as the calculations of failure probability are directly dependent on the initiation and arrest toughnesses of the materials. The existing procedures for determining reference temperatures and associated uncertainties bear reevaluation with regard to the results obtained. Remaining issues regarding fracture toughness of RPV steels include limits of applicability of small specimen measurements and the associated uncertainties, effects of high levels of embrittlement on both fracture toughness and crack-arrest toughness, potential effects of so-called "late blooming phases," the relationship between CVN toughness and fracture toughness for steels following post-irradiation heat treatment and reirradiation, and effects of dynamic loading and intergranular fracture on the shape of the fracture toughness master curve.

THE EFFECTS OF DEREGULATION OF THE ELECTRIC POWER INDUSTRY ON THE NUCLEAR PLANT OFFSITE POWER SYSTEM: AN EVALUATION

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This report presents recommendations addressing the potential effect of deregulation of the electric power industry on the nuclear plant offsite power system as requested in a staff requirements memorandum, "Briefing on Electric Grid Reliability, April 23, 1997, and Briefing on Electric Utility Restructuring, April 24, 1997," May 27, 1997. This report also describes the offsite power system, discusses the principal criteria for evaluating the effects of deregulation on the nuclear plant offsite power system and the potential impact of deregulation on the nuclear plant offsite power system, and presents a review of the various staff risk-informed and engineering-based initiatives to evaluate deregulation issues related to the nuclear plant offsite power system. Considered with the U.S. Nuclear Regulatory Commission (NRC) initiative to evaluate the regulatory effectiveness, this report is an example of the Commission evaluating an emerging issue to ensure that the regulations would be effective in maintaining an adequate level of safety.

Evaluations performed by the staff indicate that the potential increase in risk resulting from grid-related loss of offsite power (LOOP) events due to deregulation is likely to be low; however, the staff will continue to monitor grid reliability and take action, as needed. For example, the North American Electric Reliability Council (NERC) reliability assessments and site visits indicate common grid reliability concerns. While the Nuclear Regulatory Commission does not have jurisdiction over operation of the grid, Information Notice 98-07, "Offsite Power Reliability Challenges From Industry Deregulation," February 27, 1998, alerted licensees to the potentially adverse effects of deregulation of the electric power industry on the reliability of the offsite power source. Consequently, nuclear power plants are expected to prepare for these concerns by ensuring that plant features for coping with LOOP and station blackout (SBO) events are properly monitored and maintained. In addition to the appropriate command, control, and communication infrastructure with the grid-controlling entity, existing regulatory controls should ensure the reliability of emergency power generators and the adequacy of protective relays and alarms for the switchyard and emergency buses.

As noted in SECY 99-129, "Effects of Electric Power Industry Deregulation on Electric Grid Reliability and Reactor Safety," May 11, 1999, the NRC will continue to promptly assess LOOP events as part of the inspection program and also as part of the accident sequence precursor (ASP) program. For events that exceed the ASP threshold of 1E-6, further review will be performed, where appropriate, to obtain plant-specific and potential generic insights concerning the event.

In addition, review of the NERC grid-reliability forecasts and follow-up discussions, as required, appear to be the most practical means of assessing the potential impact of deregulation on the offsite power system. Continued contact with NERC, the Federal Energy

Regulatory Commission, and the Electric Power Research Institute will also enhance the NRC's understanding of potential deregulation issues related to grid reliability.

On the basis of the staff's evaluation of the initiatives completed to date, the following recommendations were developed and subsequently noted in SECY 99-129.

(1) The staff will take no further regulatory action to address grid reliability associated with the deregulation issue.

(2) To ensure that the licensing basis is maintained, the staff will follow up on the NERC and site visit concerns, risk-informed analyses, operating experience, and ASP evaluations as follows:

(a) The staff will evaluate the adequacy of (i) the existing technical guidance on offsite power and voltage issues, (ii) the degraded voltage protective relay setpoints, and (iii) the scope of the offsite power system frequency protection, including whether the existing reactor coolant pump underfrequency protection could lead to unnecessary trips. These actions will ensure that plant ac safety equipment remains protected from abnormal offsite system voltages and frequencies.

(b) The staff will investigate causes of diesel generator unreliability identified from INEL-95/0035, "Emergency Diesel Generator Power System Reliability 1987–1993," February 1996. The staff will continue to assess the reliability of the onsite diesel generators to ensure that the reliability is maintained consistent with the risk studies used to develop the SBO rule (Title 10 *Code of Federal Regulations* [CFR] Part 50 Section 63).

(c) The staff will continue to assess significant LOOP events that are reported in accordance with 10 CFR 50.72 and 50.73, for prompt review as part of the inspection program. The 10 CFR 50.73 LOOP events will also continue to be reviewed as part of the ASP program. Follow-up action will be considered, as indicated by the inspection program, for LOOP events that either meet or exceed the ASP conditional core damage probability of 1E-6, or that last longer than the national median time of approximately 30 minutes.

(d) The staff will remain cognizant of the current status of grid issues, and will assess future electric power grid reliability and its potential impact on nuclear power plants' offsite power systems through its continued contacts with NERC, the Federal Energy Regulatory Commission and the Electric Power Research Institute.

The New NRC Generic Issue Resolution Process Including Technical Results from GI-158 (Performance of Safety-Related Power Operated Valves Under Design Basis Conditions) and GI-165 (Spring-Actuated Safety and Relief Valve Reliability)

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Abstract

The Generic Issues Program first began formally in response to a Commission directive in October of 1976. In 1983, it became one of the first programs to make successful use of probabilistic risk information to aid in regulatory decisionmaking. In the 16 years since the program became quantitative, 836 issues have been processed. Of these, 106 reactor safety issues were prioritized as requiring further evaluation to determine the final resolution. Approximately a dozen generic issues remain unresolved. Although there is far less reactor licensing activity than in the 1970s, new issues continue to be identified from research and operational experience. These issues often involve complex and controversial questions of safety and regulation, and an efficient and effective means of addressing these issues is essential for regulatory effectiveness. Issues which involve a significant safety question, require swift, effective, enforceable, and cost-effective regulatory actions. Issues that are of little safety significance, must be quickly shown to be so and dismissed in an expeditious manner so as to avoid unnecessary expenditure of limited resources and to reduce regulatory uncertainty. Additionally, in the time since the generic issue program began probabilistic risk assessment techniques have advanced significantly while agency resources have continued to diminish. Accordingly, the paper discusses the steps that have been taken to enhance the effectiveness and efficiency of the generic issue resolution process. Additionally, two recently-resolved issues are discussed, along with key elements of a proposed new procedure for resolving potential generic issues.

Opportunities and Issues in the Graded Quality Assurance, Robust Fuel and Probabilistic Risk Analysis Standards Development Programs

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The STP Nuclear Operating Company is the pilot plant for the industry with the NRC in implementing a Graded Quality Assurance (GQA) program using risk insights. A robust component risk significance determination process is at the core of the GQA program. Certain cost/benefit analyses were completed by STP prior to undertaking participation in this effort. In terms of regulatory effectiveness, certain overlapping regulations have been encountered by STP that have put into question the validity of the original cost/benefit analyses threatening the continuation of the effort.

EPRI is conducting a research effort funded by the utilities to create the requirements for more robust fuel for operation in the next millennium. This Robust Fuel Program will lead, it is planned, to reactor fuel with current performance issues resolved and more operating margin at higher burnups. One of the issues currently burdening the program is an NRC Reactivity Insertion Accident (RIA) definition that is not risk-informed. In terms of regulatory effectiveness, this lack of cohesion with the NRC's PRA Policy Statement distorts the Robust Fuel Program's research plan and weakens stakeholder involvement in the program.

The industry, with full NRC staff participation, has undertaken the development of a complete set of Probabilistic Risk Analysis (PRA) standards. The American Society of Mechanical Engineers (ASME) has drafted a PRA standard for internal events. The American Nuclear Society (ANS) has begun the development of PRA standard for low power, shutdown and external events. The draft ASME standard was criticized by some who felt that all the standard's requirements applied regardless of the intended uses of the PRA. That standard is currently being redrafted.

Consensus standards derive their usefulness from their technical content and from the consensus process that brings them into being. Standards are weakened if there is a lack of confidence in the minds of the stakeholders in either the standard's technical content or the consensus process. In terms of regulatory effectiveness, the NRC staff and the industry must keep these principles in mind as they work on the standards committees to help fashion a useful product.

Reducing Unnecessary Burden for Decommissioning Rule Financial Assurance Requirements

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The Problem:

In the NRC rules for decommissioning nuclear power reactors there are requirements for licensees to periodically update the estimates for the cost of decommissioning their reactors in 10 CFR 50.75 and 50.82. These estimates are part of the financial assurance provisions included in the rules to ensure that at any time during the life of the plant, prior to actual license termination by the NRC, money will be available to complete decommissioning. One of these updating estimates is required annually, from the time of initiation of operations (10 CFR 50.75). This requirement is important because of concern for possible premature plant closure situations. These annual estimates must be performed by using a rule specified formula. The formula includes the cost of waste disposal. In the rule, reference is made to the use NUREG-1307 to obtain the parameter value needed for calculating this cost. This parameter value is updated by the NRC, when necessary, by annually examining the availability and associated monetary costs for disposal of the low-level-waste at licensed LLW disposal facilities. When the rule was originally developed, there was no concern about the cost of decommissioning waste disposal becoming a significant contributor to the total cost of decommissioning.

Since the rule became effective, the cost of LLW waste disposal at the licensed disposal facilities has increased significantly, and the cost estimate for this component has became a major factor in the total cost of reactor decommissioning. Current evaluations show that for many licensees estimates of total decommissioning cost, the cost of waste disposal has become the major cost contributor.

Industry viewed increasing LLW disposal costs as having become a major burden in demonstrating financial assurance. They focused efforts on developing alternatives for managing and disposing of the LLW waste that they generated and developed strategies for effectively reducing this cost. These economies are achieved through a combination of measures that include; (1) procedures that minimize the generation and package volume of this waste, (2) utilization of waste vendors that charge lower rates because of their efficient handling and treatment of the large amounts of waste they routinely process from a variety of sources, and (3) using reduced cost LLW disposal facilities that are licensed to accept only very low activity waste. Unfortunately, the existing regulatory structure did not have the flexibility to permit licensees to incorporate the use of these cost saving activities into their required cost estimate as a means of reducing the burden of financial assurance.

Possible Resolution of Problem:

The rule could be amended to permit the cost estimate to include the results of established cost reducing methods. However, if rule amendment were pursued, it would be unlikely to provide the near term relief sought by industry. Industry believed that a faster solution for relief was to request that the NRC use more flexible implementation guidance for the cost estimating requirements prescribed in the rule, and requested revision of NUREG-1307.

The NRC staff supported this approach and NUREG-1307 was revised to include the cost, for most of the decommissioning LLW requiring disposal, that would result if it were shipped to a generic waste vendor. For flexibility, the licensee could also choose to ship the waste directly to a LLW facility, as before. Because the changes discussed were agreed to with only about a month remaining before new cost estimates would be required from the licensees, the NRC facilitated the effort required to revise NUREG-1307. In less than a month's time, NUREG-1307, Rev. 8 was publically available for use. This NUREG was also made available on the NRC Web site for quick public access.

Changes to NUREG-1307:

NUREG-1307, Revision 8, now contains two options for estimating the cost of decommissioning waste disposal. The option selected is at the discretion of the licensee. In the first option, the annual cost adjustment for waste disposal for the available licensed LLW disposal sites to use in the rule-prescribed formula is presented based on evaluations that are similar to those used in past NUREG revisions. In the second option, the annual cost adjustment is based on a new evaluation that uses a generic cost rate for LLW disposal based on the charges a licensee would incur had most of the waste been directly shipped to a LLW vendor for disposal.

Impact on Financial Assurance Burden and Regulatory Effectiveness:

Use of the waste vendor (second) option can result in significant reduction in the decommissioning waste estimate. For example, a PWR licensee using the second option can estimate the disposal cost of the decommissioning waste (with a small amount going to Barnwell- which most licensees have as their only alternative) to be about $200 million less than the estimate based on the first option. While a rule amendment is still required to fully include the effect of all the cost reducing methods available to licensees, the modified implementation now available to the licensees in NUREG-1307 takes account of cost reducing measures that can result in significant reductions in the cost estimate. From the efficiency aspect of regulatory effectiveness, these changes allow more efficient application of industry resources by reducing the amount of funds which need to be set aside to offset decommissioning costs. Regulatory effectiveness also dictates that regulatory decisions should be made without undue delay and the facilitated revisions of NUREG-1307 to allow timely use of the new cost estimates is an example of that.

Synergistic Failure of BWR Internals

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BWR core shrouds and other reactor internals important to safety are experiencing intergranular stress corrosion cracking (IGSCC). The United States Nuclear Regulatory Commission (NRC) has followed the problem, and as part of its investigations, contracted with the Idaho National Engineering and Environmental Laboratory (INEEL) to conduct a risk assessment. The overall project objective is to assess the potential consequences and risks associated with the failure of IGSCC-susceptible BWR vessel internals, with specific consideration given to potential cascading and common mode effects.

An initial phase has been completed in which background material was gathered and evaluated, potential accident sequences were identified, and a qualitative PRA was performed to rank the sequences as having a high, medium, or low potential to significantly change the core damage frequency. A second phase is underway to perform a simplified, quantitative PRA on a representative high-power BWR/4. The existing PRA for the plant has been upgraded and modified for the project, including:

1. Introducing information from the latest IPEs
2. Including PRA branches that were not used in previous PRA studies because of low probabilities
3. Separating the main steam line, main feedwater line, and recirculation line breaks into three individual events, each with its own probability of occurrence
4. Introducing PRA branches representing IGSCC-induced failures of reactor internals components, starting with the jet pumps

The initial calculations that have been performed considered sequences initiated by jet-pump failures. To help establish which sequences would have negligible effects on core-damage frequency (CDF), and to identify the sequences that were more probable in increasing the CDF, parameter studies were conducted with the PRA model to determine which sequences could be screened as negligible contributors to CDF, and with structural analysis to determine probabilities of cascading failures. The generic BWR CDF for internal events and internal flooding, internal fires, seismic events, and low-power and shutdown operation was estimated to be on the order of 5×10^{-5} events/rx/yr. Based on the assumption that a 10% increase in CDF is significant to risk, and that there were approximately 50-100 scenarios to evaluate, a screening level for each sequence of 1×10^{-7} events/rx/yr was chosen for the preliminary calculations. Sequences involving failure of the reactor protection system and emergency core cooling system were evaluated.

In parallel, structural calculations were performed to assess damage to adjacent components that could result from jet pump failures. Thermal-hydraulic studies were conducted to calculate the flow rate through the jet pumps, and the flow velocities in the annular region surrounding the jet pumps. IGSCC history, energy required for failed jet pump parts to migrate to nearby components, and energy required to damage adjacent components were considered. The results showed that there was insufficient energy for loose jet pump parts to migrate upward to damage components such as the core spray system. Loose jet pump parts could contact adjacent jet pumps, the reactor vessel wall, the baffle plate and covers, the core shroud, and core shroud tie rods. Damage to these components would not be expected except for the following:

1. If the adjacent jet pump, baffle plate cover, and core shroud were already very severely damaged by IGSCC, these components could fail
2. The plant chosen for evaluation did not contain core shroud tie rods, and therefore the strength of the tie rod end connections was not evaluated

Preliminary results show that the cascading failure of an adjacent jet pump or baffle plate covers would not result in a significant increase in CDF. If a core shroud were very severely damaged, it is expected that inservice inspection would have detected the degradation, and repair methods such as tie rods would have been installed. If a tie rod were impacted by loose jet pump part, the rod itself would not fail, but since end connections may vary in design, these locations require further evaluation. Loose parts from a failed jet pump might migrate through the lower core plenum, and back up into the core, which could block control rod insertion or block coolant from sufficiently reaching a fuel channel. However, the preliminary studies show that there is a very low probability of this sequence affecting CDF.

The methodology is currently being extended to other major reactor internals components. A finite element model of the reactor internals has been developed and load cases for static, accident, and seismic loads are being evaluated to establish component fragilities. Inservice inspection methods are being evaluated to assist in developing estimated probabilities of crack detection. The advice of two expert panels, one on estimating crack growth rates and the other on developing the overall PRA methodology, are being used. This information will be used in the modified PRA to estimate the changes in the CDF that might result from IGSCC-induced failures of BWR internals.

Evaluation of Environmental Effects on Fatigue Life of Primary Piping

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Recent data indicate that the effects of light water reactor environments can significantly reduce the fatigue resistance of materials, and show that the American Society of Mechanical Engineers design fatigue curves for carbon, low alloy, and austenitic stainless steels may not be conservative for nuclear power plant primary system environments. Using revised fatigue curves developed by Argonne National Laboratory (ANL), the study described in this paper calculates the expected probabilities of fatigue failures and associated core damage frequencies at a 40-year and 60-year plant life for sample components in the reactor pressure boundary (piping and reactor pressure vessel components of five PWR and two BWR plants). These calculations were made possible by the development of a new version of the pc-PRAISE probabilistic fracture mechanics code that has the ability to simulate the initiation of fatigue cracks in combination with a simulation of the subsequent growth and linking of these fatigue cracks.

The calculations indicate that critical components with the highest probabilities of failure can have through-wall crack frequencies that are on the order of about $5x10^{-2}$ per year. The corresponding maximum contributions to core damage frequencies are on the order of 10^{-6} per year. These estimates are subject to many uncertainties, including the estimated values for conditional probabilities for larger leaks (small and large LOCAs). The calculated core damage frequencies for the components with the highest failure frequencies show essentially no increase in core damage frequency from 40 to 60 years.

The present study was based on the cyclic stresses that the components are projected to experience during their 40- and/or 60-year plant life. The stresses were taken from the information (NUREG/CR-6260) presented by the Idaho National Engineering and Environmental Laboratory (INEEL), which in turn were extracted from conservatively calculated values given in design stress reports for the plants. The ANL fatigue curves provide the needed statistical model of the number of cycles to crack initiation (assumed to be a 3-mm crack in a fatigue test). The number of cycles to crack initiation in the ANL equations is a function of the material type, water/air environment, temperature, dissolved oxygen content, sulfur content and strain rate. In this study, the fatigue damage due to various stress amplitudes was calculated by Miner's rule, using fatigue SN curves that account for the statistical distribution of the cycles to crack initiation. In the present calculations, the probability of crack initiation was taken to be equal to the probability that the cumulative usage factor is greater than one. It was also assumed that small initiated fatigue cracks grow based on fracture mechanics rules.

The probability that a 3-mm crack becomes a through-wall crack is computed using a Probabilistic Fracture Mechanics Code for Piping Reliability Analysis (pc-PRAISE code). New features of the code that were developed to support the present calculations included the simulation of the initiation and growth of cracks at multiple sites around a pipe circumference and the prediction of the linkage of these cracks to form potentially long through-wall cracks. The effects of through-wall stress gradients on the growth of initiated cracks were also included in the calculations.

A final part of the study estimated the consequences (i.e., core damage frequencies) of through-wall cracks in the various components addressed by the INEEL fatigue evaluations. Consequences were quantified by estimating conditional core damage probabilities given the occurrence of a through-wall crack resulting in small or large leak rates. The calculations for failure consequences accounted for the fact that through-wall cracks will usually result in only small leaks that have little or no safety significance. Larger leak rates can result in core damage, but the conditional probabilities of core damage can still be relatively low because nuclear power plants include safety systems which are designed to mitigate the consequences of leaks and breaks. Estimates of conditional core damage frequencies were based on: 1) probabilistic fracture mechanics calculations which predicted the probability that a given through-wall crack would cause various leak rates or pipe breaks, and 2) published data from probabilistic risk assessments which gave conditional probabilities of core damage given small leaks, large leaks, and pipe breaks.

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STRUCTURAL EVALUATION OF ELECTROSLEEVED TUBES UNDER SEVERE ACCIDENT TRANSIENTS

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ABSTRACT

Structural integrity of Electrosleeved steam generator tubing under severe accident transients was analyzed by analytical models, using available material properties data and results from high temperature tests conducted on Electrosleeved tubes. The Electrosleeve material is almost pure Ni and derives its strength and other useful properties from its fine-grained structure, which is stable at reactor operating temperatures. However, it undergoes rapid grain growth at the high temperatures that are expected during severe accidents, resulting in a loss of strength and a corresponding decrease in the flow stress. The magnitude of this decrease depends on the time-temperature history during the accident.

It was assumed that the material has a unique "unaged" flow stress vs. temperature curve representing the flow stress of the material before any grain growth occurs. An analytical model for predicting loss of flow stress of the Electrosleeve material with time and temperature was developed on the basis of Hall-Petch equation. A parabolic grain growth law for isothermal exposure was generalized for non-isothermal conditions. The activation energy for grain growth was obtained from data supplied by Framatome Technologies, Inc. (FTI). A flow stress model was used to predict failure of Electrosleeved tubes with throughwall axial cracks in the parent tubes.

Failure tests were conducted at ANL and FTI on internally pressurized Electrosleeved tubes with 100% throughwall machined axial notches in the parent tubes that were subjected to simulated severe accident temperature transients. The test results together with the analytical model were used to estimate the "unaged" flow stress curve of the Electrosleeved material at high temperatures. Failure times and temperatures for Electrosleeved tubes with throughwall axial cracks of various lengths in the parent tubes were calculated for several postulated severe accident transients.

Risk-Informed Considerations In the Evaluation of Steam Generator Tube Integrity

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In recent years, the nuclear industry has experienced various forms of steam generator tube degradation which have adversely affected operational performance. This issue has led to the development of improved techniques for detection and repair of tube cracking, as well as new methods for analyzing and predicting operational performance of tubes left in service. In accordance with NRC requirements, licensees perform these assessments in the context of normal operations and design basis accident conditions, and do not explicitly consider severe accidents and risk. Nevertheless, we know that steam generator tubes play an important role in certain classes of severe accidents, and are an important contributor to controlling overall plant risk. Consequently, the NRC has considered severe accidents in its evaluations of generic industry initiatives and plant-specific applications related to tube cracking. This paper describes how severe accident technical issues have been addressed in the NRC's evaluation of the risk implications of steam generator performance. In addition, the author will discuss an ongoing agency initiative to define the regulatory framework in which risk considerations can be applied to steam generator tube degradation and other licensing issues.

Outline of a RCCV Seismic Proving Test Results

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Abstract

Concrete containment vessels in nuclear power plants are the final barriers to secure radioactive materials from releasing to surroundings. CCVs are required to maintain airtight even under severe earthquakes. In the past, several static and/or dynamic tests were conducted to confirm the structural reliabilities of RCCVs under severe earthquakes. But in those tests, air-tightness, was not verified dynamically under seismic test.

Nuclear Power Engineering Corporation (NUPEC) conducted the shaking table tests of RCCV using large-scale (1/8) model with liner plate. The configuration scale is 1/8, and wall thickness is 1/10 scale considering to fail the specimen and the construction capabilities. The liner plate thickness is 1/4 scale determined by welding capabilities. The test was carried out employing the large-scale and high-performance shaking table at Tadotsu Engineering Laboratory of NUPEC in 1998 , 1999.

The shaking table test of RCCV with liner plates was conducted focusing not only structural integrity but also functional soundness. From this test, the following major results are obtained.

- The structural integrity was confirmed with the seismic excitation of design level Earthquakes and the functional soundness was also confirmed with measuring air tightness after the design level excitations
- Failure mode of the specimen was the shear mode and the shearing sliding failure occurred at the side of the A/D access hatch.
- The maximum input acceleration was about 9 times that of S2 and seismic margin of the specimen is supposed more than 5 times the S2.
- The observation result of sectional state at the concrete cored boring hole of the wall after failure. It was found that the liner plate and the concrete were considered to behave uniformly together up to the failure and functional soundness of the model was supposed maintained up to the failure.
- From the test results, the structural characteristic behavior was obtained and the transition in stiffness deterioration rate /damping factor of about 5% of the specimen at the S2 level were obtained.
- The test results and the post-test analysis results up to S2 were compared. The analytical results of FEM model for static analysis and the lumped mass model for dynamic analysis showed good agreements with the test and it was found hat these analytical methods are effective evaluation methods of the test. Furthermore, the appropriateness of the design method was confirmed from the analytical results.

Deficiencies in Regulatory Criteria for High-Burnup Fuel and NRC Research to Correct the Deficiencies

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Our attention is focused on postulated events that involve significant fuel damage. These events have the potential to cause fuel melting, if not kept within certain bounds, and melting could produce a large fission product release and significant consequences (hence risk). While there may be many pathways leading to such events, there are only two ways to melt fuel. One is to lose the coolant and the other is to get excessive power in the fuel. Selected design-basis accidents are postulated to serve as bounding examples of these kinds of events, and fuel damage limits are used to ensure that coolable core geometry is not lost, thus avoiding significant consequences.

Two main design-basis accidents are postulated to provide this bounding protection. One is a reactivity accident and the other is a loss-of-coolant accident (LOCA). Particular varieties of these design-basis accidents are evaluated for PWRs and BWRs. The fuel damage limits that are supposed to ensure a coolable core are 280 cal/g peak fuel enthalpy (Regulatory Guide 1.77) for the reactivity accidents and 17% cladding oxidation and 2200°F peak cladding temperature for the loss-of-coolant accidents (10 CFR 50.46). Related evaluation models are used to demonstrate that the regulatory criteria are not exceeded.

A few years ago, we learned from experimental work on reactivity accidents in France and Japan that cladding failure accompanied by fuel dispersal (loss of geometry) could occur at fuel enthalpies below 100 cal/g for high-burnup fuel. This is far below the current regulatory criterion. Further, the failures were occurring by a brittle mechanical fracture mechanism rather than the high temperature process seen in the original studies on fresh fuel, from which the current criterion was derived. In addition, we learned from work in Russia that different cladding alloys may behave in a completely different manner depending on their ductility. To complicate matters, the U.S. industry is now using three distinctly different PWR cladding alloys and two different BWR cladding types, all of which may have different mechanical properties in their highly irradiated state.

Early experimental work in France on loss-of-coolant accidents does not show that the oxidation and peak cladding temperature limits are inadequate, but the situation is very clouded. In extreme cases, fuel rods in commercial reactors have accumulated nearly 17% cladding oxidation during normal operation. The extent to which this reduces the amount of oxidation that can be tolerated during the accident is not known, and thus the NRC is currently taking the conservative position that the sum of transient and steady-state oxidation should be limited to 17%. Even if the oxidation and peak cladding temperature "embrittlement criteria" of 10 CFR 50.46 can be shown to be adequate at high burnup for all cladding types, it is likely that the evaluation models for oxidation, ballooning, and rupture will be affected.

Fortunately, design-basis accidents are unlikely and current analyses generally contain conservative margins. Therefore, in its high-burnup program plan, the NRC concluded that there was time to resolve these issues with long-range research programs (3-5 years). To accomplish this, we engaged in a number of formal agreements to gain access to international programs and we initiated some of our own work. A list of current NRC research activities on high-burnup fuel is shown below.

1. ANL (NRC) Hot Cell LOCA Tests of Fuel Rods and Mechanical Properties of Cladding
2. PNNL (NRC) Steady-State and Transient Fuel Rod Codes and Analysis
3. BNL (NRC) Neutron Kinetic Codes and Analysis of Plant Transients
4. Halden (Norway) Reactor Tests of Fuel Rods in Steady State and Mild Transients
5. Cabri (France) Reactivity Accident Tests of Fuel Rods and Related Programs
6. NSRR (Japan) Reactivity Accident Tests of Fuel Rods and Related Programs
7. IGR (Russia) Reactivity Accident Tests of Fuel Rods and Related Programs

Four of these research programs on reactivity accidents will be discussed in the presentations that follow. We do not have sufficient new results from ANL and PNNL to include presentations on those programs at this time, and results from the Halden project were presented at a large Halden program group meeting earlier this year.

During the past year, there have been lively discussions about how much research is needed in each of these areas and about specific details of some of the test programs. To establish a better basis for such discussions, the NRC launched a PIRT (Phenomenon Identification and Ranking Table) exercise that involves expert elicitation in a structured manner. A reactivity accident and a loss-of-coolant accident are being addressed in the PIRT exercise for PWRs and for BWRs. The first meeting of the PIRT panel of experts was held in late August, and the activity is scheduled for completion within one year. Results of the PIRT activity will be presented at a later time and published as NUREG reports.

JAERI RESEARCH ON FUEL ROD BEHAVIOR DURING ACCIDENT CONDITIONS

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SUMMARY

To provide a database for the Japanese regulatory guide for light water reactors, behavior of reactor fuels during accident conditions is being extensively studied in the Japan Atomic Energy Research Institute (JAERI). The research activities include in-pile, pulse-irradiation experiments for reactivity initiated accident (RIA), quench experiment and oxidation test for loss-of-coolant accident (LOCA), and cladding burst and ring tensile tests for RIA and LOCA. This paper describes results from RIA and LOCA experiments and related mechanical testing for high burnup cladding.

RIA-SIMULATING EXPERIMENTS (NSRR program)

In addition to the extensive efforts on high burnup PWR experiments, high burnup BWR fuels (41 to 60 MWd/kgU) are being tested in the FK test series in the Nuclear Safety Research Reactor (NSRR). We have performed two experiments, FK-4 and FK-5, with 56 MWd/kgU fuels in early 1999, and fuel failure has not been observed in these experiments. Since creep down of cladding is less significant in BWRs in comparison with that in PWRs, wider gap between fuel pellets and cladding inner surface exists in BWR fuels before pulse-irradiation in the NSRR. The wider pre-test gap provides BWR fuels with weaker PCMI loading in the early phase of the transient, and reflects smaller fuel rod deformation during the transient. In the experiments with 20 MWd/kgHM MOX fuels, significant fission gas release up to 20% was measured. Relatively large cavities having a diameter of a few ten microns were observed inside plutonium spots. The accumulated fission gas bubbles in the plutonium spots could contribute to the high fission gas release and to relatively large radial deformation of fuel rod.

LOCA EXPERIMENTS

The experimental program related to LOCA has an objective to confirm the safety margin of the high burnup fuel for the current criteria and to obtain the basic data on high burnup fuel behavior under LOCA conditions. Cladding oxidation tests with steam, tube burst tests, and integral thermal shock tests are performed mainly on the simulated high burnup fuel cladding (oxidized, hydrogenated, and/or irradiated) to evaluate the influence of each high burnup factor on the fuel behavior under LOCA conditions. In the integral thermal shock test, LOCA conditions (burst, steam oxidation, and quench) are produced and the failure boundary oxidation condition of the cladding on quenching is determined for the simulated high burnup fuel claddings. Steam oxidation tests of hydrided Zircaloy cladding samples show that the influence of hydrogen absorption on the oxidation rate varies depending on hydrogen concentration, oxidation temperature, and time.

TUBE BURST TEST

The present test is one of the separate effect tests to complement the NSRR experiments. The objective of the test is to obtain the basic information for understanding failure behavior of embrittled cladding under RIA conditions, by simulating very quick PCMI loading that occurs in the high burnup fuel rod during a pulse irradiation. Artificially hydrided Zircaloy-4 cladding samples having different hydrogen concentrations and hydrides distributions were tested at room temperature and at 620 K. At room temperature, axially extended failure opening was formed in the hydrided cladding sample. Residual hoop strain at the rupture position was obviously reduced with increase of hydrogen concentration, and it was as low as less than 1% for the cladding samples with accumulated hydrides at the outer surface. The cladding samples with accumulated hydrides failed at relatively low pressures. The results indicate that the high burnup fuel rod failure observed in both the NSRR and CABRI experiments is strongly influenced by hydrogen absorption and hydride accumulation at the cladding periphery. Temperature increase from room temperature to 620 K mostly reduced the influence of hydrogen absorption on the failure behavior, when hydrides uniformly distributed in the radial cross section of the cladding. On the other hand, the residual hoop strain at the failure position was decreased, when a thick hydrides-accumulated layer was formed at the outer surface of the cladding. The residual hoop strain was only slightly larger than that seen in the same type of the cladding failed at room temperature. The results indicate that accumulation of hydrides at the cladding periphery have the influence on the failure behavior of the high burnup fuel under a RIA condition at reactor operating temperatures.

RING TENSILE TEST

Through pre-test analysis with FEM code ABAQUS, test method and sample geometry for the ring tensile test with machined specimen were examined. Preliminary tests with artificially hydrided samples are also being performed.

Further Results and Analysis of MOX Fuel Behaviour Under Reactivity Accident Conditions in CABRI

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The status of the CABRI REP-Na programme, which aims at studying the consequences of reactivity initiated accident in PWRS (RIAs) such as control rod ejection under hot zero power condition, is regularly reported at the WRSM meetings since 1994. The main objective of this programme led by IPSN in collaboration with EDF and supported by NRC is the investigation of potential high burn-up effects on fuel behaviour and the verification of the RIA safety criteria for standard UO_2 fuel. It was launched in 1992 at the request of the French safety authority DSIN in view of the intention of Electricité de France (EDF) to increase the discharge burn-up from 47 to 52 GWd/t (mean assembly) and for future fuel managements.

In addition, the investigation of MOX fuel behaviour under RIA conditions was included in the definition of the REP-Na test matrix, in anticipation of future licensing requests concerning the behaviour of irradiated MOX fuel under RIA conditions. Although the CABRI REP-Na tests are only devoted to the study of the accident scenarios when the clad temperature increase remains limited (before boiling crisis), the seven tests with UO_2 irradiated fuel rods at different burn-up levels already identified the role of the key parameters on fuel behaviour such as clad corrosion level and state, high burn-up effect of the pellet, and energy injection level and rate [1], [2].

In the present paper we will focus on the main outcome of the three MOX fuel tests (see table 1) performed at different burn-up levels (28, 47 and 55 GWd/t), as deduced from analysis of test results including available PIE (still underway) and the interpretation gained with the SCANAIR code [3]. The MOX fuel used in the tests originated from the MIMAS fabrication process leading to a slightly heterogeneous structure with $(U,Pu)O_2$ agglomerates of 20μm mean size for the major part, embedded in the UO_2 matrix (a maximum of 2% of agglomerates having a mean size higher than 100μm).

From the preliminary analysis of the available results, no evidence was seen of a direct impact of the agglomerates with regard to local thermal effect for rod failure in spite of the high energy deposit (REP Na9). However, the very high clad straining of the 2-cycle rod (7.4 % mean maximum value in REP Na9) is explained and described by the intra-granular gas-induced swelling mechanism linked to the high energy deposition.

On the other hand, the results of all the tests clearly underline an enhanced fission gas effect compared with UO_2 fuel behaviour. A significant increase of fission gas release is found with the MOX fuel compared with UO_2 fuel at similar burn-up; this effect is related to the presence of a higher quantity of gases in inter-granular and porosity bubbles associated to the $(U,Pu)O_2$ agglomerates' behaviour under irradiation and is much increased at high burn-up. As a consequence of a rapid power transient with fuel heat-up and gas overpressure leading to grain boundary separation, a larger amount of gases can be released and be available for clad loading under gas pressure; such result is described by the SCANAIR code, taking into account an increased porosity of the agglomerates.

In contrast to the UO_2 fuel rod failures, which occurred when the clad mechanical properties were degraded by the presence of hydride accumulations, the possibility of rod failure with a sound cladding at a low corrosion level was revealed by the REP Na7 test result with MOX fuel (at 55 GWd/t). This failure, however, occurred with a mean fuel enthalpy at failure time (120 cal/g) that was higher than the expected maximum enthalpy level in commercial power reactor accident conditions. Nevertheless, the obtained failure can be explained by a high contribution of the fission gas pressure to the clad loading and suggests a high burn-up effect with MOX fuel.

Table 1: The MOX-fuel tests of the CABRI REP Na test matrix

Test (date)	*Tested rod*	*Pulse width (ms)*	*Energy at pulse end (cal/g)*	*Clad-corrosion (μZrO2)*	*Results and remarks*
Na-9 (4/97)	EDF MOX 2c span 5 28 GWd/t	34	241 (at 1.2 s) (1007 J/g)	<20	No rupture Hmax = 210 cal/g Δ Φ/Φ= 7.4 % mean max FGR = 35 % (estimation) Examination currently carried out
Na-6 (3/96)	EDF MOX 3c span 5 47 GWd/t	40	165 (at 1.2 s) (690 J/g)	35	No rupture Hmax = 148 cal/g Δ Φ/Φ (max) : 2.5 % max FGR = 21.6 %
Na-7 (2/97)	EDF MOX 4c span 5 55 GWd/t	40	175 (at 1.2 s) (732 J/g)	50	Rupture at 120 cal/g Hmax = 140 cal/g Pressure peaks Fuel dispersal Examination currently carried out

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Summary of Results on the Behavior of VVER High Burnup Fuel Rods Tested under Wide and Narrow Pulse RIA Conditions

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SUMMARY

Capsule tests performed with high burnup VVER fuel rods in the Impulse Graphite Reactor (IGR) under reactivity initiated conditions have demonstrated absence of the pellet cladding mechanical interaction (PCMI) failure [1]. High temperature mechanism of the cladding failure due to plastic deformation and rupture was fixed for these type of fuel rods only. But analysis of representativity of these tests has shown that:

- all tests were performed under wide pulse conditions;
- specific behavior of gas fission products can lead to additional stress of cladding during PCMI stage under narrow pulse conditions.

That is why the new tests with VVER fuel rods were performed in special impulse reactor (BIGR) under narrow pulses. Main parameters of these tests were:

♦ number of fuel rods	6
♦ burnup	48, 61 MWd/kg U
♦ pulse width	2.6 – 3.2 ms
♦ peak fuel enthalpy	116 – 161 cal/g

All six fuel rods did not fail after the tests. The summarized results of IGR and BIGR tests are presented in Fig. 1. Analysis of these tests allows to make the following conclusions:

- PCMI failure of the VVER high burnup fuel rods was not observed for both wide and narrow pulses;
- absence of PCMI failure under narrow pulse conditions can be explained by high ductility of Zr-1%Nb cladding and by presence of the central hole along the fuel stack in VVER fuel rods;
- we can assume that one of the fuel rods tested in the BIGR reactor was practically close to the failure threshold, because its hoop strain was ~6.5% (peak fuel enthalpy was ~160 cal/g). This correlates well with the IGR results.

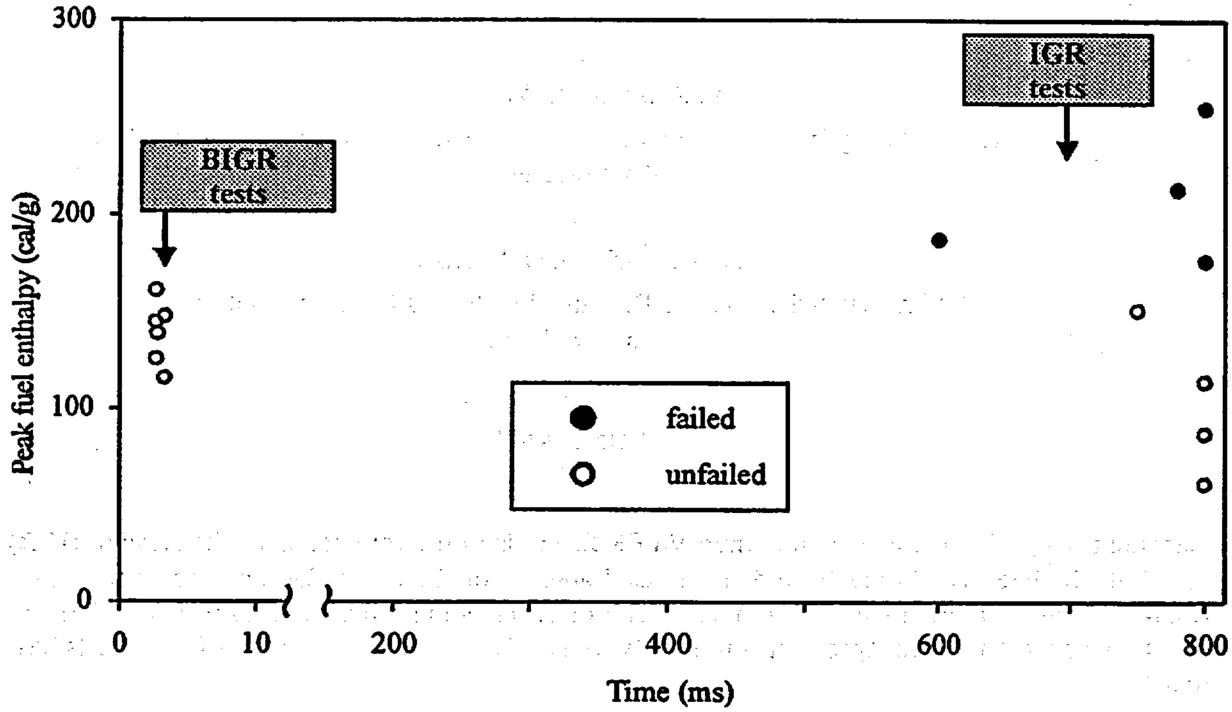

Fig. 1. Results of IGR and BIGR tests.

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Industry Assessment of Existing RIA Data

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In the "Agency Program Plan for High-Burnup Fuel" issued on July 6, 1998, in the "criteria and analysis for reactivity accidents" section, the NRC staff has concluded that "there is no reason to change currently approved burnup levels, unless the confirmatory research program demonstrates a need for change".

Additional RIA-simulation tests and post irradiation examinations have been conducted since mid-1998. Assessment of all existing RIA data shows:

- Fuel performance codes such as FALCON and SCANAIR can interpret the test results (cladding strain, fuel stack elongation, fission gas release and failure enthalpy level) fairly well. This conclusion indicates that there is a good understanding of the dominant operative mechanisms during a RIA event and the availability of reliable tools for assessing the failure criteria.
- For high burnup UO2 fuel, the failure mechanism is PCMI (pellet cladding mechanical interaction) assisted by hydride embrittlement of the fuel cladding. No fuel failures occurred in CABRI tests up to 64 GWD/T as long as the rods did not exhibit spallation. However, test rod failures have occurred on rods without oxide spallation under non-representative test conditions in NSRR.
- Experimental data support the separation of the fuel failure limit and fuel dispersal limit for all burnup levels. For high burnup fuel (up to 64 GWD/T), the enthalpy difference between these limits is greater than 30 cal/g. Therefore, the margin for the coolability safety limit is significantly higher than this value.
- MOX fuel has significantly higher level of fission gas release than the UO2 fuel. At high burnup (55 GWD/T), the experiment suggests the failure mechanism for the MOX fuel may be assisted by the fission gas pressure. However, at lower burnup and enthalpy levels, MOX fuel behaves essentially the same as the UO2 fuel.

As both the PWR Rod Ejection Accident (REA) and the BWR Rod Drop Accidents (RDA) are very low probability events, the need for and type of any future tests must be defined using a risk-informed approach in order to be consistent with the NRC probabilistic risk assessment policy. A consensus should be developed that harmonizes existing NRC staff, NRC contractor and industry studies in order to develop appropriate testing envelope. Such an approach should consider the probability of occurrence and consequences for each RIA event defined in Chapter 15 plant safety analysis reports.

The assessment of existing data and the industry strategy to define the RIA criteria and data needs will be discussed in detail.

MECHANICAL PROPERTIES OF UNIRRADIATED AND IRRADIATED Zr-1%Nb CLADDING UNDER ACCIDENT CONDITIONS

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The burnup level of 50-60 MWd/kgU, that can be reached in the commercial reactors, made it necessary to pay attention to the safety criteria of the fuel rods of the light water reactors under accident conditions. Now the wide range of investigations is being performed in this direction. Modification and improvement of the database on the fuel cladding mechanical properties in accordance with the requirements of the computer codes intended to analyze mechanical behavior of the fuel claddings under accident conditions is an important component of this work. Reassessment of the databases that are being currently used has indicated a number of drawbacks, like:

- narrow range of test conditions due to the priority of the quasi-steady-state tasks;
- absence of reliable data for the irradiated claddings of the commercial reactors;
- significant age (10-20 years) of the majority of data and correlations.

That is why a special program was initiated to get a modern database to analyze mechanical behavior of Zr-1%Nb claddings under accident conditions. The main objective of the investigation was to obtain short-term mechanical properties and parameters of the cladding failure, which were later to be included into the computer libraries of material properties. Analysis of reliability of the calculated results, obtained with the help of the developed correlations of the material properties was another objective of the investigations.

In the framework of this program we have critically analyzed the archives results of the measured mechanical properties of the unirradiated claddings; therefore, the new database was obtained for the unirradiated tubes, and fuel claddings irradiated in the commercial reactor of the VVER-1000 type up to the burnup of ~50 MWd/kgU. The database includes the results of the uniaxial tensile tests in a transverse direction, and of the biaxial burst tests of the pressurized tube samples.

Uniaxial tensile properties were obtained with the simple ring specimens according to the standard Russian procedure. A number of problems connected with the interpretation of raw load-displacement curves of that type of specimens was revealed. The question of whether the obtained data can be adequately used to calculate the stress-strain conditions of the cladding has not been answered yet, despite the significant update of the previously existed procedure. Special approach has been developed to solve this problem. The idea of the approach is to compare the results of ring tensile tests and biaxial burst tests that quite closely repeat the geometry and loading conditions of the standard fuel cladding. The paper will present results of the comparative analysis, which required the performance of new tests of both types.

Additional ring tests of unirradiated and irradiated cladding specimens were performed. Along with measurement of the traditional engineering parameters like strength and ductility, it was also necessary to determine the true stress at rupture. Cross-area reduction procedure was specifically developed for these purposes. On the other hand, the series of burst tests was performed in the temperature range of 20-450°C. Engineering ultimate strength and true burst stresses were reviewed as the major results of the tests. Unirradiated and irradiated claddings were loaded by the pressure of liquid at the rate of 1 MPa/s under the constant set temperature. A number of nondestructive and destructive post-test procedures was performed for the quantitative definition of stresses and strains in the specimens. For the correct comparison strength parameters obtained of those two types of tests were transformed into the effective stresses with the help of the existing database on the anisotropy coefficients of Zr-1%Nb claddings.

The series of burst tests was also important to determine cladding failure parameters under low temperatures typical for the pellet-cladding mechanical interaction (PCMI) stage of the reactivity initiated accident. Although the majority of burst tests were aimed at the LOCA problems, there is practically no experimental database to develop the failure criterion in the low-temperature region. In the framework of this task along with the typical tests with the loading by internal pressure only, a number of parallel tests was performed with additional axial load, simulating axial constraining of the cladding by the expanding fuel in the case of PCMI. The law of the change of the axial force was specified to support equality between tangential and axial stresses in the specimen within the whole duration of the test. Therefore, testing of unirradiated and highly irradiated claddings was performed for two values of the ratio of biaxiality, 2 and 1, which covers the expected range for the reactor PCMI case.

Description of experimental procedures, obtained results, and their analysis will be presented in the complete text of the paper.

Intercomparison of Results for a PWR Rod Ejection Accident

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Introduction

The rod ejection accident (REA) is the design-basis reactivity initiated event for a pressurized water reactor. The calculation of local fuel enthalpy during the event can be used to determine the extent of fuel damage. When this calculation is done with best-estimate methods it is important to also understand the uncertainty in the calculated results. To improve our understanding of that uncertainty, an inter-comparison of the results for an REA using different methods has been carried out. Since some of the methods used treat the fuel assembly as a homogenous region, and some use an explicit representation of each fuel pin, the difference in results was, to a large extent, a reflection of the uncertainty introduced by the simpler representation. Hence, this intercomparison partially satisfies the overall objective of determining the uncertainty in these calculations. In this summary we discuss the methods used and the definition of the REA problem. Detailed results and conclusions will be presented in the full paper.

Calculational Methodology

The calculations carried out in the USA were done with the PARCS/RELAP5 code [1], those done in Russia were done with BARS/RELAP5 [2] and those done in France used a code system (SAPHYR) composed of three codes, APOLLO2 [3], CRONOS2 [4], and FLICA4 [5].

PARCS uses a nodal approximation wherein each fuel assembly is homogenized and the power (flux) is calculated for each axial region in the assembly (or in a quadrant of the assembly if a 2x2 mesh is used within an assembly). The assembly is homogenized in the sense that the cross sections are uniform across the assembly and the thermal-hydraulic parameters are calculated for an average channel representing the assembly. The power in individual fuel rods at each time of interest can be obtained from a flux reconstruction option.

BARS uses a Green's function approach wherein each fuel pin is represented explicitly in the time-dependent calculation. The Green's functions are based on diffusion theory and the calculation is typically done using five energy groups. Although each pin is represented explicitly in the neutronics calculation, the fuel temperature for each pin is based on an assembly-average calculation.

CRONOS2 uses three-dimensional diffusion theory and can either represent an homogenized assembly with a 2x2 mesh or each pin explicitly if the data are available. As with the other approaches, the thermal-hydraulics is obtained for the assembly average (although it does have the ability to do a subchannel calculation in selected regions).

Definition of the Problem

Attention was paid to defining the problem in sufficient detail so that the differences between calculations would primarily be due to the assembly representation. For the neutronics models in PARCS and CRONOS2 two-group cross sections for each homogenized assembly were supplied. Since BARS does not use two-group cross sections, the starting point for that calculation was the core geometry and composition, with the composition corresponding to the burnup in each axial node of each assembly.

The REA was defined for the center rod at hot zero power conditions. The rod was ejected in 100 ms and had a worth fixed at $1.2 in order to be sure that the intercomparison was of the resulting power (and enthalpy) distribution rather than of the initial conditions for the event. Total reactor power rose to approximately four times nominal power during the event and local fuel enthalpy increased from 15 cal/g to approximately 50 cal/g. The pulse width at half-max was approximately 50 ms.

The intercomparison focuses on the local fuel enthalpy in the region surrounding the ejected control rod at the time at which the reactor power is at its peak value as well as other times It is of interest not only to compare the local values but also to see if the peak values of fuel enthalpy are predicted to occur in the same positions.

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Burnup Credit in the Criticality Safety Evaluation of Spent Fuel in Casks:

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Abstract

Until recently, the NRC's approval of criticality safety evaluations for spent fuel in transport and storage casks has been based on analyzing the fuel as though it were fresh and without burnable poisons. The well-known nuclide composition of fresh fuel provides a straightforward and bounding approach to the criticality safety analysis of spent fuel. As fuel is irradiated in the reactor, the nuclide composition changes and, ignoring the presence of burnable poisons, this composition change causes the reactivity of the fuel to decrease. Allowance in the criticality safety analysis for the lower reactivity of irradiated fuel is referred as burnup credit. Extensive investigations have been performed in the U.S. and other countries to understand and document the technical issues related to burnup credit. This paper discusses the technical issues of applying burnup credit to casks containing spent fuel from pressurized water reactors (PWRs) and describes the NRC staff's evolving guidance on acceptable methods and approaches for such uses of burnup credit.

Industry Views and Technical Bases of DOE's Transportation Burnup Credit Program

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Work sponsored by the U.S. Department of Energy and by EPRI

Transportation regulations under 10 CFR Part 71 include explicit requirements for assuming fresh-water in-leakage in the criticality analysis of transport packages for fissile material.

Current regulatory practice requires a demonstration of subcriticality under prescribed conditions:

- Subcriticality is assured when $k_{eff} < 1$
- The allowed k_{eff} is then reduced from 1 to account for such things as modeling and calculational biases and uncertainties
- Additionally, the allowable k_{eff} is further reduced by applying an arbitrary criticality safety margin of 5%, i.e., $\Delta k_{eff} = -0.05$
- Presence of water and fully moderated conditions are assumed
- The fuel is assumed to be fresh (new) fuel

Burnup credit only seeks a change in the fresh fuel assumption.

The issue of "Burnup Credit" (BUC) vs. the "Fresh Fuel" assumption for the evaluation of PWR spent fuel reactivity in a transportation package involves a trade-off. On one hand, consideration and credit for the reduced reactivity of the spent fuel allows for a better utilization of the package volume; this results in a greater number of assemblies per package, and, in turn, in a smaller number of shipments. On the other hand, the fresh fuel assumption provides additional margin for criticality considerations as it leads to the addition of engineered poisons within the package cavity.

Can any incremental reduction in criticality likelihood (and subsequently risks) be justified against the reduction in transportation risk deriving from using burnup credit?

NRC-sponsored work [NUREG/CR-4829, referred to as the "Modal Study") discusses the likelihood of a rail cask accident with a greater than 2% strain coupled with a concurrent submersion (Subsection 9.3.2.4). Rail shipping is particularly relevant because it is required for the dual-purpose systems that are or will be implemented at reactor sites. Under the rail shipping scenario of the Modal Study, "... this type of accident is estimated to occur once every ten million years."

The estimated frequency of a criticality event is then obtained by multiplying the frequency of the accident referred to above [i.e., 10^{-7}/year] by the likelihood that the specific package involved in the accident contain enough reactivity under the moderation and geometric conditions of the accident to result in a critical configuration [i.e., 10^{-x}/accident].

Assuming that the package system is a BUC-designed system, such a likelihood, i.e., 10^{-x}/accident, would be acceptably low if:

- Non-conservative errors associated with the specific BUC methodology are smaller than the sum of (i) the administrative margin ($\Delta k_{eff} = 0.05$) and (ii) the systematic bias in k_{eff} introduced in the methodology to account for enveloping conditions and uncertainties.

- The potential for human errors is small enough to protect against non-conservative fuel assembly insertion errors (misloadings)

Based on probability data for human error [Homes & Narver in NSS-8191.1, Transportation Accidents Risks in the Nuclear Industry], and given that (i) misloadings can introduce less reactivity as well as more reactivity, (ii) only misloadings in specific cask or canister locations have a marked effect; and (iii) two checks are required for every fuel movement, it can be estimated that the probability of a non-conservative misloading can be as high as 10^{-3} and as small as 10^{-5} for a large package. In addition, past analyses have shown that more than one misloading is required to approach criticality conditions. This brings the likelihood of having to deal with a critical configuration, given a severe enough accident, to an estimated $(10^{-3}$ to $10^{-5})^n$/accident, where "n" is the required number of non-conservative misloadings. Using the conservative assumption that only two non-conservative misloadings are required, the likelihood is (10^{-6} to 10^{-10})/accident.

Given that the frequency of an accident severe enough to result in significant damage to the package (coupled with submersion) is already very low [10^{-7}/year], the expected frequency of a critical configuration under the rail shipping scenario of the "Modal Study" is essentially zero (10^{-13} to 10^{-17}/year is a meaningless number!) Estimates of the consequences of a criticality accident are inconsequential from a risk standpoint.

Therefore, the fresh fuel assumption results in a meaningless numerical reduction in critical configuration likelihood. On the other hand, by using a BUC approach, the reduction in the number of shipments is real, and results in a measurable reduction in fatality, non-fatality, and property damage risks from the risk inherent in making more shipments.

A risk-informed approach would seek an overall reduction of the risks associated with spent fuel shipments. A conservative approach would be to adopt the DOE methodology documented in DOE/RW-0472. This methodology would, in a cost-effective manner, deliver a substantial fraction of the benefits to be derived from using burnup credit. This methodology still does not include credit for the poisoning effect of fission products, and, therefore, a very significant margin against the potential of a criticality configuration is built into it. The technical basis for the DOE Transportation Burnup Credit methodology will be presented.

TAKING BURNUP CREDIT INTO ACCOUNT IN FRENCH CRITICALITY STUDIES : THE SITUATION AS IT IS NOW AND THE PROSPECT FOR THE FUTURE

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Abstract

As the enrichment of the fuel has become higher than the limits used at the designing stages, it seemed necessary to consider the fuel depletion during irradiation to guaranty the criticality safety for highly enriched fuels transportation, storage or reprocessing. This burnup credit will make it possible to use the devices for spent fuels which are initially highly enriched.

For that purpose, a method was developed considering : (i) partial Uranium-and-Plutonium burnup credit in the criticality studies, (ii) a conservative assumption concerning the axial profile ; this actinides-only method was supported by an experimental program called HTC. The method was accepted by the French Safety Authority.

Moreover, in order to reduce again the reactivity of irradiated fuels, a French working group was set up in 1997 to define a conservative method which enables industrial companies to take burnup credit into account with some of the fission products and using a more precise profile. This work of this group has been divided into four tasks related to : the determination of (i) the composition of the fuel, (ii) a conservative profile, (iii) a conservative irradiation history, (iv) the calculation scheme. This work is also supported by experimental programs related to the validation of the fission products effects, in terms of reactivity.

THE REBUS INTERNATIONAL PROGRAM

(CRITICAL EXPERIMENT WITH SPENT-FUEL FOR BURNUP-CREDIT VALIDATION)

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SUMMARY

Actual trends in fuel management increase the fuel utilization by increasing the discharge burnup and irradiation times, resulting in an increase of the fuel initial fissile content.

On the other hand, some delay for final disposition of UO_2, and later for MOX-spent-fuel prescribes the storage of increasing quantities of spent fuel.

Burnup credit becomes a key issue in spent-fuel storage, allowing more compact racks for more enriched fuel (with costs reduction) than the classical approach which considers only fresh fuel for criticality safety evaluation.

As there is a recognized lack in experimental validation, BELGONUCLEAIRE and SCK•CEN have proposed to launch the REBUS International Program, consisting of reactivity measurements of well-characterized similar fresh and spent-fuel in a zero-power critical facility.

The REBUS idea was originally based on spent-fuel bundles (of approximately 20 fuel rods) refabricated with BR3 fuel, which have a 1-meter active length compatible with critical facility height. In addition, it has been also decided to refabricate segments of the same length with commercial fresh and irradiated fuel at very high burnups (50 to 60 GWd/t). Critical tests will be performed differentially with fresh and spent fuel to avoid systematic uncertainties.

The main challenge of this program is the radiological impact of the spent fuel to be manipulated in a critical facility which must be adapted accordingly (LR-0 in Czech Republic or VENUS in Belgium).

Δk measurements in the facility will be completed by spent-fuel post-examination measurements, including gamma scanning of each rod (or segment), actinide content measurements, and selected fission products measurements on representative samples.

Selection of bundles for critical experiments is presently as follows :

		Fuel Type	Basic Description
RE-01	Reference absorber test bundle	(fuelless)	Use of B_4C
RE-02	Fresh UO2 (BR3)	PWR-UO_2	5 w/o U-235
RE-03	Irradiated (BR3)	PWR-UO_2	5 w/o U-235 ; 30 GWd/t
RE-04	Fresh MOX (BR3)	PWR-MOX	6.9 w/o Pu-fiss.
RE-05	Irradiated MOX (BR3)	PWR-MOX	6.9 w/o Pu-fiss, 20 GWd/t
RE-06	Fresh UO2	Commercial PWR UO_2	(4 w/o U-235)
RE-07	Irradiated UO2 – High burnup	Commercial PWR UO_2	(4 w/o U-235 ; 60 GWd/t)

This paper will describe in detail :

- Objectives.
- Test bundles design.
- Experimental procedure : Facility Description – Experimental Cores – Experimental Measurements and Techniques.
- Expected Δk (theoretical predetermination).
- Hot-cell work (Post-irradiation measurements before and after critical tests).
- Schedule.

The expected database will be used for burnup credit validation and licensing (including for derivation of usual safety margins).

Implementation of the Revised Source Term at U.S. Operating Reactors

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October 1999

The NRC's reactor regulations require that a fission product release into the containment be postulated and that offsite radiological consequences be evaluated. The release of fission products into the containment (called "source term") is used for judging the acceptability of the effectiveness of engineered safety features and the containment in mitigating offsite releases. The original source term, which was based on releases from a severely damaged core, was published in 1962 by the U.S. Atomic Energy Commission in Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactors." Since that time, there have been significant advances in our understanding of the timing, magnitude, and chemical forms of the fission product release from severe reactor accidents. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," was published in February 1995, and reflects that extensive research and experience culminating in the development of a revised source term. The impetus for operating reactors to adopt the revised source term is that through its more realistic characterization of the source term, plants may modify existing restrictive plant features, (e.g., component actuation times, leakage control systems) as well as make safety enhancements. This presentation gives the status of the NRC's program to implement the revised source term at operating reactors. The program contains three elements, namely, rebaselining, pilot plant applications, and rulemaking.

The objective of rebaselining was to develop a better understanding of the impacts of implementing the revised source term at operating reactors. Having concluded the rebaselining initiative in 1998, the NRC did not identify any issues that would prevent implementation of the revised source term at operating reactors. Currently, pilot plant applications and rulemaking are proceeding in parallel. Five operating reactors have proposed pilot plant applications using the revised source term. The changes proposed by these reactors include removing main steam isolation valve leakage control systems, increasing allowable leak rates, removing charcoal and particulate filters, adding procedures to control containment pH to inhibit revolatilization of iodine from containment water, and relaxing primary containment isolation timing. The first pilot plant application (Perry) was approved in March 1999 and two others (Indian Point 2 and Grand Gulf) are currently under review. Also, a rule and regulatory guide are being prepared to codify use of the revised source term and provide guidance for its use, including updated fission product deposition modeling and insight gained from the review of the pilot plant applications.

FIRST RESULTS OF THE LAST PHEBUS F.P TEST FPT4

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The PHEBUS FP (Fission Product) international program is dedicated to the study of core meltdown in a light water reactor (PWR or BWR), and of the subsequent source term. It is more precisely devoted to the study of the FP release from fuel in degraded conditions (from core melt onset to pool formation), of the FP transport in the primary circuit of the reactor as well as the FP behavior in the containment. The program comprises 6 in-pile tests, carried out in the PHEBUS nuclear facility operated by the French Institut de Protection et de Sûreté Nucléaire (IPSN). It is supported by the IPSN, Electricité de France (EdF), The European Union (EU), the USNRC (US), COG (Canada), NUPEC and JAERI (Japan), KAERI (South Korea), HSK and PSI (Switzerland).

Three tests have been successfully performed within this program, with the last one (FPT4) on July 22nd, 1999 . The test objectives were:

- to investigate the release of low volatile and transuranium elements from a solid debris bed. This fuel geometry, which limits the cooling of the fuel, allows to reach rapidly high temperatures, leading to the releases of such elements.

- to study the transition from a solid bed to a molten pool, resulting from the liquefaction of the debris bed, and the release of fission products in this new geometry.

The initial fuel geometry in this test was different than that of FPT0 and FPT1 (bundle geometry). A debris bed has been prefabricated using fuel fragments and fully oxidized Zircaloy cladding shards. It was sitting on a debris bed made of depleted urania, that is surrounded by an hafnia neutronic shield. This design has been chosen in order to prevent a too large axial melt progression during the pool phase. The debris bed was surrounded by a thermal shroud made of thoria and zirconia, as in previous tests. Due to its fuel geometry and unlike the previous tests, FPT4 has been performed without fuel re-irradiation in the PHEBUS reactor. This led to low expected I131 release. Therefore, it has been decided to collect the released elements in several filters located in the test device, just above the debris bed, in order to obtain the best evaluation of the source term during the different phases of the test.

After reaching the nominal flow conditions through the bed, the fuel was heated up in three phases, as planned by the test protocol:

- a thermal calibration phase with four different steady state fuel temperature levels. During this whole phase, the first filter F1 remained open.

- a second phase devoted to collect the elements released in an intact bed geometry. The first plateau was performed at about 1900°C (~ 2200 K). It will allow to compare releases of volatile fission products in rod-like and in solid debris bed configurations. During this plateau the filter F1, used during the calibration phase, remained open.
  The objective of the second plateau was a maximal fuel temperature of 2430°C (~2700 K) with an intact debris bed geometry or with a very limited liquefaction of the fuel which did not induce perturbations in the releases. The shroud instrumentation showed modifications in the temperature evolutions. These events are under analysis and might be related to material displacement. These events were not foreseen..At this time, the instrumentation at the hottest point inside the debris bed was no more available. It broke in the early part of the power plateau after measuring a temperature level of about 2300°C. A second filter, F2, was opened during the ramp between the first and the second plateau of this phase and closed at the beginning of the second plateau. A third filter, F3, which was specially devoted to collect the release of low volatile elements from a debris bed at high temperature, was opened during the last thirty minutes of the power plateau.

- the last phase of the test included a power ramp up to a power plateau which lead to a large molten pool. The onset of fuel liquefaction was expected during the ramp, followed by a radial and axial progression of the molten pool during the power plateau. Several shutdown conditions had been elaborated to stop the test :
  - when the response of the instrumentation indicates that the experimental objective are reached (2.5 kg of molten fuel),
  - in order to limit the radial and axial extension of the pool.

  Two of these shutdown conditions have been fulfilled just before reaching the target power of the last plateau, and the test was stopped by the test director.

The test FTP4 was successfully performed regarding the fuel degradation objectives. The operating for the filters was as specified and no filter blockage has been observed. Post test analysis and examinations will provide many information on the fuel behavior at the end of the test and of the FP elements (species and quantities) trapped in the filters.

Insights on Application of the Alternative Source Term to Operating Plants

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The Alternative Source Term (AST) was originally developed for the Advanced Light Water Reactors. An effort to apply the AST to operating plants has been underway since late 1994. Considerable industry and NRC resources have been committed to this effort with the goal of making radiological Design Basis Accident (DBA) analysis more consistent with what is now known of radioactive releases from leaking or damaged reactor fuel.

Much has been learned by both the industry and the NRC in the course of applying the AST to operating plants. This paper summarizes what the authors believe are the important lessons learned over the last five years with specific reference to actual applications.

DBA analysis is deterministic in the sense that a specific set of appropriately conservative assumptions and analytical approaches are used to generate a set of results which are then compared to specific regulatory limits. The rules governing DBA analysis, however, are generally not too detailed; some latitude is provided in the regulatory guidance which accompanies each rule. One lesson learned with respect to AST application is that a balance must be reached and maintained between very prescriptive regulatory guidance (which narrowly defines acceptable assumptions and approaches) and regulatory guidance which recognizes (and even encourages) plant-specific innovation.

It is important to note that presently only a few AST applications have been completely or even partially reviewed. Therefore, confining the industry too strongly to a single set of acceptable assumptions and analytical approaches may be premature and even counterproductive to the goal stated above. On the other hand, industry and NRC resources may be used inefficiently if no agreed-upon set of assumptions and analytical approaches is defined at all.

This paper provides perspective on what the authors believe are adequately conservative assumptions and analytical approaches in the areas of thermal-hydraulics, activity transport and removal, and activity release to the environment for both Loss of Coolant Accident (LOCA) and non-LOCA DBAs. In areas where the proposed NRC guidance for AST application to operating plants (DG-1081) is viewed as too prescriptive, alternatives are discussed.

FISSION PRODUCT RELEASE UNDER SEVERE ACCIDENTAL CONDITIONS; GENERAL PRESENTATION OF THE PROGRAM AND SYNTHESIS OF VERCORS 1 TO 6 RESULTS

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The VERCORS experimental program is devoted to the source term determination of radioactive elements in severe PWR accident situations up to the loss of fuel integrity. It is a part of the French Nuclear Protection and Safety Institute (IPSN) program concerning nuclear accident studies and it is fund by IPSN in collaboration with Electricité de France (EdF).

Six VERCORS tests were performed between 1989 and 1994, extending eight previous HEVA tests, conducted in less severe conditions between 1983 and 1989. The present communication is focused on the results of these VERCORS tests carried out with irradiated fuel heated in a temperature range from 2100 to 2600 K. A general overview of the program is given, with its objectives and the way to perform the tests. A synthesis of experimental Fission Products (FP) release is presented and VERCORS 6 results are particularly highlighted.

Prior to the experimental sequence, the fuel sample is manufactured. Three pellets of a standard PWR reactor fuel in its original cladding, irradiated in EdF's nuclear plant, are re-irradiated in the SILOE experimental reactor for a week at low linear power in order to re-build short half-life FP inventory, without any in-pile release. The out of pile sequence is led in a shielded hot cell dedicated to this study. The fuel sample is heated to high temperature within a high frequency furnace in a steam and hydrogen environment, generally after an intermediate plateau aimed at oxidizing the cladding. Beside temperature level and duration of the final plateau, the main parameters of these tests are the composition of the fluid flow (oxidizing or reducing conditions) and the fuel burn-up (38 to 60 GWd/tU).

In cell experimental device will be described; it includes in particular the furnace, a cascade impactor for aerosols size characterization, an iodine filter and a charcoal cold trap for fission gas measurement.

During the accidental sequence, FP release kinetics is measured by three complementary gamma spectrometers. Hydrogen emission kinetics, resulting from cladding oxidation, is quantified by on line gas chromatography. After the test, gamma spectrometry is performed on the fuel sample to quantify the total FP released fraction, as well as on all the components of the loop, to locate and quantify FP deposit along the circuit lines, and draw up FP balance. In addition, some gamma emission tomographies of the fuel give useful results of residual FP location inside the UO_2 matrix and cladding. Fuel behavior is studied by radiography and ceramography analyses. Aerosols deposited on the impactor plates or circuit lines are systematically analyzed by SEM/EDS, and some complementary chemical analyses are performed on powders by XPS and XRD to determine FP chemical forms.

VERCORS 1 and VERCORS 2 were performed in mixed steam and hydrogen flow up to 2150 K:

- under a low fluid flow for VERCORS 1, in order to study the effect of FP high concentration during the aerosols transportation phase, and
- under a higher fluid flow for VERCORS 2 and with four intermediate plateaus, between 1070 and 1770 K, in order to quantify fission gas and volatile FP releases in representative conditions of a LOCA.

In both tests similar kinetics was measured for fission gas, iodine and cesium; the total released fraction reached 20 to 40% for these elements. For VERCORS 1, this fraction is a little more important, due to a longer plateau at the final high temperature and the use of a fuel with a slightly higher burn-up. Aside these elements, molybdenum, antimony, tellurium (and some trace of barium) had measurable released fraction, which is more important for VERCORS 2, due to more oxidizing conditions for molybdenum and to intermediate plateaus, giving time to fully oxidize the cladding, for antimony and tellurium.

The next four tests were all performed up to higher temperature, around 2600 K, just below fuel collapse, except VERCORS 6, which high fuel burn-up sample led to corium formation at this temperature level. The conditions were:

- a mixed steam and hydrogen atmosphere for VERCORS 3,
- pure hydrogen during the last high temperature plateau for VERCORS 4, but after a pre-oxidizing phase,
- pure steam during the last high temperature plateau for VERCORS 5, but after intermediate plateaus between 1070 and 1770 K, as for VERCORS 2, in order to confirm volatile FP release rate in LOCA conditions,
- a mixed steam and hydrogen atmosphere for VERCORS 6, like VERCORS 3, but with a 60 GWd/tU fuel sample.

The high temperature plateau was maintained 30 min for the last three tests and only 15 min for VERCORS 3, because of a blockage of the loop on the last stage of the impactor.

Due to these more severe conditions, these four complementary tests have given useful extension of the FP database. According to their releases at 2600 K, FP could have been classified in four categories:

- usual volatile FP, iodine and cesium, and in addition antimony and tellurium, with a nearly complete release at these temperature level,
- semi-volatile FP, composed of molybdenum, rhodium and barium, with significant release, about half of the volatile FP release, but with non volatile chemical forms deposited close to the fuel, and with high sensitivity to the oxidizing or reducing conditions,
- low volatile elements, composed of ruthenium, cerium, neptunium and probably strontium, with a low but measurable release, between 3% to 10%, exclusively deposited at high temperature upstream the loop, very close to the fuel,
- non volatile FP, composed of zirconium, niobium, lanthanum, neodymium and europium, with no measurable release in this temperature range.

Other elements, like uranium, non-measurable by gamma spectrometry, were detected on impactor plates by SEM/EDS, but could not be quantified in terms of released fraction.

Since 1996 a new VERCORS HT and RT program has been launched to improve the database about fission products and actinide releases during the later phase of an accident, in particular after fuel melting.

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AeRosol Trapping In STeam generator (ARTIST): an investigation of aerosol and iodine behaviour in the secondary side of a steam generator

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Modern Light Water Reactor power plants are extensively equipped with diverse safety features to provide "defence-in-depth" designed to avoid possibility of significant release of activity to the environment. Even in the unlikely event of a core melt accident when fission products are released from the fuel and transported out from the coolant circuit, almost all of fission products are expected to be retained in the reactor containment. However, in the case of a severe accident involving a steam generator tube rupture (SGTR), a there is a path by which fission products can be released directly to the environment if the safety valve remains stick open. Sequences of this kind, referred to as containment bypass, represent a significant risk to the public although their probability are low. They accordingly receive special attention in safety assessments. Probabilistic studies do not generally take account of any retention of fission products in the steam generator (SG) secondary side and typically indicate that a large fraction of the inventory would be released; degraded core SGTR sequences therefore make a large contribution to the total assessed risk from all nuclear accidents. In reality, the complex geometry of the tube bank, support plates, separator and dryer components provides a large surface area on which fission products may be retained. The presence of liquid water in the secondary side of the SG bundle region may further augment the retention. However, the controlling processes are complex and there are no reliable models or empirical data to support estimates of retention. A series of experiments, AeRosol Trapping In STeam generator (ARTIST), will be performed at Paul Scherrer Institut as a first of a kind investigation to provide a database for retention in the SG under the range of conditions that may apply in a reactor accident.

An accident sequence of particular concern is an SGTR followed by a stuck-open secondary relief valve, leading to a long term depletion of coolant and dry conditions in the core, primary and faulted secondary systems. Mitigation of the fission product release depends on mechanisms such as impaction on tubes and other structures in a dry environment. It is important, therefore, to characterise the effectiveness of the retention process for this "baseline" set of conditions. An important set of issues is raised by the Severe Accident Management (SAM) measures to refill the faulted SG using externally supplied coolant. Even if a core melt is not prevented, the presence of liquid in the SG bundle can provide good conditions for scrubbing of gaseous and in particulate form fission products. However, the adoption of such measures requires careful consideration to avoid the possibility of overfilling the SG and carry-over of contaminated water. Also, refilling of a hot dry SG might be problematic, and violent boiling may result in such rapid steam flow as to cause resuspension of previously trapped aerosol particles. Data are needed to address the questions concerning the effectiveness of SG refill and the sensitivity to timing, rate and water level.

The experimental facility is currently under construction. It will be scaled by 1:24 in area and number of tubes, and approximately 1:2 in bundle height. To maintain a correct representation of the tube bundle, separator and dryers, same tube pitch, diameter, support plate configuration and spacing will be used as in a real plant SG. A single separator and a single dryer component of full-size and standard design will be installed. The facility is being constructed in such a way as to provide flexibility of break configuration. For example it will be possible to examine a break near the tube sheet on either the hot or cold side, a top break, an offset or counterflow guillotine break, or a side-oriented "fish-eye" break. The surface to volume ratio, the residence time, the flow geometry between the tubes and through the support plates, separator and dryer will be prototypic of the real plant in all aspects important to aerosol retention. Flow of gas and aerosols to the ARTIST rig will be provided using the aerosol generation facility DRAGON, which can be used to supply a specified and controlled aerosol flow and composition, including multi-component, and flow of steam/non-condensable carrier gas in required proportions.

One of the experimental challenges is to define correctly the break flow for the scaled geometry. Aerosol retention processes operate, broadly, at two contrasting length scales - locally near the break, and globally within the whole of the SG bundle. To examine the retention processes near the break, where the emerging jet of aerosol-laden gas interacts with the nearby tubes, the flow rate should correspond to the full break flow, thus preserving the local velocity and conditions for aerosol impaction on nearby structures. The behaviour distant from the break, and in compartments of the bundle partially screened from the break by the support plates, the flow rate must be scaled by the rig-to-plant flow area so as to preserve the average velocity, the distribution and pattern of flow in the SG as a whole. The investigation is a challenging task due to the complicated geometry, the high and possibly choked flow conditions, the variety of break configurations, and the innate complexity of the aerosol mechanisms. Preliminary analyses have been performed to characterise the flow behaviour and to demonstrate the feasibility of the rig design for simulation of aerosol behaviour during an SGTR severe accident.

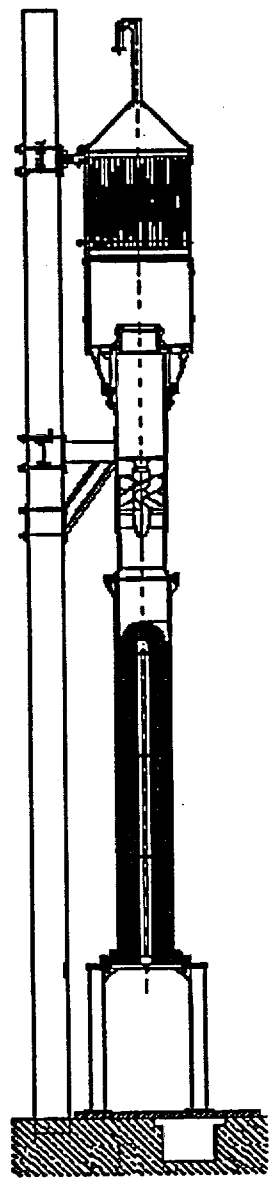

Figure 1: ARTIST facility for SGTR aerosol experiments

OVERVIEW OF DIGITAL RESEARCH OPTIONS FOR NRC

by

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ABSTRACT

The NRC established a deterministic framework for regulation of digital systems. This is documented in the Standard Review Plan, Chapter 7 and associated Regulatory Guides. This paper proposes research options that could lead to an improved technical basis to support risk-informed decisions involving digital systems.

Plants are replacing analog systems with digital systems and this raises new potential safety issues. For example, vulnerabilities of digital systems are different than analog systems. The challenge is to take advantage of the performance enhancements available with the use of digital I&C technology, without introducing any offsetting safety problems. Failure probabilities, critical sensitivities, and the failure characteristics of these systems are also different from analog systems. Modeling and analytical methods that include the combined and integrated operation of both hardware and software found in digital systems will support and improve decision-making.

The research objectives for transitioning from the deterministic framework to a more risk-informed approach include: (1) identifying or developing methods and data that enable the NRC to establish the risk important aspects of advanced I & C systems; (2) understanding potential new failure modes and the criteria for detecting these failure modes prior to failure of plant safety system function; (3) characterizing significant hardware, software and interface errors, including system interface errors, that could prevent safety system action or cause initiating events which could unduly challenge mitigation systems; and (4) modeling of digital systems that could be used to provide system reliability metrics in probabilistic risk assessments (PRA).

APPLICATIONS OF MASTER CURVE TECHNOLOGY

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Operators of pressurized water reactors (PWR) in the United States are currently required to demonstrate that their reactor pressure vessels (RPV) have adequate fracture toughness to maintain structural integrity during severe postulated accident conditions, such as pressurized thermal shock (PTS) [10CFR 50.61]. 10CFR50.61 provides a methodology for estimating the limiting fracture toughness of the RPV that accounts for the effects of irradiation damage. This methodology includes two key components: a curve that provides a lower bound to the fracture toughness of RPV steels in fracture mode transition, and a trend curve that estimates how irradiation damage shifts this lower bound curve to higher temperatures throughout the operational life of the vessel. The technical bases for this methodology were established between 15 and 30 years ago [SECY-82-465, WRC Bulletin 175]. At that time, the physical understanding and data-basis for both fracture and irradiation damage models was quite limited relative to what it is today. These limitations led to adoption of correlative and inferential technologies for these two key components of 10CFR50.61.

In the ensuing timeframe, research sponsored by both the industrial and by the regulatory communities has provided fundamental advances in understanding of the physical process that underlies the fracture of ferritic steels by transgranular cleavage, and the process of irradiation embrittlement in these materials. These advances have led to models of material behavior that are more fundamentally sound, and (consequently) are more accurate than those of 10CFR50.61. Additionally, a much larger body of empirical evidence now exists to assess the appropriateness of any candidate methodology for estimating the fracture toughness of an in-service RPV. These developments have led to acceptance of new methodologies in consensus professional standards [ASTM E1921, ASME N-629]. Because these new methodologies are based on physically appropriate models, rely on direct measurements rather than correlations, and have been benchmarked relative to considerable empirical evidence, they (generally) provide less pessimistic estimates of the state of vessel embrittlement than does 10CFR50.61. Certain nuclear licensees have therefore either used, or are considering the use of these new methodologies to estimate the fracture toughness of their vessels, and thereby obtain relief from the PTS rule. However, these applications, of necessity, are limited to those aspects of these new technologies currently accepted by consensus professional standards. In the longer term, more fundamental changes to the way we estimate the fracture toughness of an in-service RPV are possible. In this paper we review on-going research being conducted by both EPRI and NRC/RES, and describe how this work may lead to future applications of these new technologies to ensure reactor vessel integrity.

Application of Phased Array UT for Nuclear Power Plant Nondestructive Evaluation

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ABSTRACT

EPRI is using phased array technology to develop several ultrasonic inspection techniques for the electric utility industry. The objective is to increase the economic value of the inspections by decreasing inspection time and simplifying the scanning hardware, while matching or exceeding the flaw detection and sizing capabilities of conventional ultrasonic techniques. EPRI is working closely with utilities and NDE vendor companies to develop probes and procedures, to demonstrated and qualify them, to conduct first field application, and to see the array techniques into routine commercial operation. Commercial field deployments in 1999 include inspection of the stainless steel core shroud of boiling water reactors for detection and sizing of intergranular stress corrosion cracking near the welds, and inspection of turbine disks for detection of cracking in the blade attachment hooks. Also under development are rapid-scan techniques for inspection of boiling water reactor pressure vessel welds, for inspection of austenitic welds, and for inspection of ferritic piping welds.

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NOTES

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG/CP-0168
2. TITLE AND SUBTITLE Transactions of the Twenty-Seventh Water Reactor Safety Information Meeting	3. DATE REPORT PUBLISHED MONTH: September YEAR: 1999 4. FIN OR GRANT NUMBER A3988
5. AUTHOR(S) Conference Papers by various authors; Compiled by Susan Monteleone, BNL	6. TYPE OF REPORT Transactions of conference on safety research 7. PERIOD COVERED *(Inclusive Dates)* October 25-27, 1999

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

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

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

Same as Item 8 above.

10. SUPPLEMENTARY NOTES

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

11. ABSTRACT *(200 words or less)*

This contains summaries of papers to be presented at the Twenty-Seventh Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 25-27, 1999. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, U.S. NRC. 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 are also included. the summaries have been compiled in one report to provide a basis for meaningful discussion and information exchange during the course of the meeting, and are given in the order of their presentation in each session.

12 KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT
reactor safety research nuclear safety research	Unlimited 14. SECURITY CLASSIFICATION *(This Page)* Unclassified *(This Report)* Unclassified 15. NUMBER OF PAGES 16. PRICE

NRC FORM 335 (2-89)