NUREG/CP-0148

Transactions of the Twenty-Third Water Reactor Safety Information Meeting

To Be Held at
Bethesda Marriott Hotel
Bethesda, Maryland
October 23–25, 1995

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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Date Published: September 1995

Compiled by: Susan Monteleone, Meeting Coordinator

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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 23rd Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel in Bethesda, Maryland, October 23-25, 1995. 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 the order of their presentation in each session.

An asterisk [*] in place of a page number in the Table of 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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R. Meyer, Chair

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REVIEW OF EPRI HUMAN FACTORS PROGRAM

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The Electric Power Research Institute (EPRI) Human Factors Program, which is part of the EPRI Nuclear Power Group, was established in 1975. Over the years the Program has changed emphasis several times based on the shifting priorities and needs of the nuclear power industry. The Program has produced many important products that provide significant safety and economic benefits for EPRI member utilities. This presentation will provide a brief history of the Program and the nuclear industry pressures that have influenced its content. Current projects and products that have been released recently will be described.

The Study of Control Room Crew Staffing for Advanced Passive Reactor Plants

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SUMMARY

In addition to ensuring compliance of existing nuclear units with regulatory requirements, the U.S. Nuclear Regulatory Commission also conducts Design Certification reviews of proposed reactor designs to ensure the compliance of a vendor's proposed design with regulatory requirements. New plant designs have been produced by a number of vendors for which Design Certification is underway. Some of these advanced plant designs rely on new design or application approaches using passive systems to achieve increased redundancy and/or diversity of methods to achieve safety objectives. Although they use proven technologies to a large extent, these advanced passive plant designs also differ in many respects from conventional plant design.

By improvements in ease of plant operation gained through such features as passive design and digital instrumentation and control systems, vendors expect differences in the ways in which the operations staff will interact with passive plants to fulfill their responsibility in achieving plant safety objectives for foreseeable, credible events. These differences have led to a need for reconsideration of the requirements for minimum shift staffing of licensed Reactor Operators and Senior Reactor Operators contained in current federal regulations (i.e., 10 CFR 50.54(m)).

Since the advanced passive plants under consideration are not yet built there is no opportunity to observe the operation of the plant and the demands placed on the operating staff. Operating experience with these plants and systems is limited to similar systems in other plants (e.g., BWR isolation condenser designs, PWR N_2 injection accumulators, etc.) and designs incorporating similar features (e.g., backfit of digital control systems into existing analog-based control rooms, etc.). Although the basic principles of operating these new plants are similar to existing plants, the range of operating conditions and response requirements may be quite different. Hence, the demands placed on the operating crew itself may be different and this would have significant implications in the design certification process for the advanced passive plants.

The purpose of this research project is to evaluate the impact(s) of advanced passive plant design and staffing of control room crews on operator and team performance and, in doing so, to contribute to the understanding of potential safety issues and provide data to support the development of design review guidance in these areas. Two factors are being evaluated across a range of normal and challenging plant operating conditions: 1) control room crew staffing configuration, and; 2) characteristics of the operating facility itself, whether employing conventional or advanced, passive features.

This research is being carried out by conducting a study of control room crew performance. The first phase of this study was carried out at the Loviisa nuclear power station, in Loviisa, Finland. The second phase of this study will be carried out at the Halden Human-Machine Laboratory (HAMMLAB) at the OECD Halden Reactor Project in Halden, Norway. In the first phase, the Loviisa plant served as the conventional reference plant for this study; HAMMLAB will serve in the second phase as the advanced reference plant. Both facilities run a simulated plant model of the Loviisa nuclear power plant. The

control room systems at Loviisa used in this study are representative of those in conventional control rooms, as are the time constants of the plant's thermal-hydraulic performance. The control room systems used in HAMMLAB will consist of video display unit-based information and control systems, similar in form and function to those of advanced plant control rooms. In addition, some of the time constants of the Loviisa plant thermal-hydraulic model in HAMMLAB will be longer than at Loviisa, to emulate longer thermal-hydraulic time constants in advanced plants which are due to the performance of passive systems or other design features different from conventional plants.

This study uses scenarios of possible design basis events which, without significant mitigation efforts, would challenge the safety of the plant and, in some cases, the integrity of barriers designed to prevent the uncontrolled release of radiation outside the plant. Two different control room crew staffing configurations are used: a normal control room crew staffing configuration level, and a smaller control room crew staffing configuration. The normal staffing configuration is based on current federal regulation for staffing of a single operating unit at a single unit site, which is similar to the current staffing practice at the Loviisa plant. The smaller staffing configuration is based on the minimum crew size considered by an advanced passive plant vendor, or the minimum crew composition which would typically occur at Loviisa.

This paper presents the results of the study conducted thus far. Analyses of crew performance at the Loviisa power plant conducted this year have been completed. These analyses include performance measures of crews in both normal and smaller staffing configurations. Workload, situation awareness, team interactions, and other objective measures of crew and plant performance were used to show how reductions in control room crew staffing affect crew and plant performance. Previous research at the Halden Project has been reviewed to produce a series of lessons learned reports; this paper discusses how these lessons learned have been brought to bear on the design and execution of this study. This paper will also present the plans for the second phase of the project which include evaluation of crew performance in the advanced plant facility to be used for this study, and how results produced by this study can provide technical bases for the review of vendor proposals in the design certification process.

Human-System Interface Design Review Guideline: The Development of Draft Revision 1 to NUREG-0700

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One significant outgrowth of the Three Mile Island accident was that the U.S. Nuclear Regulatory Commission (NRC) required all licensees and applicants for commercial nuclear power plant operating licenses to conduct detailed control room design reviews to identify and correct human factors design problems. The NRC developed *Guidelines for Control Room Design Reviews* (NUREG-0700) in 1981 to support of these reviews. Following this activity, the NRC has continued to focus on issues for which there were uncertainties in the scientific data needed to support regulation. One such issue was the introduction into control rooms (CRs) of advanced, computer-based, human-system interface (HSI) technology; a technology which was not used in Three Mile Island-era plants. The term "advanced" refers to HSI technologies such as touch-screen controls and large-screen displays which are advanced relative to the HSIs implemented in most current nuclear power plants. These developments could have significant implications for plant safety in that they will affect the role (function) of personnel in the system, the method of information presentation, the ways in which personnel interact with the system, and the requirements imposed upon personnel to understand and supervise an increasingly complex system.

The NRC has established programs to review the human factors engineering (HFE) aspects of design and implementation of significant changes to existing CRs and advanced CR designs in order to help assure that the incorporation of advanced technology enhances the potential safety benefits and minimizes the potentially negative effects on performance and plant safety. Because the available HFE review guidance was developed more than ten years ago, well before these technological advances, it did not adequately address these new HSI technologies. Accordingly, updated guidance was needed to serve as the basis for NRC HFE reviews. Thus, Draft Revision 1 to NUREG-0700 was developed.

The document consists of an evaluation methodology and HFE guidelines which are used as criteria for part of the evaluation. The overall purpose of the HSI design review is to ensure that the HSI supports safe, efficient, and reliable personnel task performance. This is accomplished by systematically identifying and resolving design deficiencies, called human engineering discrepancies (HEDs), that could adversely affect plant safety. The scope of the HSIs included in the review is defined on the basis of their function and not their physical location, e.g., in the CR. Relevant HSIs include alarms, displays, controls, job performance aids, workstation and workplace layouts.

The review methodology includes four major phases of activity:

<u>Planning Phase</u> - An applicant's HSI design review plan is reviewed to ensure that the evaluation is adequately defined in terms of review goals and scope, review team, and technical approach.

<u>Preparatory Analysis Phase</u> - The preparatory analyses include operating experience review, function and task analysis, and HSI inventory and characterization. These analyses provide valuable information which supports the establishment of requirements for the HSI design process. These analyses also provide the

technical basis and criteria for the conduct of HSI design verifications and validation; i.e., they identify human performance issues and task requirements with which the HSI should be evaluated.

HSI Design Verifications and Validation Phase - HSI design verifications and validation should be performed to ensure that HEDs have been appropriately identified and documented. Because no one method is likely to be sufficiently comprehensive, it may be necessary to perform a series of analyses:

HSI Task Support Verification. This evaluation verifies that the HSI supports all identified personnel task requirements as defined by task analyses. HEDs are identified for: (1) personnel task requirements that are not fully supported by the HSI, and (2) the presence of HSI components which may not be needed to support personnel tasks.

HFE Design Verification. This evaluation verifies that the HSI is designed and implemented to account for human capabilities and limitations. HEDs are identified if the design is inconsistent with HFE guidelines. NUREG-0700, Revision 1 provides HFE design review guidelines that address both advanced and conventional HSIs. Topics include information display, user-system interaction (e.g., dialog formats and navigation), process control and input devices, alarms, analysis and decision aids, inter-personnel communication, workplace design, and local control stations.

Integrated System Validation. This evaluation ensures that the integrated HSI design enables the accomplishment of task performance requirements for safe operation. HEDs are identified if task performance criteria are not met or if the HSI imposes a high burden on plant personnel.

HED Resolution Phase - This phase should ensure that HEDs have been assessed and that important ones have been resolved.

In addition to a hard-copy document, the guidance has been developed as an interactive, computer-based review aid on a portable computer to facilitate guidance use and to correct some of the usability issues associated with the original NUREG-0700.

The guidance development included a test and evaluation program which contributed to its development. In addition, Revision 1 was offered for public comment and the feedback was used to improve the document.

This paper will address the development of Draft Revision 1 to NUREG-0700 and will describe the resulting review methodology, design review guidelines, and interactive review tool. The results of the public comment process will be discussed as well.

STRUCTURAL & SEISMIC ENGINEERING INTRODUCTION

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Research activity in Structural and Seismic Engineering concentrates on assuring that a sufficient level of understanding exists to assess the safety of licensed facilities if subjected to extreme environmental loads or loads due to accidents beyond the design basis. In order to satisfy this need our recent efforts have focussed on the effects of very large earthquakes and on the capacity of containment structures to sustain pressures well beyond their design level. Our future activities will have additional emphasis on the response and capacity of structures that may have suffered some local damage or degradation due to aging effects.

This year four papers are being presented. Two describe some recently completed studies motivated by an interest in both the likelihood of very large earthquakes and their effects on structures and components. These papers are indicative of the range of methodology necessary for effective study of such a complex phenomenon. One paper is devoted to the use of models based on probability theory to estimate the effects of rare events; the other relates the results of field studies of an actual event to see how well analytical models can represent observations. The other two papers are, essentially, progress reports on current projects aimed at improving the understanding of actual safety margins for piping systems and containments. The paper on piping systems gives a status report on attempts to utilize results of a completed cooperative testing program to improve design rules. The final paper reports on progress in a major cooperative program in which the USNRC and the Ministry of Trade and Industry (MITI) of Japan are sponsoring tests performed by the Nuclear Power Corporation (NUPEC) of Japan and Sandia National Laboratories.

Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts

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In the late 1980s, the methodology for performing probabilistic seismic hazard analysis (PSHA) was exercised extensively for eastern-U.S. nuclear power plant sites by the Electric Power Research Institute (EPRI) and Lawrence Livermore National Laboratory (LLNL) under NRC sponsorship. Unfortunately, the seismic-hazard-curve results of these two studies differed substantially for many of the eastern reactor sites, which has motivated all concerned to revisit the approaches taken. This project is that revisitation.

The NRC, EPRI, and the U.S. Department of Energy (DOE) jointly supported a two-year project that began in the spring of 1993, with the goal of developing a recommended methodology, including implementation guidelines, suitable for performing PSHA.

To accomplish this objective, an independent committee of technical experts, the Senior Seismic Hazard Analysis Committee, was established under joint NRC, DOE, and EPRI sponsorship.* Technical Support Panels sponsored by NRC, DOE, and EPRI performed various analyses and studies as defined by the Committee. The final report provides the desired implementation guidelines, including a recommended methodology, suitable for the performance of PSHA. It can be used in seismic regulation of nuclear power plants and other critical facilities.

This paper will discuss some of the major findings and recommendations contained within the final report, including the approach taken and some of the key issues that have been confronted.

As mentioned, the specific objective of this project has been to provide methodological guidance on how to perform a PSHA. Both technical guidance and procedural guidance are provided, with a strong emphasis on the latter. The reason for this is that, in the course of the Committee's review, it was concluded that many of the major potential pitfalls in executing a successful PSHA are procedural rather than technical in character. These pitfalls can be laid at the feet of the observation that one of the most difficult challenges

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* The members of the Committee include Robert Budnitz (chair), George Apostolakis, David Boore, Lloyd Cluff, Kevin Coppersmith, Allin Cornell, and Peter Morris.

for a PSHA analyst is properly representing the wide diversity of expert judgments about the technical issues in PSHA into an acceptable analytical result, including addressing the large uncertainties. This conclusion, in turn, explains the heavy emphasis on procedural guidance.

This also explains why the Committee came to believe that how a PSHA is structured is as critical to its success as the technical aspects -- perhaps more critical because the procedural pitfalls can sometimes be harder to avoid and harder to uncover in an independent review than the pitfalls in the technical aspects.

In light of the above, this paper will concentrate on providing a brief overview of the most important findings, conclusions, and recommendations in the procedural area. In doing so, it is recognized that several very important pieces of technical guidance concerning the earth-sciences aspects of PSHA will not be touched upon here. As important as they are, the interested expert will perforce need to turn to the full report to find these.

The areas where key procedural guidance is provided, and that will be touched upon in this paper, include the different roles of experts, such as the expert as proponent, as evaluator, and as integrator; the different types of technical consensus and their roles in PSHA; what is meant by technical integration; different levels of technical complexity for various issues, and how the procedural guidance works differently at each level; the role of technical facilitator/integrator (TFI), and new concept developed by the Committee for coping with the most complex technical issues that require eliciting judgments from a diverse panel of experts; how the TFI should use experts to assist in arriving at the appropriate representation of the technical community's understanding of a given technical issue; how weighting or weighing of various different expert interpretations should be done; and how peer review should be structured best.

AN ASSESSMENT OF SEISMIC MARGINS IN NUCLEAR PLANT PIPING

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SUMMARY

The Energy Technology Engineering Center (ETEC) is providing assistance to the U.S. NRC in developing regulatory positions on the seismic analysis of piping. As part of this effort, ETEC performed reviews of the ASME Code, Section III, piping seismic design criteria as revised by the 1994 Addenda. These revised criteria were based on the results of the PFDR Program.

The PFDR Program was implemented by General Electric (GE) and included simulated dynamic seismic, hydrodynamic, and water hammer testing and analyses of piping components and systems. Seismic tests were performed on 33 simple cantilever configurations of piping components and two prototypic piping systems. Material specimen tests were also performed. Based on these analyses and tests, revised criteria for the seismic design of ASME Code, Section III, Class 1, 2, and 3 piping systems were developed by EPRI and GE.

The revised ASME Code piping seismic design criteria in the 1994 Addenda were based on the rules developed by EPRI and GE, and on evaluations by the ASME Special Task Group on Integrated Piping Criteria and the Technical Core Group of the Advanced Reactor Corporation. The results of the PFDR Program were included in these evaluations.

This paper summarizes the current status of ETEC reviews of the revised ASME Code piping seismic design criteria in the 1994 Addenda and some of their bases. In addition, interim reviews by the Peer Review Group (PRG), which was established to provide an overview of the results of the ETEC reviews, are also summarized.

A RECONNAISSANCE REPORT - HYOGO-KEN NAMBU EARTHQUAKE

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SUMMARY

An eight member U.S. Nuclear Regulatory Commission (NRC) and Department of Energy (DOE) team visited Japan from February 11, 1995 through February 19, 1995 to participate in effort to gather data on the January 17, 1995 Hyogo-ken Nambu earthquake. The team comprised of NRC and DOE staff members and several consultants.

The team's focus was to evaluate performance of the industrial facilities, with emphasis on power generating and distribution facilities, as both NRC and DOE have made extensive use of experienced based data to evaluate components and equipment in power plants and other facilities. The team met with the Kansai Electric to discuss overall performance of the power generating and distribution facilities, effects on the nuclear plants because of the grid disturbances, design criteria used, steps necessary for recovery, and lessons learned. The team visited a thermal power station and a substation to observe the damage as well as to discuss operational aspects. The team had interest in not only evaluating failures but also to learn about successes.

The team was also interested in performance of bridge and building structures as well as ground related failures. The team's observations regarding failure modes and possible causes are discussed in the paper. An assessment of various codes used in building, bridges and power plant facilities has also been made.

In this paper, the team's observations are described first along with some general insights. Performance of industrial facilities with emphasis on power plants is discussed. A considerable amount of strong ground motion data has been obtained and analyzed. Results of ground motion analyses are presented. These results include comparison with several ground motion models and results of soil amplification and liquefaction studies. Insights are also drawn with respect to near-field earthquakes which are characterized by potentially damaging large-amplitude velocity pulses and associated large displacements. Implications to the design are discussed.

PLAN ON TESTS TO FAILURE OF STEEL AND PRE-STRESSED CONCRETE CONTAINMENT VESSEL MODELS

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SUMMARY

Light-water-reactor (LWR) containment buildings, which are constructed from steel, reinforced concrete, or prestressed concrete, are the last engineered barrier to prevent the release of radioactive materials to the environment. Although some experiments have been done to aid in assessing the capacity of containments, not all types have been tested.

In this regard, the Nuclear Power Engineering Corporation (NUPEC), the United States Nuclear Regulatory Commission (NRC), and Sandia National Laboratories (SNL) have been involved in a cooperative research program on structural integrity of various containments. The program will include pressure tests on scaled containment vessels of two different types; a steel containment vessel (SCV) and a prestressed concrete containment vessel (PCCV). The SCV model represents some features of an improved BWR Mark-II containment vessel in Japan. The PCCV model represents a typical two buttress PCCV in actual PWR plant in Japan.

The objectives of these tests are to measure the failure pressure, to observe the mode of failure, and to record the containment structural response up to failure. Pre- and post-test analyses will be performed to predict and evaluate the test results and to validate the analytical methods that will be used to evaluate the structural behavior of the actual containments under severe accident conditions.

ADVANCED I&C HARDWARE AND SOFTWARE SESSION INTRODUCTION

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Research activity in Control and Instrumentation concentrates in providing the technical basis for developing regulatory guidelines and criteria to evaluate the interface between the plant and the human user. Also, addressed is the technical basis for qualification of digital hardware from perspective of safe operations and maintenance. The current program is for advanced I&C systems being applied to current and advanced plants and covers hardware qualifications, human capabilities, software quality, and total system considerations.

In the development of a microprocessor-based safety-critical system, the software system design plays a significant role in the reliability of the total system. The first paper discusses Haldens experience and lessons learned in the development of high integrity software as well as other grades of software. Also, the paper will focus on the experience gained throughout the complete software life cycle starting with the software planning phase and ending with software operation and maintenance.

Fiber optics sensing technologies have been considered for the development of advanced instrumentation and control (I&C) systems for the next generation of reactors and in older plants which are retrofitted with new I&C systems. The next paper describes the results of a six-month Phase I study to establish the state-of-the art in fiber optic pressure sensing and describes the design and principle of operation of various fiber optic pressure sensors.

As part of an effort to clarify and enhance regulatory guidance concerning environmental qualification of safety-related instrumentation and control systems, research has been conducted to identify, evaluate, and resolve environmental capability issues posed by microprocessor-based equipment.
Research conducted as part of two projects under this program is described in the next two papers in this session.

The third paper presents part of the ongoing research to identify and confirm the appropriate qualification requirements for microprocessor-based I&C systems. In particular, the research being reported supports an assessment of the significance of smoke exposure as an environmental stressor. The paper describes the investigative approach and the recent results from smoke exposure testing of representative digital components. These results are significant in determining whether smoke is a stressor that requires inclusion in the environmental qualification.

The fourth paper describes a portion of the effort to provide clear guidance for regulatory review to ensure electromagnetic compatibility among I&C systems at nuclear power plant. To establish reasonable acceptance criteria for nuclear power plant application, it was necessary to determine the characteristic electromagnetic operating environment in nuclear power plants.

In support of the nuclear industry, the Power Research Institute (EPRI) has taken the lead in the research for a new pressure sensor, and has found that fiber optic pressure sensors offer reasonable alternative. The next paper presented by EPRI describes the research activities to establish criteria under which commercially available fiber optic components (i.e. fiber optic cables, connective projects, light sources, light detectors etc.) can be used in nuclear plant instrument applications.

In the last paper, the National Academy of Science identifies and discusses the issues that are being addressed in a study and workshop on the safety and reliability of digital instrumentation and control systems technology in nuclear power plant applications as part of a coherent and effective approach to the regulation of the application of digital safety control systems.

Lessons Learned from Experience with Development and Quality Assurance of Software Systems at the Halden Project

by

Thorbjørn J. Bjørlo, Øivind Berg, Morten Pehrsen, Gustav Dahll, Terje Sivertsen

The OECD Halden Reactor Project has developed a considerable number of software systems within the research programmes. These programmes have comprised a wide range of topics, like studies of software for safety-critical applications, development of different operator support systems, and software systems for building and implementing graphical user interfaces. The systems have ranged from simple prototypes to installations in process plants. In the development of these software systems, Halden has gained much experience in quality assurance of different types of software. This paper will summarise the accumulated experience at the Halden Project in quality assurance of software systems.

The different software systems being developed at the Halden Project may be grouped into three categories. These are plant-specific software systems (one-of-a-kind deliveries), generic software products, and safety-critical software systems. This classification has been found convenient as the categories have different requirements to the quality assurance process. In addition, this paper will address the experience from use of software development tools and proprietary software systems at Halden.

The paper will also focus on the experience gained from the complete software life cycle starting with the software planning phase and ending with software operation and maintenance. The main findings may be summarised as follows.

Good project planning is essential for successful implementation of a software project. A waterfall software development model with configuration phases, baselines and unit milestones provides a solid foundation for the detailed development plans. A software quality assurance plan with reviews and audits must be included to ensure that established procedures are followed. Detailed time schedules for all phases - design, implementation, testing, and integration - have proven to be a necessity.

A large part of the effort involved in a formal development project is invested in the production and assessment of the specification. Investing effort in system specification, clarifying the functional requirements and focusing on the specific needs of the end-users by discussing and reviewing the specifications has been most beneficial. Execution of a formal specification is an effective means for detecting specification errors and increase the comprehensibility of the specification, and can be performed incrementally during the production of the specification.

In the design phase, considering details such as functionality, data structures, interfaces, algorithms and tests, before actually writing the code, is deemed extremely valuable. The detailed design documents are commonly handed over to the customer for final discussion and acceptance. This is normally the last chance to reveal any misinterpretations regarding the requirements.

In development of safety-critical software use of formal methods have been found beneficial. There are, however, important non-functional requirements for which use of formal methods provides little support, like requirements to technical performance, e.g., software execution times. Formal methods should therefore be supplemented with other V&V techniques. For efficient use of theorem proving informal software development the specification language must be supported by a powerful theorem prover.

In the testing phase, various checklists have proven valuable in the reviews and audits of the software. Walk-through of code by another programmer is extremely effective for revealing any errors at this stage in the development. The use of commercial test tools for detecting programming errors have proved to be of significant importance, and such tools are heavily used in Halden.

Integration in a project of large complexity is liable to be the phase with highest risk in the development process, so it is important to start integration as early as possible. The easiest way of achieving this is to make use of a build strategy. This implies a number of executable subsets of the final system, which can form integration milestones for progress checking purposes.

Configuration management is an important mechanism for identifying, controlling and tracking the versions of each software item. Integration request and change proposal procedures used during the configuration management and change control, are some of the most important procedures within the software life cycle.

In recent years the trend has been to utilise higher-level development tools, e.g., user interface management systems and case tools, to implement operator support systems. High-level tools close the gap between the design and the programming, and guide the developer in generating a software system true to the design. The major advantage of using high-level development tools is that the developer does not have to worry about technical programming solutions, thus the possibilities for introducing programming errors in the program code are limited. Another benefit from using such tools is that the implementation effort is reduced. However, one should remember that some of the high-level tools are extensive systems requiring extensive coursing and the consultation of experts in the initial stages of use.

With respect to proprietary software for safety related systems, the general impression, based upon information received from companies producing software in the instrumentation and control area, is that they follow a software development practice of high quality standard. An argument for the use of commercial proprietary software in safety-related applications is that the wide user experience grants high reliability.

Assessment of Fiber Optic Sensors and Other Advanced Sensing Technologies for Nuclear Power Plants

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As a result of problems such as drift in nuclear plant pressure sensors and the recent oil loss syndrome in some models of Rosemount pressure transmitters, the nuclear industry has become interested in fiber optic pressure sensors. Fiber optic sensing technologies have been considered for the development of advanced instrumentation and control (I&C) systems for the next generation of reactors and in older plants which are retrofitted with new I&C systems.

The full paper will present the results of a six-month Phase I study to establish the state-of-the-art in fiber optic pressure sensing and describe the design and principle of operation of various fiber optic pressure sensors. The paper will also present a number of new techniques, mostly based on neural network modeling, that are emerging for testing of nuclear plant equipment. This will provide a complete assessment of advanced sensors and related technologies.

Fiber optic pressure sensors provide EMI/RFI immunity, are generally more accurate than conventional pressure sensors, offer smaller size in both the sensor and the cabling, provide faster dynamic response, and allow multiplexing of signals which enables several process parameters to be measured at once. Fiber optic sensors can be used in flammable environments because they will not easily induce ignition. Also, the optical fibers themselves are chemically inert, which makes them suitable for corrosive environments and prevents them from affecting the process.

Fiber optic pressure sensors are typically more expensive than conventional pressure sensors and are not as readily available. Today, none of the traditional suppliers of nuclear-grade sensors supply fiber optic pressure sensors for safety-related applications in nuclear power plants. The susceptibility of fiber optic pressure sensors to nuclear radiation is a problem that may preclude the use of these sensors in high radiation environments of a plant.

The project concluded that fiber optic pressure sensors are still in the research and development stage and only a few manufacturers exist in the United States and abroad which supply suitable fiber optic pressure sensors for industrial applications. Presently, fiber optic pressure sensors are mostly used in special applications for which conventional sensors are not able to meet the requirements.

Preliminary Studies on the Impact of Smoke on Digital Equipment*

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Last year the USNRC initiated a program at Sandia National Laboratories (SNL) to determine the impact of smoke on advanced instrumentation and on safety. In recognition of the fact that reliability of safety-related equipment during or shortly after a fire in a nuclear power plant is more important than long-term effects, we are concentrating on short-term failures. Our studies are approaching the question of reliability in two ways. In cooperation with Oak Ridge National Laboratory, we have exposed parts of a microprocessor-based reactor trip system to smoke. In the second approach we are investigating aspects of circuit bridging in detail, as a function of the types of components used, coatings or coverings for printed circuit boards, and the environmental conditions that can result from various fire scenarios.

To date, we have run two system-type tests in conjunction with Oak Ridge. First, we exposed a printed circuit board with an attached analog-to-digital converter to three types of cable insulation smoke. The printed circuit board was connected to an active host computer located outside the smoke exposure chamber. The host computer ran a simple program to determine if the circuit board and converter were acting normally. In all cases these exposures were fairly heavy; the ratio of fuel to volume would be comparable to the exposure of a piece of equipment located in the same cabinet as a fire. Intermittent failures occurred in the case of the PVC cable insulation fire, but because of the preliminary nature of the tests, no conclusions could be drawn other than that circuit bridging was a likely suspect.

The second set of tests was performed on a mockup of a reactor trip channel. The system consisted of appropriate computers and a multiplexing unit networked together in a fiber distributed data interchange (FDDI) environment. Components incorporating technologies (e.g. optical fiber transmitters and receivers) that are likely to be used retrofits of operating plants were used in assembling the trip system. Different parts of the trip system were exposed to smoke and fire suppression agents according to likely smoke exposure scenarios. Some problems occurred in the computers for higher levels of smoke and for unenclosed fiber optic transmitters. Except for the largest smoke density used, soot tended to collect in the computer fan and power supply rather than the boards inside of the computer.

The detailed investigation of circuit bridging began during the summer of 1995 and is continuing. SNL is investigating the effect of various chip packages (plastic, ceramic or metal), chip mounting schemes (surface mount vs. through-hole), conformal coatings and enclosures. Instead of measuring the loss of metal, a new standard probe which measures the change in surface insulation resistance will be used. The change in surface insulation resistance can be better associated with circuit bridging

* This work was supported by the U.S. Nuclear Regulatory Commission and was jointly performed by Sandia National Laboratory and Oak Ridge National Laboratory. Sandia National Laboratory is operated for the U.S. Department of Energy under contract DE-AC04-94AL85000. Oak Ridge National Laboratory is managed by Lockheed Martin Energy Systems, Inc., for the U.S. Department of Energy under contract DE-AC05-84OR21400.

than the previous corrosion probe, which measured metal loss. Measurements will include resistance between contacts for different chips and performance changes. Fourteen different smoke exposure scenarios will be investigated in which environmental conditions such as humidity, amount of smoke, heat of fire, and CO_2 are varied. Both long-term and short-term failures will be investigated.

Plans for 1996 include the exposure of functional boards to determine the effects of smoke on high-voltage/low-current, high-current/low-voltage, high-frequency, and high-speed digital circuits. These tests will relate the change in resistance between contacts of a chip to the actual performance of a circuit.

DEVELOPMENT OF ELECTROMAGNETIC OPERATING ENVELOPES FOR NUCLEAR POWER PLANTS*

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Safety-related instrumentation and control (I&C) systems in advanced nuclear power plants are expected to make use of both analog and digital equipment and will be significantly different from the totally analog-based I&C systems currently in use. Operational experience with digital technology and advanced analog electronics in the U.S. nuclear industry is quite limited and there is concern about the possibility of upsets and malfunctions in safety-related I&C systems due to electromagnetic interference (EMI). Consequently, Oak Ridge National Laboratory (ORNL) staff have been tasked by the U.S. Nuclear Regulatory Commission (NRC) Office of Regulatory Research to develop the technical basis for evaluating EMI effects in safety-related I&C systems. Test criteria and test methods have been established to minimize EMI problems in nuclear power plants and are documented in NUREG/CR-5941, *Technical Basis for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related I&C Systems*. However, little is currently known about the prevailing ambient electromagnetic environment in nuclear power plants, and this lack of information makes it difficult to specify electromagnetic operating envelopes—that is, the level of interference that systems should be able to withstand without performance degradation—with a high degree of confidence. This paper details the development effort by ORNL to profile the radiated and conducted electromagnetic emission levels at selected plant sites and establish electromagnetic operating envelopes suitable for the nuclear power plant environment.

ORNL staff are presently performing long-term electromagnetic measurements at selected plant sites. The electromagnetic measurement data will be used to profile the ambient nuclear power plant environment and establish acceptable electromagnetic operating envelopes. Observations at plant sites began in August 1994 and will conclude in November 1995. In the meanwhile, interim electromagnetic operating envelopes have been recommended by ORNL staff to augment the test methods suggested in NUREG/CR-5941. The interim operating envelopes are based on acceptance criteria designated for similar environments and are documented in NUREG/CR-6304, *Interim Electromagnetic Operating Envelopes for Safety-Related I&C Systems in Nuclear Power Plants*. These envelopes are consistent with acceptance criteria recommended by the Electric Power Research Institute (EPRI) for digital upgrades of I&C systems in nuclear power plants.

Some information is available through EPRI on the electromagnetic environment in nuclear power plants. EPRI sponsored short-term surveys of the electromagnetic environment at six plant sites in 1994. However, the electromagnetic measurement methodology utilized by ORNL staff differs from the EPRI survey techniques in several ways. The ORNL measurements are long term, featuring at least a week of continuous, round-the-clock measurements at several observation points. The measurements are unattended, allowing the observation of actual electromagnetic fields generated by

*Research sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, under Interagency Agreement 1886-8946-5A and performed at the Oak Ridge National Laboratory, managed by Lockheed Martin Energy Systems, Inc., for the U.S. Department of Energy under contract DE-AC05-84OR21400.

typical plant operations. In addition, the measurements feature simultaneous time and frequency localization, allowing the capture of peak effects without sacrificing the frequency information typically lost in long-term unattended measurements.

A detailed analysis is presently being conducted on the ambient electromagnetic measurement data collected thus far at plant sites. Results from the analysis conducted to date indicate several preliminary conclusions. First, although the ORNL measurement methodology looks at the ambient electromagnetic environment from a different perspective than the EPRI surveys, the ORNL measurement data collected thus far confirm the EPRI findings. Second, routine operations at nuclear power plant sites can trigger notable EMI events but at an extremely low rate of occurrence. Third, although the EPRI surveys did not attempt to collect radiated fields data in the frequency band 50 to 100 kHz, the ORNL measurements indicate significant radiated field levels in this band for all of the selected plant sites.

Testing of Fiber Optic Components in Nuclear Plant Environments

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Over the past several years, the Electric Power Research Institute (EPRI) has funded several projects to evaluate the performance of commercially available fiber optic cables, connective devices, light sources, and light detectors under environmental conditions representative of normal and abnormal nuclear power plant operating conditions. Future projects are planned to evaluate commercially available fiber optic sensors and to install and evaluate performance of instrument loops comprised of fiber optic components in nuclear power plant applications.

The objective of this research is to assess the viability of fiber optic components for replacement and upgrade of nuclear power plant instrument systems. Fiber optic instrument channels offer many potential advantages: commercial availability of parts and technical support, small physical size and weight, immunity to electromagnetic interference, relatively low power requirements, and high bandwidth capabilities. As existing nuclear power plants continue to replace and upgrade I&C systems, fiber optics will offer a low-cost alternative technology which also provides additional information processing capabilities. This work is funded and managed under the Operations & Maintenance Cost Control research target of EPRI's Nuclear Power Group. The work is being performed by faculty and students in the Mechanical and Nuclear Engineering Departments of The Ohio State University.

Currently, this research is divided into four activities, one of which is completed, one of which will be completed in 1995, and two of which are planned. The initial project, published as EPRI report TR-100367, "Optical Fibers in Radiation Environments", was completed in 1992. This project examined performance of various commercially available communications grade optical fiber under exposure to mixed neutron/gamma and gamma radiation. The second project, underway since 1993, will be completed this year. This project is evaluating performance of both optical fibers as well as connective devices, transmitters, and receivers under mixed neutron/gamma and gamma radiation, and under a high pressure and temperature steam environment. Two additional projects are planned. One will address environmental testing of commercially available fiber optic sensors, and the other will entail a demonstration of a fiber optic instrument loop in a nuclear power plant. Each of these projects will be described in this paper. An overview of results obtained to date will be presented.

The results from the first project, published in EPRI TR-100367, focused exclusively on communications grade optical fiber and measurement of its performance characteristics under mixed neutron/gamma and under gamma irradiation. A review of existing research was performed prior to testing. Six types of optical fiber were tested comprising several different attributes: three different wavelengths, single mode and step/graded index, doped and undoped cores. Doses on the order of 4000 Gray were delivered. Mixed gamma/neutron exposures were produced in the 10 kWt Ohio State University Research Reactor (OSURR). Gamma exposures were produced in the University of Cincinnati ^{60}Co irradiation facility. Results were measured in terms of added signal attenuation as a function of dose. Results were also compiled for different temperatures and for post-irradiation self-annealing.

The second project, currently underway, is a substantial expansion of the first project. It consists of two major parts: testing of fiber optic components and benchtop testing of prototype sensor channels. The majority of this project is focused on component testing. Several different types of transmitters, receivers, couplers, splices, and connectors are being tested. Environmental tests consist of mixed neutron/gamma irradiation, gamma irradiation, and exposure to high temperature and pressure steam in a pressure cell. Mixed gamma/neutron exposures are produced in the OSURR facility mentioned above. Gamma exposures are produced in the Ohio State University Nuclear Reactor Laboratory Cobalt Irradiator Facility (OSUNRL-CIF). The steam environment testing is being performed at Ohio State in a pressure cell. Test conditions replicate a main steam line break profile as defined in the EPRI Equipment Qualification Manual (TR-100516), which is consistent with the requirements of IEEE 323. Doses on the order of 10^4 Gray and 10^{17} n/cm^2 have been delivered. For transmitters and receivers, results are measured in terms of rise and fall times, output power, and sensitivity. For connective devices, results are measured in terms of coupling and insertion losses. A second phase of this project involves assembly and integral testing of prototype instrument channels under radiation and post-accident steam environments. Each channel consists of a receiver, transmitter, splices, a coupler, and a sensor. Two channels are being tested, one with an optical strain sensor and the other with a conventional differential pressure transmitter and electro-optical signal conversion. This project will be completed in 1995.

Initial funding and planning have been obtained for a third project to design and install a fiber optic instrument loop at a nuclear power plant as a demonstration of fiber optic technology. A plant has been identified, and the initial project scope and schedule are being determined. Tentatively, installation is planned for 1997. A fourth project to perform testing similar to that described above on commercially available fiber optic sensors is also planned. Funding is not yet approved for this project.

The overall objective of these research activities is to establish criteria under which commercially available fiber optic components can be used in nuclear plant instrument applications, especially those for which radiation or potentially severe post-accident conditions could exist.

APPLICATION OF DIGITAL I&C TO NUCLEAR POWER PLANT OPERATIONS AND SAFETY: RESULTS OF PHASE 1 OF A STUDY BY THE NATIONAL ACADEMY OF SCIENCES

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The National Academy of Sciences/National Research Council is conducting a two-phased study for the Nuclear Regulatory Commission on the application of digital instrumentation and control technology. In the just-completed phase 1 effort, the Committee formed to conduct this study has defined the important safety and reliability issues that arise from the introduction of digital I&C technology in nuclear power plan operations and published its initial report. In this presentation, the Committee chairman will discuss the study background and history, the phase 1 task, the Committee membership, study approach/methodology, and the phase 1 report's findings and conclusions. Phase 2 tasks and plans will also be presented.

HIGH BURNUP FUEL BEHAVIOR
INTRODUCTION

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A decade ago, high burnup was thought to be around 40 GWd/t and all fuel damage licensing criteria and related analysis methods for fuel behavior were validated with data out to approximately that burnup. In the late 1980s and early 1990s, however, burnup levels in U.S. commercial plants were pushed as high as 60 GWd/t for economic reasons, and requests have been made for even higher burnups. Corresponding increases in burnup are also being made in other countries. While fuel manufacturers utilized new data out to the higher burnup levels, the NRC did not start reviewing the licensing criteria and modifying its own computer codes until late 1993.

Shortly after the NRC began this work as a routine activity, data from a test in the CABRI reactor in France raised serious questions about some of the licensing criteria -- the fuel damage thresholds used for analyzing reactivity-initiated accidents (RIAs). Similar RIA testing was being performed by the Japan Atomic Energy Research Institute and by the RRC-Kurchatov Institute in Russia. U.S. testing in this particular area had ended in the early 1980s with the highest burnups tested around 39 GWd/t.

To address the high-burnup fuel issues, the NRC put together a research effort in this area comprised of several contracts, a university grant, and a number of foreign cooperative agreements. These efforts are listed below. In addition, NRC and EPRI are cooperating informally and sharing some information in this area.

<u>CONTRACTS</u>

PNL High Burnup Fuel Models
INEL FRAPCON Code Improvements, and RIA Data Assessment
BNL Analysis of Reactivity Transients
ANL Cladding Metallurgy (theoretical) at High Burnup

<u>GRANT</u>

PSU Fracture Mechanisms (experimental) in Zircaloy

<u>FOREIGN COOPERATIVE AGREEMENTS</u>

Halden (Norway) Fuel Program
CEA-IPSN (France) High Burnup Fuel Behavior
JAERI (Japan) RIA Research
RRC-Kurchatov Institute (Russia) High Burnup Fuel Test Data

The products of this research will be a set of reaffirmed or modified fuel-related licensing criteria that are valid out to about 65 GWd/t and modified computer codes that can be used to perform licensing audits and other studies out to that burnup. In developing these criteria and codes, care will be taken to distinguish between standard fuel rod types and improved types that are designed for high-burnup operation. In this manner, protection can be provided against excessive fuel damage while allowing appropriate credit for improved designs. Significant results have been obtained during the past year and will be given in the following papers. Many of these results, particularly from the foreign programs, will be made publicly available for the first time at this meeting.

NEW RESULTS FROM PULSE TESTS
IN THE CABRI REACTOR

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SUMMARY:

The unexpected low level of the fuel enthalpy at failure in the first CABRI RIA test REP Na 1 which was performed in November 1993 [1] was certainly a key event for the evaluation of high-burn-up fuel behaviour under reactivity accident conditions.

It was a challenging task to come to a coherent understanding of the origin of the fuel rod failure. An important programme of examination work and theoritical investigations has been performed to each this goal. First of all the numerous factors of non-representativity had to be considered and investigated. Subsequently hints and proofs had to be gathered in order to establish a fully plausible explanation of the observations from on-line test diagnostics and post test examinations.

It is believed that this objective has been reached now. The reinforced clad loading (PCMI) resulting from fission gas driven transient fuel swelling and the specific state of clad corrosion of the REP-Na 1 test pin are the factors which explain the failure behaviour.

Beyond REP-Na 2 and 3 which have already been presented, two further tests will have been performed at the time of this meeting. REP-Na 5, a rapid-pulse-test performed in May 95 with a low-corrosion test pin at 64 GWd/t fabricated from a rod section at the intergrid level 2/3 did not fail for a fuel enthalpy of 104 cal/g (at 0.4 s).

The test REP-Na 4 (intergrid 5/6, 62 Gwd/t) is programmed to be performed by the end of July 1995. This test is the first one performed at a reduced power ramp rate in order to reach a better simulation of the energy increase rate expected from reactor kinetics calculations. Under these reduced conditions the fuel temperature history of high-burn-up fuel changes significantly, especially in the RIM zone. Under fast ramp conditions this fuel region heats up almost adiabatically whereas even slightly reduced ramp rate conditions allow for sufficient heat transfer to lower the maximum RIM temperature significantly. The result of this experiment is expected with high interest.

Three MOX fuel tests and one final UO_2 test will close the present test series in the sodium loop of the CABRI reactor by the end of 96.

The implementation of a pressurised-water-loop into CABRI is presently under discussion. In fact the available experimental conditions do not allow to explore all the phases of the RIA accident scenario.

Uncertainties in the knowledge of transient clad to coolant heat transfer of highly irradiated ZIRCALOY do not allow presently to rule out this scenario even at reduced ramp rates. An analytical programme in the PATRICIA loop of the CEA research center at Grenoble is in progress in order to improve this situation.

In addition, the transient fission gas behaviour under real conditions of the system pressure is another field characterised by large uncertainties. In consequence, the IPSN specialists are of the opinion that the coupling of all the phenomena must be allowed for by global experiments with high burn-up fuel in a PWR environment because potential post-DNB phenomena (rupture, fuel dispersion and FCI) are not accessible by separate effect tests.

[1]F.SCHMITZ et al.

Investigation of the behaviour of high burn up PWR fuel under RIA conditions in the CABRI test reactor.
22nd WRSM - October 1994, Bethesda.

NEW RESULTS FROM THE NSRR EXPERIMENTS WITH HIGH BURNUP FUEL

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ABSTRACT

The in-pile fuel irradiation experiments under simulated reactivity initiated accident (RIA) conditions have been performed in the Nuclear Safety Research Reactor (NSRR) of Japan Atomic Energy Research Institute (JAERI) since 1989 with test fuel samples pre-irradiated in the Japan Materials Testing Reactor (JMTR rods) of JAERI or in Japanese power reactors (PWR rods and BWR rods). So far, 37 tests were conducted (19 JMTR rod tests, 13 PWR rod tests and 5 BWR rod tests) and detailed post-test examination including non-destructive and destructive tests is going on for each tested rod.

Recent effort in the NSRR program is concentrated on the study of high burnup fuel behavior. At present, 4 tests called HBO test series were completed with the high burnup PWR rods (fuel burnup : 50Mwd/kgU, peak fuel enthalpy : 50 - 75cal/g). These PWR rods were refabricated from the commercial PWR rod with conventional type cladding material and extended burnup (4 cycle operation) for test purpose. As reported in the last WRSM, the first test HBO-1 performed at 73cal/g showed cladding failure at a lower fuel enthalpy with very long axial cracks. All of the fuel pellets was recovered from the bottom of the test capsule as relatively fine particles.

The time of cladding failure was confirmed at around the end of power burst and before the rapid cladding temperature rise (at approximately 60 cal/g) by the transient records during the experiment. This suggested the fuel failure due to pellet-cladding mechanical interaction (PCMI) caused by rapid expansion of fuel pellet and severe creep down of the cladding tube during irradiation in the PWR. The detailed post-test examination for the ruptured cladding tube showed severe hydride deposition near the cladding surface and generation of many small cracks vertical to the surface in this area. Wall-through cracks were originated from some of these crack tips and showed ductile fracture in the inner cladding region. Residual cladding hoop strain estimated at the failed axial position was approximately 2%.

This behavior was re-examined by the third test HBO-3 tested at 74 cal/g. The cladding tube of the PWR rod used in this test had thinner oxidation thickness at the surface and, therefore, smaller hydride deposition. The HBO-3 rod showed approximately 1.5% of residual cladding hoop strain but did not fail. Residual cladding hoop strains measured in HBO-1 and -2 tests were well correlated with those data obtained by the other PWR rod tests with lower fuel burnup. This suggested that local hydride deposition in HBO-1 rod cladding tube had a major effect on the cladding failure at a relatively low fuel enthalpy. This idea was supported by the results of ring tensile tests for the cladding tube specimens taken from the adjacent locations to those for refabricating test rods and some failure cases in the tests with the JMTR rods. In the JMTR rod tests, wall-through cracks observed in the failed rod were originated from the crack tips in the locally deposited hydride on the cladding outer surface and showed ductile fracture in the inner region.

This hypothesized failure mechanism for the high burnup PWR rod under a simulated RIA condition should be confirmed by additional in-pile tests in the NSRR for the test rods with various levels of water-side corrosion and different rod design such as cladding material, and by the out-of-pile tests on the separate effects relating to this kind of fuel failure. Such short and long-term test programs are going on in the JAERI.

Quite high fission gas release rates (FGRs) during transients were another important finding in the high burnup PWR fuel tests. Indeed, these FGRs are much larger than those observed in the other PWR rod tests where the fuel burnups of used rods are about 40 MWd/kgU. The difference in fuel burnup of approximately 10 MWd/kgU brought big difference in FGR. This might be due to the formation of rim zone in the high burnup PWR rod fuel pellet.

Mechanical energy generation during an RIA is also another important research item. As stated above, all of the fuel in the rod tested in the HBO-1 was recovered from the bottom of the test capsule as fine powders. This suggests that a fuel pellet with a high burnup can be fragmented easily and largely into fine powders due to still undetermined phenomenon (or phenomena). Extensive grain boundary separation caused by the rapid thermal expansion of fission gas existed and retained there might be one of possible mechanism. Large openings of the cladding tube and the large FGR observed in the HBO-1 test might have led some amount of direct fuel ejection from the openings. However, any apparent pressure generation was not observed in this test. This may suggest that essential part of the fuel escaped from the position just above the lower end fitting weld where another failure was observed or that the temperature of fuel fragmented and released was not high enough (60 to 70cal/g in terms of fuel enthalpy) to cause fuel-coolant interaction or large steam generation. The recent tests with JMTR rods at higher energy depositions showed fuel ejection from the openings in the cladding tube and relatively large pressure generation. This test series will give us more clear knowledge on the effects of fuel burnup on the mechanism of mechanical energy generation due to fuel rod destruction.

RECENT VIEW TO THE RESULTS OF PULSE TESTS IN THE IGR REACTOR WITH HIGH BURN-UP FUEL

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Within the time period of 1990-1992 the specialists of RRC "KI" in cooperation with a number of research institutes of the former USSR prepared for and carried out an experimental program directed towards a comparative study of the behavior of fuel elements under RIA conditions with fresh fuel and with the fuel of commercial Russian reactors of PWR type having average burnup of 48 MWd/kgU. Testing of 43 fuel elements (13 fuel elements with high burn-up fuel, 10 fuel elements with the preirradiated cladding and fresh fuel, and 20 non-irradiated fuel elements) was carried out in the IGR pulse reactor with a half width of the reactor power pulse of about 0.6 sec. Type of tests: capsule, no flow rate, with the standard initial conditions in the capsule of 20°C, 0.1 MPa. Two types of coolant, water and air, were used in the tests. Fig. 1 presents the results of tests for 23 fuel elements with burnt and fresh fuel inside preirradiated cladding.

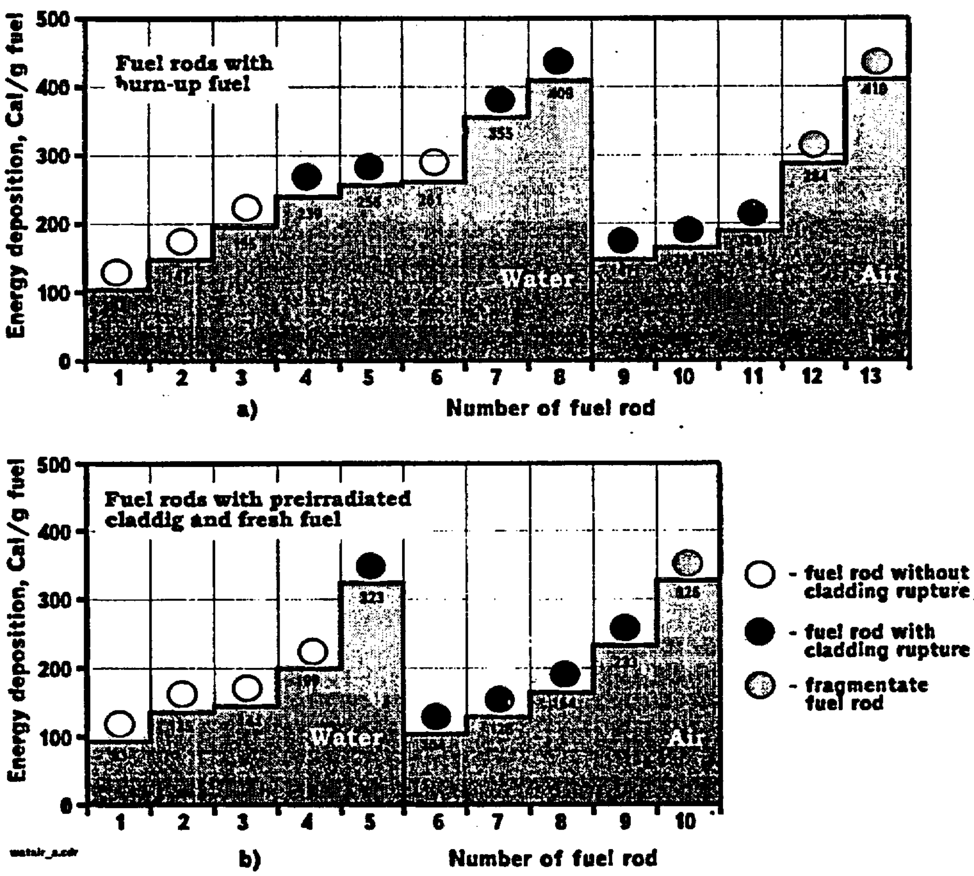

Fig.1 Summarized Results on High Burn-up Fuel Rods Tests under RIA Conditions

Preliminary analysis showed that the damage mechanisms for the three types of tested fuel elements did not differ in quality. Ductile deformation of the balloning type followed by cladding rupture (initial gas pressure inside all the types of fuel elements was 2.0 MPa) was the threshold damage mechanism for cladding of the fuel elements. No low temperature brittle damage of the cladding was identified up to reaching of the second threshold of fuel element damage, i.e., the fragmentation threshold.

The following values are the threshold damaging energy deposition, which result in loss of tightness of the claddings:

- fuel elements with high burn-up fuel cooled with water - 240 cal/g·fuel;
- fuel elements with preirradiated and non-irradiated cladding and fresh fuel cooled with water - 320 cal/g·fuel;
- all types of fuel elements cooled with air - 90 cal/g·fuel.

The tresholds of all types of fuel elements for fragmentation are 420 cal/g·fuel when cooled with water and 280 cal/g·fuel when cooled with air.

The experimental results obtained in the IGR reactor are an important contribution to the world data base which characterizes the behaviour of PWR fuel elements under RIA conditions. It should be kept in mind however that before safety standards criteria for a high burn-up fuel element under RIA conditions are revised it is necessary to carry out a comparison analysis of the results of tests in IGR, PBF, NSRR, CABRI pulse reactors. The aim of such an analysis should be the account of differences in the design of fuel elements (Zr-1%Nb, Zircaloy, the central hole in the Russian fuel pellets) and test conditions (a half pulse width, coolant type, burn-up, pressure drop on the fuel element cladding).

HIGH BURNUP EFFECTS IN VVER-1000 FUEL RODS

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The report presents the main examination results of the VVER-1000 fuel irradiated in normal operating conditions of power reactors. Data are given on such parameters as fuel assembly and fuel rod shape, thickness of cladding and oxide film on fuel rod claddings, state of fuel pellets and size of fuel-cladding gap.

The fuel claddings were found to have initial failures which could be a result of their interaction with solid coolant particles (fretting-corrosion) and structural members of spacer grids.

The values of the main parameters indicate that the VVER-1000 fuel is able to achieve burnup to 50 MWd/kgU at steady-state conditions, and there is a potential for its extension.

The main purpose of the VVER-1000 fuel examination was to study the efficiency of their operation at steady-state regimes during two or three cycles and to assess the potential for burnup and operation length extension. For this purpose, the examination specified the properties of the materials and their influence on each other in power reactor operating conditions. On the basis of this examination, the fuel assembly design was improved and the calculation codes were verified. Furthermore, the reasons and consequences of the fuel failures were studied.

EPRI ASSESSMENT OF THE EFFECTS OF REACTIVITY TRANSIENTS ON HIGH BURNUP FUEL

O. Ozer, R. L. Yang - EPRI
Y.R. Rashid, R.O. Montgomery - ANATECH

Results from recent RIA simulation experiments have raised interest in the response of high-burnup fuel to reactivity transients. Although the experiments are not directly representative of in-reactor conditions, they have helped to focus on the need for a better understanding of the phenomena that affect the material properties of fuel at high burnup. Appropriate representations of such phenomena are needed to explain the observed response of fuel during the RIA simulation tests and to evaluate the relevancy of these tests to in-reactor fuel performance.

Fuel and cladding undergo significant microstructural changes as a consequence of extended operation. For the fuel pellet, these changes result in a decreased thermal conductivity. Moreover, the buildup of plutonium isotopes near the periphery of fuel pellets results in a radially-peaked power profile and a non-uniform burnup distribution which can be ~2 times greater than the pellet average. This eventually leads to the formation of a narrow porous rim along the pellet periphery.

The radial distribution in the power profile, the decrease in thermal conductivity and the formation of a pellet rim do not result in unusual changes in the steady-state performance of fuel. However, during a rapid transient, these effects must be accounted for explicitly, since they affect the location and amount of temperature peaking, the extent of pellet-cladding mechanical interaction and the amount of cladding strain.

EPRI's evaluation of fuel responses in RIA experiments, indicates that the above high-burnup pellet effects tend to increase cladding strain, however even for burnups in the low 60 GWD/MTU range the contribution of the rim is small compared to the strain caused by the thermal expansion of the pellet. This contradicts some earlier hypotheses which attributed fuel failures to a "rim explosion" effect. Even for the case of the RIA simulation experiments the calculated strains are small and in a range that should be accommodated by cladding with sufficient ductility.

Predicting the extent of cladding ductility loss is somewhat more difficult since in addition to fluence, above a certain level, cladding ductility becomes strongly affected by hydriding. Hydriding is the result of hydrogen pickup during waterside corrosion which in turn depends on cladding composition, the manufacturing process, and operating history. Moreover, hydrides tend to migrate to local cold spots which form after layers of corrosion start to delaminate and spall, thus forming potential crack initiation sites.

Therefore, to evaluate the response of cladding to reactivity transients at high burnup a capability to estimate the effects of cladding corrosion and hydriding on ductility loss is needed.

EPRI is utilizing such a capability within a 2-dimensional transient fuel performance code to evaluate the observed fuel responses during the RIA simulation experiments.

Power Excursion Analysis for BWRs At High Burnup[1]

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There were three specific objectives in this study. One was to identify boiling water reactor (BWR) transients/accidents in which there is significant energy deposition in the fuel. Another was to analyze the response of BWRs to the rod drop accident (RDA) and other transients in which there is a power excursion. The last objective was to investigate the effect that the use of high burnup fuel may have on the analysis of events leading to power excursions.

The review of BWR transients/accidents to identify potentially significant power excursions made use of analysis already carried out for Safety Analysis Reports. The RDA is the most important accident of this type. In addition, a recirculation flow control failure (with increasing flow) could lead to significant energy deposition. Although this event has been studied from the point of view of satisfying the criterion relevant to critical heat flux, it should also be studied in more detail with energy deposition in mind. Calculations with spatial detail would enable the peak fuel enthalpy to be determined taking into account the distribution throughout the core. The review also concluded that if another transient were to be studied, it could be any of the overpressurization events (caused by closure of different steamline valves) as, for a particular model, it is not known a priori which event would lead to the highest enthalpy rise.

It was noted in this review that because fuel enthalpy at operating conditions is significant, it is important to make the distinction between the peak enthalpy increase and the peak total enthalpy and to consider both in the context of fuel behavior.

Calculations were carried out for several BWR events: a rod drop accident (RDA), a closure of main steam isolation valves, and a recirculation pump controller failure. These calculations were done with the RAMONA-4B code and a model for a BWR/4 with a core having bundle burnups up to 30 GWd/t. The RDA calculations were also repeated for a pseudo high burnup core with bundle burnups up to 60 GWd/t. The latter model was generated as a first approximation to a high burnup core.

The RDA calculations assumed initial conditions such that a control rod worth approximately 950 pcm dropped out of the core. The maximum increase in fuel enthalpy in the core was low relative to existing acceptance criteria for this event. However, what was of concern was the effect of different modeling assumptions and the increase in fuel enthalpy as a function of burnup. The latter would be of particular interest if acceptance criteria were changed to depend on burnup. The results of the calculations were consistent with the expectation that the peak fuel enthalpy in any location of any bundle would be a complicated function of the (dropped) control rod worth, the distance of the bundle from the control rod, and the burnup at that location. The implication of this is that high burnup in a fuel bundle does not inherently limit the enthalpy rise.

[1]This work was performed under the auspices of the U. S. Nuclear Regulatory Commission.

If acceptance criteria were made a function of burnup, then the case with the highest fuel enthalpy in the core might no longer be the limiting case, i.e., the enthalpy and burnup at each location would have to be compared with the criteria. Hence, more analysis than is currently required to satisfy licensing requirements would have to be done. The new analysis would probably use more of a best-estimate approach rather than the conservative approach which is frequently used in current practice. A best-estimate approach would require looking at all the control rod patterns possible during the startup of a reactor to determine the limiting case. This would take into account the withdrawal sequence, e.g., the banked position withdrawal sequence, and any situation in which a control rod could drop roughly half-way or more through the core. It would also be necessary to take into account patterns caused by having a single failure during the withdrawal sequencing.

The initial coolant temperature and pressure for the RDA would have to be consistent (or more conservative) than those expected at the time during startup corresponding to the particular pattern being used. The amount of subcooling and the temperature of the coolant are important for several reasons. An increase in subcooling delays the time at which coolant voiding occurs, and the corresponding negative reactivity helps shut down the power excursion. (Note that the timing of the steam voids is also dependent on the void generation model being used and these models have not been obtained for RDA conditions.) High subcooling and low temperature lead to an addition of positive reactivity during the period that the moderator heats up. The positive moderator temperature feedback is higher in bundles with higher burnup and, therefore, could impact the resulting fuel enthalpy. Keeping the coolant subcooled also decreases the heat transfer out of the fuel relative to the 2-phase case. This tends to maximize the enthalpy increase after the initial power rise is over.

The calculations of thermal-hydraulic transients for this study included one initiated by closure of main steam isolation valves and one initiated by a recirculation flow controller failure. The results show that the fuel enthalpy increase is small and takes place over a long time interval relative to the case of an RDA. The significance of these numbers will be assessed as more data on fuel behavior becomes available.

In calculations with high burnup fuel, the rim effect will increase the uncertainty. The rim effect is the large increase in plutonium concentration and power along the surface of the fuel rod. Reactor physics models that generate cross sections make assumptions about the temperature and power distributions across the pellet which might be in error due to the enhancement of this effect with burnup. Furthermore, current models may not track enough of the higher actinides that may be present in high burnup fuel. The rim effect also introduces a spatial distribution of thermal properties that may be important. Some fuel heat conduction models may make assumptions about the distribution of energy deposition within the pellet that may be inconsistent with the rim effect. Note too that the thermal properties change with burnup and currently used models may have to be updated to reflect these changes.

In summary, this study has looked at various aspects of the analysis of power excursions in BWR cores with high burnup fuel. If the current acceptance criteria for these events changes in the future, or for any other reason there is a need for best-estimate methods, then there must be changes in the way analysis is currently used for licensing purposes. This study has identified many of the aspects that must be kept in mind when investigating these events.

REVIEW OF HALDEN REACTOR PROJECT HIGH BURNUP FUEL DATA THAT CAN BE USED IN SAFETY ANALYSES

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The fuels and materials testing programs carried out at the OECD Halden Reactor Project are aimed at providing data in support of a mechanistic understanding of phenomena, especially as related to high burnup fuel. The investigations are focused on identifying long term property changes and irradiation, and instrumentation techniques developed over the years which enable the assessment of fuel behavior and properties in-pile.

The fuel-cladding gap has an influence on both thermal and mechanical behavior. Improved gap conductance due to gap closure at high exposure is observed even in the case of a strong contamination with released fission gas. On the other hand, pellet-cladding mechanical interaction, which is measured with cladding elongation detectors and diameter gauges, is re-established after a phase with little interaction and is increasing. These developments are exemplified with data showing changes of fuel temperature, hydraulic diameter and cladding elongation with burnup.

Fuel swelling and cladding primary and secondary creep have been successfully measured in-pile. For example, they provide data for the possible cladding lift-off to be accounted for at high burnup.

Fuel conductivity degradation is observed as a gradual temperature increase with burnup. This affects stored heat, fission gas release and temperature dependent fuel behavior in general.

The Halden Project's data base on fission gas release shows that the phenomenon is associated with an accumulation of gas atoms at the grain boundaries to a critical concentration before appreciable release occurs. This is accompanied by an increase of the surface-to-volume ratio measured in-pile in gas flow experiments. A typical observation at high burnup is also that a burst release of fission gas may occur during a power decrease. This implies that released gas is trapped in fuel cracks, exerting pressure on the fuel fragments and the cladding.

Gas flow and pressure equilibration experiments have shown that axial communication is severely restricted at high burnup. Therefore, gas in fuel cracks and the gap cannot easily escape to the plena and fill gas flow from the plena to a ballooning spot may be impeded.

New High Burnup Fuel Models for NRC's Licensing Audit Code, FRAPCON

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SUMMARY

Fuel behavior models have recently been updated within the U.S. Nuclear Regulatory Commission steady-state FRAPCON code used for auditing of fuel vendor/utility codes and analyses. These modeling updates have concentrated on providing a best estimate prediction of steady-state fuel behavior up to the maximum burnup levels of current data. Evaluation of 15 existing separate-effects models identified 9 models that needed updating for improved prediction of fuel behavior at high burnup levels. The 9 models that have been improved are fission gas release (FGR), fuel thermal conductivity (accounting for high burnup effects and burnable poison additions), fuel swelling, fuel relocation, radial power distribution, contact conductance, cladding corrosion, cladding mechanical properties, and cladding axial growth. The model modifications are summarized in the following.

The previous version of FRAPCON had several FGR models, all of these models have been removed except the ANS 5.4 model. The ANS 5.4 model was retained because it provided the best overall predictions of steady-state FGR even though it underpredicts FGR data from power ramp experiments. In addition, the Forsberg-Massih FGR model has been adopted and modified to predict both steady-state and power ramp FGR data at high burnups. A comparison of the modified Forsberg-Massih FGR model to steady-state FGR data up to 62 GWd/MTU, and power ramp FGR data up to 45 GWd/MTU, shows good agreement with a standard deviation of 6.2% release (absolute) for the high temperature release (>6% release) data set.

The fuel thermal conductivity model has been updated to include burnup degradation as proposed by Lucuta and gadolinia effects as proposed by Massih in the phonon term of the model. Comparisons of the FRAPCON code using the Lucuta thermal burnup degradation term to thermocouple-measured fuel centerline temperatures from experimental fuel rods demonstrates good agreement up to rod-average burnup levels of 60 GWd/MTU. Predictions from the modified FRAPCON thermal conductivity model compare well to urania-gadolinia thermal conductivity data from specimens with up to 8 wt% gadolinia additions in the UO_2 matrix. The effect of gadolinia on fuel specific heat has also been added using the Neuman-Kopp rule of mixtures.

The solid fission product swelling model was developed by fitting a linear burnup dependency to commercial fuel rod swelling data with pellet-average burnups up to 70 GWd/MTU and provides a 15% higher solid swelling rate relative to the previous FRAPCON swelling model. The gaseous swelling model in the previous FRAPCON was found to overpredict fuel swelling in experimental rods that operated at or near peak fuel temperatures for commercial fuel rods, i.e., at rod powers near the commercial technical specification limits. The gaseous fuel swelling model was removed from FRAPCON.

The relocation and cracking models were removed and replaced with the relocation model from the GAPCON-THERMAL-2 Revision 2 code and solid pellet thermal conductivity. A comparison of the

FRAPCON predicted early-in-life fuel temperatures to thermocouple-measured fuel temperatures in experimental fuel rods, with as-fabricated diametral fuel-to-cladding gaps between 80 μm to 380 μm, has demonstrated that this modeling change provides a best estimate prediction of centerline fuel temperatures and stored energy with a wide range of gap sizes.

The radial power model previously used in NRC codes did not correctly predict the high edge peaking at high burnup as measured by experimental data. Therefore, the radial power distribution model proposed by K. Lassmann from the European Institute for Transuranium Elements has been adopted in FRAPCON. Comparisons of this model to measured radial concentration profiles of plutonium and neodymium from fuel pellets with pellet-average burnups between 25 and 83 GWd/MTU indicate that this model does a reasonably good job at all burnup levels.

The Mikic-Todreas model for fuel-cladding contact conductance in FRAPCON has been modified to agree with experimental data from a previous NRC-sponsored program. The Mikic-Todreas contact model is increased by a factor of 2 when interface pressures exceed 1000 lb_f/in^2 in order to agree with the NRC experimental data.

A corrosion model for PWR fuel cladding proposed by Garzarolli et al., that predicts the observed accelerated corrosion at high burnup levels, has been adopted in FRAPCON. The previous corrosion model underpredicted corrosion at high burnups. In addition, the hydrogen pickup fraction of 6% assumed in the previous FRAPCON has been increased to 15% based on a fit to high burnup commercial reactor fuel rod data.

The previous mechanical property models do not predict the hydride and fast fluence effects observed in high burnup cladding. Current mechanical property data from PWR cladding have waterside corrosion thicknesses up to 100 μm, excess hydrogen (levels greater than the hydrogen solubility limit) up to 600 ppm, and fast neutron fluence levels up to 12 x 10^{21} n/cm^2. The mechanical property models have been updated to include the effects of hydriding (excess hydrogen) in the cladding due to corrosion and modifications to the original fluence dependent terms. Predictions from the updated mechanical property model compare well to the high burnup data.

The original cladding axial growth model in FRAPCON significantly underpredicts the recent high burnup data. A growth model proposed by Franklin was adopted and compares well to data obtained from high burnup PWR cladding with fast fluences up to 12 x 10^{21} n/cm^2. Predictions from the Franklin model multiplied by 0.5 compare well to high burnup BWR cladding (Zircaloy-2) up to 10 x 10^{21} n/cm^2.

In conclusion, the results of these individual model modifications have been shown to compare well with individual high burnup data. The inclusion of these models into the FRAPCON code is complete and an integral code assessment is currently underway.

RESOLUTION OF THE DIRECT CONTAINMENT HEATING ISSUE FOR ALL WESTINGHOUSE PLANTS WITH LARGE DRY CONTAINMENTS OR SUBATMOSPHERIC CONTAINMENTS

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In a light-water reactor core melt accident, if the reactor pressure vessel (RPV) fails while the reactor coolant system (RCS) is at high pressure, the expulsion of molten core debris may pressurize the reactor containment building (RCB) beyond its failure pressure. A failure in the bottom head of the RPV, followed by melt expulsion and blowdown of the RCS, will entrain molten core debris in the high-velocity steam blowdown gas. This chain of events is called a high-pressure melt ejection (HPME). Four mechanisms may cause a rapid increase in pressure and temperature in the reactor containment: (1) blowdown of the RCS, (2) efficient debris-to-gas heat transfer, (3) exothermic metal/steam and metal/oxygen reactions, and (4) hydrogen combustion. These processes, which lead to increased loads on the containment building, are collectively referred to as direct containment heating (DCH). It is necessary to understand factors that enhance or mitigate DCH because the pressure load imposed on the RCB may lead to early failure of the containment.

NUREG/CR-6075, "The Probability of Containment Failure by Direct Containment Heating in Zion," was the first step in resolving the DCH issue. It assessed the probability of containment failure by DCH for the Zion nuclear power plant (NPP). It underwent an extensive review by a panel of 13 experts representing national laboratories, universities, and industry. The reviewers provided written comments; the authors responded to these comments; and finally, the reviewers wrote rebuttals to the authors' responses. From the peer review process, two areas of residual concern were identified: initial conditions and the validity of the model. Two working group meetings addressed these unresolved issues. A supplement to NUREG/CR-6075 was written to document the peer review process, address residual concerns about initial conditions and model validity, and document modeling enhancements.

Four new splinter scenarios were proposed for Zion in the working group meetings. The new scenarios either bound the scenarios in NUREG/CR-6075 or stress greater consistency in the conditions at vessel breach. Two high-pressure scenarios resulting from operator intervention were defined. Scenario V is characterized by coejection of large quantities of water (75 mt) at 16 MPa, and Scenario VI is characterized by coejection of 10 mt of water at 8 MPa. The expected melt composition is predominantly oxidic. Two low-pressure scenarios were also defined. These are characterized by melts with a larger metallic component and small amounts of coejected water.

In order to ensure consistent initial conditions for each scenario, the working group members stressed the use of insights from system-level codes, specifically SCDAP/RELAP5 and CONTAIN. Existing SCDAP/RELAP5 calculations for short-term station blackout scenarios for Zion, Surry, Calvert Cliffs, and Arkansas Nuclear One Unit 2 all indicate that failure of the hot leg or surge line and resulting depressurization of the primary system occur well before core relocation and lower head failure in all cases analyzed. Calculations were continued until lower head failure and showed that only a small amount of metallic debris relocates to the lower plenum. Little or no melting of upper plenum steel was observed, and there was very little relocation of metallic core blockages into the lower plenum. In addition, these analyses showed that RCS pressure could remain high only if the vessel was reflooded. These insights were used to develop the distributions for the four new scenarios defined in the supplement to NUREG/CR-6075.

NUREG/CR-6109 used the methodology, which was based on comparisons of containment loads with containment strength, developed for NUREG/CR-6075 and its supplement to assess the conditional containment failure probability for the Surry NPP. The scenarios described in NUREG/CR-6075, Supplement 1, were considered in NUREG/CR-6109. The methodology used for NUREG/CR-6075 to quantify initial conditions was repeated with specific input from Surry and with the insights gained from existing SCDAP/RELAP5 calculations for the Surry NPP.

There are several tools for calculating DCH loads. In NUREG/CR-6075, the two-cell equilibrium (TCE) model and the convection-limited containment heating (CLCH) model were used. These models were validated against the extensive DCH experimental database and gave similar results because the basic modeling assumptions are the same. Only the TCE model was used to compute containment loads in NUREG/CR-6109 and in NUREG/CR-6338. In addition, the CONTAIN code has also been used to calculate DCH loads. For comparison, load calculations were performed for specific sets of input parameters with the CONTAIN code and with the TCE model in NUREG/CR-6109. The calculations were performed for Scenarios V, Va, and VI at the upper end of the mass distributions and with likely hydrogen concentrations. The loads computed with CONTAIN were comparable to or less than the loads calculated with the TCE model for comparable DCH scenarios.

The conditional (on core damage) containment failure probability (CCFP) can be divided into two components: (1) the likelihood of being at high pressure at vessel failure, and (2) the probability that the containment will fail given DCH. NUREG/CR-6075 and its supplement resolved the DCH issue for Zion based on containment loads only, i.e., the load distributions were convoluted with the containment strength distribution to calculate containment failure probabilities without regard to the likelihood of being at high pressure at vessel breach. The conclusion in NUREG/CR-6075 for Zion was that there were no intersections of the load distributions and the containment strength distributions, and thus the DCH issue was resolved for the Zion NPP. The results of the load evaluations for Surry were similar to those for Zion: there were no intersections of the load distributions with the containment strength distribution, and thus the DCH issue for Surry can also be resolved on containment loads alone. Furthermore, the likelihood of high RCS pressures at vessel breach was evaluated for Surry for a limited number of sequences. The probability of RCS pressures greater than 1.38 MPa for all station blackout scenarios without power recovery or operator intervention was found to be low ($p \sim 0.077$). This probability could have been factored into the containment failure probability for Surry if there had been substantial intersections of the load and strength distributions.

NUREG/CR-6338 addresses the DCH issue for all Westinghouse plants with dry containments, which include 34 plants with large dry containments and 7 plants with subatmospheric containments. Westinghouse plants with ice condenser containments are excluded. The methodology developed in NUREG/CR-6075 and NUREG/CR-6075, Supplement 1, was used to perform a load versus strength evaluation for each of these plants using plant-specific data gathered from IPEs, FSARs, and when necessary, direct contacts with plant personnel. The same enveloping accident scenarios (splinters) that were used in NUREG/CR-6075, Supplement 1, and NUREG/CR-6109 were used for these plant evaluations; these scenarios establish important input parameters for the loads calculations, e.g. the RCS pressure at vessel breach, the RPV breach size, the containment pressure and composition at vessel breach, etc. The melt mass and composition distributions developed for Zion (a four-loop plant) in NUREG/CR-6075, Supplement 1, were used for all of the four-loop plants. For all of the three-loop plants, the melt mass and composition developed for Surry (a three-loop plant) in NUREG/CR-6109 were used. For two-loop plants, the prescription given in NUREG/CR-6075, Supplement 1, was used to develop the melt mass and composition distributions. These quantifications are given in this report.

Plant-specific data were gathered for each of the Westinghouse plants with dry containments for the loads versus strength evaluations. As much as possible, similar plants were grouped to facilitate the DCH quantifications. Drawings from all 41 Westinghouse plants were reviewed so that cavities could be grouped for cavity dispersal and coherence quantifications and so that lower compartment configurations could be grouped to facilitate the quantifications of the debris transport through the subcompartments to the containment dome. The likelihood of water being present in the cavity at vessel breach is also assessed because cavity water may have an impact on DCH loads. Cavities are grouped according to whether they are dry, wet, or deeply flooded and are categorized as either excavated or free standing.

The containment fragility curve was extracted and digitized from the IPE for each plant and the fragility quantifications are summarized in NUREG/CR-6338. The TCE/LHS code was used to perform a load versus strength evaluation using a Monte Carlo simulation to determine the CCFP for each of the Westinghouse plants with dry containments. The results of these calculations show that the CCFP based on the mean fragility curves is less than 0.01 for each plant analyzed in this study. Thus, DCH is considered resolved for all Westinghouse plants, excluding only plants with ice condenser containments, and no additional analyses are required.

We are seeking some additional confirmatory information on certain two-loop plants (Ginna, Kewaunee, and Prairie Island 1 & 2) to allow more accurate estimates of the subcompartment debris transport fractions. This information has been requested from the utilities and will be factored into the final report.

STATUS OF THE FARO/KROTOS MELT-COOLANT INTERACTIONS TESTS

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ABSTRACT

The FARO programme of JRC-Ispra includes three main activities centred on the FARO facility: large scale FARO melt quenching experiments, small scale KROTOS FCI tests, code development and test analysis. The paper presents and discusses the most recent developments in each activity.

The FARO tests have been designed to provide the experimental data base with information on melt jet/water quenching and premixing from tests performed with large masses of real corium in prototypical conditions. The data are used to understand the physical processes that govern quenching and to validate computer models.

Basically, 150 kg of UO_2-type melt are poured into water of depth up to 2 m. Five tests have been performed so far, four of which have previously been reported (22nd WRSM). The fifth test (L-19) is reported here. It involved 155 kg of w% 80 UO_2 + 20 ZrO_2 at 3073 K quenched in 338-kg, 1-m-depth water at saturation at 5.0 MPa (i.e., 537 K). The results are compared with those of two former tests (L-06 and L-08) performed in similar conditions (1-m-depth water) but with reduced quantities of melt (18 and 44 kg, repectively). Test L-14, performed with a similar quantity of melt (125 kg) but in 2-m-depth water is also considered for comparison.

The KROTOS FCI tests aim at providing benchmark data to examine the effect of fuel-coolant initial conditions and mixing on steam explosion energetics. Experiments, fundamental in nature, are performed in a well-controlled one-dimensional geometry. A fuel simulant melt, e.g. Sn (7.5 kg at 1370 K) or Al_2O_3 (1.5 kg at 2650 K) or a prototypical material melt, UO_2/ZrO_2 mixture (3 kg at 3100 K), is poured into a water column of ≤120 cm in height, 9.5 or 20 cm in diameter. The melt penetration depth is measured and estimates for the integral void fraction can be deduced from level-meter data. An explosion can be triggered from the base of the water column using a gas trigger device. The pressures are measured in the test section and the explosion expansion work can be deduced from the expansion volume pressurisation. The post-test debris analyses provide additional qualitative information about explosion efficiency.

As reported previously (22nd WRSM), no spontaneous energetic FCI has been observed in the corium tests without the trigger. This is in contrast with the very energetic FCI observed in the previously reported Al_2O_3 tests which had the same initial conditions (pressure and subcooling). To investigate further the observed differences, a new set of tests have been conducted with Al_2O_3. In these tests the effect of melt superheat, water subcooling and ambient pressure on Al_2O_3/water system behaviour have been tested. The results demonstrated that spontaneous explosions will occur in the Al_2O_3/water system over a wide range of melt superheats (150-750 K). On the other hand, the effect of lower subcooling of water on suppressing the explosions was confirmed also in the larger diameter test section (20 cm). Finally, a test was made to see if a moderately higher overpressure (0.1 MPa) could already suppress the explosion. However, since the water temperature was held at 293 K, the higher pressure also meant that the water was more subcooled. A violent explosion did occur in this test, thus the pressure was still too low for suppressing a steam explosion in the alumina/water system.

Modelling and test analysis activities have been focused on development of the computer codes COMETA and TEXAS, and analyses of the FARO and KROTOS tests with them. COMETA (COre MElt Thermal-hydraulic Analysis) is a JRC-Ispra code developed for the prediction of the thermal-hydraulic behaviour of the FARO facility, design verification, definition of operational procedures and test interpretation. The code is composed of a two-phase flow field, which is described by 6+n equations (mass, momentum and energy for each phase and n mass conservation equations for n non-condensable gases) and a corium field with 3 phases: the jet, the droplets and the debris. The two-phase field is described in Eulerian while the corium field in Lagrangian coordinates.

The latest additions and modifications to the models in the codes COMETA and TEXAS will be considered. Pre- and post-test analyses have continued for the FARO tests, in particular FARO test L-19, using both these codes. Inter-code comparisons and comparison with experimental data will be reported and discussed, especially the important features that occur during the different stages in the melt/water heat exchange process.

Difficulties still arise in reproducing the KROTOS UO_2- ZrO_2 experimental data and the main reasons for this will be outlined. Of the two above mentioned codes only TEXAS contains an explosion model. Some TEXAS results for the latest KROTOS Al_2O_3 test, in which a violent steam explosion did occur, will be presented and compared with experiment. During some KROTOS experiments there are large deformations of the bottom plate and hold-down bolts. Use is made of the 2-D axisymmetric code SEURBNUK-EURDYN to analyze these deformations of the test section and some results will be presented.

The FARO programme is being performed in the frame of a Technical Exchange Arrangement between CEC and US-NRC.

AN OVERVIEW OF FUEL-COOLANT INTERACTION (FCI) RESEARCH AT NRC

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Fuel-Coolant Interaction (FCI) is a process by which molten fuel transfers energy to the surrounding coolant. During a postulated severe accident, the time scale for this mode of energy transfer may range from milliseconds to tens of seconds. Interactions occurring in milliseconds range could lead to energetic steam explosions which, in turn, could challenge reactor vessel and containment integrity as well as create a leakage path for radiological releases. It is in this context that the FCI is considered a severe accident issue of potential risk significance, and its resolution is sought to understand its impact on safety. This paper gives an overview of the NRC-sponsored research on FCI in support of the above issue resolution.

The α-mode of containment failure, identified in the 1975 Reactor Safety Study (WASH-1400) as the failure mode induced by in-vessel steam explosion-generated missiles, was the primary focus of NRC research in the early days. In 1985, the first Steam Explosion Review Group (SERG-1) workshop was convened by NRC to discuss and systematically evaluate the α-mode failure issue. At the conclusion of the workshop, there was a consensus among the experts that the occurrence of an explosion of sufficient energetics which could lead to an α-mode containment failure had a low probability ($<10^{-3}$ given a core melt accident). However, the experts also concluded that additional research would be necessary to develop a more complete understanding of the fundamental processes involved in FCI (i.e., premixing, triggering, and propagation) so that the α-mode failure probability could be quantified with a high level of confidence.

Much of the NRC-sponsored FCI research since 1985 was aimed at enhancing the technical basis for understanding the α-mode failure issue, estimating the bounds of potential energetics, determining the conditions under which energetic interactions could occur, and resolving residual uncertainties in our understanding of fundamental processes involved in FCI. Recently completed work at the University of California, Santa Barbara examined the water depletion phenomenon during premixing, and also the fragmentation kinetics during propagation. The significance of the water depletion phenomenon is that it puts bounding limits (i.e., limits to mixing) on interacting water, steam, and melt masses in a multiphase system thereby providing a strong technical argument in support of a very low probability for the α-mode failure. While the "limits to mixing" argument is sufficient in addressing the α-mode failure issue, the knowledge of fragmentation kinetics is essential in studying propagation and escalation of pressure waves in the context of shock loading and potential failure of adjoining structures.

The ongoing research program at the University of Wisconsin is investigating the effects of various FCI parameters on steam explosion energetics using a one-dimensional geometry and using tin as a simulant. Results obtained from these experiments thus far indicate that the melt-coolant mass ratio is an important parameter affecting the energetics. However, generalization of these results to reactor cases requires a careful consideration of scaling (both geometric and material), and a more expanded data base that includes, for example, an oxidic simulant.

The NRC participates in the cooperative FCI research program, FARO, under an agreement with the Joint Research Center (JRC) of the Commission of the European Communities at Ispra. To date, the program addressed mixing and quenching of large masses of prototypic melt jets into water at high pressures (FARO) as well as small-scale experiments to examine the effects of melt-coolant initial conditions and mixing on explosion energetics (KROTOS). Results from FARO tests indicate that the presence of zircaloy in the melt, even in small quantity, enhances melt quenching and decreases thermal loading on adjoining structures, but increases steam generation and vessel pressurization. Results from KROTOS tests indicate absence of an energetic interaction between a prototypic melt and water under a wide range of subcooled conditions, but presence of such an interaction between an oxidic simulant (aluminum oxide) and water. Once again, material scaling is an issue here that needs to be addressed.

Given the significant progress made in FCI research over the last ten years, the NRC convened the Second Steam Explosion Review Group (SERG-2) workshop in June 1995. The purpose of the workshop was to review the current status of the broader FCI issues, to revisit the α-mode failure issue and to re-assess the failure probability estimates. With regard to the α-mode failure issue, the previous probability estimates (assessed by SERG-1 experts) were reconfirmed and, in most cases were re-assessed to be lower ($<10^{-4}$ given a core melt accident). A consensus view emerged that this particular manifestation of in-vessel FCI posed no significant risk to containment integrity. There was also general agreement that FCI issues requiring further attention and resolution are localized FCIs and resulting shock loading of adjoining structures, chemical augmentation of explosion energetics, and a better understanding of triggering.

Shock loading and potential failure of structures are particularly relevant to ALWR designs: lower head loading in the case of AP600 design which employs the external cooling concept using a flooded cavity, and pedestal loading from ex-vessel steam explosions in the case of SBWR design. Our current understanding of premixing and propagation, developed in the course of resolving the α-mode failure issue, should be applicable to ALWR-specific issues but this must be demonstrated analytically or otherwise. In this context, it is worth mentioning that the NRC-supported analytical program on FCI resulted in the PM-ALPHA and ESPROSE codes for premixing and propagation, respectively, and the integrated fuel-coolant interaction code, IFCI.

The issue of possible chemical augmentation of steam explosion energetics due to the presence of metal (zircaloy) in the melt is being addressed through a research program at the Argonne National Laboratory. This program is complementary to the FARO program in that it will provide data on chemical energetics considering a high metal fraction in the melt. The future FARO program will examine melt quenching at low pressures, melt spreading under water, and steam explosion potential in deep and shallow water pools. The future University of Wisconsin program will examine the scaling issue (both material and geometric) by considering an oxidic simulant for one-dimensional shock propagation study.

PROGRESS ON THE MELCOR CODE

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Considerable progress has been made at Sandia in the past year on the MELCOR code for integrated severe nuclear reactor accident analysis. We have implemented a number of new models, and have made important improvements in some of the existing models. We have also extended the assessment base by comparisons with a number of experiments. Finally, we have undertaken a number of calculational studies of specific accident sequences at nuclear power plants. This paper will summarize this recent progress and indicate how the work has addressed the findings of the MELCOR Peer Review committee.

NEW MODELS

Fission Product Reactions with Surfaces. Fission product releases from the primary system depend not only on generation and deposition mechanisms, but also on re-vaporization of volatile fission products from surfaces after deposition has occurred. Chemisorption at these surfaces can significantly inhibit re-vaporization. New models for chemisorption have been incorporated in MELCOR, and in addition, some improvements in the equations of state of fission products have been made to give more realistic re-vaporization behavior.

Fission Product Aqueous Chemistry. Accurate tracking of iodine inventories in the atmosphere and water pools requires improved modeling of aqueous chemical processes. Work has been initiated on a new model that treats diffusion through the pool surface boundary layer, radiolytic chemical reactions, and selected equilibrium chemistry. A key feature is tracking the pool pH, which is known to be a strong determinant of iodine chemistry equilibrium.

BWR Core Spray. The currently released version of MELCOR cannot adequately treat the effects of the core spray in BWRs, but this mechanism is currently being considered by the industry as a potential accident management tool. We have begun work on a model that will include the effects of quench front movement, countercurrent flow restrictions and related effects.

MODEL IMPROVEMENTS

Core Flow Blockage and Reverse Flow Axial Gradient. Perhaps the most challenging improvements to MELCOR is our modified treatment of the effect on coolant flow of blockages due to core melt movement. Previous versions of MELCOR have handled this important phenomenon poorly because of inherent limitations in the underlying models coupling core behavior (modeled in COR) and hydrodynamic/thermodynamic behavior (modeled in CVH). A significant effort was required to allow smooth transitions from fully open channels to fully blocked flow, and (potentially) the reverse sequence. A closely related model improvement was to extend MELCOR's sub-grid axial gradient feature to conditions under which flow is reversed (e.g., in natural circulation modes).

Boron-Carbide Steam Reactions. We completed an upgrade of our treatment of chemical/physical interactions among water, B4C, and stainless steel. These processes are important for tracking iodine inventories in BWRs.

Fission Product Vapor Scrubbing by Water Pools. We have upgraded the earlier treatment of fission product scrubbing by replacing the SPARC-87 models with the more recent SPARC-90 models. Of particular importance is the proper treatment of vapor removal (not just aerosols). This improvement should allow more accurate predictions of decontamination factors for sequences in which vapor scrubbing is important.

ASSESSMENTS

Phebus FPT-0 . We have completed preliminary calculations of the first experiment in the Phebus FP series. The focus of these calculations is to understand the validity of MELCOR models for transport and deposition of fission products in the primary system.

VANAM (ISP-37). This is an integral containment behavior experiment, involving multiple volumes, steam injection, heat sources, and multi-component aerosol sources. We have completed the required pre-test calculations for the International Standard Problem, and will continue to be involved in the ISP activities throughout the program.

Fission Product Scrubbing by Water Pools. We have used data from a series of experiments sponsored by EPRI to evaluate the new models for fission product scrubbing (vapor and aerosol). These experiments involved injecting steam, non-condensable and aerosols into water pools.

Westinghouse Large Scale Tests. In support design certification, Westinghouse has conducted an extensive series of tests to evaluate the effectiveness of the passive containment cooling system of the AP600 reactor. The MELCOR code has been used to assess the related models for containment thermalhydraulics.

Accident Sequence Analyses. In view of the current interest in steam generator tube rupture (SGTR), we have conducted a series of studies of SGTR-induced accidents in the Surry plant. These calculations serve not only to assess current MELCOR models, but also to provide thermal-hydraulic boundary conditions for detailed fission product transport calculations with the VICTORIA code (see paper by Bixler, et al. in this conference). We have also completed an extensive series of accident sequence calculations for the Westinghouse AP600 reactor in support of the current ALWR certification process.

INVESTIGATION OF A STEAM GENERATOR TUBE RUPTURE SEQUENCE USING VICTORIA*

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VICTORIA-92 is a mechanistic computer code designed to analyze fission product behavior within the reactor coolant system (RCS) during a severe reactor accident. It provides detailed predictions of the release of radionuclides and nonradioactive materials from the core and transport of these materials within the RCS. These predictions account for the chemical and aerosol processes that affect radionuclide behavior. Coupling of detailed chemistry and aerosol packages is a unique feature of VICTORIA; it allows exploration of issues involving deposition, revaporization, and re-entrainment that cannot be resolved with other codes.

The purpose of the work reported here is primarily to determine the attenuation of fission products in the RCS and on the secondary side of the steam generator in an accident initiated by a steam generator tube rupture (SGTR). As a class, bypass sequences were identified in NUREG-1150 as being risk dominant for the Surry and Sequoyah pressurized water reactor (PWR) plants. SGTR sequences are the most probable of the bypass sequences studied in NUREG-1150. However, at the time that the work supporting NUREG-1150 was performed, the capability did not exist to adequately account for attenuation of fission products in the primary and secondary sides of the steam generator and in the pipework leading from the steam generator and venting into an auxiliary building. Assumptions that were made on attenuation of fission products in the steam generator and exiting pipework resulted in relatively small retentions there. Following the publication of NUREG-1150, the possibility was brought out that, if fission product attenuation in the steam generator were significant, this class of accident sequences might be shown not to be risk dominant after all. The work presented here addresses this outstanding issue. More recent information from the Surry IPE also identifies SGTR sequences as significant contributors to core damage frequency and indicates that they would result in relatively large fission product releases to the environment.

The Surry plants, which are 3-loop PWRs of Westinghouse design, were chosen as the basis for this investigation. Thermal-hydraulic data were calculated using MELCOR. These data were subsequently used as input to VICTORIA, which was used to analyze fission product release from the fuel and transport of these fission products through the RCS leading to the broken steam generator tube, through

*This work was supported by the United States Nuclear Regulatory Commission and was performed at Sandia National Laboratories, which is operated by the U.S. Department of Energy under contract DE-AC04-94AL85000.

the secondary side of the steam generator, out through the pipework leading from the steam generator to an atmospheric dump valve, which is assumed to stick open during the early portion of the accident, and into an auxiliary building. This analysis utilizes specific models of aerosol deposition in the steam separators and dryers that reside in the upper portion of the steam generator, as well as the more widely used models of aerosol deposition in pipes with bends. Moreover, VICTORIA accounts for fission products that deposit when temperatures are cooler and may revaporize later in the transient as temperatures rise, e.g., as the steam generator heats up.

The results of the VICTORIA analysis show where the fission products deposit within the RCS, the steam generator, and exiting pipework and what fraction of the fuel inventory is released into the auxiliary building. These results are used to evaluate the validity of the assumptions originally used in the work supporting NUREG-1150. Also, these results are compared with MELCOR and MAAP fission product release predictions for SGTR sequences.

THE SEVERE ACCIDENT RESEARCH PROGRAM PHEBUS F.P.: FIRST RESULTS AND FUTURE TESTS

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I. INTRODUCTION

PHEBUS FP is an international program investigating in integral in-pile experiments, on a reduced scale, LWR severe accident phenomena such as core degradation and the subsequent behavior of the released fission products. Core degradation is conducted up to the rather advanced stage of molten pool formation. Regarding the fission products, their release from the degraded fuel, their transport in the reactor cooling system, their deposition in the containment are being investigated with a special emphasis on the production of volatile iodine in the containment. The study of the less volatile fission product behavior under the extremely high temperature conditions reached in PHEBUS is also a major objective of the program.

Facility and test conduct have been described by several publications [1,2] and are not repeated here.

More than 35 organizations from Europe and overseas contribute to the scientific and technological development of the program. Its results improve the data base for

- accident management
- risk assessment
- licensing criteria
- backfitting of operating plants
- instrumentation and safety features of future plants,

through the development and validation of analytical tools used world-wide for LWR safety assessment.

II. FIRST TEST RESULTS

The first test of the program, FPT0, using trace-irradiated fuel, was performed in December 1993. A large effort was undertaken and is still in progress to analyze the numerous measurements recorded during the test and to examine the results of the post-irradiation examinations (tomographs, metallographies) and of the post-test analyses (physical and chemical analyses of the samplings).

The interpretation efforts have mainly concentrated on the bundle section [3,4]. Indeed, the radio-tomographs taken along the fissile length, have clearly shown that FPT0 reached a more advanced stage of bundle degradation than any previous experiment. About 50% of the fuel relocated, creating a large cavity and under it an accumulation of liquefied material at the level of the lower grid. Codes such as ICARE 2 and SCDAP are still unable to recalculate correctly the large bundle degradation observed, which has taken place rather early in the experiment at an estimated temperature of 2400 - 2500 K. Fuel dissolution by zirconium-rich mixtures formed during the cladding oxidation phase, perhaps enhanced by the presence of silver and steel from the control rod, seems to be the mechanism which governed the FPT0 bundle degradation, although the steam rich (i.e. oxidizing) atmosphere of the test was expected to limit the amount of metallic fuel dissolution.

The fuel bundle post-irradiation examination (PIE) is going on. It confirmed so far the amount of fuel relocation and the formation of a molten zone.

Regarding fission product release, FPT0 strongly indicates that temperature is not the only first order parameter controlling the rate of emission. Fuel dissolution and, to a larger extent, geometry changes during the degradation process also have an important effect.

Fission product retention in the primary circuit remained below the precalculated figures. Detailed evaluation of on-line and sampling instrument data is presently still ongoing, but about 25 activation and fission product isotopes (including those of less volatile FP) have been identified and quantified.

More astonishing is the behavior of iodine whose concentration in the atmosphere of the containment was at least two decades above that calculated by codes. It is clear that iodine radiolysis in the sump cannot account for such a production of volatile iodine. In addition the reaction with silver aerosols in the aqueous phase would tend to limit even more this production. Alternative mechanisms are being investigated. One of them could be gaseous iodine originating from the circuit.

III. FUTURE TESTS

Consequently, it is one of the main objectives of the coming FPT1 test to check the behavior of iodine in the circuit and in the containment under more prototypical conditions and with an improved instrumentation. To this end, the test will be carried out using irradiated BR3 fuel, providing a fission product to structural material emission ratio about 50 - 100 times greater than in FPT0. In addition the much larger quantity of iodine expected to be transported and to reach the containment should make it possible to determine more easily its chemical forms by means of advanced analytical techniques.

The PHEBUS test matrix is revised on a periodic basis, within the international scientific PHEBUS community, taking into account, a.o., the needs and priorities of reactor safety analysis and licensing. The late phase of core degradation is obviously a domain of large uncertainties and this is the reason why one of the next tests after FPT1, FPT4, will use a prefabricated bed of irradiated fuel which will be heated up to fuel melting and pool formation. It will enable to investigate the release and the transport of low-volatile elements (Ru, Sr, Ce, Pu....).
It is also planned to perform an air ingress test, FPT5, to investigate the degradation of the fuel rod remnants and the subsequent fission product release (especially Ru) after vessel rupture.

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PRIMARY SYSTEMS INTEGRITY
INTRODUCTION

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Assuring the integrity of the primary pressure boundary is essential to assuring the safe operation of nuclear power plants. As plants have aged, several issues have been identified that warrant examination to assure this integrity. For example, problems have emerged with embrittlement of the reactor pressure vessel, cracking of BWR core shrouds, cracking of PWR vessel head penetrations, and various forms of degradation in steam generator tubes.

The NRC has initiated research programs addressing the broad spectrum of materials aging and degradation issues for the primary pressure boundary components. The pressure vessel integrity program includes research on fracture mechanics evaluation methods, embrittlement estimation methods, embrittlement mechanisms, and experimental validation of both the fracture analysis methods and the embrittlement estimation methods. The irradiation effects research also addresses the effects of thermal annealing on reducing levels of embrittlement, and the reembrittlement rates for annealed materials.

The environmentally assisted cracking research addresses environmental effects on fatigue life and fatigue crack growth, irradiation assisted stress corrosion cracking, and environmentally assisted cracking of nickel alloys. The steam generator tube integrity program addresses inspection techniques, corrosion effects, and tube integrity analysis methods, and burst and leak tests. Finally, the inspection procedures and techniques program addresses techniques for inspecting pressure vessel and piping components, as well as efforts to quantify the initial flaw size distribution and density for typical reactor pressure vessels.

The presentations in Session 6 include a summary of the aging and degradation problems in the pressure vessel and steam generator pressure boundary, and summaries of the research on environmentally assisted cracking, steam generator tube integrity, thermal annealing, and the pressure vessel integrity research program. These presentations were selected to provide a brief overview of the nature of the problems and the specific research programs that have been implemented to address those problems.

Environmentally Assisted Cracking of LWR Materials*

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The program on Environmentally Assisted Cracking of Light Water Reactor Materials is currently focused on four tasks: fatigue initiation in pressure vessel and piping steels, fatigue and environmentally assisted crack growth in cast duplex and austenitic stainless steels, irradiation–assisted stress corrosion cracking of austenitic stainless steels, and environmentally assisted crack growth in high–nickel alloys.

Fatigue Initiation in Pressure Vessel and Piping Steels

Recent test data in Japan and the U.S. have shown potentially significant effects of light water reactor (LWR) environments on the fatigue resistance of carbon and low–alloy steels. Interim fatigue design curves that account for environmental effects were presented in NUREG/CR–5999. A more rigorous statistical analysis of the available data was developed in NUREG/CR–6237. These correlations are in excellent agreement with available laboratory data for loading histories with constant strain amplitudes and constant strain rates during the tensile portion of the loading cycle. However, actual loading histories are far more complex. Exploratory fatigue tests are being conducted with waveforms where the slow strain rate is applied during only a fraction of the tensile loading cycle. The results of such tests will be used to develop a "damage rule" that can be used to predict life under complex loading histories. Tests have shown that a minimum strain is required for environmentally assisted decrease in fatigue life. This threshold strain range appears to be ≈0.36% for the present heats of carbon and low–alloy steels. Present results suggest that slow strain rates during any portion of the tensile–loading cycle above the threshold strain are equally damaging to fatigue life.

Low–cycle fatigue tests were conducted on A302–Gr B low–alloy steel to verify the current predictions of modest decreases of fatigue life in PWR environments for very–high–sulfur materials. This steel had shown strong environmental enhancement of crack growth rate (CGR) and was presumed to be a "worst case" material based on comparison with the large body of CGR data available. While large effects of orientation and strain rate on fatigue life in air were observed, even this material showed only a marginal effect of PWR water on fatigue life.

Fatigue and Environmentally Assisted Crack Growth

Our studies of fatigue and environmentally assisted crack growth in cast duplex and austenitic stainless steels are intended to provide a technical basis for updating the correlations given in NUREG/CR–6176 for fatigue crack growth of austenitic stainless steels in LWR environments. Fracture–mechanics CGR tests were conducted on 1T–compact tension specimens of Type 316NG and 304 stainless steel (SS) and as–received and thermally aged CF–3 cast SS to investigate threshold stress intensity factors K_{th}^{EAC} for environmentally assisted cracking (EAC) in high–purity oxygenated water at

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* Job Code A2212; NRC Program Manager: Dr. M. McNeil

289°C. The CGRs were compared with the predictions of the correlations for wrought SSs in air in Section XI of the ASME Code and the correlations in oxygenated water from NUREG/CR–6176. Threshold behavior was clearly observed, and the dependence of K_{th}^{EAC} and ΔK_{th}^{EAC}, where $\Delta K_{th}^{EAC} = K_{th}^{EAC}(1 - R)$, on load ratio for specimens of Types 347, 316NG, and sensitized 304 SS, and for thermally aged CF–3 and CF–8 grades of cast SS, was determined.

Previous work at ANL, based on the results of slow–strain–rate tensile (SSRT) tests on austenitic stainless steels, had suggested that crack propagation is largely controlled by the rate of cathodic reduction of dissolved oxygen (DO), with a concomitant anodic dissolution process at the crack tip, and a model was developed to predict the dependence of CGRs on dissolved oxygen. Fracture–mechanics CGR tests on a variety of materials, e.g., Alloy 600, Alloy 690, sensitized Type 304, nonsensitized Type 316NG, and CF–3, CF–8, and CF–8M grades of cast SSs, show that in all the materials the CGRs in oxygenated water associated with EAC are approximately the same, despite significant differences in material chemistry, microstructure, and mode of crack propagation. This is consistent with the hypothesis that crack propagation is largely controlled by the rate of cathodic reduction of DO.

In NUREG/CR–6176, it was observed that there were few data on CGRs in deaerated water at CGRs of 10^{-10} m·s^{-1} or less, which are of the most interest. Fracture–mechanics CGR tests were conducted on compact–tension specimens of sensitized Type 304, Type 316NG, mill–annealed Alloy 600, and mill–annealed Alloy 690 in deaerated water containing B, Li, and dissolved H_2 at low concentrations at 289°C to provide data at low CGRs. The correlations in NUREG/CR–6176 provide conservative estimates of the observed growth rates. Revised correlations, which are based on the model for the dependence of CGRs on dissolved oxygen, give results in better agreement with the observed CGRs.

Irradiation Assisted Stress Corrosion Cracking

SSRT testing in simulated BWR environments on high- and commercial-purity (HP and CP) Type 304 SS has been performed to determine the effects of water chemistry on susceptibility to irradiation–assisted stress corrosion cracking (IASCC). The effect of DO level and electrochemical potential (ECP) on IASCC appears to be different for the HP and CP materials. The HP heats were less sensitive to DO level and ECP and were more susceptible to IASCC than the CP heats for all DO and fluence levels. No IASCC was observed in the CP heats for ECP <–140 mV SHE and DO <0.01 ppm.

To obtain some insight into the mechanism(s) of IASCC, we attempted to correlate the susceptibility determined by the SSRT tests with results of microchemical analysis of grain boundaries by Auger electron spectroscopy. The more susceptible HP heats were characterized by lower concentrations of Cr, Ni, and Li and higher concentrations of F and N on grain boundaries than those of the less susceptible CP heats. Although it is not clear at this time which elements and what metallurgical processes are most responsible for susceptibility to IASCC, the higher susceptibility and the weaker dependence on water chemistry of the HP heats is consistent with the hypothesis that there is a synergism between grain–boundary Cr depletion and F contamination in which F atoms play a catalytic role leading to accelerated intergranular SCC. A similar mechanism has been invoked to explain intergranular cracking in thermally sensitized materials and welds contaminated by F–containing welding fluxes.

Steam Generator Tube Integrity Program*

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Steam generators have historically been among the most troublesome of the major components in commercial pressurized water reactor (PWR) nuclear power plants around the world. Corrosion problems have afflicted steam generators from the very introduction of PWR technology. Shippingport, the first commercial PWR operated in the United States, developed leaking cracks in two Type 304 stainless steel (SS) steam generator tubes after only 150 h of operation. Because austenitic SS steam generator tubes were found to be susceptible to stress corrosion cracking (SCC) from both chlorides and free caustic, the decision was made in the late 1960s to instead use Alloy 600 tubes in the United States and most of Europe and Alloy 800 tubes in Germany. However, these changes did not eliminate corrosion–related degradation in steam generator tubing.

Degradation of steam generator tubes has resulted from corrosion and wastage, pitting, denting, SCC, and intergranular attack. Cracking has been experienced on both the primary and secondary sides of the steam generator tubes. Both axial cracks and circumferential cracks have occurred. Cracks are most prevalent in crevices (tube sheet, tube support plates), under sludge piles (top of tube sheet) and other deposits, and in areas of high strain or residual stress such as dented regions, roll transitions, and tight U-bend radius locations. Cracks have also occurred, to a lesser degree, in free-span zones of steam generator tubes, in tubes repaired by sleeves, and in tube plugs. Cracking of various forms and at various locations is the degradation type of most interest currently.

The important information needed for evaluating integrity and assessing various defect–specific management schemes includes the reliability of the screening inspection for identifying specific flaws and conditions, the probability of flaw detection for specific flaws as a function of flaw size or other characterization, the accuracy of flaw sizing or other flaw severity parameters, flaw initiation times, flaw progression and propagation rates as a function of the environment and location, and models relating failure pressure, failure mode, and leak rate to flaw severity and nondestructive measurements.

The NRC has conducted research on steam generator tube integrity and inspection since 1977. Completed research on steam generator tube integrity has produced an extensive data base on failure pressures (burst and collapse mode), leak rates, and eddy current inspection results from variously degraded steam generator tubes. From this data base predictive models for the integrity of degraded tubes were formulated.

The purpose of the new program is to produce and/or update the information, data, and predictive modeling discussed above, but focussing on the degradation types and inspection methods that are of current interest (i.e., various forms of cracks, multifrequency eddy current test equipment, modern probes and equipment, voltages and other parameters, and complementary inspection methods such as ultrasonic

* Job Code: W6487-5; NRC Program Manager: Dr. J. Muscara

testing). This new research will allow evaluations of various defect-specific management schemes and will support current NRC rulemaking needs and activities.

The areas addressed by the program include (a) evaluation of personnel, procedures, and equipment used for in-service inspection (ISI) of steam generator tubes, and recommendations for criteria and requirements to improve the reliability and accuracy of ISI; (b) validation and improvement of correlations and models for evaluating integrity and leakage of degraded steam generator tubes; and (c) validation and improvement of correlations and models for predicting degradation generation and progression in steam generator tubes as a function of aging, including such effects of the operational environment as temperature, dry-out and concentration conditions, stresses, and primary- and secondary-side water chemistry.

The studies in the program will focus primarily on Alloy 600 steam generator tubing in the mill-annealed condition, because this is the tubing material that is (and will be) present in plants that have not replaced steam generators and is more susceptible to cracking than replacement materials such as thermally treated Alloy 600 or 690. Although most steam generators that use mill-annealed Alloy 600 will probably require replacement eventually, the behavior of this material will be of concern for many more years. Testing will also be performed on thermally treated Alloy 600 and 690. Although these materials are expected to be much less susceptible to degradation than the mill-annealed Alloy 600, the ability to predict their behaviors is still necessary.

The bulk of the materials used in the program will be exposed in laboratory testing to simulated operating conditions and environments. Because some of the laboratory environmental conditions represent accelerated conditions, and because service degradation, tubing conditions, and in-service operating and inspection conditions cannot always be faithfully represented in laboratory conditions and specimens, this program seeks to and will use service-degraded tubing and perform in-situ inspections in generators removed from service for correlation with and validation of the experimental data, integrity- and degradation-predictive models, and inspection capability. Comparisons will be made with the morphology and character of service-degraded flaws to help ensure that the flaws produced in the laboratory and used for studies on inspection reliability, and pressure- and leak-testing will be as realistic as possible. The reliability of flaw detection and accuracy of flaw-sizing data will also be evaluated with typical in-service inspection personnel, procedures, and equipment.

Embrittlement Recovery Due to Annealing of Reactor Pressure Vessel Steels

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Summary

Embrittlement of reactor pressure vessels (RPVs) can be reduced by thermal annealing at temperatures higher than the normal operating conditions. Although such an annealing process has not been applied to any commercial plants in the United States, one US Army reactor, the BR3 plant in Belgium, and several plants in eastern Europe have been successfully annealed. All available Charpy annealing data were collected and analyzed in this project to develop quantitative models for estimating the recovery in 30 ft-lb (41 J) Charpy transition temperature (TT) and Charpy upper shelf energy (USE) over a range of potential annealing conditions.

Models for estimating the degree of recovery due to post-irradiation annealing have been developed previously. All of these preliminary models were calibrated to high flux test reactor data, and their relevance to low flux power reactor situations is limited. They also do not reflect the physical interaction between the temperature of annealing, flux, and the hardening features that are affected by annealing.

In the current work, pattern recognition, transformation analysis, residual studies, and the current understanding of the mechanisms involved in the annealing process were used to guide the selection of the most sensitive variables and correlating parameters and to determine the optimal functional forms for fitting the data. The resulting models were fitted by nonlinear least squares. The use of advanced tools, the larger data base now available, and insight from the surrogate hardness data produced improved models for quantitative evaluation of the effects of annealing. The quality of models fitted in this project was evaluated by considering both the Charpy annealing data used for fitting and a surrogate hardness data base. The standard errors of the resulting recovery models relative to calibration data are comparable to the uncertainty in unirradiated Charpy data.

The USE recovery model may predict over-recovery of USE due to annealing (i.e., USE after annealing exceeding the USE for the unirradiated material), as was evident in the data. The TT recovery model does not allow over-recovery, because no examples of

over-recovery were observed in the data.

The models have several physical features built into them, including nonlinear effects of annealing temperature as well as matrix defect and copper-rich precipitate hardening features. The TT model also has a limit on effective Cu based on solubility considerations and an effect of flux for annealing temperatures below 750°F. The flux effect is tentative, having been calibrated to a mix of hardness and Charpy data, but it is consistent with the large hardness data base with a range in flux spanning more than two orders of magnitude. The flux effect, like the Cu effect, varies with annealing temperature because of the different responses of the various hardening features. The TT model is consistent with an extensive hardness data base, limited shift data not used for fitting, and the current understanding of both embrittlement and annealing mechanisms. The model also provides some important insights lacking in previous treatments, suggesting that flux effects and secondary composition variables can be neglected at high annealing temperatures. While requiring additional confirmation, this is a conclusion of enormous importance. The results can help guide future research studies aimed at refining the predictions and optimizing annealing treatments, considering the effects of both recovery and re-irradiation embrittlement. This work also demonstrates that microhardness recovery is a good surrogate for transition temperature shift recovery and that there is a high level of consistency between the observed annealing trends and fundamental models of embrittlement and recovery processes.

An embrittlement data base is being analyzed under the same project to develop improved models for estimating irradiation embrittlement. The analysis data base is derived from data reported in the Power Reactor Embrittlement Data Base compiled and maintained by Oak Ridge National Laboratory. As in the annealing study, current understanding of the mechanisms involved in embrittlement of RPV steels is combined with advanced analysis techniques to guide the formulation of new models for shift in transition temperature and drop in Charpy USE due to irradiation. The status of this effort will be summarized briefly.

REACTOR PRESSURE VESSEL INTEGRITY RESEARCH AT THE OAK RIDGE NATIONAL LABORATORY

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Maintaining the integrity of the reactor pressure vessel (RPV) in a light-water-cooled nuclear power plant is crucial in preventing and controlling severe accidents which have the potential for major contamination release. The RPV is the only key safety-related component of the plant for which a duplicate or redundant backup system does not exist. It is therefore imperative to understand and be able to predict the capabilities and limitations of the integrity inherent in the RPV. This requires a quantitative understanding of the initial condition of the vessel and the amount of irradiation-induced degradation of the RPV's fracture resistance, a methodology for assessing the impact of current toughness on integrity, and a means of deciding under what conditions the vessel can continue to operate safely. For this reason, the U.S. Nuclear Regulatory Commission (USNRC) has established three related research programs at the Oak Ridge National Laboratory (ORNL) to provide for the development and confirmation of the methods used for (1) assessing the effects of irradiation on the RPV materials, (2) establishing the irradiation exposure conditions within the RPV, and (3) analyzing the overall structural and fracture analysis of RPVs.

ORNL Embrittlement Data Base (EDB) and Dosimetry Evaluation (DE) Program

The objective is to develop, maintain, and upgrade computerized data bases, calculational procedures, and standards relating to RPV fluence spectra determinations and embrittlement assessments. Uncertainties associated with the predictive methods and data bases are determined. As part of this program, the nuclear radiation embrittlement information from radiation embrittlement research on nuclear RPV steels and from power reactor surveillance reports is maintained in a data base published on a periodic basis. The EDB work consists of verifying the quality of the EDB, providing user-friendly software to access and process the data, and exploring and confirming embrittlement prediction models. The DE work consists of maintaining and upgrading validated neutron and gamma radiation transport procedures, maintaining cross-section libraries with the latest evaluated nuclear data, and maintaining and updating validated dosimetry procedures and data bases. The information available from this program provides data for assisting the Office of Nuclear Reactor Regulation, with support from the Office of Nuclear Regulatory Research, to effectively monitor current procedures and data bases used by vendors, utilities, and service laboratories in the pressure vessel irradiation surveillance program. Particularly significant recent accomplishments include the completion of update 6 of the Power Reactor Embrittlement Data Base (PR-EDB), the completion of the LSL-M2 dosimetry cross-section data base update, the modification of the BUGLE-93 cross-section library to include two thermal neutron groups with upscattering, and the development of a self-shielded 199-neutron-42-photon-group library.

Heavy-Section Steel Irradiation (HSSI) Program

The goal of this program is to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior and, in particular, the fracture toughness properties of typical pressure vessel steels as they relate to light-water RPV integrity. Effects of specimen size; material chemistry; product form and microstructure; irradiation fluence, flux, temperature, and spectrum; and postirradiation annealing are being examined on a wide range of fracture properties, and results are being transferred into national and international codes and standards activities. Particularly significant accomplishments recently completed include the development and use of fracture data from the irradiated low-upper shelf (LUS) Midland Weld WF-70 to benchmark American Society of Mechanical Engineers (ASME) methods currently in use. The ASME lower-bound KIc curves were shown to be very conservative whereas lower-bound curves of the newly proposed master curve approach bounded the data more accurately. Testing of precracked Charpy specimens as three-point bend bars was used to evaluate their potential for direct measurement of fracture toughness in reactor surveillance capsules. Initial evaluation of notched round bar specimens fabricated from weld 72W material for potential surveillance capsule applications agreed well with the 1T compact data in the middle of the transition regime but showed higher toughness at lower test temperatures. Comparative effects of annealing of Charpy and fracture mechanics specimens from the LUS welds from the Midland beltline and nozzle course welds, as well as Heavy-Section Steel Technology (HSST) Plate 02 and HSSI weld 73W were examined. Characterization of long-term (~ 100,000 h) thermally aged and neutron-irradiated surveillance materials by atom probe field ion microscopy indicated that thermal aging did not result in any copper clustering or compositional changes in the ferrite matrix.

Heavy-Section Steel Technology Program

The goal of the HSST Program is the development, validation, and application of technology for the assessment of fracture prevention margins in commercial RPVs. Its scope includes development and experimental validation of analysis methods, development of testing techniques and generation of materials property data, integration of analysis methods and materials property data, and the transfer of the results to national and international codes and standards bodies. The major focus of the HSST Program is on behavior of shallow flaws, since the initiation probabilities associated with these flaws under pressurized thermal shock (PTS) loading conditions can strongly influence the predicted failure probabilities for RPVs. Behavior of shallow flaws during a PTS transient would be influenced by the material/fracture properties and mechanical interactions associated with the cladding, heat-affected zone, and the near-surface material. Thus, coordinated experimental and analytical studies are being conducted to provide a quantitative description of effects of the cladding overlay on fracture behavior of shallow surface flaws, including a quantitative description of cladding effects on the fracture behavior of shallow-finite-length surface flaws in RPVs under uniaxial and biaxial loading conditions and a basis for improved treatment of surface-flaw geometries in fracture assessment procedures applied to PTS and pressure temperature (P-T) limit transients. Changes in the conditional probability of cleavage initiation associated with selected changes in a clad RPV showed reductions in probabilities of initiation. These were achieved by combining effects of semicircular flaw geometry, clad toughness, and clad yield stress. Most pronounced effects were for shallower flaws where the minimum average reduction, over the range of shallow flaw depths considered, was about one order of magnitude.

Results from the ORNL RPV integrity studies provide information needed to aid in resolving major regulatory issues facing the USNRC which involve RPV integrity issues. These issues include PTS, operating P-T limits, low-temperature overpressurization, the specialized problems associated with LUS welds, and the transfer of data from small-scale surveillance specimens to application in RPV structural integrity assessments.

PRELIMINARY RESULTS OF THE XR2-1 EXPERIMENT EXAMINING METALLIC CORE-MELT BEHAVIOR IN DRY-CORE BWR ACCIDENTS: THE EX-REACTOR EXPERIMENTS

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The Ex-Reactor experiments are being conducted at Sandia National Laboratories to aid in the resolution of a major uncertainty in the core melt progression process associated with a severe accident in a boiling water reactor (BWR) where loss of reactor core coolant inventory has occurred. Specifically, the class of accidents of concern are those that involve core melting under dry core conditions, such as an unrecovered Station Blackout accident with manual vessel depressurization. The intentional vessel depressurization maneuver cools down the core to gain additional time before the onset of severe fuel damage so that other vessel recovery options can be exercised, such activation of the low pressure injection system. In the event that core cooling is not regained, the severe core damage processes resulting from the continued core heatup takes place under comparatively "dry core" conditions, because of the low water level in the reactor vessel and the very low steam flow through the core. This is in contrast with the "wet core" conditions typical of a TMI-like accident where vessel depressurization does not occur, and water remains in the lower core region as fuel damage and material melting takes place.

The central issue addressed by the Ex-Reactor experiments concerns the tendency of the initially melting and draining metallic core materials(stainless steel control blades and zircaloy fuel canisters and cladding) to form blockages in the lower core region as occurred in the TMI-2 accident, or alternatively, for these materials to drain continuously into the lower vessel head region because of the dry-core conditions. The Ex-Reactor experiments address this issue by simulating the conditions in the lower BWR reactor core at the time of initial metallic melt relocation.

The general approach taken in the Ex-Reactor experiments is to simulate the lower 1/2 to 1 meter of the BWR core geometry in full scale at the time that the molten metallic core materials are beginning to drain from the upper regions of the core into the lowermost regions. A test section is constructed, including important geometrical features such as zircaloy fuel canister walls, B4C-filled stainless steel control blade structure, fuel canister nose pieces, lower core plate, bladed and unbladed inter-canister gaps, and so on. Prototypic materials are used in the conduct of these experiments. The lateral scale of the XR2-1 test section is designed to represent all of the lateral degrees of freedom with respect to the draining of molten materials, and was selected based on symmetry principles (the XR2 test section is approximately a 1/8 symmetric section extracted from the 4-canister repeating array of the BWR core).

The tests are conducted by pouring molten metallic materials (molten control blade steel and molten zircaloy components - fuel rod cladding and channel boxes) into the upper, open end of the heated test section, thereby simulating the melting and draining of the upper core metallic materials in the overheating reactor. The test sections are heated so that a prototypic axial thermal gradient is imposed over the length of the test section. The test sections are instrumented with thermocouples to measure the thermal gradient prior to introducing the molten metals, and to characterize the melt flow and blockage behavior of the melt flowing into the test section.In addition to thermocouples, a real-time x-ray imaging system provides a video image of the melt flow as it enters the test section, showing flow behavior and blockage formation. The object is to characterize the nature of any blockages that are formed as the melt enters the test geometry, and to provide information on melt drainage pathways through the lower core region.

The flow of molten materials into the package is provided by an inductively powered melter system, situated above the test section. An important aspect of the Ex-Reactor experiments is that the total amount of metallic melt delivered to the test section is representative of that available from the entire axial extent of the core above the lower 1/2 to 1 meter region, and so, the tests include the full amount of incoming melt mass and enthalpy that would be typical of actual accident conditions. This feature, together with the 1/8th symmetry unit cell cross section design, and the use of actual full scale BWR core components and prototypic materials, provides essentially full scale duplication of the dominating core behavior during the crucial melt progression events occurring at this time in the accident.

A major effort in the conduct of the Ex-Reactor program has been the technological development of a unique and innovative radiant cavity melt delivery system. The purpose of the system is to deliver molten control blade alloy and zircaloy to the top of the test bundle in the proper later locations and at prototypic melting rates. The device makes use of a ~3000K radiant cavity to melt individual stainless steel/B_4C or zircaloy wires,which are fed into the cavity by a wire drive mechanism. The wires are melted as they enter the cavity and drain into the top of the test section at the desired locations. The melt delivery rate is precisely controlled by the wire feeding rate into the radiant cavity. The inductively heated radiant cavity provides up to 200 kW which both melts the wire and provides the heat source for establishing the thermal gradient on the test section situated below the melter.

The XR2 tests are expected to show either that the draining metallic zircaloy melt is so aggressive in eroding the lower core structures that melt drainage into the lower plenum is anticipated (continuous melt drainage pathway), or will indicate that formation of robust in-core blockages leading to a TMI-2-like scenario is instead favored. In the event that blockage, and not melt drainage, is observed, the configuration revealed by the experiments can be further analyzed by other analytical means to evaluate the timing of any subsequent metallic blockage meltout or ceramic melt release. Preliminary results of test XR2-1 will be presented.

Steady-State Observations and Theoretical Modeling of Critical Heat Flux Phenomena on a Downward Facing Hemispherical Surface*

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Summary

Observations of the downward facing boiling process on the outer surface of a heated hemispherical vessel were performed at various heat flux levels under steady state conditions in a subscale boundary layer boiling (SBLB) test facility. The objectives were to seek a fundamental understanding of the boiling phenomena, particularly the behavior of the two-phase boundary layer near the critical heat flux (CHF) state, and to determine the spatial variation of the local critical heat flux along the vessel outer surface. This technical information represents one of the essential elements that are needed in assessing the efficacy of external cooling of core melt by cavity flooding as a severe accident management strategy.

The SBLB facility, developed specifically to simulate the phenomena of downward facing boiling on the external bottom surface of a reactor vessel, consisted of a pressurized water tank with a condenser unit, an interchangeable test vessel, a data acquisition system, and a high-speed photographic system. The test vessel employed in the present experiments was made of aluminum and had two separate parts, i.e., a heated lower hemispherical part and an unheated upper cylindrical part. The upper part was 12" (0.305 m) in diameter and 24" (0.61 m) in height, whereas the lower part was 12" (0.305 m) in diameter and 6" (0.152 m) in height. These two parts were used to simulate the lower head and the cylindrical wall section of a reactor vessel. The lower hemispherical part was further divided into five segments, each having the same surface area. Uniformly spaced independent heating elements were installed on the interior side of each segment to simulate decay heating of the vessel wall by a corium pool. The power supply to the heating elements in each segment was controlled separately, with the resulting local heat flux covering the anticipated range up to 1.0 MW/m^2. This design allowed for the attainment of the critical heat flux condition in a local portion of the vessel outer surface with the neighboring portions still in the nucleate boiling regime.

Special precautions were taken in conducting the experiments as the heat flux level approached the CHF point. Local meltdown of the test vessel would occur if the power to the heating elements were not discontinued before local dryout took place on the vessel outer surface. To prevent this from happening, a temperature control system was employed. Thermocouples were embedded at various locations within the wall on the interior side of the lower segments. The temperature responses of these thermocouples were fed into a computer and then checked against a set point value. If any of the temperatures inside the test vessel wall exceeded the set point, the power to all the heating elements would be discontinued by triggering the solid state relays that were part of the heater circuits. To approach the CHF state, the local heat flux level was raised by a small step each time. At a level below the critical heat flux, the local wall

temperature would rise rather moderately toward a new steady-state value. Once the critical heat flux limit was exceeded, an abrupt increase in the local wall temperature beyond the set point value would take place. The occurrence of the local CHF state was detected by observing the time response of the local wall temperature.

Flow observations and heat transfer measurements were made under steady-state conditions, covering the entire range of heat fluxes in the nucleate boiling regime. At high heat flux levels, a cyclic ejection process was clearly observed. Large and flattened vapor masses, being squeezed up against the wall by the buoyancy force, were found to grow periodically in the bottom center region of the test vessel. They were then ejected violently upward in random directions. The large vapor masses carried away the local vapor bubbles downstream and transformed into elongated vapor slugs. A close-up view of the vapor slugs revealed the existence of a thin liquid film underneath each elongated vapor mass. Apparently, it was this thin liquid film that prevented local dryout of the heating surface from occurring. Steady-state boiling data were obtained and were found to be consistently higher than those determined from transient quenching experiments, particularly at low heat flux levels. This was probably due to the fact that the time duration associated with the final stage of quenching was too short for the two-phase flow to develop. The nucleate boiling portion of the boiling curve deduced from the transient experiments was in error, especially at low heat fluxes. However, the difference between the steady-state and transient data became smaller as the heat flux level was increased. The local CHF value obtained in the steady-state experiments appeared to be very close to those deduced from the quenching data. The two sets of results tended to merge together as the condition of critical heat flux was approached.

Based upon the experimental observations, key parameters and flow quantities that contribute to the occurrence of the CHF state were identified. This information was employed to develop an advanced hydrodynamic CHF model for boundary layer boiling on a downward facing hemispherical vessel. The model considered the existence of a micro-layer underneath an elongated vapor mass or slug that grows on the vessel outer surface. The micro-layer was treated as a thin liquid film with numerous vapor jets that penetrate through the entire liquid film. The size of the vapor jets was on the same micro-scale as the liquid film and had the characteristic dimensions dictated by the Helmholtz instability. Fresh liquid was supplied to the micro-layer by the two-phase boundary layer flow. A boundary layer analysis, treating the two-phase motion as a separated flow, was performed to determine the local liquid feeding velocity. The local CHF limit was reached if, before the departure of the vapor slug, the liquid film was depleted by boiling due to an inadequate supply of fresh liquid to the micro-layer. The model provides a physical explanation for the spatial variation of the critical heat flux along the vessel outer surface observed in the SBLB experiments and the weak dependency of the local CHF values on the physical size of the vessel. Note that in all the existing CHF models developed by previous investigators, the critical heat flux was assumed constant, independent of the spatial location. The present model represents the first attempt to discern the spatial variation of the critical heat flux along the heating surface and is the only model applicable to the reactor situation.

* This work was supported by the U. S. Nuclear Regulatory Commission under Contract No. NRC-04-93-061.

HYDROGEN DETONATION AND DETONATION TRANSITION DATA FROM THE HIGH-TEMPERATURE COMBUSTION FACILITY*

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The BNL High-Temperature Combustion Facility (HTCF) is an experimental research tool capable of investigating the effects of initial thermodynamic state on the high-speed combustion characteristics of reactive gas mixtures. The overall experimental program has been designed to provide data: to help characterize the influence of elevated gas-mixture temperature(and pressure) on the inherent sensitivity of hydrogen-air-steam mixtures to undergo stable detonations; on the potential for flames accelerating in these mixtures to transit into a detonation; on the effects of venting on the detonation-transition process; on the phenomena of initiating detonations by jets of hot reactant products issuing into these mixtures; and on the capability of detonations propagating in one confined space to transmit into a connected, larger confined space. This summary presents results obtained from the completion of two test series, i.e., the intrinsic detonability test series and the deflagration-to detonation (DDT) test series. A brief description of the facility is provided.

The central feature of this facility is the large detonation vessel (LDV), which is a 27-cm diameter, 21.3-m long cylindrically cast, stainless steel detonation tube. The tube is composed of seven modular, flanged, 3.05-m long, ASME-certified, high-pressure vessels. Four other test sections have been fabricated, which can be individually attached in between these vessels in a serial manner, to investigate the effects of gas venting on the characteristics of high-speed combustion. These sections allow test gas and combustion-product gases to be vented from the tube as a result of the propagating flame front. The Maximum Allowable Working Pressure of all test vessel sections of the LDV is approximately 10 MPa (100 atm). Each section of the vessel can be heated independently to 700K, using a combination of ceramic heating blankets that surround each modular section of the LDV. Various combinations of hydrogen-air-steam test mixtures, with initial temperatures and pressures respectively ranging between 300K to 700K and 100 KPa to 300 KPa, can be quickly produced and injected into the heated detonation tube, just prior to ignition. Ignition of the test mixture can be generated in one of several ways: directly by a high-energy spark discharge or by glow-plug igniters; and indirectly by the transmission of blast waves into the test gas mixture from an oxyacetylene gas driver. Accommodations for using high-explosive (HE) discharges for initiating detonation in the test gas mixture were factored into the design of the LDV. Instrumentation and a data acquisition system permit the measurements of initial/final gas composition, dynamic pressure and temperature, flame velocity, detonation wave speed, and detonation cell width.

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*This work was performed under the auspices of the U. S. Nuclear Regulatory Commission. This program is a joint research program involving the U. S. Nuclear Regulatory Commission and the Nuclear Power Engineering Corporation (NUPEC) of Japan.

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The first series of tests in the HTCF were conducted to characterize the sensitivity of a hydrogen-air mixture to undergo stable detonations and to investigate the effects of initial temperature (and pressure) and steam addition on detonation cell structure. Basic dynamic detonation parameters measured were: detonation cell size, using the classic smoked-foil technique, and detonation wave speed, using ion and/or photodiode probes. Comparison of measured wave speed with the calculated Chapman-Jouguet (CJ) velocity was primarily employed to determine if stable detonations were achieved at those locations containing the smoked foil. Data on detonation cell width and wave speed are compared with predictions that are based on the 1-dimensional Zel'dovich, von-Neumann, Doring (ZND) gas dynamic model. Data generated from tests performed at temperatures of 300K, 400K, 500K, and 650K and steam concentrations between 10 percent and 50 percent (by volume) are presented, and the trends in the data are compared with the 1-dimensional model predictions. The results from this series of tests show the following:

- For the hydrogen-air-steam mixtures tested, increasing the initial test temperature decreases the detonation cell size, i.e., the mixture sensitivity increases.
- For a given initial temperature, steam dilution increases cell size, i.e., decreases sensitivity.
- The mitigating effects of steam are reduced with increasing initial temperature.
- Initial pressure has a small effect on the detonation cell size.
- The ZND model predicts the dominant trends in cell size with mixture composition, initial temperature, and pressure.

The second series of completed tests investigated the effects of initial temperature on the process of deflagration-to-detonation transition (DDT). A glow plug was used to ignite the test mixture, and a series of orifice plates, mounted axially within the tube, was employed to enhance the flame-acceleration process by creating turbulence upstream of the advancing flame. Trends in the measured wave speed with mixture concentration, for given initial temperatures, were examined to determine those mixture compositions where the onset of noticeable changes in the flame-acceleration process occurred. Depending on the measured wave velocity, these regions were classified as either the "choking" regime or the "quasi-detonation" regime. Quasi-detonations were obtained in hydrogen-air mixtures as low as 11 percent at initial temperatures of 650K. However, it appears that the DDT limit criteria, derived from tests conducted at room temperature, cannot be fully extended to encompass high-temperature test conditions. Possible reasons for this observation will be discussed.

RECENT EXPERIMENTAL AND ANALYTICAL RESULTS ON HYDROGEN COMBUSTION AT KURCHATOV INSTITUTE*

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A review of hydrogen combustion research at Kurchatov Institute is presented. Results on the loads from different combustion and explosion modes and on spontaneous detonation scaling methodology are summarized.

3D computer codes 3ET and B02 have been developed for description of loads from detonations. Series of large scale H2 detonation experiments have been carried out in RUT facility (16 - 25 % H2, two initiator locations). Experimental results form a data base on detonation loads on reactor-relevant scale and complex 3D geometry. B02 code was evaluated against experimental data. Good agreement of main loading parameters was observed for fully developed detonations. Effect of DDT location on loads has been studied experimentally (UTR facility) and numerically in 1D geometry. It was found, those peak overpressures depend strongly on DDT location, and can be significantly higher than that from detonation. Impulses depend mainly on the mixture volume. Data of large scale detonation, deflagration and DDT experiments (RUT facility) confirm these observations. Experimental data on turbulent flame propagation and on resulting loads (including DDT events) were received in large-scale RUT experiments.

Criterion for spontaneous detonation onset possibility and its application to severe accidents in a nuclear power plant is discussed. Theoretical and experimental results on spontaneous detonation onset conditions are summarized. Three series of large scale turbulent jet initiation experiments have been carried out in KOPER facility (50 m^3 and 150 m^3). Series of jet initiation experiments in initially confined H2 - air mixtures have been carried out in KOPER facility (20 - 46 m^3). Turbulent deflagration/DDT experiments were carried out in large scale confined volume of 480 m^3 in RUT facility. Transition to detonation was observed at min. of 12.5% H2. Results showed, that the characteristic volume size should be used for conservative estimates in accident analysis. Series of experiments on detonation transition from one mixture to another of lower sensitivity has been carried in DRIVER facility. The experiments were aimed on the estimation of the minimum size of a detonation kernel. The received results are in a good agreement with the 7 cell width criterion.

Results of combined hydrogen injection/ignition experiments are presented. The experiments are aimed on the investigation of possible consequences of deliberate ignition at dynamic conditions. Experiments include the large-scale tests on the effects of igniter location, ignition time, injection rate (0.1 - 1 kg/s) and injection point on the combustion mode. The possibility of initiation of local detonations due to ignition at dynamic conditions was observed in the tests. The experiments showed a good local mixing and large-scale hydrogen concentration nonuniformities. Multiple explosions at continuous injection were observed. Analysis of the experimental data showed applicability of 7 cell width criterion to dynamic conditions. The sum of the results on the scaling of spontaneous detonations is discussed in connection with the strategy of hydrogen mitigation at severe accidents.

* Research sponsored by the US NRC and KfK, Germany

SCDAP/RELAP5 Code Development and Assessment*

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The SCDAP/RELAP5 computer code is designed to describe the overall reactor coolant system (RCS) thermal-hydraulic response, core damage progression, and fission product release and transport during severe accidents. The code is being developed at the Idaho National Engineering Laboratory (INEL) under the primary sponsorship of the Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission (NRC). The current production version of the code is SCDAP/RELAP5/MOD3.1. Although MOD3.1 contains a number of significant improvements, a set of new models to treat the behavior of the fuel and cladding during reflood has had the most dramatic impact on the code's ability to predict the accelerated heating and melting of hot, damaged fuel assemblies during water injection. Although the detailed mechanisms for these processes are not completely understood, these models, describing the cracking/spalling of oxidized fuel rod cladding during reflood, and the resulting oxidation of the underlying Zircaloy and relocating liquefied U-Zr-O, provide a reasonable estimate of the experimentally-observed bundle temperatures, hydrogen production, and changes in bundle geometry. This paper provides a brief description of the new models and initial results of a systematic assessment of these models using data from electrically-heated experiments conducted in the German CORA facility and nuclear heated experiments conducted in the Idaho Power Burst and Loss of Fluid Test Facilities. A brief summary of other ongoing model development activities is also included.

The new models for SCDAP/RELAP5/MOD3.1 have been introduced in two steps. The initial version of SCDAP/RELAP5/MOD3.1, which was released in 1994, contained two bounding correlational-based models to treat the cracking/spalling of the oxidized fuel rod cladding. One model, a local cracking model, cracked the protective oxide layer at a given elevation when oxidation-embrittled Zircaloy cladding cooled rapidly to a temperature between 1150 to 1560 K. The other model, a global cracking model, cracked the protective oxide layer over the full height of the fuel rod cladding as soon as water was added. In both cases, enhanced oxidation and heating then occurred due to the oxidation of the hot Zircaloy layer exposed by the cracking of the protective oxide. Mass diffusion of steam to the surface of the cladding was the oxidation-rate limiting process once the oxide was cracked. The flooding rates and heat transfer during the reflood process were described using standard RELAP5 thermal-hydraulic constitutive models. After a systematic assessment of these models, an additional model was then added to MOD3.1 to treat the oxidation of the liquefied U-Zr-O moving down the outer

* Work supported by the U.S. Nuclear regulatory Commission, Office of Research, under DOE Contract No. DE-AC07-76ID01570.

surface of the fuel rod cladding. During reflood, molten U-Zr-O is formed during the initial increase in temperature resulting from the oxidation of the hot Zircaloy layer exposed by oxide cracking. This model, based upon the observation of U-Zr-O rivulet and droplet flows in the German CORA experiments, describes the motion of droplets of liquefied U-Zr-O and the resulting oxidation of those drops.

The assessment of the cracking/spalling models following the general release of MOD3.1 in 1994 indicated a significant improvement in SCDAP/RELAP5 predictions of the response of the experimental bundles during reflood. Where previous versions of the code had predicted little, if any, increase in bundle oxidation and heating during reflood, SCDAP/RELAP5/MOD3.1 indicated a significant increase in the oxidation of the bundles during the initial phases of the reflood process. In addition, MOD3.1 predicted that the oxidation of the TMI-2 core during the B-pump transient, following initial core uncovery, was a significant factor in the formation of a core blockage and the overall pressure response during this phase of the accident. However, the assessment revealed that the total impact of the oxidation process during reflood was still under predicted. It appeared that the total hydrogen being produced during reflood experiments was still too low by approximately a factor of two. In addition, the increase in bundle temperatures and melt relocation during reflood also appeared to be systematically underpredicted. In the case of TMI-2, the extent of the blockage and subsequent ceramic molten pool in the core was consequently underpredicted, leading to the eventual cooling of the molten pool in the core and no relocation of ceramic melt into the lower plenum.

As a result of those assessment results, the new model describing the oxidation of the liquefied U-Zr-O was added to MOD3.1 and a systematic assessment of the combined models was undertaken. Although the assessment will continue through the summer of 1995, initial results indicate that the combination of the U-Zr-O oxidation model and oxide cracking models appears to better predict the overall response of the experimental bundles during reflood. For example, the assessment results for one experiment, CORA-13, indicates that the U-Zr-O oxidation model in combination with the local cracking model matches the measured total hydrogen production, bundle temperature response during reflood, and final geometry of the bundle well within the estimated experimental uncertainties for those parameters.

Although the assessment of the new reflood and oxidation models has been one of the primary activities in the overall development and assessment of SCDAP/RELAP5, the development of new models to treat the late phase of a severe accident has also been an important activity over the past year. These new models will be released under the designation SCDAP/RELAP5/MOD3.2. Detailed model designs were developed, and peer reviewed, for the (a) formation, heating, and melting of debris beds, (b) molten pool natural circulation heat transfer and crust failure, (c) upper plenum structure heating and melting, and (d) debris bed thermal-hydraulics. New or improved models to treat the unique features of ALWRs including the effects of ex-vessel flooding have also been incorporated during the last year. Oak Ridge National Laboratory has the lead role in developing the upper plenum heating and melting models for BWR applications.

Recent SCDAP/RELAP5 Improvements for BWR Severe Accident Simulations *†

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SUMMARY

Work began at Oak Ridge National Laboratory (ORNL) during 1991 to incorporate a boiling water reactor (BWR) control blade/channel box interaction and melt relocation model within the SCDAP/RELAP5 code. This model was developed originally at ORNL as a result of posttest analyses of several BWR severe accident experiments and accounts for material interactions that occur between the B_4C and stainless steel of the control blade and the Zircaloy of the channel box. SCDAP/RELAP5 has been developed primarily at the Idaho National Engineering Laboratory to provide best-estimate predictive capability for use in light water reactor severe accident applications.

The basic features of the BWR control blade/channel box model within SCDAP/RELAP5 were described in a previous paper at the 1993 Water Reactor Safety Information Meeting. Subsequently, several new capabilities and improvements have been added to the model and are described in this paper along with an example simulation of a short-term station blackout (STSB) accident sequence for the Browns Ferry Nuclear Plant design.

The control blade/channel box model improvements include: (1) implementation of timestep repetition, (2) modifications to improve execution times for BWR simulations, (3) replacement of hard-wired material property correlations with equivalent correlations from the MATPRO library of material properties, (4) modifications to allow molten control blade/channel box material to spread radially into the fuel bundle, and (5) modifications to allow molten control blade/channel box material to relocate downward into the lower plenum. Items 1 and 2 have minimal impact on the predicted results for BWR simulations, but they reduce the frequency of water property failures and improve the usability of the code. Item 3 makes the code easier to maintain by eliminating redundancy. Items 4 and 5 are new capabilities that represent the relocation of molten material during core degradation in a more realistic manner.

During a BWR severe accident, core degradation begins with control blade liquefaction at a temperature of ~1505 K. A molten B_4C/stainless steel mixture relocates downward and solidifies at a lower elevation to form a blockage in the region between the control blade and the channel box. After failure of the adjacent channel box wall because of a Zircaloy

* Research sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission under Interagency Agreement No. 1886-8620-2W with the U.S. Department of Energy under contract DE-AC05-84OR21400 with Lockheed Martin Energy Systems, Inc.

interaction with stainless steel, additional molten control blade material (1) is diverted by the blockage horizontally through the opening in the channel box wall, (2) moves downward, and (3) solidifies on the fuel-bundle side of the wall. The new radial spreading logic determines when a second blockage forms in the region between the channel box and the first row of fuel rods. This second blockage then diverts additional molten control blade and/or channel box material horizontally into the fuel bundle where it relocates down the outer surface of the fuel cladding.

Downward-moving molten material that does not solidify before reaching the lowest elevation of the control blade or channel box surfaces relocates below the bottom of the defined core. If a COUPLE finite element mesh is defined by the user to represent lower plenum debris, then the new lower plenum relocation logic transfers this molten control blade/channel box material directly into the lower plenum debris bed. (This relocating material does not interact with the structures in the core plate region because SCDAP/RELAP5 does not currently include a model to represent the behavior of the core plate during a severe accident.)

The control blade/channel box model improvements are included in the current production version of SCDAP/RELAP5 and have been tested using a Browns Ferry input deck that represents a STSB accident sequence. This input deck simulates the entire reactor coolant system of a General Electric Company BWR/4 from the feedwater inlet to the turbine inlet. The reactor coolant system is divided into 197 hydrodynamic volumes that represent the pressure vessel, two recirculation loops, the feedwater piping, the control rod drive cooling water, and the steam piping. The volumes and structures in the active core are divided into four radial rings and thirteen axial nodes. The lower head and any lower plenum debris are represented by a COUPLE finite element mesh with a total of 54 elements.

The initial condition for the Browns Ferry simulation is steady-state operation of the reactor at full power. The STSB accident sequence is initiated by a loss of off-site AC power combined with failure of the emergency diesel generators. After reactor scram and closure of the main steam isolation valves, the water in the vessel boils slowly and is released from the vessel through cyclic actuations of the safety/relief valves. When the vessel water level falls to one-third of the active core height, the Automatic Depressurization System (ADS) is manually initiated and the vessel water inventory flashes to a level well below the core plate. Subsequent core degradation occurs under dry conditions with no steam generation in the core region to feed oxidation reactions. The calculation is executed to the time of creep rupture failure of the lower head.

EQUIPMENT OPERABILITY AND AGING INTRODUCTION

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For the reasons of safety and economics Equipment Operability and Aging-Related Issues continue to draw the attention of plant operators and regulators. Ensuring the operability of safety-related equipment to perform its intended function is important for maintaining safety in operating nuclear power plants of all ages. This requires careful evaluation of the effects of aging, and programs to efficiently manage that aging.

During the past ten years, the U.S. NRC Office of Nuclear Regulatory Research has pursued research programs to address technical safety issues related to (i) primary system integrity, (ii) structural and seismic engineering, (iii) equipment operability involving motor-operated valves, check valves, and pumps, and (iv) aging and condition monitoring of risk significant electrical and mechanical components including cables.

Session 2 includes technical papers on the results of the NRC's containment and seismic related research work, while the topics covered in Session 6 include the results of NRC's research effort on RPV and steam generator pressure boundary, environmentally assisted cracking of materials, and annealing of reactor pressure vessels.

In Session 8 the results of some of our research effort pertaining to the operability of motor-operated valves and check valves and aging evaluation and condition monitoring of electrical cables will be discussed. Mr. George Hubbard will provide a status of the NRR's EQ task action plan.

Other participants in the session include representatives from the U.S. Department of Energy and the Duke Power Company and the discussions of their programs related to aging management and condition monitoring of electrical cables. This session will provide an overview of the types of aging-related issues being evaluated and the work underway to resolve those issues.

Condition Monitoring and Testing for Operability of Check Valves and Pumps*

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ABSTRACT

A detailed analysis of historical failure data available through the Institute of Nuclear Power Operations' Nuclear Plant Reliability Data System (NPRDS) has been conducted for both check valves and pumps. This analysis, which originated as part of the Nuclear Regulatory Commission's (NRC's) effort to evaluate the effects of age and wear on nuclear systems components, involved the manual review and characterization of hundreds of component failure records according to parameters inherent in the NPRDS database and supplemented by those defined by the analyst for each component type. For example, failure information relative to component size, age, manufacturer, system of service and NSSS vendor was readily available from the NPRDS database and could be compared relatively easily. Determination of parameters such as failure mode, failure cause, extent of degradation, failure area, and detection method, however, had to be determined based on manual review of individual failure narratives.

This paper discusses the results of the analyses of historical check valve failure data from 1984 through 1992 and pump failure data from 1990 through 1993. A comparison of the findings of the analyses is made, and emphasis is placed on evaluation of the effectiveness of certain failure detection methods for each component type. Generally speaking, while it was observed that check valve degradation or failure was likely to be detected by code or regulatory required testing, it was discovered that pump degradation or failure was most likely to be discovered by voluntarily implemented plant programs.

Failure rates were found to be strongly influenced by the valve or pump application. The type of plant (BWR vs. PWR) was also found to significantly influence both the overall failure rate and the method of detection.

* Research sponsored by the Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission under Interagency Agreement DOE DE-AC05-840R21400 with the U. S. Department of Energy under contract No. 1886-8082-8B with Lockheed Martin Energy Systems, Inc.

CORROSION EFFECTS ON FRICTION FACTORS

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The ability of valves to perform their safety functions is an ongoing issue for nuclear power plants. The U.S. Nuclear Regulatory Commission (NRC) and the Electric Power Research Institute have conducted considerable testing to provide bases for predicting the forces to close and open these safety-related valves when subjected to their design bases differential pressure. These tests used new or refurbished valves. A question to be answered is "will aging as a result of some period of operation cause these forces to increase?" Also, in-plant tests are sometimes used to establish the required operating forces and the torque switch settings for motor operated valves. The question to be answered for these tests is "should a margin be added to the torque switch setting to account for a possible increase in operating forces due to aging?" The objective of this project is to begin answering these questions.

Many of the safety-related valves are gate valves. As a start, we look at the forces that need to be overcome to operate these valves. The forces are:

$$F_{\text{stem}} = F_{\text{packing}} \pm F_{\text{stem rejection}} \pm F_{\text{net vertical pressure on disc}} + F_{\text{friction}}$$

For most valves, friction is the largest of these forces. The friction occurs between the sliding surfaces, which are the disc and the guides when the valves are mostly open and the disc and body seats as the valve nears closure. Changes in the characteristics of these surfaces with age could increase the forces required to operate the valves. Corrosion, which occurs to some degree even in the high-purity water of nuclear power plants, is a likely mechanism that could change the characteristics of these surfaces. Therefore, our initial investigations are focusing on the effect of corrosion on friction factors.

The largest force usually occurs when the disc is fully in the flow stream just prior to seating. In this case, the friction is between the disc and body seats. Both seats are usually Stellite 6 material. Therefore, much of our initial work has been to investigate the effect of corrosion on the friction factors of Stellite 6. However, if the friction factor between the disc and the guides were to increase sufficiently because of corrosion, the largest force could occur when the disc is still riding on the guides. Also, for some valves the largest force already occurs when the disc is still riding on the guides. These are valves with relatively large clearances that allow the disc to tip. The tipping, if excessive, can lead to interference as the disc enters the body seat, which may cause considerable metal deformation and result in high closing forces. The interference is not related to aging and is not included in this study. Tipping can also cause a relatively large net vertical force from changing pressure distribution around the disc. This increased vertical force can occur when then the disc is still riding on the guides. In this case, tipping reduces the contact area, thereby raising the contact stress which, if sufficient, can lead to damage of the sliding surfaces. The damage could accentuate the increase in stem force required to operate the valve. The guides and discs for many of the valves are carbon steel; therefore, our project investigates the effect of corrosion on the friction factors and the contact stress that would damage carbon steel. We have only preliminary results for carbon steel and most of the work remains to be done.

To begin the project, we searched nuclear plant operational databases, reviewed literature, and contacted experts for information relevant to the effect of corrosion on friction. The information we obtained was insufficient to resolve the issue; therefore, we concluded that testing would be necessary.

Our first approach was to conduct relatively simple tests with available equipment to determine if corrosion would affect friction. We corroded half of the Stellite 6 and carbon steel test specimens by installing them in an autoclave with simulated boiling water reactor (BWR) temperature, pressure, and chemistry. We applied a small electrical potential to accelerate the corrosion. We conducted friction tests on the corroded as well as the uncorroded specimens at ambient conditions dry and with a spray of deionized water. Results of the tests showed an increase in friction factors for the corroded specimens. The increases for Stellite 6 were from 0.18 to 0.33 with high contact stress dry, and 0.20 to 0.47 with low contact stress wet. The increase for carbon steel wet was from 0.23 to 0.48. Also, the tests showed that the corroded carbon steel surfaces were more susceptible to galling.

These preliminary tests showed that corrosion is likely to affect the friction factors. However, the friction tests did not duplicate in-plant conditions and the friction test apparatus may have accentuated galling. As a result, questions were raised about the effect of the magnitude of the contact stress, temperature, relative velocities of the sliding surfaces, and the apparatus geometry. To resolve the questions and duplicate in-plant conditions would require that the specimens be tested while submerged in high temperature pure water, have typical contact stress, and typical relative velocity between the sliding surfaces. We concluded that to answer the question "would corrosion have a similar effect on the friction factors of in-plant valves" would require additional tests.

Second-phase tests are currently underway that will better duplicate the in-plant conditions. The corrosion films are being established as before. However, the friction tests are being conducted in an autoclave simulating BWR primary coolant temperature, pressure, and water chemistry. Also, the friction test apparatus simulates the closing speed of actual valves (10–16 in./min), simulates typical contact stress for the sliding surfaces, and has a symmetrical design that avoids accentuating galling.

Some second-phase friction testing using the more representative apparatus has been completed for Stellite 6. Tests for carbon steel will follow. Preliminary results for Stellite 6 tests confirm the findings from the earlier tests. Friction factors increased from approximately 0.2 for uncorroded specimens to approximately 0.4 for corroded specimens. Tests with contact stress of 10 ksi and 40 ksi had similar results, except for the 40 ksi test the friction factor decreased after a few strokes. This decrease may indicate that the higher contact stress is effective in removing the corrosion film, which may accentuate the expected decrease in friction factors with increased contact stress. Tests were conducted with oxide film thickness of 0.5, 2–3, and approximately 4 micrometers. The results were similar for the three thicknesses.

We compared the results of our tests with results from full-scale valve tests. INEL has conducted several full-scale tests for the NRC. These tests show that for subcooling less than 70°F, the friction factor is approximately 0.4. Other industry tests and in-plant tests confirm that friction factors of 0.4 are reasonable. The results are somewhat in conflict. Our specimen tests show a marked increase for corroded surfaces. On the other hand, the higher value for the corroded surface is in line with the values for full-scale tests with new or refurbished valves. Additional tests may resolve the apparent conflict. However, an explanation may be that the corrosion film develops rapidly, so that even with the limited exposure of new and refurbished valves, the film has developed sufficiently so that even these tests are measuring friction factors of corroded surfaces.

ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT
Task Action Plan

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A review of environmental qualification (EQ) requirements for license renewal and failures of qualified cables during research tests led to the development of the EQ Task Action Plan (TAP), which was issued in July 1993. The EQ TAP was developed to address: (1) staff concerns regarding the differences in EQ requirements for older and newer plants; (2) concerns raised by some research tests which indicate that qualification of some electric cables may have been non-conservative; and (3) concerns that programmatic problems identified in the staff Fire Protection Reassessment Report might also exist in the NRC EQ Program. The EQ TAP is intended to address these concerns and includes meetings with industry, a program review of EQ, data collection and analysis, a risk assessment, and research on aging and condition monitoring.

Since the development of the EQ TAP, the staff has met with the Nuclear Energy Institute, the Nuclear Utility Group on Equipment Qualification, EPRI, and licensees to discuss activities under the EQ TAP.

The staff completed the program review of EQ in June, 1995. The final report that summarizes the results of the program review is under management review and will be released to the public when complete. The program review involved a look back at the basis for the different requirements and a review of the adequacy of the requirements and their implementation. The staff conducted surveys, met with industry representatives, conducted an extensive document research effort, and documented its findings. The staff issued internal reports on the following topics: License renewal background information, the Fire Protection Reassessment Report, the survey of NRC and industry EQ experts, existing program requirements, NRC EQ inspection practices, and licensee implementation practices.

Data collection and analysis activities are continuing. In 1994, the staff reviewed operating experience to determine whether there are significant problems with EQ in the industry and to focus research on those problems. The staff visited sites to gather information on licensee EQ activities. The staff issued reports on equipment replacement and operating experience. The staff is reviewing past and ongoing EQ-related work, including literature from qualification tests and research. In addition, to gain an international perspective on EQ practices and requirements, the staff met with EQ experts in Germany and France in December, 1993, and in Sweden and the United Kingdom in 1994. The staff also participated in a technical committee meeting at the International Atomic Energy Agency in the fall of 1994. The staff issued a report on the impact of the new source term (NUREG-1465) on environmentally qualified equipment at operating power plants in early 1995.

As part of its activities to support the EQ TAP, the Office of Nuclear Regulatory Research (RES) held a public workshop in November 1993 and used the information received at the workshop to develop a Research Program Plan.

RES issued its revised EQ program plan in March, 1995, which provides for a cable condition monitoring program, a cable testing program, and an EQ database in support of the EQ TAP. Brookhaven National Laboratory (BNL) is assisting RES in implementing major elements of the program plan. BNL is developing cable testing and cable acquisition programs and has identified some limited sources of naturally aged cable for the program. The cable test plan includes testing of new, naturally aged, and artificially aged cables and evaluation of condition monitoring techniques that could give insights into methods for determining how cable is actually aging and performing in plants. The plan includes LOCA testing of some cables under design-basis event conditions.

As activities of the program review and data collection proceed or are completed, the staff will make changes to the research program as necessary. Following completion of the program review and data collection effort, staff activities will focus on research in the areas of accelerated aging, condition monitoring techniques, and accident testing. Research activities will extend over the next few years.

Results of a Literature Review on the Environmental Qualification of Low-voltage Electric Cables

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ABSTRACT

In the design of nuclear power plants in the U.S., safety-related electric equipment must be qualified to ensure it will withstand the effects of a design basis accident and still be able to perform its function. The requirement for environmental qualification (EQ) originates from the General Design Criteria in the Code of Federal Regulations. The method of performing the qualification of this equipment has evolved over the years, starting with the DOR Guidelines and NUREG-0588 Category I and Category II requirements, up to the current EQ Rule, 10 CFR 50.49. While the EQ methods described in these three documents have the same objective, there are some differences for which a clear technical basis has not been established. One difference is the preaging requirements for the components prior to LOCA testing.

In addition to the differences in EQ methods, other qualification issues have been raised by the U.S. NRC which need to be addressed. These issues are related to the sources of conservatism and uncertainty in the IEEE Standard 323-1974 qualification process, which is the current standard used. To address these issues, the NRC Office of Nuclear Reactor Regulation (NRR) implemented a Task Action Plan, and the Office of Nuclear Reactor Research (RES) initiated a research program. The current focus of this program is on the qualification of low-voltage instrumentation and control cable. This component was selected since it is not typically replaced on a routine basis, and degradation of this component can result in false or misleading signals being sent to plant operators.

As the first step in developing a research plan, NRC sponsored a public workshop on EQ in November 1993. Panels of experts from across the country attended and provided input, which was used in the development of a research program plan. One of the main conclusions drawn from the workshop is that a great deal of work has already been performed in the area of EQ, and it should be reviewed before any new work is performed.

Brookhaven National Laboratory (BNL) was selected as the lead lab assisting NRC/RES in the EQ Research Program. In response to the public workshop, NRC requested that BNL perform a literature review to identify past and ongoing work in the area of environmental qualification that may be of use in resolving the issues raised. The purpose of this review was to determine which issues could be resolved by past or ongoing work, and which required more research. The results of the review will be used as input to optimize the scope of future EQ research.

In conducting the literature review, BNL examined over 400 documents from a variety of sources. This includes technical reports and papers related to cable qualification and polymer research performed by Sandia National Laboratory, EPRI, and others. Also, actual cable manufacturers qualification test reports were obtained and reviewed. EQ experts were consulted and their input was obtained. In addition, the NUS EQ database and the INEL EQ database were reviewed to identify the information in them and evaluate their usefulness. Research from other countries was also reviewed, including France, Japan, Canada, Germany, and the United Kingdom.

To properly focus the review, the issues to be addressed in this research program were identified and documented. These can be categorized into seven broad issues. For each of these issues, a number of specific sub-issues were then identified, and individual dossiers were prepared for each one. The dossier provides background information on the sub-issue, a listing of the specific questions to be answered for the sub-issue, and the results of the literature review related to that sub-issue. This paper discusses the issues identified to be resolved, along with the results of the literature review performed by BNL to date. The results of the literature review, along with the dossiers will be published as a NUREG/CR report.

CABLE CONDITION MONITORING PROGRAM AT OCONEE NUCLEAR STATION

By: T.J. Al-Hussaini, Senior Engineer
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The majority of cables used at Duke Power Oconee Nuclear Station were purchased between 1968 and 1971. During this period, the industry qualification standards for equipment used in Nuclear Plant were in the process of being developed. Even though qualification testing was performed on Oconee cable at that time based on available data, no natural aged cable data in nuclear plant environment was available. In order to study the effect of the containment environment on the safety-related cable, a cable condition monitoring program was instituted.

Six cables, representing the type of cable used inside Oconee containment, were selected to be monitored on a periodic basis. These cables were installed inside Oconee Unit #2 containment. At five year intervals over the life of the plant, a five foot segment from each cable sample is removed and tested. The purpose of the test is to measure and document the mechanical and the electrical properties of the cable insulation. The data obtained is analyzed and compared to that obtained from the previous tests for trending. This data will be used in the future in conjunction with the original cable qualification data to support the station license renewal program. This paper will summarize the result of twenty years of testing.

DOE-SPONSORED CABLE AGING RESEARCH AT SANDIA NATIONAL LABORATORIES

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ABSTRACT

Cables have been identified as critical components requiring detailed technical evaluation for extending the lifetime of LWRs beyond 40 years. This paper will highlight some of the cable aging studies currently underway at Sandia.

One important issue relevant to the aging of cables is the reliability of the Arrhenius thermal aging methodology. In agreement with numerous thermal aging literature results, we find that ultimate tensile elongation results are often amenable with the Arrhenius approach. On the other hand, ultimate tensile strength results available from the same experimental procedure are decidedly non-Arrhenius, calling into question the reliability of this methodology. Data from our unique modulus profiling apparatus resolved this inconsistency by showing that complex, diffusion-limited oxidation effects (involving surface hardening) are typically present under accelerated aging conditions. Since excellent correlations usually exist between the surface modulus and the tensile elongation, Arrhenius is valid for elongation whenever surface-initiated cracks immediately propagate through the sample. Tensile strength, on the other hand, is non-Arrhenius because it is dependent on the total cross section. These results both increase the confidence in Arrhenius extrapolations and offer compelling evidence that the indentor approach (sensitive to "surface" hardness) is a promising NDE technique for thermal-dominated regions of nuclear power plants.

Another significant aging issue involves the importance of and procedures for handling synergistic and dose-rate effects. We have derived and successfully tested a combined environments aging method which allows us to predict the lifetimes of many types of cable materials. We have obtained naturally aged cables from Yankee Rowe, which we will use to further test both our combined environments model and the Arrhenius approach.

Finally, we will discuss some interesting combined environment results on several polyolefin cable insulation materials, in which mechanical degradation at constant dose rate occurs more rapidly as the aging temperature is lowered. This fascinating phenomenon contradicts the fundamental aging principal that an increase in environmental stress will lead to an increase in degradation rate.

AGING MANAGEMENT GUIDELINE FOR ELECTRICAL CABLE AND TERMINATIONS

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ABSTRACT

The Aging Management Guideline (AMG) for Electrical Cable and Terminations, sponsored by the DOE Plant Lifetime Improvement (PLIM) Program, provides an analysis of the potential age-related degradation mechanisms for low-voltage (600 V and below) and medium-voltage (2-kV through 15-kV) extruded cables and associated terminations. The AMG is intended to provide plant operators with a basis for verifying that effective means for managing the effects of aging on cable systems are in place, as well as acceptable guidelines for developing aging management programs for current and license renewal periods. General classifications of equipment identified in 10CFR54 and covered in the study included 1) safety-related cables, 2) non-safety-related whose failure could prevent satisfactory accomplishment of a safety-related function, and 3) those relied on in safety analyses or plant evaluations for compliance with the NRC regulations (such as environmental qualification or fire protection). Additionally, all cables important to plant operation or continuity of power production, even if not included in one of the above categories, were considered.

As part of the study's methodology, industry data on aging and failure of these components (including LER and NPRDS data), and the maintenance activities performed on cable systems were evaluated. The principal aging mechanisms, and effects resulting from environmental and operating stresses on these systems were identified and correlated with actual plant experience. Maintenance procedures and condition monitoring/testing methodologies used by plant operators were evaluated to determine if the effects of these aging mechanisms were being detected and managed. Other currently available and developmental testing and condition monitoring techniques were also identified. Where an aging mechanism was not fully managed or not considered, additional plant-specific activities to manage the aging mechanism were recommended.

Historical failure and host utility data examined during the study indicated that bulk runs of low- and medium-voltage cable are generally very reliable and fail at a very low rate. More significant degradation and failure rates are indicated for low-voltage cable segments near end devices or in proximity to process piping or components. A relatively high incidence of failures associated with low-voltage panel wiring was identified, stemming largely from maintenance or other non-aging related factors. Medium-voltage cable appears to be more susceptible to the effects of moisture than low-voltage cable due largely to increased voltage stress. Furthermore, installation damage may have a significant impact on cable longevity, especially for medium-voltage systems. Cable terminations also indicated a high degree of reliability. Oxidation of connectors and contact surfaces in impedance-sensitive instrumentation circuits appeared to be one of the most frequent types of termination failure.

Finally, a methodology for selecting cables and terminations for potential inclusion in a plant's aging management program was developed. General considerations relevant to such a program include the initial limitation of the population of equipment included in the aging management program, and focusing on specific cable types used in certain applications (e.g., low-voltage cables connected to heavy continuous loads or routed through "hotspot" areas, or medium-voltage cables subject to wetting or submersion). A program to determine baseline aging condition using a non-destructive test or evaluation method, and review of existing cable installation practices and procedures are also recommended.

THERMAL HYDRAULIC RESEARCH INTRODUCTION

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Calculation of coolant temperatures, pressures, and flow rates in reactor and plant systems is the backbone of a safety analysis for any nuclear plant design. Traditionally, NRC has had a strong independent program of thermal-hydraulic data confirmation and analytical studies, usually involving the RELAP and TRAC computer codes. This program has not only provided sound technical skills at NRC for reviewing safety analyses, but it has also uncovered a number of safety issues such as the sizing of pressure relief valves, extensive thermal stratification, CMTs refill, and long term oscillations.

For the past several years, NRC's research in this area has been concentrated on the new passive LWR designs under review for design certification. These new designs, the Westinghouse AP-600 and the General Electric SBWR, use gravity fed systems with low pressures and flow rates that present difficult challenges for thermal-hydraulic computer codes. To address these designs, (a) new plant models have been developed for code input, (b) modeling improvements in the codes have been made to handle the new flow regimes, (c) a few independent tests have been conducted to supplement the data base for code validation and to check the applicant's experimental results, and (d) numerous calculations have been performed to check the codes against data and to study the inherent characteristics of the plant designs.

A lot of the NRC's work involves cooperation with other researchers. In some cases, as with the GIRAFFE and PANDA test facilities, direct access to data has been arranged to enable good quality analysis of results. In others, as with the ROSA and OSU facilities, NRC is running its own tests in facilities built by others. And in one case, a new scaled test loop is being constructed for the NRC at Purdue University. These and several related efforts are described in the papers presented in this session.

It is in the interest of the public and the utilities, who are the potential operators of the new reactor designs, for the NRC to press hard on the designers of these plants to assure that the designs are robust and incorporate no hidden safety concerns. However, there are also more immediate benefits from this work. As we extend the code capabilities to handle low flow and low pressure conditions, the codes can then be used to assess situations where the older codes had excessive conservatism, e.g., at low power and shutdown conditions. In addition, as these codes are assessed against a much larger database, conservatism in the codes are reduced, the codes can be used to optimize the performance of the plant, e.g., power rate increase, PTS issue.

Finally, the methodology used for the evaluation of the advanced ALWR experimental data provides a mechanistic way that can be easily used to resolve operating reactor issues. The method focuses attention on the major issue while setting temporarily aside these issues that are not absolutely necessary to resolve the main issue.

GIRAFFE TEST RESULTS SUMMARY

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As a long-term containment cooling system with passive safety feature for advanced light water reactors, the shell-and-tube condenser type passive containment cooling system (PCCS) is adopted in the SBWR design, which removes the decay heat following a LOCA without any electrical power supply. During a LOCA, a steam-noncondensable mixture flows into the PCC vertical heat exchanger tubes, condensate drains to the reactor pressure vessel by the gravitational force, and the noncondensable is vented to the suppression chamber via noncondensable vent line.

Toshiba has been taking the lead in the research and development of the PCCS and constructed the integral system test facility, "GIRAFFE" (Gravity-Driven Integral Full-Height Test for Passive Heat Removal) to demonstrate the PCCS feasibility. GIRAFFE models the SBWR in full-height to correctly present the gravity head driving forces with a 1/400 volumetric scale. In order to investigate the basic heat removal performance of the PCC heat exchanger as well as various thermal-hydraulic transient phenomena in the primary containment vessel following an accident, extensive tests have been conducted including steady state tests and system response integral tests.

From the thermal hydraulic viewpoint, a key phenomenon to affect the PCCS heat removal performance is steam condensation in the presence of a noncondensable gas retained in the PCC heat exchanger tube. As the first step of the GIRAFFE test program, steady state fundamental heat transfer tests were conducted to investigate the amount of heat transfer degradation caused by noncondensable gases in the PCC tubes. The test results showed that the steam condensation degradation in the PCC heat exchanger tubes is slightly milder than the published data and analysis for forced convective condensation*. This was due to the renewal of the noncondensable concentration boundary layer by the liquid film wavy behaviors.

The amount of noncondensable absorbed into the PCC heat exchanger is attributed to the noncondensable distribution in the containment vessel, which is also influenced by the break location. After the favorable noncondensable gas venting characteristics via noncondensable gas vent line were confirmed, system response tests were conducted to investigate the PCCS heat removal performance during LOCAs with changing the pipe break location.

From these system response test results with wide break spectra, it has been confirmed that the PCCS has the capability to remove decay heat sufficient to suppress the containment pressure below the design limit, although the PCCS heat removal performance somewhat changes depending on the break locations. In all cases, the drywell pressure decreased gradually after taking the peak value.

*E.M. Sparrow, et al., "Forced Convection Condensation in the Presence of Noncondensables and Interfacial Resistance," Int. J. Heat Mass Transfer, Vol. 10, p. 1829, 1967.

In addition to the basic system response tests mentioned above, some potential issues which might degrade the PCCS heat removal rate have been investigated. If the vacuum breaker leakage occurs, some amount of steam/noncondensable mixture discharges directly from drywell to wetwell airspace and affects the noncondensable distribution in the containment. The emergency core coolant injection also influences the steam-gas mixture flow in the containment. The thermal stratification layer can be produced at the suppression pool water surface by the excessive steam flow through the PCCS and affects the containment pressure response. Using the GIRAFFE, all effects of these phenomena were examined and it was confirmed that the primary containment vessel pressure can be maintained below the design limit even with these effects. The effect of the PCC heat exchanger tube length on its heat removal performance was also examined.

Due to such favorable heat removal capability, it is expected that the PCCS can maintain the integrity of the containment vessel even under a severe accident condition. As the first step to investigate the PCCS heat removal performance under a severe accident condition, the light noncondensable gas (i.e. hydrogen) effect has been clarified using helium gas. The system response test result showed that the helium gas effect on the PCCS performance is comparable to the nitrogen gas effect and the primary containment vessel peak pressure was very similar to that measured in the nitrogen test.

From these GIRAFFE test program results under wide test conditions, the feasibility of the PCCS has been verified for the passive decay heat removal system for next generation light water reactors.

AP600 SBLOCA PHENOMENA INTERACTIONS AND TRANSIENT BEHAVIOR

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Summary

The Westinghouse Corporation is seeking US Nuclear Regulatory Commission (USNRC) design certification for a new generation of pressurized water reactor (PWR) called Advanced Passive 600 (AP600), which relies on gravity-driven safety systems rather than pumped safety systems for mitigation of accidents. The new AP600 safety systems introduce sufficient phenomena and phenomena interactions not present in the current generation PWRs that the USNRC had to undertake a research program in support of independent audit analysis of postulated accidents. This program consists essentially of three major elements:

- improvements to the RELAP5 code,
- development of experimental data base, and
- code assessment.

Because of a relatively short certification period and evolving design, this program is unprecedented in its common focus of experimental and analytical efforts on the general accident response of a single plant design. The efforts of this program are conducted in many cases in parallel instead of in the traditional sequential approach. However, the program is conducted in a consistent and systematic way using the Code Scaling Applicability and Uncertainty (CSAU) principles as a basic framework.

Early in the program, it was recognized (Ref. 1) that the existing data base associated with current PWR designs was insufficient for code assessment and to support understanding of the AP600 behavior during accidents. The USNRC has identified that integral system behavior and particularly interactions between the new passive safety systems and the reactor coolant system are unique and of prime importance from the safety point of view. This lead to the USNRC's ROSA-AP600 confirmatory experimental program and testing at the Oregon State University (OSU) APEX integral facility. Similarly, Westinghouse has addressed the new features of the design creating the integral test program at OSU and adapted the SPES facility in Italy for high pressure integral system testing. Also, Westinghouse has established and conducted series of separate effects tests, which included systems such as Passive Residual Heat Removal (PRHR), Core Make-up Tank (CMT), and Automatic Depressurization System (ADS). Results of the Westinghouse tests are included in the USNRC program to provide an expanded data base for evaluation of the analytical methods. This paper addresses only a part of the USNRC program and focuses on results of small break LOCA analyses performed to date at the Idaho National Engineering Laboratory (INEL) in support of validation of the RELAP5 code for AP600 analyses.

Most of small break LOCA transients will activate passive safety systems and will lead to ADS actuation and passive long-term cooling. During these transients, the forces that drive the safety injection are, in contrast to the current plants, very small, and performance of the safety systems is tightly coupled with processes in the reactor coolant system. The development and assessment of current thermal-hydraulic system codes were not particularly focused on conditions such as small driving forces, low flows and pressures or phenomena such as natural circulation or condensation, since these issues were of secondary

importance for the safety of current plants. In support of RELAP5 development and assessment for the AP600 application, the INEL has undertaken analyses of four SBLOCA transients, performing analyses of the AP600 system and analyses of experiments conducted on three test facilities of different scales: ROSA-AP600, SPES and OSU-APEX. The four transients were: 1-inch cold leg break, 2-inch pressure balance line break, direct vessel injection line break, and inadvertent ADS opening. The analyses include in-depth review of the experimental data, to provide solid understanding of the observed phenomena, and simulations of the tests using RELAP5 to assess the code capabilities to represent the phenomena expected in AP600. In this paper, we discuss analysis results of the first transient, the 1-inch cold leg break. The experimental data and code analysis show that the reactor vessel coolant inventory arrives at a minimum after the stage four of the ADS is actuated and the inventory recovers via gravity-draining from the In-Containment Refueling Water Storage Tank (IRWST). The coolant depletion from the system does not result in core heatup. The RELAP5 analyses indicate that the code is doing a credible job in simulating the transient. It is recognized that it does not fully represent local level detail, but the dominant processes and system trends are well represented. The lessons learned from the 1-inch break integrated analyses were applied to the code and model improvements that should enhance the code capabilities.

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Results from the NRC Testing Program at Oregon State University

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The Department of Energy (DOE) and Westinghouse Electric Corporation cooperated on the construction of an integral test facility that models the AP600 design. The facility was build at Oregon State University and given the name APEX. Title to the facility is held by DOE. Westinghouse performed a series of experiments in APEX as part of its experimental program in support of AP600 design certification. The NRC subsequently contracted with OSU to run a series of experiments following the completion by Westinghouse of its experimental program.

The facility is designed around a basic concept: study the passive safety features of the AP600 under natural circulation and depressurization conditions. The facility scaling objectives were to: identify important processes; establish the priorities for preserving important processes; obtain the similarity groups that should be preserved between the facility and the full-scale prototype for these processes; establish the priorities for preserving the similarity groups to assure that important processes have been identified and addressed; provide specifications for test facility designs; and quantify the biases due to distortions. The facility design and testing program takes account of the following:

o A spectrum of transients are possible, i.e.. small break LOCAs in various locations (e.g. P cold leg, C cold leg, hot leg, PBL line DVI line) orientations (top, side, bottom), SGTR(s), large breaks, secondary side breaks, transients, etc.;

o The AP600 system has multiple loops capable of competing flow and interactions i.e. PRHR loop, two CMT loops, two primary loops, ADS 1-3 loop, two ADS4 loops;

o The AP600 system has multiple components acting as mass and energy sources and sinks. Mass sources include CMTs, accumulators, IRWST, sump, non-safety systems while mass sinks include the break, ADS 1-3, ADS4a and ADS4b. Energy sources: core, structures, steam generators. Energy sinks consist of the break, ADS 1-3, ADS4s, steam generators, CMTs, and the PRHR.

The facility is a scaled representation of the AP600 at 1:4 height and 1:192 volume scale. Most facility dimensions follow from three basic decisions: (1) 1:4 height; (2) 1:7 diameter scale; and (3) pressure scale (400 psi). Considerable effort was devoted to geometric similitude. Following these basic decisions, the next set of design decisions follow from Friction Number scaling.

The following summarizes the 13 NRC tests conducted during the first year of the contract. The tests fall into five categories:

1. Design basis accident scenarios intended to provide counterpart to ROSA for very small breaks and transients not covered by the Westinghouse test program (3 tests).

2. Beyond design basis accident scenarios which are also counterpart to ROSA tests (5 tests).

3. Tests to examine scaling issues associated with depressurization from the Westinghouse program (2 tests).

4. Tests intended to help understand the data from the Westinghouse program (2 tests).

5. PIRT driven tests (1 test).

Test 1 is a one inch cold leg break with failures of all ADS stages 1, 2, and 3 valves to open. Both of the fourth stage ADS trains are fully functional. This is beyond design basis and is counterpart to ROSA test AP-CL-05. The objective was to determine the effect of reduced depressurization capability on the system to depressurize to IRWST injection. The test stayed at high pressure for an extended period of time. This extended period at high pressure was important since all the Westinghouse tests were larger breaks, i.e. they break alone was sufficient to remove decay heat. Therefore, limited information was available from OSU on extended PRHR natural circulation, CMT recirculation and heatup, and thermal stratification in the primary system. This test extended the range of experimental results and provided data counterpart to ROSA.

Test 2 is a station blackout and is counterpart to ROSA test AP-SB-01. The purpose is to evaluate the long term PRHR performance including heatup of the IRWST to saturation. The test evaluates ADS performance when the discharge is to a saturated pool. The IRWST injection begins with the saturated so the period of subcooling of the primary system is not present. The test has extended duration of high pressure.

Test 3 is a two inch cold leg break repeat of the Westinghouse test SB1. The purpose was to remove a check valve in the sump exit, which is an artifact of the facility, to determine its effect on oscillations during the long term cooling phase. The test met its objectives and we were able to conclusively prove that the oscillation prior to sump injection was a facility artifact.

Test 4 is also a two inch cold leg break repeat of SB1. The purpose was to bypass the break separator, which is an artifact of the facility, to determine its effect on oscillations during the long term cooling phase. Suspicion was directed at the break separator from an examination of the Westinghouse data, however, a mechanistic explanation to connect the two was never found. The test met its objective, showing that the oscillation upon return to saturation was not a result of the break separator. In the meantime a mechanism for the oscillation was developed and explored in NRC12.

Test 5 is a one-half inch cold leg break and is counterpart to ROSA test AP-CL-04. The purpose is to explore CMT and PRHR natural circulation during extended operation at high pressure characteristic of very small breaks, similar to NRC1 and NRC2. A break of this size is smaller than was considered by the Westinghouse testing.

Test 6 is a one inch cold leg break at full pressure and is counterpart to ROSA test AP-CL-03. The purpose is to operate the OSU facility at full pressure using initial conditions determined by AP-CL-03. In this test, ADS1 actuated at 3400 seconds at a pressure of 420 psig. Thus it is possible to operate OSU at full pressure rather than as a scaled pressure facility and initialize the conditions to be representative of ROSA at 3400s. This test avoided pressure scaling and provides a different representation of the ADS phase of the transient than the other tests which were based entirely on scaled pressure.

Test 7 was a one inch cold leg break without nitrogen injection. This is counterpart to Westinghouse SB23. The purpose was to determine the effect of noncondensables on late depressurization and long term cooling, particularly with regard to PRHR operation and CMT refilling, as well as the possibility to refill other volumes in the absence of noncondensables. As such, it is directed at understanding of phenomenology. It provides information of nitrogen effects, which are identified in the PIRT.

Test 8 is a one inch cold leg break with failure of CMT drain valves to open. This is beyond design basis and is counterpart to a planned ROSA test. ADS is manually actuated after a time delay. The purpose is to determine the inventory history and ADS depressurization with reduced coolant injection.

Test 9 is a loss of feedwater with failure of PRHR and is beyond design basis and counterpart to a planned ROSA test. The purpose is to explore system performance absent a principal safety system and to determine CMT performance during extended operation at pressures up to the safety valve setpoint.

Test 10 is a one inch break with failure of 3/4 ADS4 valves to open. It is beyond design basis counterpart to a ROSA test and is intended to extend the region examined in Westinghouse test SB7. That test had 2/4 ADS4 valves failed and the data showed a decrease in margin to depressurize to allow IRWST injection. This examines design margin for one of the more important AP600 design features, namely the ability to depressurize to allow IRWST injection.

Test 11 is a two inch cold leg break with rescaled ADS1-3. Examination of the Westinghouse test data revealed that ADS1-3 was overscaled for late phase depressurization. The sizing of ADS1-3 was reduced to obtain better scaling.

Test 12 was a two inch cold leg break in the PRHR side of the plant and was the first test with the break on this side and the results show the effect of break location. The break drained away the cold returning fluid from the PRHR. The test was run without CMT refill. During the oscillation upon return to saturation, the break was closed at the point in the oscillation when ADS4 flow was at a minimum. The ADS4 flow proceeded through a normal increase in flow, through a maxima, and then settled out immediately at an expected nominal value. When the break was reopened, the oscillation did not reestablish itself.

Test 13 is currently reserved pending further examination of data from the current testing programs and review of the latest AP600 PRA.

RESULTS OF WESTINGHOUSE TESTS AND ANALYSIS FOR AP600

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SUMMARY

The AP600 is a new reactor design which incorporates increased simplification with enhanced safety. Some of the features of the design include: reduced kw/ft, use of canned motor pumps for increased reliability, reduced numbers of valves and associated piping, and the use of passive safety systems to provide core and containment cooling in case of postulated accidents. The use of gravity driven core and containment cooling results in simplification of the reactor safety systems with enhanced overall safety since the passive systems are not dependent on emergency electrical power to perform their safety function.

The passive safety systems perform the same function as the active systems in current Westinghouse PWRs except that the passive systems utilize gravity driven flows to accomplish the necessary core and containment cooling. The passive systems are able to remove the reactor decay energy and to maintain the containment within its design pressure by using relatively small gravity heads to drive the required flows.

In order to verify that the safety analysis computer codes and predict the performance of the passive safety system with the same confidence as the active systems which exist on operating PWRs, a detailed experimental and analysis program has been underway since 1989. The AP600 Test and Analysis Program utilized basic research experiments, engineering tests, separate effects tests on specific components and integral systems tests at different scales to validate the safety analysis computer codes.

The basic research experiments were performed for the passive containment design and investigated the surface coating, its wettability and ability to spread water films. Basic research experiments were also performed with the University of Wisconsin on condensation heat transfer in the presence of non-condensible gases, as well as condensation on surfaces at different angles.

Engineering experiments were performed to determine the feasibility of different components, their manufacturability, and their expected performance. Engineering tests were also performed to help guide the design of specific components. Examples of the Engineering tests include the tests on the high inertia bearing for the canned motor pump, examining the pump to pump performance when attached to the steam generator head, and investigating the flow distribution in the reactor vessel lower plenum and the impact of the structures on the flow distribution.

The experiments which were used to validate the safety analysis computer codes were the separate effects experiments and the integral systems experiments. The separate effects experiments were performed on a specific component and included tests on a scaled Core Makeup Tank (CMT) and tests on a full size simulation of the Automatic Depressurization system. there were also separate effects tests on a heated plate simulating a portion of the containment shell, air flow tests on the containment baffle and a full scale sector test on the water distribution for the containment shell.

The integral systems tests included the full height, full pressure SPES-2 facility which examined small break LOCA, steam generator tube ruptures, and main steam line break transients; the reduced pressure, reduced height Oregon State University long term cooling tests which examined the small break LOCA behavior and its transition into long term cooling. There were also two scaled containment experiments which were performed. The initial experiment used a three-foot diameter vessel which was 25-feet in height. this experiment examined the stability of the liquid film on the outside of the containment shell as well as the condensation heat transfer inside the tank. The second experiment was the large-scale containment experiment which examined, in more detail,, the gravity driven cooling inside the containment for postulated LOCA transients as well as streamline break transients.

This paper will present the results of comparisons of the safety analysis computer codes to the CMT separate effects tests as well as comparisons to the SPES and OSU integral systems tests. The comparisons will indicate that the passive safety system performance can be predicted in a satisfactory fashion.

The PANDA Tests for SBWR Certification

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In 1991, the Paul Scherrer Institute (PSI) initiated the ALPHA project for the experimental and analytical investigation of the long-term decay heat removal from the containment of the next generation of "passive" ALWRs. The dynamic containment response of such systems, as well as containment phenomena, are investigated. The ALPHA project includes integral system tests in the large-scale (1:25 in volume) PANDA facility; the smaller-scale separate-effects LINX series of tests related to various passive containment mixing, stratification, and condensation phenomena in the presence of non-condensable gases; the AIDA tests on the behavior of aerosols in Passive Containment Cooling Systems (PCCS); and supporting analytical work. The project has been, so far, mainly directed to the investigation of the General Electric (GE) SBWR PCCS and related phenomena.

The PANDA integral-test results were initially expected to bring only confirmatory information for the certification of the SBWR by the US NRC. Recent developments have made the first series of experiments to be conducted in PANDA a *required* experimental element in the certification process; thus, the tests are now performed according to the NQA-1 Quality Assurance procedure.

Other elements of the international program closely linked to the ALPHA project and the SBWR are:

- Single-tube condensation experiments at the UC-Berkeley and at MIT.
- The smaller scale (1:400) integral test facility GIRAFFE, operated by Toshiba.
- The full-scale PCCS condenser qualification PANTHERS experiments performed by SIET in Italy.

Tests in all major PSI facilities started in 1995. In addition, small-scale experiments and numerous analyses were conducted to better understand basic phenomena and SBWR system behavior, to provide preliminary data for the development of computational models, etc.

The PANDA general experimental philosophy, facility design, scaling, and measurement concepts were defined in early 1991. The heavily instrumented facility includes, beyond the classical measurements, non-condensable fraction sensors, phase detectors, etc., a total of some 590 channels. The data acquisition system can sample all channels continuously with a frequency of 0.5 Hz and for short periods of time with a "burst" frequency of 5 Hz. The facility is operated and controlled remotely and interactively by a computer-screen-based system.

The very first series of PANDA experiments conducted at the beginning of 1995 were steady-state PCCS condenser performance tests, as counterpart tests to those conducted at the PANTHERS facility. Extensive facility characterization tests were completed in July 1995: the facility leak rates, heat losses, as well as the pressure-drop–flow-rate characteristics of the various lines were obtained. These are needed for the accurate description of the facility in computations. The actual transient system behavior tests are underway.

In relation to the SBWR certification effort, the PANDA examines system response during the long-term containment cooling period. The objectives of the transient PANDA tests are to demonstrate that:

- The containment performance is similar in a larger scale system to that previously demonstrated with the GIRAFFE tests.
- Any non-uniform distributions in the containment do not create significant adverse effects.
- There are no adverse effects associated with multi-unit PCCS operation and interactions with other reactor systems.
- The tests also extend the data base available for code qualification.

The transient test series at PANDA include two Main Steam Line Break (MSLB) tests. One test is similar to a GIRAFFE MSLB test with uniform Drywell conditions, while the second maximizes the influence of Drywell asymmetries on the operation of the PCCS condensers. Uniform and asymmetric Drywell conditions can be created in PANDA by varying the fraction of steam flow that is injected into each of the interconnected Drywell vessels and by modifying the number of condenser units directly connected to each vessel.

In relation to scaling, both "top-down" and "bottom-up" scaling considerations and criteria were developed. General, "top-down," scaling criteria are derived by considering the processes controlling the state of classes of containment sub-systems (e.g., containment volumes, pipes, etc.). Close examination of specific phenomena or system components (e.g., thermal plumes, vents, etc.) leads to "bottom-up" scaling rules.

As part of the SBWR certification process, blind pre-test calculations are performed and submitted to the US NRC.(in collaboration with IIE in Mexico, KEMA in the Netherlands, and GE) using the TRACG code. Pre-test calculations for the steady-state PANDA PCCS condenser tests showed very good agreement between the calculated condenser performance and the test data, the TRACG calculations being only slightly conservative.

NRC CONFIRMATORY AP600 SAFETY SYSTEM PHASE I TESTING IN THE ROSA/AP600 TEST FACILITY

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SUMMARY

Westinghouse Electric Corporation has submitted the Advanced Passive 600 MWe (AP600) nuclear power plant design to the NRC for design certification. In contrast to the current generation of reactors, this new design features passive safety systems for mitigating accidents and operational transients. Since these passive safety systems rely on gravity-driven flow, the driving forces for the safety functions are small compared to those available under conventional pumped systems. Thus, the performance of these new safety systems may be adversely affected by small variations in thermal hydraulic conditions. Also, the computer analyses of the passive safety systems pose a challenge for current thermal-hydraulic system analysis codes in that the current codes were not sufficiently assessed for conditions of low pressure and low driving heads and for the system interactions that may occur among the multiple flow paths used in the AP600 design. Therefore, integral effects test data have been obtained under the ROSA/AP600 Confirmatory Test Phase I Program for evaluation of AP600 safety system performance and for independent assessment and validation of computer analysis codes. Other integral effects test data were also obtained from the NRC low pressure confirmatory test program at the Oregon State University (OSU) test facility and by Westinghouse from its integral test programs in the SPES-2 (Simulatore Per Esperienze di Sicurezza-2) and OSU (Oregon State University) test facilities. SPES-2 is a full-pressure, full-height test facility in Italy but much smaller in scale (1/395 by volume) than ROSA which represents a 1/30 volume-scale for AP600. The OSU facility is a low pressure, reduced height facility with a considerably smaller volumetric scale (1/200 by volume) as compared to ROSA. NRC confirmatory safety system testing is not required for design certification but would provide additional technical bases for the NRC licensing decisions.

The ROSA facility is a full pressure test facility and thus is particularly good for studying the system behavior under high pressure conditions. The types of accident scenarios chosen for the ROSA/AP600 test matrix were those which would have high pressure conditions for a relatively long period of time during which the system behavior and the interactions among subsystems could be studied. Each accident scenario chosen encompassed a multitude of phenomena, some of which occurred at the same time while others appeared sequentially. Any one test was not chosen to see a particular phenomenon but to examine many phenomena. These tests were integral effects tests and not separate effects tests and covered both design basis and beyond-design basis accident scenarios. In all cases the important objective of the test was to obtain data for the computer code assessment under AP600-like conditions.

Out of 14 tests conducted under the Phase I program, 7 tests were on small-break loss-of-coolant accidents (SBLOCAs) at the cold leg with a break size varying 1/2" to 2", 2 tests on 1" and 2" breaks in the pressure balance line (PBL) to the core makeup tank (CMT), 1 test on a double-ended guillotine break (DEGB) at the direct vessel injection (DVI) line, 1 test on inadvertent actuation of automatic depressurization system (ADS), 1 test on steam generator tube rupture (SGTR), 1 test on main steam line break (MSLB), and 1 test on station blackout. These tests covered the areas of primary concern regarding whether system interactions and other factors under a variety of conditions would prevent depressurization from taking place fast enough to avoid core heatup before an uninterrupted, steady injection from the In-containment Refueling Water Storage Tank (IRWST) is established. Another objective of the confirmatory testing was to understand the passive safety system behavior under different accident conditions to assure that a proper guidance be given to plant operators for various accident conditions.

All 14 tests showed that the core would be effectively cooled, and there would be no danger of heating up the core. However, three areas have been identified for a closer examination. They are:

- Condensation-induced pressure oscillation and water hammer,
- Large thermal gradient in the cold-leg where the passive residual heat removal (PRHR) flow is returned, and
- System-wide oscillations initiating shortly after ADS4 actuation and persisting until steam generation in the core is substantially reduced as a result of cold water injection into the core from the IRWST.

For the scenarios investigated, a concern about the possible violent condensation taking place in the top region of CMTs did not materialize. Instead, the ROSA/AP600 tests showed that recirculation between cold legs and core makeup tanks (CMTs) warmed up cold liquid in CMTs which eventually flashed as system depressurized to allow CMTs to drain. If flashing was not enough to overcome the downstream pressure, the draining would be delayed until the cold leg was uncovered such that steam could flow up to the top of the CMT through a pressure balance line. In this latter case, a violent condensation did not occur because the top part of the CMT liquid had already been warmed up.

The tests demonstrated that the PRHR cooled the system very effectively and kept it subcooled until near the end of the depressurization process. As such, it had a dominant effect on the overall system behavior.

The detailed analyses are underway, and the results will be reported in a future paper.

TRAC-BF1 WATER LEVEL TRACKING ALGORITHM AND SBWR LOOP TYPE OSCILLATION

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Summary

Recently, a possible large amplitude power oscillation (150% rated power) in the Simplified Boiling Water Reactor (SBWR) was predicted by the TRAC-BF1 code [1]. The power oscillation occurred when the downcomer water level reached 19.2 meters above the bottom of the vessel. The power oscillation frequency was found to be about 0.3 HZ. It has been suspected that the power oscillation could be caused by a manometer type loop system instability. Since natural circulation is used in the SBWR, the flow is very sensitive to the water level. Accurate modelling of the water level behavior is important in understanding SBWR behavior.

Recent numerical experiments at Penn State demonstrate that for the international manometer benchmark problem [2], all the safety analysis codes (i.e, TRAC-BF1, TRAC-PF1, RELAP-5 and CATHARE) have excessive numerical damping at or near the liquid level [3]. Proper treatment of the levels can largely eliminate numerical damping and produce a calculational prediction fairly close to physical behavior. These safety codes assume uniform void fraction for every mesh cell. This assumption however fails for a cell, once a void fraction front exists in it. The position and velocity of front must be computed and some physically sensible conditions must be imposed in the solution for the mesh cells around the void fraction front [4]. The position and propagation velocity of the front is calculated in terms of neighboring cell void fractions. Convective flux quantities in the flow equations are redefined when a void fraction front exists. Inertia terms that divide the net forces acting on each field in momentum equations are also redefined for the cell with a void fraction front and their neighboring cells. The results of modelling the water level oscillatory behavior for the international manometer benchmark problem have been greatly improved after these conditions were imposed on the TRAC-BF1 flow equations based on the position and velocity of void fraction front obtained by a simple tracking scheme. The calculated transient water level for manometers with uniform, non-uniform, and symmetric finite-difference grids match the analytical solution very well.

The above modifications apply however to modelling adiabatic and isothermal one-dimensional test problems only. Clearly, further development is needed to improve the scheme in order to model multi-dimensional flows when mass and energy exchanges are present from one phase to another. It will then be possible to redo the SBWR power instability analysis.

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PROBABILISTIC RISK ASSESSMENT TOPICS
INTRODUCTION

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The NRC's research program in probabilistic risk assessment includes a spectrum of activities, from basic research to direct regulatory applications. More specifically, the research program includes the following:

1. development and demonstration of methods which improve existing techniques or fill gaps in the state of PRA technology,

2. development and demonstration of advanced models and tools for use by the NRC staff and others performing risk assessments,

3. review of risk assessments performed by licensees or the staff, and

4. support to other agency staff in risk assessment and statistics.

This session will focus on the first two areas - methods development and advanced model and tool development. (Session 11, Individual Plant Examinations, includes six papers covering the third area of risk assessment research - reviews.)

A total of five papers are included in this session. The first two papers cover projects which are filling gaps in the present state of PRA technology. these gaps, which were brought to light in recent staff risk studies, are: incorporating additional types of human errors in risk assessments, and estimating the uncertainties in key parameters in offsite consequence models.

The third and fourth papers address the development of risk models for staff use. Staff in NRC's Offices of Analysis and Evaluation of Operational Data (AEOD) and Nuclear Reactor Regulation (NRR) have used such models for a number of years to assess the significance of operational events. The two papers discuss recent work to upgrade existing models which focus on the probability of core damage (given an event) and to add new models which incorporate containment, offsite consequence, and risk information.

The final paper addresses tool development, in this case in the form of computer software, closely related to the risk models noted just above. A new module to the SAPHIRE computer code - GEM - has been developed to permit AEOD and NRR staff to more quickly analyze the significance of operational events, using the new models.

Status of Development of an Improved HRA Method: A Technique for Human Error Analysis (ATHEANA)

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A multidisciplinary HRA framework is currently being developed that more realistically represents the roles of humans in both the initiation, prevention, and mitigation of accidents at nuclear power plants.[1,2] This framework will support the characterization of EOCs and human dependencies as well as guide their representation (i.e., modeling and quantification) in PRAs.

The graphic description of the framework, illustrating the inter-relationships between unsafe human actions, their impact on the plant, the incorporation of their impact in the PRA model, and the influences of the plant conditions and PSFs on human reliability is presented in Figure 1. The framework includes elements from the plant operations and engineering perspective, the PRA perspective, the human factors engineering perspective, and the behavioral sciences perspective, all of which contribute to our understanding of human reliability and its associated influences, and has emerged from the review of significant operational events at NPPs by a multidisciplinary project team representing all of these disciplines. The elements included are the minimum necessary set to describe the causes and contributions of human errors in, for example, major NPP events.

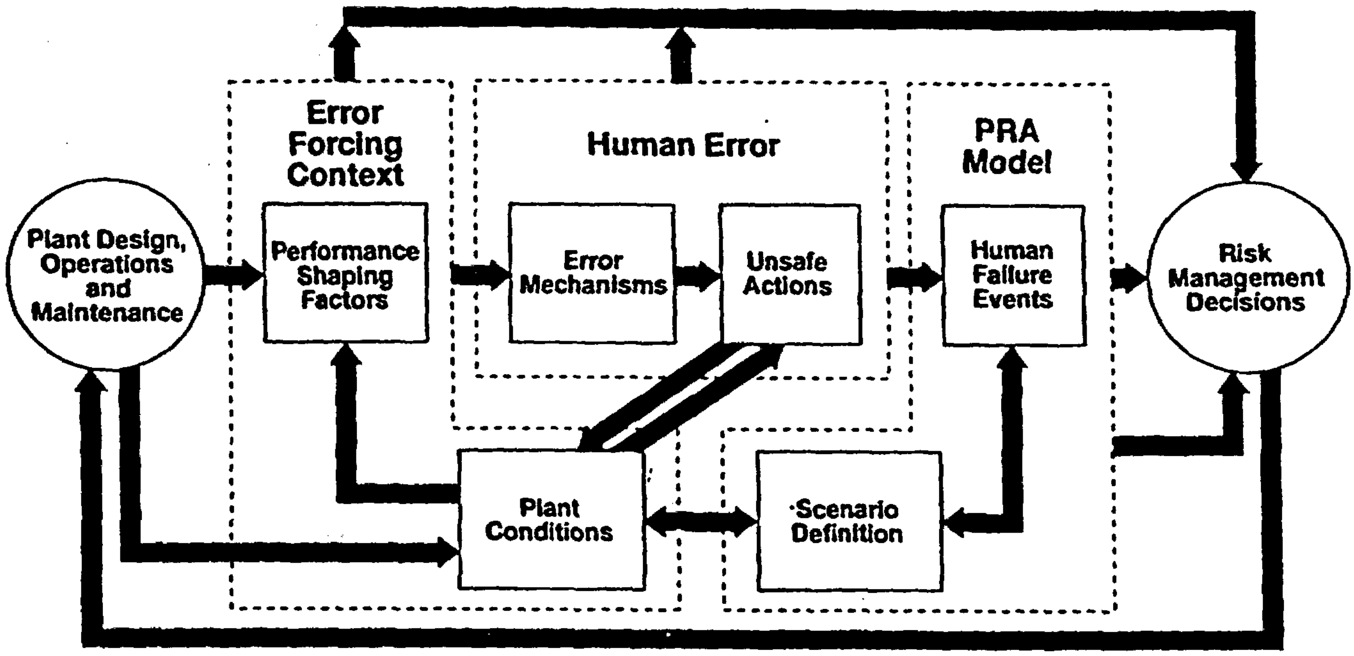

Figure 1. Multidisciplinary HRA Framework

The HRA-related disciplines of the framework, i.e., human factors, behavioral science and plant engineering disciplines, are reflected by the elements on the left side of the figure, namely PSFs, plant conditions, and error mechanisms. These elements are representative of underlying causes of unsafe actions and hence explain why a person may perform an unsafe action. The unsafe action and human failure event elements represent the point of integration between the HRA and PRA. The elements on the right side of the figure represent the PRA perspective with which the HRA-related elements must ultimately be integrated. The PRA traditionally focuses on the consequences of the unsafe action, which it describes as a human error that is represented by a human failure event. The human failure event is included in the PRA model associated with a particular plant state which defines the specific accident scenarios that the PRA model represents.

The framework has served as the basis for retrospective analysis of real operating event histories. That retrospective analysis has identified the context in which severe events can occur; specifically, the plant conditions, significant human performance shaping factors (PSFs), and dependencies that "set up" operators for failure. Based upon applying the framework to the analysis of operating events, serious events have been found to involve both unexpected plant conditions and unfavorable PSFs which comprise an error forcing context. The error forcing context typically represents an unanalyzed plant condition that is beyond, for example, normal operator training and/or procedure PSFs. When the unanalyzed plant condition is combined with unfavorable PSFs, a human error mechanism related to, for example, inappropriate situation assessment (i.e., a misdiagnosis) is activated. Consequently, subsequent mistakes (i.e., errors of commission), and ultimately, an accident with catastrophic consequences can result. Recent discussions with those who have analyzed transportation and aviation accidents (e.g., National Transportation Safety Board) indicate that an error forcing context is most often present in serious accidents involving human operational control in these industries. These contexts are not explicitly modeled in existing PRAs due to constraints imposed by the current "state-of-the-art" in HRA methodology.

Previous HRA methods have focused on addressing the question: "What is the chance of random operator error (e.g., operator fails to...) under nominal conditions?" With respect to understanding, modeling and quantifying human reliability, i.e., based on operating experience, a more appropriate question to pursue is: "What is the chance of occurrence of an error-forcing-context such that operator error is very likely?" How to use the framework to perform this type of 'prospective analysis', i.e., identify and define the context so that important EOCs and dependencies can be identified and predicted, is currently under investigation.

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[1]NUREG/CR-6093, An Analysis of Operational Experience During Low Power and Shutdown and A Plan for Addressing Human Reliability Assessment Issues, U.S. Nuclear Regulatory Commission, Washington, D.C., 1994.

[2]NUREG/CR-6265, Multidisciplinary Framework for Analyzing Errors of Commission and Dependencies in Human Reliability Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C., in press.

UNCERTAINTIES ON OFFSITE CONSEQUENCE ANALYSIS

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This paper describes an ongoing project supported jointly by the U. S. Nuclear Regulatory Commission (NRC) and the Commission of the European Communities (CEC). The objectives of this joint effort are to formulate a state-of-the-art approach, and systematically obtain much of the quantitative information necessary to perform uncertainty analysis using their respective probabilistic consequence codes, MACCS and COSYMA. The distributions utilized in consequence uncertainty studies prior to this joint project have been developed primarily by consequence code developers rather than the phenomenological experts in the many different scientific disciplines that constitute a consequence analysis. Both commissions were aware of the key role of uncertainty in decisions involving prioritization of activities and research, and they were interested in initiating a comprehensive assessment of the uncertainty in consequence calculations used for risk assessments and regulatory purposes. The two commissions, therefore, decided to pool their resources.

Formal expert judgment techniques were selected to develop distributions over important consequence analysis input parameters for which the experimental database does not provide all the necessary information, and the analytical models used for extrapolations are not indisputably correct. Less resource intensive methods will be used for the development of the remaining needed distributions. The state-of-the-art approach formulated by the joint project staff was based on two important ground rules: (1) the current code models would not be changed because both the NRC and the CEC were interested in the uncertainties in the predictions produced by MACCS and COSYMA, respectively, and (2) the experts would only be asked to assess physical quantities which could be hypothetically measured in experiments. Benefits of these ground rules were: (1) the codes have already been developed and applied in U.S. and European risk assessments, and (2) eliciting physical quantities avoids ambiguity in variable definitions and, more importantly, the elicited physical quantities are not tied to any particular model and thus the results will have a much wider application.

To ensure the quality of the elicited information, a formal expert judgment elicitation procedure, built on the process developed for and used in the NUREG-1150 study, was followed. Refinements were implemented based on the experience and knowledge gained from several formal expert judgment elicitation exercises performed in the U.S. and Europe since the NUREG-1150 study. The philosophy of the project was to allow the experts to use whatever modeling techniques or experimental results they believed were appropriate to assess the problems. The only constraints were: (1) the initial conditions in the elicitation questions could only be specified to the same level of detail as in MACCS and COSYMA; (2) code input distributions could be generated from the elicited assessments; and (3) the rationale behind the assessments was thoroughly documented by the experts. The joint approach for this project is summarized below:

1. **Definition of elicitation variables:** Elicitation variables are the variables presented to the experts for assessment. When the important code input parameters are not physical variables, the probability assessment team is required to select physically measurable elicitation variables from which distributions over code input parameters can be developed.
2. **Selection of experts:** The objective of the expert selection process is to engage the best

experts available in the phenomenological areas of interest. A large list of experts is compiled from the literature by requesting nominations from experts in and organizations familiar with the phenomenological area. The experts are contacted and curriculum vitae (CV) are requested. The CVs are evaluated and experts are selected by an impartial selection panel.

3. **Preparation of experts for elicitation:** The experts are introduced to the purposes of the study and background material on consequence codes and the science of probability elicitation. Training is conducted to introduce the experts to the psychological biases in judgment formation, and to give them feedbacks on their performance in assessing probability distributions. Following the preparation meeting, the experts spend several weeks preparing responses to the elicitation questions and a written statement explaining their information sources and rationale.

4. **Elicitation:** On the first day of the elicitation meeting, the experts deliver presentations explaining how they addressed the issues without giving their quantitative assessments. The elicitation of each expert is conducted privately with a specialist in probability assessment and a project specialist in the consequence codes.

5. **Processing of judgments:** The first step in processing the elicited information is the aggregation of the assessments from the individual experts into a single distribution for each elicitation variable. In the aggregation process, all experts are assigned equal weight, i.e., all experts on each respective panel are treated as being equally credible. If the elicitation variables are not code input variables, it is necessary to mathematically transform the aggregated distributions to obtain distributions over the appropriate code input parameters.

The above approach has been applied in the phenomenological areas of atmospheric dispersion and deposition, behavior of deposited material and related doses, and ingestion pathways. A detailed review of the dispersion and deposition elicitation exercise, and the development of the uncertainty distributions over the related code input parameters, have been published in a three-volume joint report, NUREG/CR-6244, EUR 15855 EN, in January 1995. The documentation reviewing the food pathway and behavior of deposited material panels is currently being prepared by the project staff, and will be published by the end of 1995. Expert panels in the phenomenological areas of dosimetry, early health effects and late health effects are being planned, and will be completed by mid-1996. Since the input parameters in the remaining areas in the consequence calculations, such as protective actions, are specific to the U.S. or Europe, joint expert panels will not be held. However, the same approach developed in this joint project will be followed if the commissions decide to convene expert panels in these areas at a later date.

The uncertainty distributions which have been developed in this project represent state-of-the-art knowledge in the areas assessed. The distributions elicited to date were developed by the experts from a variety of information sources (analytical, experimental, observational, etc.) for physical quantities. The aggregated distributions, therefore, include variations due to different modeling approaches and perspectives, and are anticipated to have many future applications beyond the current joint study. Furthermore, formal expert judgment elicitation has proven to be a valuable vehicle to synthesize the best available information. With a thoughtfully designed elicitation approach, expert judgment elicitation can play an important role and possibly becomes the only alternative technique to assemble the required information when it is impractical to perform experiments or the available experimental results do not lead to an unambiguous and a non-controversial conclusion.

ADVANCED ACCIDENT SEQUENCE PRECURSOR ANALYSIS LEVEL 1 MODELS

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SUMMARY

The Idaho National Engineering Laboratory (INEL) has been involved in the development of plant-specific Accident Sequence Precursor (ASP) models for the past two years. Last year 75 individual plant models, that represent 107 operating plants, were developed and delivered to the NRC. These models were developed for use with the SAPHIRE suite of PRA computer codes. They contained event tree/linked fault tree Level I risk models for the following initiating events: general transient, loss-of-offsite-power, steam generator tube rupture, small loss-of-coolant-accident, and anticipated transient without scram. These models were generally train-level models "rolled up" from more detailed traditional PRA system models. They included common cause failures and operator actions. They did not include any support system failures other than basic emergency AC power or maintenance/testing unavailabilities.

This year the ASP models were revised based on review comments from the NRC and an independent peer review. These models were released as Revision 1. The Office of Research (RES) has sponsored several projects at the INEL this fiscal year to further enhance the capabilities of the ASP models. The first project, W6467, incorporates more detailed plant information into the models concerning plant response to station blackout conditions, more recent information on battery life, and other unique features gleaned from an Office of Nuclear Reactor Regulation (NRR) quick review of the Individual Plant Examination (IPE) submittals. These models are being released as Revision 2. All Revision 2 models are scheduled to be completed by the end of November 1995. The second project, W6355, is a feasibility study for model development of low power and shutdown (LP&S) internal and external events, and full power flooding extensions to the ASP models. This project will establish criteria for selection of LP&S and full power flooding initiator operational events for analysis within the ASP program. Prototype models for each pertinent initiating event (loss of shutdown cooling; loss of inventory control; LP&S fire, flood, and seismic; etc.) will be developed. These LP&S models will reflect the changing success criteria as a plant goes through all the various plant operating states (POSs) from power operation through shutdown, draining for mid-loop operations, refueling, and back up to power operations again. The full power and LP&S external event models will capture the key elements of initiating event frequencies and locations, propagation paths and barriers, and mitigating responses. A third project, W6340, is developing extensions to the ASP models addressing the Level 2 and 3 PRA issues. This work is being presented in a separate paper for this conference. Another project concerns development of enhancements to SAPHIRE; and this work is also being presented in a separate paper for this conference. In relation to the ASP program, a new SAPHIRE module, GEM, was developed as a specific user interface for performing ASP evaluations. This module greatly simplifies the analysis process for determining the conditional core damage probability for a given combination of initiating events and equipment failures or degradations.

ADVANCED ACCIDENT SEQUENCE PRECURSOR ANALYSIS LEVEL 2 MODELS

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The U. S. Nuclear Regulatory Commission Accident Sequence Precursor (ASP) program pursues the ultimate objective of performing risk significance evaluations on operational events (precursors) occurring in commercial nuclear power plants (NPPs). To achieve this objective, the Office of Nuclear Regulatory Research is supporting the development of simple probabilistic risk assessment (PRA) models for every NPP in the U.S. Presently, only simple Level 1 plant models have been developed, which estimate core damage frequencies. In order to provide a true risk perspective, the consequences associated with postulated core damage accidents also need to be considered. Therefore, simple plant models that analyze the response of the NPP containment structure in the context of a core damage accident, estimate the magnitude of a radioactive release (source term) to the environment, commonly referred to as Level 2, and calculate the consequences (typically health and financial effects on the surrounding population) associated with a particular source term, commonly referred to as Level 3, are necessary.

This paper documents the current ASP Level 2/3 model development effort. The first step follows the procedure used in the Level 1 portion, of classifying all NPPs into groups or classes with a single plant being selected from each group as the subject of the initial model development effort. The objectives of the Level 2/3 plant-group model development are to demonstrate: (1) appropriate interfaces between the Level 1 models and the Level 2 models; (2) simplified Level 2 models; (3) source terms (STs) estimates; (4) consequence estimates; and (5) integration of the Level 2/3 models into the existing ASP software. Each of these issues will be addressed in the following paragraphs. Furthermore, during the plant-group model development, to the extent feasible, information and methods developed and collected in the course of the NUREG-1150 study are utilized.

Since the existing ASP Level 1 event trees simply identify whether or not a particular accident sequence results in core damage, they do not satisfy the informational requirements of the Level 2 models. In addition, some Level 2 issues involve consideration of containment systems which are not modeled in the Level 1 event trees. It is important to include the containment systems models with the Level 1 event tree in order to account for dependencies, e.g., support systems such as ac power, between the Level 1 systems and the containment systems. This necessitates the use of bridge trees, event trees attached to the end of the Level 1 event trees that model containment systems. For each bridge-tree endstate, a PDS vector, as used in the NUREG-1150 study, is generated using a set of logic (IF-THEN) rules on the combined Level 1-bridge tree accident sequences. These logic rules question the success and/or availability of certain plant systems and functions, which figure into the development and quantification of the containment event trees (CETs).

The CETs are greatly simplified versions of the NUREG-1150 accident progression event trees (APETs). Basically, the events deemed most relevant to the identification of a source term are extracted from the APET and organized into a graphical event tree (the CET). In most cases, the split fractions for the CET are generated by rolling up the detailed APET (using the EVNTRE code) to the corresponding intermediate level. In some cases, logic models are developed and included in the CET in the form of fault trees supporting a particular top event appearing on the CET.

Developing appropriate source terms for the accident scenarios identified through the CETs presents two issues needing resolution. First, since the ASP models intend to include more than the five plants analyzed in the NUREG-1150 study, the information base for the XSOR codes may not be adequate. As a possible alternative to the XSOR codes, a parametric source term code (PST) is being developed. PST models in simple terms the transport of radioactive material through the various volumes in the pathway from the reactor vessel to the environment. Conservation of curies is maintained through the use of transport fractions into and out of each volume in the pathway. Existing literature is used to estimate the different transport fractions. PST will be benchmarked against the XSOR codes for the NUREG-1150 plants during the course of the ASP Level 2/3 plant-group model development. The second source term issue to be resolved is the transfer of information from the CET to XSOR or PST. As in the NUREG-1150 study, a source term vector is formed by examining each endstate on the CET to generate the appropriate identifiers in each position in the vector. A set of logic (IF-THEN) rules are written to test for the appearance of certain events in the CET scenario. When these characteristics are found, a specific identifier is written into the source term vector. PST is written to read these vectors, and to automatically generate the corresponding source term.

MACCS is used to calculate the offsite consequences associated with each source term. Initially, the consequence measures are estimated for a generic site, and include: 50-mile population dose (effective dose equivalent), 50-mile population thyroid dose, average individual early fatality risk within 1 mile from the site boundary, and average individual latent cancer fatality risk within 10 miles from the site boundary. Once calculated, these consequence results will not change for a given severe accident scenario, i.e., source term. Hence, they are only calculated once and then hardwired onto the corresponding CET endstates.

Ultimately, all parts of the ASP model (including Level 1 and Level 2/3) will be constructed and run using IRRAS. Although the GEM module is used to perform ASP evaluations, GEM is basically a shell that simplifies the interface between the ASP user and IRRAS. A number of modifications to IRRAS were needed in order to accommodate the development of the Level 2/3 model and the linking of the different ASP models. These modifications created the ability to automatically utilize the output from one event tree (i.e., PDS from the Level 1 event tree) as an initiating event for a second event tree (i.e., the CET). Also, the capability in IRRAS to generate endstate identifiers (e.g., PDS and source term vectors) through a set of generic logic (IF-THEN) rules was enhanced.

Subsequent to the completion of the Level 2/3 plant-group models, the adequacy of these models will be examined based on plant information and experimental data which have become available after the NUREG-1150 study, such as the IPE program. It is expected that these plant-group models will be further refined to account for the important characteristics that differentiate the NPPs.

NEW DEVELOPMENTS IN THE SAPHIRE COMPUTER CODES*

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SUMMARY

The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) refers to a suite of computer programs that were developed to create and analyze a probabilistic risk assessment (PRA) of a nuclear power plant. The programs in this suite include: Models and Results Data Base (MAR-D) software, Integrated Reliability and Risk Analysis System (IRRAS) software, Systems Analysis and Risk Assessment (SARA) software, Fault tree, Event tree, and Piping and instrumentation diagram (FEP) graphical editor and the Graphical Evaluation Module (GEM). Each of these programs performs a specific function in taking a PRA from the conceptual state all the way to publication.

Many recent enhancements to this suite of codes have been made. This presentation will provide an overview of these features and capabilities. The presentation will include a discussion of the new GEM module. This module greatly reduces and simplifies the work necessary to use the SAPHIRE code in event assessment applications. An overview of the features provided in the new Windows version will also be provided. This version is a full Windows 32-bit implementation and offers many new and exciting features. A separate computer demonstration will be held to allow interested participants to get a preview of these features. The new capabilities that have been added since version 5.0 will be covered. Some of these major new features include the ability to store an unlimited number of basic events, gates, systems, sequences, etc.; the addition of improved reporting capabilities to allow the user to generate and "scroll" through custom reports; the addition of multi-variable importance measures; and the simplification of the user interface. Although originally designed as a Level 1 suite of codes, capabilities have recently been added to SAPHIRE to allow the user to apply the code in Level 2 analyses. These features will be discussed in detail during the presentation.

The modifications and capabilities added to this version of SAPHIRE significantly extend the code in many important areas. Together, these extensions represent a major step forward in PC-based risk analysis tools. The presentation will provide the attendees with a current up-to-date status of these important PRA analysis tools.

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* Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under DOE Idaho Operations Office Contract DE-AC07-76ID01570.

INDIVIDUAL PLANT EXAMINATION SESSION INTRODUCTORY REMARKS

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The USNRC initiated the Individual Plant Examination Program in 1988. Its objective is for each licensee to systematically examine their plants for plant-specific vulnerabilities to severe accidents and, if found, to take appropriate actions to reduce or eliminate the vulnerabilities. At present, almost all of the IPE submittals have been completed and submitted to the NRC for review. The NRC review of the submittals is well underway.

The USNRC also initiated two follow-up programs. One is to identify patterns and extract insights from the IPE submittals to support regulatory efforts. The emphasis of the program is to search for any potential generic significance arising from plant features. In addition, the program will quantitatively assess the impact of the proposed plant changes and modifications on core damage frequency and containment performance. The USNRC believes that the nuclear industry will benefit from what we have learned from this program and would like to share the information with you. Some of the preliminary results of this program are to present in this session.

The other follow-up program is the development of a database to store information about plant design, core damage frequency and containment performance in a structured and formal manner. It is designed so that the user can extract information and can make detailed inquiries regarding these characteristics across a defined class of plants. The USNRC plans to release the database for public use when it is completed. A discussion of the database will be presented in this session.

Core Damage Frequency Perspectives for BWR 3/4 and Westinghouse 4-Loop Plants Based on IPE Results

Mary Drouin (NRC), Susan Dingman (SNL), Jeff LaChance (SAIC), Allen Camp (SNL)

In performing a PRA, an analysis of the As-Built, As-Operated Plant is performed by collecting data and information on the plant design (i.e. redundancy and independence of systems), plant performance (i.e. reliability and availability of systems), and operational activities (i.e. maintenance and emergency). This information is synthesized using defined boundary conditions, assumptions and modeling processes. One product of this PRA process is perspectives on the Core Damage Frequency (CDF). The purpose of this paper, then , is to 1) discuss the technical approach used to derive CDF insights from the IPEs, and 2) present the CDF insights resulting from this process for BWR 3/4 and Westinghouse 4-Loop plants.

First, the technical approach used to summarize information from all IPE submittals to gain CDF insights will be discussed. To obtain CDF insights, a four step process was used: 1) summary of plant core damage frequencies, 2) summary of dominant accident types, 3) summary of dominant factors contributing to core damage, and 4) generic implications of the results.

Second, the CDF insights will be discussed. These insights will be presented for BWR 3/4 and Westinghouse 4-Loop plants. More specifically, calculated CDFs for each of the accident classes (i.e. SBO, ATWS, LOCA) will be discussed. Important factors which contribute to the variability in the individual plant contributions to the accident classes will be presented. These factors include unique plant characteristics and design, reliance on plant specific vs. generic data, and assumptions or boundary conditions defined by a particular plant. Thus, similarities and differences in results observed in the IPE submittals will be discussed.

Severe Accident Progression Perspectives for Mark I and Ice Condenser Containments Based on IPE Results

Mary Drouin (NRC), C.C. Lin (BNL), John Lehner (BNL), Trevor Pratt (BNL)

In addition to the perspectives on core damage frequency found in the IPE submittals, accident progression perspectives were obtained which consisted of insights related to the containment failure modes, the releases associated with those failure modes, and the factors responsible for the types of containment failures and release sizes reported. These factors involve actual containment design characteristics and plant specific hardware or operational features, as well as similarities and differences in assumptions and modeling techniques. The purpose of this paper, then, is to 1) discuss the technical approach used to derive severe accident insights from the IPEs, and 2) present severe accident progression insights resulting from this process for Mark I and Ice Condenser Containments.

First, the technical approach used to summarize information from all IPE submittals to gain severe accident progression insights will be discussed. To obtain severe accident insights, a four step process was used: 1) collect and summarize accident progression results, 2) identify similarities and differences in the accident progression results, 3) assess the similarities and differences in the accident progression results, and 4) derive generic vs. plant specific accident progression perspectives.

Second, the severe accident progression perspectives for Mark I and Ice Condenser Containments will be discussed. While there was significant variability in the approaches chosen by the licensees, the essential information regarding containment failure modes, release type and size, and factors driving the analyses could be found in all submittals. Specifically, important parameters related to accident progression and containment performance which could be obtained from the submittals were 1) frequency and conditional probability of early containment failure and bypass, 2) frequency and conditional probability of late containment failures, and 3) magnitude of importance source term releases.

Finally, the insights for severe accident progression will be presented. This discussion includes: 1) general containment performance results, 2) plant features and phenomena important to accident progression, 3) liner melt-through and containment venting, 4) early containment failure, 5) containment bypass, 6) late containment failure, and 7) large early fission product release. The reasons for the similarities and differences in the results from different IPEs will be provided and discussed.

Human Action Perspectives Based on IPE Results

Kitty Thompson (NRC), Mary Drouin (NRC),
Erasmia Lois (NRC), John Forester (SNL)

The determination and selection of human actions for incorporation into the event and fault tree models and the quantification of their human error probabilities (HEPs) can have an important impact on the resulting estimates of Core Damage Frequency (CDF) and Risk. The purpose of this paper, then, is to identify and discuss 1) the important human actions found across plants and the variability in identified actions due to plant design or characteristics, 2) the variations in modeling, assumptions, and quantification methods that can create "real" vs. artifactual differences in the results of Human Reliability Analysis (HRA), and 3) the HEPS for specific human actions obtained from the IPEs to demonstrate the variability of results as a function of "real" or artifactual differences.

First, human actions found to be important in the IPE submittals will be identified and discussed to understand whether there is a general consistency in which human actions were identified as important across plants. For example, the most important human actions identified for BWRs involve manual depressurization, alignment of containment cooling, initiation of SLC, containment venting, and level control in ATWS. Further, differences in important human actions due to plant type and unique plant design or characteristics will be discussed to understand these causes of variability in identified important human actions.

Second, discussions will focus on variability in identified human actions and HEPs which is introduced by sequence specific attributes, dependencies, HRA methods and performance shaping factors (PSFs) modeled, assumptions about the PSFs and biases of both the analyst performing the HRA and the plant personnel from which selected information and judgements are obtained. While some of these factors introduce "real", or appropriate differences, others have the potential for creating inappropriate variability. For example, a particular methodology used in several IPEs, which is apparently a modified version of THERP, is distinguished by its lack of emphasis on modeling the diagnosis portion of a task, while creating a PSF referred to as "slack time" which allows substantial credit to be given for potential recovery of initially failed operator actions.

Finally, these causes of variability will be discussed within the context of specific human actions to demonstrate the variability in the derived HEPs from the IPEs and to discuss whether this variability is due, in general, to "real" or artifactual influences. For example, switchover to recirculation is an important operator action in PWRs. The difference between the lowest and highest HEP values between plants is several order of magnitude, with the greatest variability occurring for plants with automatic initiation. In general, it appears that the more detailed and thorough the analysis, in which procedures and training were thoroughly reviewed, the lower the HEPs. Thus it seems that the variability is due to both "real" plant differences (automatic vs. manual initiation) as well as to differences in methodology.

Plant Vulnerability and Improvement Perspectives Based on IPE Results

Richard Clark (NRC), John Lane (NRC), Alan Kolackowski (SAIC)

One of the most significant goals of the IPE process was to identify any unique plant vulnerabilities and take appropriate actions to address these vulnerabilities. It is clear from the submittals, however, that most licensees went beyond this limited intent and identified other improvements worthy of consideration or even implementation. Hence, the IPE program served as a catalyst or further improving the overall safety of nuclear power plants as a result of numerous improvements either implemented, planned, or under evaluation as to their cost-effectiveness. The purpose of this paper is to summarize and discuss plant improvements as reported in the IPE submittals.

Plant improvements were categorized as to 1) whether they were already credited in the submittal and implemented and so are already reflected in the results and insights previously summarized in this report, and 2) the type of improvement by characterizing them as operational or maintenance-related or design-impacted improvements. Additionally and where particularly significant, the reduction in core damage frequency estimated by the licensee as a result of implementing the improvement was also noted.

The potentially generic vs. plant unique nature of the improvements has implications and potential significance for future regulatory actions. For instance, some improvements may be worthy of further investigation for industry-wide implementation; others may be important to a select group of plants; etc. Hence, as part of the improvements assessment discussion, the extent to which similar improvements were identified at numerous plants was also noted.

IPE Results as Compared with NUREG-1150

Ed Chow (NRC), Trevor Pratt (BNL), Allen Camp (SNL),
C.C. Lin (BNL), John Lehner (BNL)

In 1990 the NRC published NUREG-1150 which assessed the risks for five U.S. nuclear power plants: Unit 1 of Surry Power Station, Unit 1 of Zion Nuclear Plant, Unit 1 of Sequoyah Nuclear Power Plant, Unit 2 of the Peach Bottom Atomic Power Station, and Unit 1 of the Grand Gulf Nuclear Station. The purpose of this paper is to provide a comparison of the results and perspectives from NUREG-1150 to those from the IPEs.

First, consideration is given as to whether or not the NUREG-1150 CDF results are consistent with those found in the IPEs. The results indicate that the NUREG-1150 CDF results fall within the range of the IPE CDF results. Both NUREG-1150 and the IPEs have shown that the relative contributions of accident sequences to the CDF are plant specific. Therefore, the accident sequence which dominates in one plant may not be dominant in another. However, the results show that the mix of contributors is consistent with the results found in NUREG-1150. That is, for the PWRs, station blackout, transients and LOCAs tend to be important contributors, while for the BWRs, station blackout and transients tend to be the most important.

After evaluating the accident sequences leading to core damage and calculating the CDF, both the NUREG-1150 and IPE programs evaluated the ability of the containments to prevent the release of radioactivity. Thus, the containment performance results obtained from the NUREG-1150 and IPE programs were compared fro consistency. In general, the events that contribute to the IPE frequencies are similar to those that contributed in the NUREG-1150 study. For example, direct containment heating is an important failure mode for PWRs with large dry and subatmospheric containments, hydrogen combustion is important for PWR ice condensers and BWR Mark III containments, and liner melt-through is important for BWR Mark I containments.

Finally, perspectives derived from NUREG-1150 were compared with those obtained from the IPE results. Each of the NUREG-1150 perspectives were summarized and then a determination was made whether or not the IPE results supported those findings.

IPE DATABASE :PLANT DESIGN, CORE DAMAGE FREQUENCY AND CONTAINMENT PERFORMANCE INFORMATION

J. Lehner, C. Lin, W. T. Pratt (BNL), T. Su (NRC)

ABSTRACT

Substantial progress has been made on the IPE Database over the last two years. The entries into the Database from the internal events examination have been completed for all IPEs based on the information contained in the IPE submittals. Another datafile, containing high level information, has been completed and is being coupled to the rest of the database files. This additional file contains summary information found in each IPE submittal regarding initiating event frequencies and the total contribution to core damage from each initiating event. Because of the way information was reported in the submittals, the previously existing accident sequence files of the Database contain only partial results regarding the contribution to core damage from various initiators. The summary information file remedies this situation.

Future improvements of the IPE Database are currently focussed on the development of a user friendly version of the Database which is menu driven and will allow the user to ask queries of varying complexity without being familiar with dBase IV. In addition, information from the external events examination, the IPEEE, will eventually be incorporated into the Database.

Generic Safety Issue Program at NRC

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

Generic Issues are defined by NRC to be those issues which are applicable to all, several, or a class of reactors or reactor-related facilities. The issues can arise from engineering, systems or severe accident concerns, and analysis of operating events, and are managed and resolved by a variety of mechanisms. Generic issues are classified as safety Issues (GSI), Regulatory Impact Issues (RI), which are not safety related but would modify current NRC guidance especially on cost; Environmental Issues (EI) involving impact on items protected by the National Environmental Protection Act; and Licensing Issues (LI) which seek to improve the staff's knowledge in an area for better assessments of safety, and to improve NRC operational effectiveness and efficiency. The generic issue process in NRC consists of six phases: Identification, Prioritization, Resolution, Imposition, Implementation, and Verification. In the prioritization stage, a quantitative assessment of the safety benefits (risk reduction) and NRC and utility impacts (costs) are described. Based on the extent of potential risk reduction to the public and the impact/value ratio developed from this assessment, a priority of HIGH, MEDIUM, LOW, NEARLY RESOLVED, or DROP is assigned to each issue. Following prioritization, resolution of the HIGH, MEDIUM, and NEARLY RESOLVED issues is usually achieved through rulemaking, development or revision of Regulatory Guides, modifications to Technical Specifications or the Standard Review Plan, and publishing of Generic Letters or Information Notices.

Resolution of generic issues began at NRC in the mid-to-late 1970's upon direction from both the Commission and Congress. The program plan was published in NUREG-0410 in January 1978, whereupon the Commission issued a Policy Statement on the "Program for Resolution of Generic Issues Related to Nuclear Power Plants." The early attempts at categorization of issues from "significant" to "little or no importance" was soon overwhelmed by the large number of issues raised by the Three Mile Island Unit 2 (TMI-2) accident. In addition to the TMI-2 issues, a number of Task Action Plan Items were also identified around this same time, identified by the prefixes "A," "B," "C," and "D." Furthermore, a series of Unresolved Safety Issues (USIs) were also identified and listed. Finally, a series of issues related to the Human Factors Program Plan as well as a series of Chernobyl Issues were identified for study. The quantitative prioritization system described above was approved by the Commission in November 1983, and was published in NUREG-0933 "A Prioritization of Generic Safety Issues;" this document continues to have the updated versions of the prioritization criteria approved by the Commission, and is the repository for documentation on all issues identified plus their resolution status. Finally, in April 1989, the Commission recognized that the earlier Policy Statement did not match the current program, so it was withdrawn.

The NRC Office of Nuclear Regulatory Research currently has responsibility for maintaining NUREG-0933 as the official documentation for NRC generic issues. Many other technical issues of a generic nature but which have not reached the level of recognition of a "Generic Safety Issue" are being studied and handled in the Office of Nuclear Regulatory Research, in the Office for Analysis and Evaluation of Operatal Data, and in the Office of Nuclear Reactor Regulation. All the issues being worked on in RES, both approved GSIs and studies leading up to such approval status, are tracked in the Generic Issue Management Control System (GIMCS), which is updated quarterly. Issues handled by the Research office require more detailed risk analyses and are of a longer-term nature. Most of NRR's issues are short term or are compliance issues, and are now being tracked and documented in a new, NRR Monthly Status Report. Close coordination and cooperation exists between NRR, AEOD, and RES for all six phases of the generic issue process.

An Overview of the BWR ECCS Strainer Blockage Issue

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The safety issue concern regarding blockage of boiling water reactor (BWR) emergency core cooling system (ECCS) suction strainer by debris following a loss-of-coolant accident is that the deleterious effects of debris blockage will imperil the long-term cooling function of the ECCS. Power reactor licensees in the United States (USA) are required to ensure long-term cooling to the reactor core in accordance with Code of Federal Regulations (CFR), specifically 10 CFR 50.46 "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." The United States Nuclear Regulatory Commission (NRC) first addressed this concern as part of the resolution of Unresolved Safety Issue (USI) A-43 "Containment Emergency Sump Performance." The resolution was published in October 1985. The USI A-43 evaluation and resolution focused primely on pressurized water reactors, but its results were considered applicable to BWRs. In 1992 a debris blockage event at the Barsebäck Nuclear Power Plant Unit 2 in Sweden and in 1993 the debris blockage events at the Perry Nuclear Power Plant in the USA, raised concerns about the applicability of the evaluation and resolution of USI A-43 to BWRs. During the Barsebäck event, two ECCS suction strainers were blocked by mineral wool debris; mineral wool insulation was dislodged by steam from a pilot-operated relief valve. Approximately seventy minutes into the event, the operators received indications of high differential pressure across the suction strainers. The high differential pressure was caused by the accumulation of mineral wool debris on the strainers. During the two blockage events at suction strainers were blocked by a relatively thin layer of debris consisting of fiberglass fibers, corrosion products, and other debris. Structural damage occurred to one of the strainers during one of the two events. In addition to illustrating the deleterious effects of debris blockage of a suction strainer, the Perry event raised a concern about the contribution of filtration of particulate debris (i.e., corrosion products) to the severity of debris blockage, which was not addressed during the USI A-43 study.

Because of the 1992 Barsebäck event, the 1993 Perry events, and the activities of Swedish and Finnish regulatory authorities, who began addressing this issue after the Barsebäck event, the NRC began an evaluation of the effects of debris blockage of BWR suction strainers on long-term cooling function of the ECCS system in BWRs. In 1993 the NRC completed a deterministic scoping analysis of the effect of debris blockage at BWRs. During the scoping analysis the NRC conducted a survey of BWRs for information important to an evaluation of the susceptibility of BWRs to debris blockage, such as strainer surface area, flow rate through the strainer, and insulation installed in the containment. The information from the survey and a modified and simplified methodology used in the USI A-43 analysis was used to perform the scoping analysis. The results of this analysis indicated that BWRS were susceptible to the deleterious effects of debris blockage.

In late 1993 the NRC embarked on a more rigorous evaluation of the potential of strainer blockage at BWRs to cause loss of net positive suction head. The evaluation used a similar methodology and a level of analysis equal to the one used during the USI A-43 study. The original goals of the evaluation (assessment) were:

1. Estimate the likelihood that debris blockage of a suction strainer following a LOCA will result in core damage,
2. Develop a model for calculating debris generation, debris transport, and the head loss caused by debris blockage, and
3. Recommend possible solutions to the problem, if one exists.

During the course of the study the goals were expanded to include:

1. Develop a calculational model (i.e., computer code) that can be used by the NRC staff to independently assess licensees' analysis.
2. Assess whether reflective metallic insulation (RMI) will block suction strainers.

Later in the study the probabilistic aspects of the evaluation were de-emphasized and the deterministic aspects (i.e., head loss estimates, debris generation estimates, debris transport estimates) were elevated. The results of the study, which are documented in NUREG/CR-6224 " Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," indicate that the ability of the ECCS to provide cooling water to the reactor core following a design based LOCA will be lost within ten minutes.

<u>Regulatory Perspective on the BWR ECCS Strainer Blockage Issue</u>

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The basic regulatory issue concerning blockage of boiling water reactor (BWR) emergency core cooling system (ECCS) suction strainers by debris generated during a loss-of-coolant accident (LOCA) is that blockage of the strainers is highly likely to occur shortly after the start of the accident. Section 50.46 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.46) requires that licensees design their ECCS systems to meet five criteria, one of which is to provide long-term cooling capability of sufficient duration following a successful system initiation so that the core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core. The ECCS is designed to meet this criterion, assuming the worst single failure. Experience gained from operating events and detailed analysis demonstrate that excessive buildup of debris from thermal insulation, corrosion products, and other particulates on ECCS pump strainers is highly likely to occur, creating the potential for a common-cause failure of the ECCS, which could prevent the ECCS from providing long-term cooling following a LOCA. Action by licensees, therefore, is required in order to ensure compliance with the regulations; specifically, to ensure that long-term cooling can be provided in accordance with 10 CFR 50.46. As a result, the staff issued NRC Bulletin 95-XX requesting licensees take appropriate actions to ensure that their ECCS would be able to provide long-term cooling in accordance with 10 CFR 50.46.

The staff based their action on a number of insights gained from operating plant events, and from detailed studies and experiments. The Barsebăck event demonstrated that a pipe break can generate and transport large quantities of insulation debris to the suppression pool where they can be deposited onto strainer surfaces and potentially cause the ECCS to lose NPSH. The Perry events further demonstrated that fibrous insulation debris combined with corrosion products present in the suppression pool (sludge) can exacerbate the problem. This phenomenon was confirmed in the staff study which showed that the calculated loss of NPSH could occur soon (less than 10 minutes) after ECCS initiation. The effect of filtering sludge from the suppression pool water by fibrous debris deposited on the strainer surface was further confirmed in NRC-sponsored testing conducted at the Alden Research Laboratory which demonstrated that the pressure drop across the strainer was greatly increased by this filtering effect. Additional testing sponsored by the NRC at Alden Research Laboratory demonstrated that the energy conveyed to the suppression pool during the "chugging" phase of a LOCA is sufficient to ensure that the fibrous debris and sludge are well-mixed and evenly distributed in the suppression pool, and can remain suspended for a sufficiently long period of time to allow large quantities to be deposited onto the strainer surfaces.

The staff also evaluated the different plant designs and concluded that this problem is applicable to all domestic BWRs. Three reasons formed the basis for the staff's conclusion: (1) there does not appear to be any features specific to a particular plant, class of plants, or containment type which would mitigate or prevent the generation, transport to the suppression pool, or deposition on the ECCS strainers of sufficient material to clog the strainers, (2) parametric analyses performed in support of the NUREG/CR-6224 study using parameter ranges which bound most domestic BWRs failed to find parameter ranges which would prevent BWRs with other containment types from being susceptible to this problem, and (3) the NUREG/CR-6224 study was conducted on a Mark I; Barsebäck had a strainer clogging event and is similar in design to a Mark II; and Perry, a Mark III, has also had strainer clogging events.

The staff concluded that there are three basic options for resolving this issue, although licensees are free to propose alternate solutions. NRC Bulletin 95-XX requested licensees to: (1) install a large capacity passive strainer design of sufficient size that the worst expected debris loading would not cause the ECCS pumps to lose NPSH, (2) install a self-cleaning strainer design which continuously cleans the strainer surface and is capable of withstanding the worst case debris concentrations in the suppression pool, or (3) install a backflush system which can be actuated to clean the strainer surface and prevent the ECCS pumps from losing net positive suction head. Each of these options would also require supplementary actions by the licensee in order to comply with the intent of 10 CFR 50.46. For instance, a large capacity passive strainer would require actions by the licensee to ensure that the potential debris loading of the strainer never exceeds its design capacity. An example of a measure that may be required to ensure this is that a licensee may have to clean their suppression pool on a regular basis in order to ensure that sludge concentrations in the pool do not exceed the design basis of the strainer.

The CSNI/PWG-1 International Task Group on ECCS Reliability

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On July 28, 1992 a small steam line LOCA in the Barsebäck 2 reactor occurred when a safety relief valve inadvertently opened. The steam jet stripped the adjacent pipework from fibrous insulation. A part of the insulation was transported to the wetwell pool and clogged the intake strainers for the drywell spray system after about one hour. Although the incident in itself was not very serious it revealed a weakness in the defence-in-depth concept which under other circumstances could have led to failure of the emergency core cooling to provide water to the core. Moreover, the weakness had been realized earlier and facilities to clean the strainers during operation had been provided.

As in many other countries, applicable parts of the regulatory guide on strainer requirements for PWR, issued in 1974 by the USNRC, had also been used for judgement of ECCS performance of Swedish plants. However, data indicated that the guide was not sufficient, and national experimental programs were carried out in the late seventies to determine the performance of the strainers. It was found that strainer clogging, if occurring at all, would at least not occur during the first ten hours after a LOCA. Since operation of the emergency core cooling equipment would be needed for a long time, backflushing capabilities and monitors of pressure drop were installed in older Swedish BWR plants with small strainer area. Because of the backflushing capability compliance with the revision of the regulatory guide issued in 1985 was anticipated. The question of strainer clogging was considered to have been resolved until the incident happened in Barsebäck, which showed that clogging could occur significantly earlier than the expected 10 hours.

SKI required that measures to prevent strainer clogging should be taken for the five oldest BWRs which had strainers of small area, before they were allowed to start again. This led to intensive research and development projects. It was first identified that the earlier obtained data on strainer clogging had been misleading since parameters which were essential for clogging had been neglected. For example, in the old experiments new rockwool which had been mechanically cut to pieces had been tested. Such material first floats on the water surface and thereafter sinks rapidly to the bottom. The new experiments using debris which was removed from the reactors or aged by temperature, showed that the material tended to remain suspended in the water and thus available for strainer clogging. Also material which was fragmented by steam jets produced much higher head losses. The observations at Barsebäck and also experiments with steam jets indicated that the destruction zone could be more extended than anticipated.

In order to provide feedback of the Swedish experience, a workshop on the strainer clogging issue was held in Stockholm on January 25-26, 1994, under the auspices of CSNI/PWG-1. Also international experience and actions were presented. The objectives of the workshop were to give an overview of decisions and work performed on the issue, to address the actual safety issues with regard to the reliability of ECC recirculation, and to discuss further actions needed. The workshop revealed a rather confusing picture of the knowledge base and also examples of conflicting information. It was therefore decided to form a working group with the specific objective to establish an internationally agreed knowledge base for assessing the reliability of emergency core cooling water recirculation systems.

The specific tasks given to the group were:

Critical review and compilation of available experimental and other data related to the performance of ECC water recirculation systems, including formation and behavior of various types of debris contaminating the water.

Assessment of the applicability of the data base. Identification of major uncertainties, lacking information, and data.

Proposal for additional research and experiments together with pointing out those uncertainties which should rather be accommodated in terms of conservative design features.

The work of the group was divided on five major areas: Debris generation, Drywell transport, Suppression pool transport, Strainer pressure drop, and Other effects.

For the debris generation the group concludes that plant specific studies are needed. In addition to the amount of dislodged material also the character of the dislodged material is important since, for example, mixtures of particles and fibrous material generally will produce larger pressure drop than pure materials. Currently used models for evaluation of amount of dislodged material seem to be mostly applicable to flashing water. Steam jets produce destruction zones which are much narrower and much longer as compared to jets of flashing water.

Debris is transported through drywell by blast forces, blowdown forces and by washdown. Plant specific analyses are needed to determine the retention in drywell. The uncertainty is high, and experiments performed have shown retention factors which are much higher than, for instance, observed in Barsebäck. It is therefore difficult to draw conclusions of high confidence which are necessary for safety assessment from these experiments.

Debris transport in the wetwell pool is controlled by sedimentation and resuspension which are dependent on parameters like character of the debris and turbulence. Fibrous material which undergoes loads from condensation in the wetwell pool could be further fragmented and thereby remain suspended in the water.

The pressure drop over a debris bed is dependent on parameters like composition of the material and size distribution, thickness, velocity of approach and temperature. It is in general possible to reasonably well predict the behavior of pure materials. The methodology for pressure drop prediction over beds of mixtures of fibrous and particulate debris has large uncertainties. Caution has to be exercised when testing different types of debris so that the test and characteristics of the materials are representative.

The report concludes that also other effects like vent path clogging, missiles from encapsulated insulation debris and strainer penetration should be considered on a plant specific basis. The investigations on these areas have not been systematic and the uncertainties are large.

EXPERIMENTS OF ECCS STRAINER BLOCKAGE AND DEBRIS SETTLING IN SUPPRESSION POOLS

by George E. Hecker, Alan B. Johnson, Prahlad Murthy, and M. Padmanabhan
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If a rupture occurs in a nuclear power station pipe that leads to or from the reactor pressure vessel, the resultant Loss of Coolant Accident (LOCA) would initiate a chain of events which involving complex flow phenomena. In a Boiling Water Reactor (BWR), the steam or liquid pipe break pressurizes the dry well, forcing the inert containment gases and steam through downcomers into the suppression pool, thoroughly mixing any particulates and pipe insulation debris carried with the gas flow to the pool. As the steam flow decreases, its unsteady condensation at the end of the downcomers ("Condensation Oscillation" and "Chugging") produces continued water motion in the suppression pool and downcomers. During the "blowdown" event, high pressure and then low pressure pumps automatically start injecting water from the suppression pool into the reactor to keep its temperature under control. Proper functioning of this Emergency Core Cooling System (ECCS) is critical for the first 30 minutes or so, before operators have time to consider and align alternative sources of cooling water.

A major concern for proper operation of the ECCS is the effects of fragmented insulation and plant particulates (oxidation products, concrete dust, paint chips, dirt, etc.) on the head loss at pump suction strainers. Sufficient loss could exceed the NPSH margin, causing cavitation with a resultant loss of pump capacity and longevity. The head loss increases with the mass of debris accumulated on the pump strainers (among other things), which in turn is dependent on the debris concentration versus time in the suppression pool.

This paper describes two sets of experiments to quantify the strainer head loss, the resulting data being used as input to a computer code (developed by Science & Engineering Associates, Inc.) to predict strainer head loss versus time in the actual plant. One set of experiments considered the mixing and settling of fibrous insulation debris and fine iron oxide particles in the suppression pool during and after chugging. These tests used a reduced scale facility which duplicated the kinetic energy per unit water volume to define the concentration of the actual materials in the pool versus time. Such data allows calculation of the mass of debris on the strainer versus time.

The second set of experiments measured the head loss across a representative strainer plate for a range of masses of fibrous insulation debris and iron oxide particles. Other parameters that influenced the head loss were the approach flow velocity and the water temperature. To allow the mass of oxide particles on the strainer plate to be estimated from the pool concentrations, filtering efficiencies (trapping of the iron oxide particles by the fiber bed) were also measured.

The knowledge gained from these experiments allows the expected head loss versus time in the actual plants to be calculated with greater confidence.

The Strainer Blockage Assessment Methodology Used in the BLOCKAGE Code

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On July 28, 1992, a spurious opening of a safety valve at Barsebäck Unit 2, a Swedish BWR, resulted in the clogging within one hour of the ECCS pump strainers. Instances of clogging of ECCS pump strainers have also occurred at US BWRs. Given these precursor events, NRC staff initiated analyses to estimate potential for loss of NPSH of the ECCS pumps in a BWR due to clogging of suction strainers by a combination of fibrous and particulate material. A BWR/4 with a Mark 1 containment was selected as the reference plant for a parametric study of BWR strainer blockage due to LOCA generated debris documented in NUREG/CR-6224.

A strainer blockage assessment methodology was developed as part of the NUREG/CR-6224 and codified in a computer code named BLOCKAGE. Important aspects of the methodology include:

- The development of a three region spherical debris generation model (DGM) which accounts for the lower operating pressures of BWRs and the congested layout of BWR drywells. Destruction factors are assigned to each region to account for the degree of destruction caused by the overpressure pressure pulse and the blowdown following a double ended guillotine break.

- The development of a time dependent drywell transport model to estimate the fraction of fibrous and particulate debris reaching the suppression pool.

- The development of a time dependent suppression pool model to estimate the type and volume of fibrous or particulate debris reaching the strainer. The model accounts for:

  a) re-suspension of particulates contained in the bottom of the suppression pool at the time of the DEGB,
  b) gravitational sedimentation (or settling) of the particulate and fibrous debris, and continued deposition of fibrous and particulate debris on the strainer.

- The development of a head loss model to estimate the pressure drop across the strainer due to fibrous and particulate debris bed build-up.

The models developed were integrated into a single strainer blockage computer code which was used in NUREG/CR-6224 to evaluate whether or not a pipe break at each of the welds located in the primary system piping and the main steam lines of the reference plant resulted in a head loss larger than the available ECCS NPSH margin. This paper describes the methodology codified in the BLOCKAGE code, as well as highlights of the code user interface and outputs.

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

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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-0148

2. TITLE AND SUBTITLE

Transactions of the Twenty-Third Water Reactor Safety Information Meeting

3. DATE REPORT PUBLISHED

MONTH	YEAR
September	1995

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 res.

7. PERIOD COVERED (Inclusive Dates)

October 23-25, 1995

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

C. Bonsby, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report contains summaries of papers on reactor safety research to be presented at the 23rd Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel, Bethesda, Maryland, October 23-25, 1995. 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.)*

reactor safety research
nuclear safety research

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page) Unclassified

(This Report) Unclassified

15. NUMBER OF PAGES

16. PRICE

TWENTY-THIRD WATER REACTOR SAFETY INFORMATION MEETING

Bethesda Marriott Hotel, Bethesda, Maryland

October 23-25, 1995

SESSION SCHEDULE

PLENARY SESSION in Congressional Ballroom
Opening Remarks: Shirley Ann Jackson, Chairman, NRC
Panel Discussion: "Current Industry Issues and Their Relation to NRC Research"
Moderator: D.L. Morrison, Director, RES
W.T. Russell, Director, NRR;
R.A. Fenech, VP Nuclear Operations, Palisades Plant;
and J.T. Beckham, VP Hatch Project

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Mon. PM	3 Advanced I&C Hardware & Software C. Antonescu	4 High Burnup Fuel Behavior R. Meyer
Tues. AM	5 Severe Accident Research I C. Tinkler	6 Primary Systems Integrity M. Mayfield
Tues. PM	7 Severe Accident Research II A. Rubin	8 Equipment Operability & Aging J. Vora
Wed. AM	9 Thermal Hydraulic Research W. Hodges	10 Probabilistic Risk Assessment Topics M. Cunningham
Wed. PM	11 Individual Plant Examination T. Su	12 ECCS Strainer Blockage Research and Regulatory Issues C. Serpan

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