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Transactions of the Twenty-Eighth Water Reactor Safety Information Meeting

To Be Held at
Bethesda Marriott Hotel
Bethesda, Maryland
October 23-25, 2000

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

Proceedings prepared by
Brookhaven National Laboratory

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PREFACE

This report contains summaries of papers on reactor safety research to be presented at the 28th Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel in Bethesda, Maryland, October 23-25, 2000. They briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, U.S. NRC. Also included are summaries of invited papers concerning nuclear safety issues from U.S. government laboratories, the electric utilities, the nuclear industry, and from foreign governments and industry. The abstracts have been compiled here to provide a basis for meaningful discussion of information exchanged during the course of the meeting, and are in the order of their presentation on each day of the meeting.

An asterisk [*] in place of a page number in the Contents indicates no submission in time for inclusion in these Transactions.

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

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Rapporteur: M. Mayfield

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Practical Applications of PRA to Improve Plant Regulation

James W. Johnson
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Introduction/Background:

The NRC Strategic Plan identifies four performance goals that support the achievement of the overall agency strategic goal of preventing radiation-related deaths and illnesses, promoting the common defense and security, and protecting the environment in the use of civilian nuclear reactors. These performance goals are: (1) maintain safety, protection of the environment, and the common defense and security, (2) increase public confidence, (3) make NRC activities and decisions more effective, efficient, and realistic, and (4) reduce unnecessary regulatory burden on stakeholders. Risk-informed approaches can be useful in their implementing strategies. In addition, NRC is working to transition to a risk-informed regulatory structure. Risk-informed approaches are often undergirded by Probabilistic Risk Analysis (PRA). The development of PRAs requires detailed modeling of complex phenomena and consideration of uncertainties.

Discussion of Current Issues:

The NRC is embarking upon a major initiative to make the transition from the current deterministic based regulations to risk-informed regulations. The current set of deterministic requirements was developed with no explicit consideration of quantitative risk assessment. Recent successes with risk analysis results suggest that risk-informed regulation can contribute to maintaining safety while reducing unnecessary regulatory burden. Risk-informed regulation can also support NRC efforts to make its activities more effective, efficient and realistic. The transition from the current set of regulatory requirements to risk-informed regulations will disclose a number of technical and policy issues and will require several years to accomplish. This session will focus on two major questions: (1) How is regulatory decision-making impacted by uncertain PRA results? and (2) What are the key technical impediments to moving to risk-informed regulation?

The NRC current activities are being focused on risk-informing Part 50. A draft framework document has been developed to guide the process of risk-informing Part 50. The framework has three basic steps: (1) select Regulatory Requirements to be risk-Informed, (2) develop risk-informed options, and (3) evaluate options. The framework includes quantitative objectives and guidelines for risk-informing existing technical requirements. The intent is to develop risk-informed regulations, which retain deterministic characteristics, in such a way that compliance provides reasonable assurance that the public health and safety is protected. The framework will be discussed as well as its application to 10 CFR 50.44. Technical issues will be identified and discussed. The treatment of uncertainty in PRA results will be highlighted as well as the role of the defense-in-depth concept in a risk-informed regulatory structure. Industry perspectives on the use of PRA results in the regulatory arena will also be highlighted and discussed.

Future Activities:

The NRC will draw on the experiences gleaned from its efforts to risk-inform 10 CFR 50.44 as it prepares to broaden the scope of its activities to risk-inform Part 50. The next major application will be risk-informing 10 CFR 50.46. The effort will likely take several years to complete. Close interaction with the industry and public at each step of the process is planned. Interactions with ASME, industry, and the public to develop PRA standards for broader application of PRA results will continue. The availability of a standard should reduce NRC review time and offer more definitive requirements to ensure quality in PRAs.

Risk-Informing Technical Requirements

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Introduction

The United States Nuclear Regulatory Commission (NRC) has made use of probabilistic risk analysis (PRA) information for many years. A key milestone in this use was the issuance of the Commission's 1995 PRA Policy Statement, which indicated that: "the use of PRA technology should be increased in all regulatory matters ...in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy."

Using this guidance, the NRC staff is now working to modify its basic nuclear reactor safety regulations, contained in 10CFR50, to make these regulations impose regulatory burdens on licensees that are commensurate with their safety importance. One part of this work involves making changes to specific requirements in the body of regulations. As discussed below, the staff is now studying the Part 50 technical requirements to identify areas of unnecessary conservatism and needed additional safety requirements.

The staff's work to risk-inform the Part 50 technical requirements is, of course, dependent on the quality of PRA information being used by NRC and its licensees. While today's PRA library of information is useful for many applications, there remain technical impediments and uncertainties which, until overcome, constrain the potential uses of this information. Two of these impediments - the lack of PRA standards and the gaps in PRA technology - have been the subject of considerable work in the past several years, and are discussed further below.

Risk-Informed 10CFR50 Technical Requirements

In one part of its program to risk-inform 10CFR50, the staff is studying the Part 50 technical requirements to identify areas of unnecessary conservatism and potential additional safety requirements. The staff has developed, and is now using, a general framework for identifying and prioritizing potential changes. An early result of this work was the identification of potentially valuable changes to requirements contained in 10CFR50.44 ("Standards for combustible gas control system in light-water-cooled power reactors"). The identified changes offer the opportunity to both improve plant safety and reduce licensee burden. The staff expects to make recommendations on such changes to the Commission in August 2000. In addition, the staff is evaluating the potential value of changing the requirements contained in 10CFR50.46 ("Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors"), and modifying and consolidating the 10CFR50 requirements for "special treatment" of important systems, structures, and components.

The staff is continuing to use the framework to identify other potentially important changes to 10CFR50. It expects to have an initial set of recommendations on such changes (in addition to that for 10CFR50.44, and including potential changes to 10CFR50.46 and special treatment requirements) in late 2000.

Role of PRA Standards

In any regulatory decision making process, the goal is to make a sound safety decision based on technically defensible information. Therefore, for situations where a regulatory decision relies upon risk perspectives as one source of information, there needs to be confidence in the PRA results from which the perspectives are derived. Consequently, the PRA needs to have the proper scope and technical attributes to give an appropriate level of confidence.

Consensus PRA standards can be used to define the needed scope and technical attributes, and an industry peer review program can provide an assessment of the weaknesses of a PRA. Such standards have been under development for the last few years by the American Society of Mechanical Engineers (ASME), the American Nuclear Society (ANS), and the National Fire Protection Association (NFPA). Industry peer review programs have also been undergoing development during this time. The staff is now reviewing, or expects to soon review, industry peer review programs, and the ASME and ANS PRA standards, as well as the PRA portion of the NFPA fire protection standard, in this light. To support this review, the staff is developing acceptance criteria for the technical requirements and peer review process.

Filling Gaps in PRA Technology

As noted above, one important impediment to the greater use of risk information in regulatory decision making is the existence of gaps in currently available PRA methods and data. One function of NRC's research program in PRA is to identify such gaps and perform research to fill them. More specifically, the staff now has work underway to improve:

- Human reliability analysis methods. It has been accepted for some time that failures in human performance are one of the principal sources of risk. Although techniques have been used in the past to quantify the likelihood of both pre-accident and post-accident human error, one of the remaining questions is how to treat "errors of commission." This question has been the subject of recent NRC and international work.

- Fire risk analysis methods. Experience from major fire events around the world has shown that serious fires can pose an important challenge to nuclear safety. Moreover, fire PRAs show that fire-initiated accident scenarios can be significant contributors to the calculated risk at many U.S. plants. Because the results of fire PRAs are subject to considerable uncertainty, the NRC has undertaken a research program to improve fire PRA methods and data.

- Treatment of aging effects in PRAs. The staff is exploring how the use of mechanistic models of failure can be better integrated into PRAs. These models appear to be especially useful when addressing the effects of aging on risk. The staff has recently completed a feasibility analysis focused on one aging mechanism: the flow accelerated corrosion of piping. This analysis demonstrates a relatively simple approach for addressing aging within the structure of current PRAs. It also leads to a number of questions concerning the treatment of various plant activities (e.g., inspection) and the quantification of uncertainties.

Use of PRA Results in Regulatory Decision-Making

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In making any risk-informed regulatory decision, the staff's goal is to ensure that an appropriate level of safety is maintained, and, in particular, if any risk increases do occur as a result of a design, operational, or regulatory change, they are small and consistent with the Commission's Safety Goal Policy Statement. Practical guidance for the use of PRA in risk-informed decisions on plant specific changes to the licensing basis was provided in Regulatory Guide 1.174 and the associated Standard Review Plan Chapter 19. The approach described in those documents was developed taking into account the state of the art in risk assessment technology, and in particular reflected the need to address explicitly the uncertainties associated with PRA results. It also provided guidance on the quality required of a PRA. Most of what was developed there can be taken over to address regulatory decision-making in a more general sense. This paper discusses first how the staff intends to address the uncertainties in PRA results during its decision-making, and second how it intends to address the issue of variability in the quality of PRA results.

As discussed in Reg Guide 1.174, a PRA provides only one part of the information used to make such a decision. Typically the PRA results will be used to provide a means to assess that "proposed increases in risk, and their cumulative effect, are small and do not cause the NRC Safety Goals to be exceeded". This is done by comparison of an evaluation of the change in CDF (core damage frequency) and LERF (large early release frequency) with corresponding acceptance guidelines. The role of the uncertainty analysis is to give a measure of confidence that the acceptance guidelines have indeed been met. The approach adopted in Reg Guide 1.174 is that the impact of parameter uncertainties and those model uncertainty that are explicitly addressed in the structure of the PRA logic model be addressed by generating mean values for comparison with the acceptance guidelines, and, for other model uncertainties, that sensitivity analyses be performed to demonstrate that the decision would not be changed for reasonable alternate model hypotheses. This approach has the advantage that those alternate models that can influence the decision can be identified and whether or not the acceptance guidelines are met can be assessed on the merits of the alternate models.

Since the importance of the PRA results will vary from decision to decision, the quality of the PRA must be judged in the context of the decision-making process, and on the way the PRA results are used to justify the decision. The quality of the PRA, coupled with an understanding of the sources of uncertainty and how they impact the results is what determines the confidence we can have in the results it generates. The less confident the decision-maker is in the results, the more he has to rely on compensatory measures to ensure that safety is maintained. These measures include an increased reliance on the more traditional approaches such as relying on defense-in-depth or adequate safety margins, which will restrict the degree of implementation of the application. Another approach is to institute performance monitoring to make sure that any plant changes do not result in unexpected degradation of performance. There will, therefore, generally be a trade-off between the benefit to be obtained from the application, in terms of relaxation of requirements for example, and the quality of the risk information. The better the quality of PRA information, the more benefit that should be expected.

TRANSITION TO RISK-INFORMED REGULATION

Robert A. Bari
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SUMMARY

Laboratory initiatives and contributions in support of the transition to risk-informed regulation are presented and discussed. Key accomplishments are noted and their impacts on current approaches to regulatory activities are summarized. Particular attention is given to how regulatory decision-making is impacted by uncertainty in PRA results and to the key technical impediments to moving to risk-informed regulation. Some challenges are presented for the research community in facilitating the practical applications of probabilistic risk assessment in the regulatory area.

Industry Perspectives on the Role of PRA

Steve Floyd
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The stated purpose of the paper is to address the following issues:

1. How is regulatory decision-making impacted by uncertain PRA results?
2. What are the key technical impediments to moving to risk-informed regulation?

With regard to the first item, it is important to remember that all regulatory approaches, whether deterministic or probabilistic, contain inherent uncertainties. Because of its quantitative nature, PRA tends to elucidate uncertainties in a more direct manner than deterministic methods. Regulatory decision-making involving PRA is generally impacted by uncertainties when the regulatory application involves relaxation of some current deterministic requirement. The rationalist approach to defense in depth argues that DID provisions should compensate for those areas where uncertainties are significant to the application. For most applications, the result is inclusion of some conservatism in the final form of the approved application. For example, risk insights alone would justify elimination of ASME Section XI in-service inspection requirements for reactor coolant system piping welds, but due to uncertainties in failure frequencies, and concern with heretofore unknown failure mechanisms, a proportion of the current inspection requirements is retained. Uncertainties are generally less of an issue when the PRA insights are used to establish new requirements.

With regard to the second issue, the most significant impediments appear to be cultural rather than technical. There is a large degree of comfort with the current regulatory scheme, which is effective if not efficient. Acceptance of change is difficult, and PRA is complex enough to introduce many academic and technical considerations that are not obvious in the deterministic framework. Because it is not limited to design basis considerations, PRA must address phenomenological and technical issues for which experience and information are limited or nonexistent. Human reliability, fire modeling and propagation, and shutdown risk are some of the more challenging technical areas. The use of risk-informed versus risk-based approaches is paramount to successful application of PRA, as a perfect state of knowledge cannot be achieved.

Dry Cask Storage and Transportation of Spent Nuclear Fuel

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Introduction/Background

To support the Office of Nuclear Materials Safety and Safeguards/Spent Fuel Project Office (NMSS/SFPO), the Office of Nuclear Regulatory Research has developed a research program to investigate a range of technical issues concerning the dry storage and transportation of spent nuclear fuel. Several tasks of this research program (Tasks 1, 2, & 3) will provide data to augment the technical basis for the renewal of license and certificates of compliance for the dry storage systems for spent nuclear fuel and high level radioactive waste at independent spent-fuel storage installations. Task 4, the development a probabilistic risk assessment (PRA) of dry storage of spent nuclear fuel, is needed to comply with the Commission directives to develop safety goals and to risk-inform 10 CFR Parts 71 & 72, as well as to gain public confidence in shipment and dry storage of spent nuclear fuel. Task 5, the Package Performance Study (PPS), will reevaluation of the level of protection provided by NRC certified spent fuel transportation designs under accident conditions.

Discussion of Current Activities

Task 1. Characterization of Condition and Material Behavior of Spent Nuclear Fuel (< 45GWd/MTu) and Dry Cask Components

The objectives of this research are: (1) to determine the long-term integrity of dry cask storage systems and of spent nuclear fuel under dry storage conditions; and (2) to provide data to augment the technical bases and criteria for evaluating the safety of spent fuel storage and for extending dry cask storage license. To accomplish these objectives, a Castor - V/21 steel cask, containing spent fuel that has been in dry storage for about 15 years, and its contents have undergone a detailed visual examination; also fuel rods have been removed from one assembly and will be subjected to testing to ascertain their physical properties. The NRC is soliciting interest in performing a similar study of a VSC-17 concrete cask and its contents. An additional study under this task is to determine if a possible zinc-zircaloy interaction takes place under conditions representative of those experienced in the storage of spent nuclear fuel in a dry cask. A zinc-zircaloy interaction forms brittle intermetallics, which can degrade the mechanical properties of the spent fuel rods.

Task 2. Source Term Issues - Development of Criticality Safety Technology for Licensing Review.

This research covers three aspects: (1) revision of an existing software package to enable independent processing of cross-section evaluations for use in criticality safety reviews utilizing state-of-the-art procedures, (2) development of the technical basis for guidance related to the licensing review of spent nuclear fuel storage and transportation systems that use burnup credit (i.e., reactivity loss due to burnup) in the criticality safety analysis, and (3) investigation of the adequacy of predicted source terms for high-burnup spent nuclear fuel and development of guidelines on bounding values and/or appropriate models and analysis methods that can be used.

Task 3. Seismic Capacity of Dry Storage Casks to Tipping & Sliding

The objective of this project is to provide the technical basis for the evaluation of the seismic safety of a dry cask storage system. The issues of concern include: how the casks behave seismically; how much sliding and tipping are likely to take place; how the casks impact each other; and how the internals would be affected seismically. The first phase of this effort entailed the collection and review of information and previous analyses, the evaluation of relevant study parameters, and the identification of areas requiring further investigation. The second phase includes generic and plant-specific analyses to develop easy-to-use tools, such as tables and nomograms, which characterize limiting conditions for parameters important to the seismic stability of dry cask systems. The third phase, if warranted will include necessary testing to validate the analyses performed in Phase II.

Task 4. PRA of Dry Storage and Transportation of Spent Nuclear Fuel

The objective of this task is to develop a PRA of the HI-STORM dry storage cask from which the SFPO can assess various options for developing safety goals and for risk-informing 10 CFR Parts 71 and 72, as well as risk-informing both the prioritization and inspection programs. The PRA aspects of the PPS, mentioned below in Task 5, will be integrated into this effort.

Task 5. PPS - Study of Spent Nuclear Fuel Cask Response to Severe Transportation Accidents - Phase 1

Phase I of the PPS is a scoping study; the objectives of this phase are to: (1) evaluate the need to revisit the conclusions of the 1987 modal study, (2) identify possible follow-on research, and (3) provide additional information to enhance public confidence in spent nuclear fuel transport. The first two objectives were completed by SNL as a contractor to NRC, the third objective was accomplished through a series of public workshops/meetings at which the NRC staff explained the purpose of the PPS and sought public input on their concerns on the transportation of spent nuclear fuels. It is anticipated that a conclusion of Phase I will be a recommendation to proceed with Phases II thru IV of the PPS. See Future Activities below for Phases II thru IV.

<u>Future Activities</u>

Task 1. Material Properties & Stability of Cask & Contents for Fuel > 45GWd/MTu

A cask demonstration project for high burnup spent fuel is planned; the inspection of the cask and its contents will be similar to the study described above in Task 1 for spent fuel with a burnup < 45GWd/MTu.

Task 2. PPS Phase II thru IV

Phase II corresponds to development of an experimental and analysis plans for: (1) an impact test, (2) a fire test and (3) a PRA for severe transportation accident scenarios; Phase III is the implementation of the plan; and Phase IV is the documenting and reporting the results of the tests and the analyses.

INSPECTION OF THE CASTOR-V/21 CASK AND CONTENTS

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Most nuclear power plants in the United States were not originally designed with storage capacity for the spent fuel generated over their operating life. Utilities have developed independent spent fuel storage installations as a means of expanding their spent fuel storage capacity until the geologic repository is available to accept spent fuel for permanent storage. The U.S. Nuclear Regulatory Commission promulgated Part 72 for the independent storage of spent nuclear fuel and high-level radioactive waste outside reactor spent fuel pools. The license term for an independent spent fuel storage installation must not exceed 20 years, and a few licenses are approaching this time limit. In preparation for possible license renewal, the NRC is developing the technical basis for extended storage in existing sites. Verification of past performance of selected components of these systems is required as part of that technical basis.

Under a demonstration program that has been underway since the mid-1980's, the Department of Energy has managed some quantities of commercial spent nuclear fuel in four dry storage casks at the Idaho National Engineering and Environmental Laboratory, Test Area North facilities in Idaho. The NRC, Electric Power Research Institute, and the Department of Energy have a mutual interest in performing research on dry cask storage characterization. The objectives of this cooperative research program are to (1) obtain confirmation of the predicted long-term integrity of dry cask storage systems and spent nuclear fuel under dry storage conditions, and (2) provide data to augment the technical bases and criteria for evaluating the safety of spent-fuel storage and for extending dry cask storage licenses.

The results from the visual inspections performed on the Gesellschaft fuer Nuklear Service Castor-V/21 cask exterior and interior, and the stored fuel assemblies are described. The Castor-V/21 is a nodular cast-iron cask containing spent fuel assemblies from the Surry nuclear power plant. The fuel assemblies have been out-of-reactor for approximately 20 years, and in this cask for approximately 15 years. Several rods have been withdrawn from one of the fuel assemblies and will be subjected to detailed non-destructive, destructive, and mechanical examinations to provide quantitative and qualitative information concerning the integrity of the fuel. Because this cask has been in use for such a substantial amount of time, information about the current condition of the cask and the contained fuel would be of great potential use in establishing a technical basis for dry cask storage system license renewal.

Keywords: dry cask storage, dry cask license renewal, spent nuclear fuel examinations, Castor-V/21 cask inspection, Surry spent fuel examinations

Research Supporting Implementation of Burnup Credit in the Criticality Safety Assessment of Transport and Storage Casks

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The Nuclear Regulatory Commission (NRC) Office of Regulatory Research (RES) initiated a program to support effective implementation of burnup credit in the criticality safety assessment of transport and dry storage casks. The goal is to develop technical bases that can be used to provide criteria and guidance for use in licensing activities. The majority of the technical work is being performed at Oak Ridge National Laboratory under contract with NRC/RES. The program is being conducted in a phased approach with the initial focus on unresolved issues related to use of actinide-only burnup credit in PWR transport and dry storage casks. The work will gradually expand to investigate credit for fission products in PWR casks, application to BWR casks, and application to long-term disposal. This summary will review the status of the progress to date and identify planned activities and priorities.

The NRC/RES project provided the NRC Spent Fuel Project Office (SFPO) with confirmatory research that supported the issuance of the Interim Staff Guidance - 8 (ISG-8)[1] which provides recommendations for use of burnup credit with PWR spent fuel in transport and dry storage casks. Since that time the RES program has been working to develop expanded guidance relative to selected elements of the ISG-8, implement software enhancements that can facilitate computational analyses, and develop the technical basis for the SFPO to use in considering revision of ISG-8 to allow added flexibility and/or expanded applicability. A baseline report[2] was developed to review the status of burnup credit and provide a strawman prioritization for areas where additional guidance, information, and/or improved understanding were judged to be beneficial to effective implementation of burnup credit in transport and dry storage casks.

Work is near completion on a reference report that uses current cask designs (rail and truck) to provide a consistent basis for demonstrating the magnitude of the various negative reactivity components as a function of burnup, initial enrichment, and cooling time. Similarly a computational benchmark to help licensees calibrate the analysis of fission product margin for their cask design (as recommended by ISG-8) has been recently drafted. Technical guidance that is intended as a supplement or reference for a standard review plan will be prepared in 2001. An automated process for coupling the depletion/decay process to the criticality analysis has been developed to support initial license reviews. Eventually the analysis tool will be released as a module of the SCALE code system.[3] Initial recommendations and associated technical basis for potential near-term modifications to the ISG-8 have been developed in the following areas: use of cooling times other than 5 y and allowance for use of burnup credit with PWR fuel containing burnable poison rods and/or integral burnable absorbers. An approach for modification or removal of the loading offset (the added burnup margin required for fuel with initial enrichments above 4.0 wt%) has also been proposed an work is proceeding to develop the technical justification. The issues related to selection of the appropriate axial profile for use in the safety assessment being explored in order to develop criteria and/or recommendations that are technically credible, practical, and cost effective while maintaining needed safety margins.

The NRC research program is working to obtain input from domestic and international experts and organizations with experience in burnup credit research, experiments, criticality safety practice, and operations of transport and dry storage casks. One primary tool for this input is the expert panel convened to participate in a process of developing Phenomena Identification and Ranking Tables (PIRT). The main goal of the PIRT panel is to identify phenomena, parameters, procedures, etc. that influence the determination of k-eff for spent fuel in a cask environment, provide a graded (e.g., high importance, moderate importance, low importance) ranking of the phenomena and, as appropriate, judge the uncertainty associated with each phenomena. Besides its primary objective, the PIRT process can also facilitate a beneficial exchange of information and ideas that will hopefully lead to improved understanding of the issues and practical approaches for effective implementation of burnup credit within the licensing process. The progress of the PIRT panel can be followed by reviewing the following web site: www.nrc.gov/RES/pirt/BUC).

Another important facet of the NRC/RES program is the identification and assessment of past, planned, and potential experiments that can support improved understanding and/or implementation of burnup credit. Currently the NRC/RES is actively participating in the REBUS experimental program[4] and is discussing with the French the various avenues available for potential use of portions of their experimental data. To assist in this assessment, sensitivity/uncertainty (S/U) methods discussed in Ref. 5 are being used to provide information on the strengths and potential limitations of various types of experiments relative to validation needs for burnup credit. Existing fresh fuel (UO_2-fuel and mixed-oxide) critical experiments, reactor critical configurations, reactivity worth experiments, and measured chemical assay data are being studied with the S/U prototypic methods. Initial results of this assessment are being documented and will be reported in the presentation.

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High Burnup Fuel

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In the introductory presentation last year, we described deficiencies in some regulatory criteria with regard to fuel irradiation up to the current limit of 62 GWd/t burnup (average in the peak rod) and we identified NRC research efforts to address related issues.[1] The issues were related to fuel damage limits and models used in analyzing (1) postulated rod-ejection accidents in PWRs, (2) power oscillations without scram in BWRs, and (3) loss-of-coolant accidents (LOCAs) in both types of plants.[2] The goal of resolving these issues is to maintain safety margins that exist for low-burnup fuel. During the past year, there have been a couple of new developments that add to these issues, and there has been some significant progress in programs that are already underway. First, we have become aware of a rather urgent need for data on fuel cladding behavior under dry storage conditions to support licensing actions for spent fuel. We are adding such work to our ongoing program at Argonne National Laboratory and we are trying to keep abreast of similar work abroad. One such paper from a German program will be presented in this session. Second, we issued a new regulatory guide on radiological source terms (R.G. 1.183), yet we had considerable difficulty determining high-burnup releases of short-lived isotopes during normal operation for analysis of a fuel handling accident. Another international paper in this session, from the Halden Project, will address work that is going on to improve an industry standard such that high-burnup releases of radiological fission product species can be calculated.

We are just completing an activity to develop phenomenon identification and ranking tables (PIRTs) for the three accident types mentioned above. The staff has drawn insights from those PIRTs and used them to plot a course for resolving the issues. A draft NUREG report describing the PIRT results and draft staff papers describing insights and actions can be seen on our web site at www.nrc.gov/RES/PIRT. We do not have enough time in this session for a presentation on that material. Another advanced effort that we cannot present in this session because of time limitations is on the FRAPTRAN transient fuel rod code. Modification and assessment have recently been completed and the code is undergoing peer review at this time. Information on this code is available on posters set up in this room, and you can talk with the code developers during breaks. The information on FRAPTRAN will be included as a paper in the Transactions and Proceedings for this meeting.

Work in France and Japan to address reactivity accidents is progressing well. The Cabri Water Loop project in France is being launched this year in cooperation with OECD. A brief status report on that international program will be given. In the NSRR test reactor in Japan, new data have shown rather surprising effects of fuel-to-cladding bonding at high burnup on the behavior of BWR cladding. That work will be presented.

In our own program at Argonne National Laboratory, which is addressing LOCA and dry storage issues, we have recently received high-burnup fuel rods from a U.S. power plant and testing is well underway. That work will be presented. A related paper on the history of LOCA embrittlement criteria is being presented also because that review revealed a very important principle, which many of us had forgotten, and this has led to an interesting modification of the program.

Each presenter in this session was asked the general question: What kinds of research are needed from the fuel research program to ensure that safety margins for high-burnup fuel are maintained? In these presentations, you will be able to see how the research being performed will help NRC maintain safety margins in areas related to that work.

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Fission Gas Release Measurements in Relation to ANS Standard Modelling of Radiological Releases

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Summary

Early data on fission gas release (FGR) were obtained solely from Post Irradiation Examination (PIE), when discharged fuel rods were punctured and the internal gas extracted and analyzed by techniques such as mass spectrometry. PIE is extremely useful when applied to commercially irradiated fuel, as it provides data for a prototypic irradiation and fuel loading strategy. However, it is of only limited use when required for developing an understanding of the mechanisms and factors involved in the FGR process. This gap was filled by the development of in-pile instrumentation to provide on-line information from which the kinetics of release can be determined.

Two types of experiments have been used in the Halden reactor to investigate the release of fission gases from LWR fuel. The first employs internal pressure sensors from which the kinetics and quantity of stable gases can be measured during irradiation. The second is the use of sweep gases to carry released fission gases from the fuel rod to a detector situated outside the reactor. With this equipment, it is possible to measure, using gamma spectroscopy, both radioactive and stable fission product release. In conjunction with fuel centerline thermocouples to measure fuel temperatures, these techniques have been successful in improving our understanding of the release process and the factors affecting it. The data generated have been used in many member countries to develop models and validate fuel performance codes used in reactor safety assessments.

In the sweep gas experiments, gas lines are attached to both ends of the fuel rod. This allows the introduction of a gas to pass through the free volume and carry entrained gases released from the fuel to a gamma detector situated outside but adjacent to the reactor. In this way, stable fission gas release can be inferred from measurements on the long lived isotope ^{85}Kr with a half life of ~10 years as well as the release of radioactive fission products, e.g. the radiologically important isotope ^{131}I from measurements of short lived krypton and xenon with half lives spanning ~90 secs to around 5 days. The release of ^{131}I is of greatest importance primarily due to its long half-life. Iodine release data are obtained by measuring the decay product ^{131m}Xe. Estimates of ^{131}I inventories are also obtained by measurements on ^{85m}Kr. Experiments to measure the release of short lived rare gases have made a significant contribution to our understanding of stable gas release, in particular in the determination of the in-pile diffusion coefficient and the processes occurring at grain boundaries.

The American Nuclear Society has recently formed a work group to review the current ANS-5.4 model with a view to improving it. Halden Project will take active part in this activity and supply experimental data.

This paper will discuss some of the more important measurements made with in-pile instrumentation and sweep gas experiments performed in the Halden reactor with particular reference to the release of radioactive fission gases.

Short-Time Creep and Rupture Tests on High Burnup Fuel Rod Cladding

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Summary

The general tendency to higher discharge burnups of LWR fuel assemblies has also to be considered at the back end of the fuel cycle. In Germany, dry storage is a settled technology for interim storage. Two central interim storage sites are licensed and further installations near the reactor sites have been applied for by the electric power utilities. The current peak discharge burnup already exceeds 50 MWd/kgU. Early in the 90's, new experiments to support cask licensing were initiated to keep up with the already growing burnup at that time. The experiments were performed by order and in co-operation with the German electric power utilities.

A central point of investigation to avoid systematic rod failure during dry storage is the creep of the rod cladding. The current licensing basis specifies an allowable 1% strain level. For the cladding, growing burnup will result in an increase in neutron fluence and corrosion layer thickness and as a consequence in higher hydrogen content. Since the mechanical properties of Zircaloy change with neutron fluence and hydrogen content, it is necessary to confirm the cladding ductility of the fuel rods with growing burnup.

This paper describes the experimental setup and the results of the short-time creep and rupture tests that were performed to assess the strain potential of cladding of high burnt fuel rods under conditions of dry storage. The device was developed and installed at the Institute for Transuranium Elements (ITU) at Karlsruhe, Germany. In total, 21 samples with a length of about 200 mm were tested. The fuel was mechanically removed and the sample connected to an oil pressuring system. The oil pressure was controlled by the movement of a piston, which acted as pressure booster to achieve pressures of up to 100 MPa. The piston was connected to a linear variable differential transformer (LVDT) allowing to record the amount of oil pumped into the sample during the creep phase. After removal of the sample from the furnace, the diameter of the samples could be measured mechanically. A measuring head with a knife system was used to determine the sample diameter as a function of the axial position. The uniform plastic strain of a sample was calculated by means of the axial diameter distribution of the individual sample before testing.

The tests comprised irradiated corrosion-optimized Zircaloy-4 cladding samples from fuel rods with burnups of up to 64 MWd/kgU equivalent to neutron fluences of up to 12×10^{21} cm^{-2} (E_n>1MeV). The corrosion-optimized Zircaloy-4 cladding was of a fast creeping type to envelop all commercially used materials with smaller thermal creep. The oxide layers on the cladding ranged from 10 to 100 microns. The inpile creep down of the cladding amounted to about 0.8 to 0.6 % at the end of reactor insertion. The maximum creep down was about 0.8 to 0.9 % at medium rod burnups of 20 to 40 MWd/kgU. The smaller creep down at high burnups resulted from a slight back straining of 0.2 % due to the swelling of the fuel inside the rod.

To simulate the dry storage scenario, a creep test was used with a high creep rate at maximum dry storage temperatures in the cask. The tests were carried out at temperatures of 573 and 643 K at cladding stresses of about 400 and 600 MPa. The stresses, much higher than those occurring in a fuel rod, were chosen to reach circumferential elongations of about 2 % within an envisaged testing time of 3-4 days.

To assess the influence of the higher hydrogen content on the cladding ductility, a separate ductility test followed the creep test. The test was performed at low temperatures (423 K), since at that temperatures the influence of the hydrogen on the cladding ductility is more pronounced than at higher temperatures. A cladding stress of about 100 MPa was chosen to simulate maximum hoop stresses during dry storage.

The creep tests showed considerable uniform plastic strains at these high burnups. It was demonstrated that around 600 K a uniform plastic strain of at least 2 % is reached without cladding failure.

The low temperature tests at 423 K and 100 MPa hoop stress for up to 5 days revealed no cladding failure under these conditions of reduced cladding ductility. The precipitation of the hydrogen in the cladding after testing was examined by metallography. In addition to tangentially orientated hydrogen platelets, which is a typical precipitation pattern in the cladding after reactor operation, also short radially orientated platelets were found.

Definition and Status of the CABRI International Program with a Sodium Loop and a Water Loop

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Economic considerations are currently leading most utilities to increase the burnup of the UO_2 fuel in their power plants. For instance in France, Electricité de France (EdF) recently received authorization from the safety authority to burn its fuel up to 52 GWd/t (fuel assembly average) and anticipates a further burnup increase to 60 GWd/t in future years. For the same economic considerations, some utilities have introduced mixed-oxide (MOX) fuel, and EdF for example has been using MOX fuel for 10 years.

One of the key requirements of increasing burnup has been the need to improve cladding resistance to corrosion. This has led to the development of new cladding alloys like ZIRLO and M5. However, increasing burnup and introducing MOX fuel have also created a need to (a) verify the adequacy of safety criteria that were previously defined for lower fuel burnups, or to modify those criteria, and (b) demonstrate good behavior of the fuel during normal operation and during design-basis accidents such as loss-of-coolant accidents (LOCAs) and reactivity-initiated accidents (RIAs).

With this background, the Institute de Protection et de Sûreté Nucléaire (IPSN) initiated a research program in 1993, called Cabri-REP-Na, to study the behavior of high-burnup UO_2 and MOX fuel under RIA conditions. This program was conducted in collaboration with EdF and with participation of the USNRC. The RIA conditions were simulated with a very rapid injection of energy in fuel rods that were previously irradiated in a power plant. The tests were performed in a sodium loop in the Cabri test reactor. Seven tests were performed with UO_2 fuel and three tests were performed with MOX fuel between 1993 and 1998. The following parameters were studied:

- Rod burnups from 33-64 GWd/t
- Cladding corrosion from 4-130 μ ZrO_2
- Corrosion conditions from uniform to spalled with hydride blisters
- Energy deposition from 95-210 cal/g
- Pulse widths from 10-80 msec

The Cabri-REP-Na tests were complemented by separate-effect tests and by development of the SCANAIR computer code to interpret test results, perform sensitivity studies, and translate the results to reactor conditions.

Low energy failures have been observed in this program, and the data suggest that present safety criteria are no longer valid for high-burnup UO_2 fuel with highly corroded Zircaloy cladding. This program has also identified several key parameters that influence fuel behavior during RIA transients. These are pulse width, cladding corrosion, oxide spalling, and fission gas dynamics. The program has also shown the possibility of transient oxide spalling above certain oxide thicknesses, and calculations show the possibility of local departure from nucleate boiling (DNB) for a brief period.

Questions remain about the effect of DNB on cladding failure, the influence of internal rod pressure, and the possibility of fuel-coolant interactions after failure. These questions result from the lack of representative conditions in the sodium environment. The complexity of these phenomena and their important coupling make it difficult to have confidence in the current results without experimental confirmation with integral tests under representative PWR conditions. Prototypical PWR conditions are particularly important for the qualification of any further increases in fuel burnup in power reactors. These are the reasons that IPSN has decided to replace the present sodium loop in Cabri with a pressurized water loop (PWL) and to propose an international program called Cabri-PWL.

Twelve tests have been proposed for the Cabri-PWL program and they include high-burnup fuel tests combined with mechanistic tests to provide the understanding necessary to extrapolate to a broad spectrum of reactor conditions. Six test series have been identified:

S0 — two tests in the sodium loop using advanced fuels
S1 — two tests in the water loop with the same advanced fuels to provide a link to Cabri-REP-Na
S2 — tests with ultra high burnup fuel (80-100 GWd/t)
S3 — tests specifically designed to improve the understanding of RIA phenomena
S4 — tests with MOX fuel
S5 — complementary tests (open)

These integral tests will be coupled with separate-effect tests (mechanical testing, fission gas behavior experiments) and code development to facilitate translation to power reactor conditions and the development of safety criteria or limits.

To facilitate installation of the pressurized water loop, the Cabri reactor will undergo complete renovation to extend its lifetime for the long term. However, to accommodate the needs of EdF and IPSN, several additional tests will be conducted in the sodium loop. Two tests will be performed in the sodium loop in 2000 (5-cycle M5 fuel and 5-cycle MOX fuel), and another test window will be opened in 2002 to perform the two S0 tests for the Cabri-PWL project and potential domestic tests. The reactor will then be shut down from 2003-2004 for completion of the renovation and installation of the water loop. Completion of the ten tests (series S1-S5) in the Cabri-PWL program will be in 2005-2007. This strategy provides preliminary results on very-high-burnup advanced fuel as early as 2002.

Organization of the Cabri-PWL program is being done under the auspices of OECD/NEA and initial meetings of the Steering Committee and the Technical Advisory Group were held in 2000. A number of countries have expressed interest in participating in this program (USA, UK, Germany, Spain, Sweden, etc.), and first agreements will be signed in the latter part of 2000.

In summary, the Cabri-REP-Na program, begun in 1993, has already provided important results on the behavior of high-burnup UO_2 fuel and MOX fuel under RIA conditions. Cabri will continue in the framework of a broad international collaboration to provide data under more representative reactor conditions that will help the industry and their respective regulatory agencies realize the economic benefits of higher burnups and improved fuel types.

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HIGH BURNUP BWR FUEL RESPONSE TO REACTIVITY TRANSIENTS AND A COMPARISON WITH PWR FUEL RESPONSE

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SUMMARY

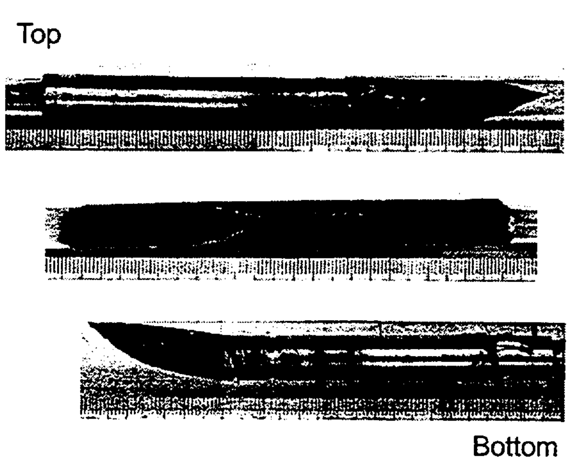
The recent two BWR fuel experiments in the NSRR, tests FK-6 and FK-7, resulted in significant cladding failure and fuel dispersal. The fuel rods in both tests were 8x8 Step II type rods with Zr-liner cladding at a burnup of 61 MWd/kgU irradiated for 5 cycles in the 2nd Fukushima plant unit 2. The fuel rods failed during the pulse irradiations at fuel enthalpies of 293 J/g (70 cal/g) for FK-6 and 260 J/g (62 cal/g) for FK-7. The expected peak fuel enthalpies were 548 J/g (131 cal/g) and 540 J/g (129 cal/g), respectively. A difference between FK-6 and FK-7 in terms of test conditions was the initial rod internal pressure, 0.1 MPa for FK-6 and 1.5 MPa (simulating EOL gas pressure) for FK-7. The cladding was broken apart into three pieces in the both tests, and all of the fuel pellets were finely fragmented and dispersed into the capsule water. Fuel particles collected from the capsule water were sieved, and the results showed that about half of fuel pellets became particles smaller than 0.1 mm.

In the first five tests of the FK test series, tests FK-1 through FK-5, cladding failure did not occur. Tests FK-1, FK-2 and FK-3 were conducted with 8x8BJ Step I type rods with Zr-liner cladding at burnups of 41 to 45 MWd/kgU irradiated for 5 cycles. The subsequent two tests, FK-4 and FK-5, used 8x8 Step II type rods with Zr-liner cladding at a burnup of 56 MWd/kgU irradiated for 4 cycles. The Step II fuel rod has a narrower pellet-to-cladding gap and higher fuel density than in the Step I rod. General behavior of the Step-II rod in FK-4 and FK-5, however, was quite similar to that of the Step I rod. The cladding conditions in terms of oxide thickness and hydrogen content were similar to those of the Step I rod, about 20 microns oxide and 60 ppm hydrogen. The cladding was ductile enough to survive PCMI loading during the pulse irradiation in the FK-1 through FK-5. The wider pellet-to-cladding gap due to the smaller creep down in BWRs could cause the PCMI loading to be milder than in the 50 MWd/kgU PWR fuel experiments.

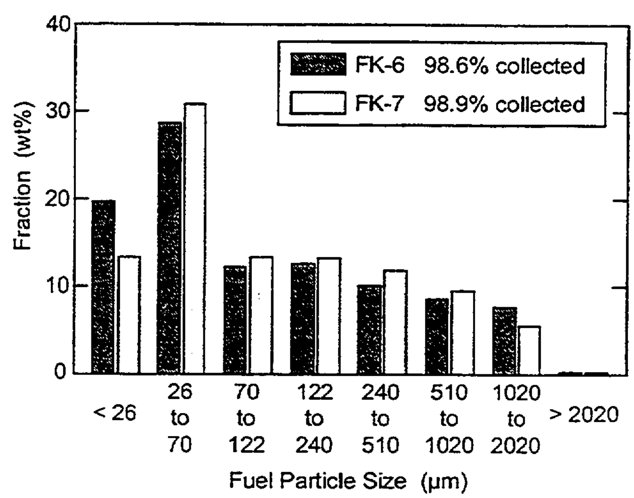
On the other hand, in the FK-6 and FK-7, extensive bonding occurred between the Zr-liner and the fuel pellets, and the pellet-to-cladding gap was completely closed before the pulse irradiations. Post-test fuel examinations and data analyses are in progress for FK-6 and FK-7. Although the data are preliminary, the results suggest that an occurrence of an intense PCMI loading due to the bonding. Hydrogen concentration in the cladding is not high, probably about 100 ppm, but hydride clusters are not circumferentially oriented. Radially located hydride clusters may have an influence on the cracking of the cladding.

NSRR/High burnup BWR fuel tests (FK test series) with PWR fuel tests resulted in failure

Test ID	Test Fuel	Fuel Burnup (MWd/kgU)	Fill gas pressure of test rod (MPa)	Peak Enthalpy (J/g)	Result
FK-1	BWR, 8x8BJ Step I, 5 cycles Zr-liner	45	0.3	544	No failure
FK-2		45	0.3	293	No failure
FK-3		41	0.3	607	No failure
FK-4	BWR, 8x8 Step II, 4 cycles Zr-liner	56	0.5	586	No failure
FK-5		56	0.5	293	No failure
FK-6	BWR, 8x8 Step II, 5 cycles Zr-liner	61	0.1	548	Failed at 293 J/g (70 cal/g), 100% fuel dispersed
FK-7		61	1.5	540	Failed at 260 J/g (62 cal/g), 100% fuel dispersed
HBO-1	PWR, 17x17 1.5%Sn Zry-4	50.4	0.1	306	Failed at 251 J/g (60 cal/g), 100% fuel dispersed
HBO-5	PWR, 17x17 1.5%Sn Zry-4	44	0.1	335	Failed at 322 J/g (77 cal/g), 5% fuel dispersed
TK-2	PWR, 17x17 1.3%Sn Zry-4	48	0.1	448	Failed at 251 J/g (60 cal/g) 7% fuel dispersed



Post-test appearance of FK-6 rod



Particle size of dispersed fuels

THE HISTORY OF LOCA EMBRITTLEMENT CRITERIA

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SUMMARY

Because of major advantages in fuel-cycle costs, reactor operation, and waste management, the current trend in the nuclear industry is to increase fuel discharge burnup. At high burnup, fuel rods fabricated from conventional Zircaloys often exhibit significant degradation, which is especially marked in PWR rods fabricated from standard Zircaloy-4, in which significant oxidation, hydriding, and oxide spallation can occur. Thus, many fuel vendors have developed and proposed the use of new alloys, such as low-Sn Zircaloy-4, Zirlo, M5, MDA, duplex cladding, and Zr-lined Zircaloy-2. Performance of these alloys under loss-of-coolant-accident (LOCA) situations, especially at high burnup, is poorly understood. Therefore, LOCA-related behavior of various types of high-burnup fuel cladding is being actively investigated in several countries. However, to correctly interpret the results of such investigations, and, if necessary, to establish embrittlement thresholds that are applicable to high-burnup operation, it appears necessary to accurately understand the history and relevant databases of current LOCA embrittlement criteria. In this paper, documented records of the 1973 Emergency Core Cooling System (ECCS) Rule-Making Hearing were carefully examined to clarify the rationale and data bases used to establish current criteria. A large amount of data, obtained for zero- or low-burnup fuel cladding and reported in literature only after the 1973 rule making hearing, were also evaluated with respect to the current criteria established in 1973.

In accordance with the recommendations of the Ergen Task Force, the primary aim of the current embrittlement criteria was to maintain coolability and preserve the heat-transfer area and coolant-flow geometry, not only during the quench phase of a LOCA but also during the longer-term postquench phase when mechanical properties of the cladding are strongly influenced by hydrogen uptake and hydride precipitation. Continued risk of cladding fragmentation, which may occur after quench below the Leidenfrost temperature under hydraulic or seismic loading was fully recognized. The major findings of our investigation can be summarized as follows:

- The staff and commissioners in the 1973 hearing were reluctant to neglect the effect of mechanical constraints on thermal-shock fragmentation. Subsequent test results from the Phebus program and JAERI appear to justify this position.
- The staff and commissioners were of the opinion that retention of ductility was the best guarantee against potential fragmentation under various types of loadings such as thermal shock and from hydraulic and seismic forces.
- Results from unconstrained or partially constrained quench tests (under simple thermal shock) were considered only corroborative and reassuring by the staff and commissioners; their use for regulatory purposes was not accepted. Results from this type of tests, conducted later in JAERI, ANL, Mitsubishi, IPSN, and EdF for cladding oxidation temperatures lower than or equal to 1204°C, show a large margin when compared to the 17% equivalent cladding reacted (ECR) criterion.
- In establishing the 1204°C (2200°F) peak-cladding-temperature criterion, the primary limiting factor was postquench cladding ductility. Potential for runaway Zr oxidation was a secondary

consideration. The criterion was selected largely on the basis of results of low-temperature slow-strain-rate compression tests conducted in ORNL on ring specimens oxidized in steam at <1320°C. For oxidation temperatures higher than about 1204°C, the embrittlement mechanism was identified as beta-phase solid-solution hardening at oxygen concentrations of >0.7 wt.%. Results from postquench impact, ring compression, and axial-tensile tests at ANL and from postquench handling failure of fuel rods tested in the Power Burst Facility, reported in 1980s, appear to justify the rationale and selection of the criterion. The threshold oxygen concentration of 0.7 wt.%, which corresponds to the approximate solubility limit of oxygen in the hydrogen-rich beta phase at about 1200-1230°C, was the key parameter used later at ORNL, AECL, and ANL to develop several embrittlement criteria.

- The 17%-oxidation criterion was selected on the same basis as the 1204°C criterion, i.e., the results of the ORNL slow-ring-compression tests. The primary criterion was that zero-ductility temperature shall be no higher than 135°C (275°F), i.e., the saturation temperature during reflood. The 17% criterion, also consistent with the thermal-shock failure boundary established by Hesson and Scatena for oxidation temperatures of >1500°C, was derived only in association with the use of the Baker-Just oxidation correlation. Results obtained in the 1980s for unirradiated cladding from ANL impact tests, JAERI constrained quench tests, and Williford's statistical analysis were found to be consistent with this criterion. Considering the fact that oxygen solubility limit in beta phase is a function of temperature and hydrogen content, a criterion developed at ANL that specifies that the thickness of beta-phase layer that contains <0.7 wt.% oxygen shall be greater than 0.3 mm appears to be consistent with the 17%-oxidation limit and a peak-cladding-temperature limit of about 1200-1230°C.

HIGH TEMPERATURE STEAM OXIDATION OF ZIRCALOY CLADDING FROM HIGH BURNUP FUEL RODS*

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SUMMARY

The ANL Cladding Metallurgy at High Burnup program is being conducted to provide data in support of efforts to model the behavior of high burnup fuel rods during Loss of Coolant Accident (LOCA) and Reactivity-Initiated Accident (RIA) events and to assess directly the LOCA criteria licensing limits for high burnup fuel. The program is sponsored by the USNRC, and the Electric Power Research Institute (EPRI) provides the fuel rods for testing. One demonstration TMI-1 PWR rod (≈50 GWd/MTU) and seven high burnup (≤57 GWd/MTU) Limerick BWR fuel rods have been provided for the test program; a similar number of H.B. Robinson PWR fuel rods (≈70 GWd/MTU) will be provided at a later time. Both USNRC and EPRI participate in the development of the test plan to ensure that regulatory and industrial issues are adequately addressed. The major program tasks are characterization of fuel and cladding, studies of cladding high temperature steam oxidation kinetics, LOCA-criteria testing, and studies of cladding mechanical properties (uniaxial tensile, plane strain, biaxial, and bending). The focus of this paper is on the results of the oxidation kinetics study and its impact on LOCA-criteria test planning.

To ensure adequate ductility during Emergency Core Cooling System (ECCS) quench and during possible post-LOCA seismic events, the current LOCA licensing criteria (10 CFR50.46) limit the peak cladding temperature to 2200°F (1204°C) and the peak Equivalent Cladding Reacted (ECR) to 17% during high temperature steam oxidation. In addition, as discussed in NRC Information Notice 98-29, the ECR is based on the total oxidation, including oxide layers formed during normal reactor operation. For PWR cladding, high burnup operation may result in coolant-side oxidation thicknesses of up to ≈100 μm, corresponding to ≈10-14% ECR. This leaves very little margin for LOCA transient oxidation. Although this approach may ensure an adequate safety margin for high burnup fuel, the in-reactor-formed oxide layer may not affect all of the mechanisms responsible for cladding ductility loss during ECCS quench, and its inclusion may be overly conservative. The primary high burnup phenomena that may affect cladding response during ballooning and burst, steam oxidation and quench are greater loss of base metal thickness during normal operation, increased hydrogen pickup (≈500-700 wppm at 100 μm oxide thickness), greater change in microstructure and precipitate morphology at the higher fluences, and tighter fuel-cladding bond. The first three phenomena may decrease the ductility of the cladding by decreasing the effective thickness of the prior-β-phase layer and by increasing the H and O transport to that layer during steam oxidation. A tighter fuel-cladding bond may influence ballooning shape and burst extent, as well as induce additional stresses on the cladding during quench. The LOCA-criteria tests will be conducted with fuel rod segments with the fuel intact to ensure that the effects of the fuel-cladding bond are included. The oxidation studies are performed on defueled cladding samples to determine ECR vs. time. Using these oxidation data, LOCA-criteria test times will be determined to correspond to the current criteria limits (ECR=17% based on total oxidation at 1204°C) and to larger and smaller ECR values such that the failure threshold is bracketed in the tests.

* Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES).

The test plan for oxidation studies of high burnup BWR and PWR fuel rod cladding specifies ranges of temperature (900-1300°C) and test time (0-300 minutes). In addition to providing data for LOCA-criteria test planning, the oxidation tests also provide fundamental data for modeling the effects of high burnup operation on high temperature steam oxidation kinetics. Of particular interest in these studies is the influence of the in-reactor-formed oxide layers and associated hydrogen pickup on the oxidation kinetics and phase boundary evolution during steam oxidation. For the TMI-1 demonstration cladding and the high burnup BWR and PWR cladding, fast fluences (E>1 MeV) range from $9\text{-}13\times10^{26}$ n/m^2, outer diameter (OD) oxide layer thicknesses range from 10-110 µm, and hydrogen contents range from about 50 to 700 wppm. In order to determine the effects of these parameters on oxidation kinetics, unirradiated archival BWR and PWR cladding samples are tested concurrently with high burnup samples to allow a direct comparison of the results.

The work to date on the oxidation studies has focused on test planning, thermal benchmarking, metallurgical benchmarking, testing of unirradiated Zircaloy-2 and -4, and testing of irradiated TMI-1 PWR and Limerick BWR cladding. In the test setup the cladding samples are mounted in a quartz tube inside a quad-elliptic radiant furnace. Steam flows in the annulus between the specimen OD and the quartz tube. Oxidation of the sample ID is minimized by flowing Ar inside the sample, as well as by inserting zirconia gaskets (under compression) between the test specimen and the alumina-spacers that separate the sample from the Inconel holders. Thermal benchmarking has been performed using five thermocouples (two above and two below the 25-mm sample; one inside the sample). This arrangement allows assessment of the axial and circumferential temperature variations within a 50-mm span. The thermal benchmarking results demonstrate uniform temperature profiles for a range of steam flow rates, Ar purge flow rates, and temperature ramp rates. Metallographic benchmarking has consisted of oxide, α-phase and prior-β-phase thickness measurements vs. axial and circumferential locations for unirradiated cladding. With the exception of small end effects, the results indicate uniform oxidation and phase layer thicknesses consistent with best estimate correlations. Most of the benchmarking has been performed for 1204°C tests. Samples (25 mm) of archival and irradiated Limerick cladding have been tested initially in sequence at several 1204°C hold times. Subsequent tests have been performed with pairs of unirradiated and irradiated cladding samples. Oxidation results will be reported for unirradiated and irradiated Limerick BWR cladding and for TMI-1 PWR cladding. Although most of these results are for tests at 1204°C, additional results will be presented for tests at 1000°C. The measurements consist of: pre-test characterization of ID and OD oxide layer thicknesses and oxygen and hydrogen concentrations; and post-test weight gain, oxide and phase layer thicknesses, and oxygen and hydrogen concentrations.

DEVELOPMENT AND ASSESSMENT OF THE FRAPTRAN TRANSIENT FUEL ROD CODE

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The FRAPTRAN computer code is being developed for the U.S. Nuclear Regulatory Commission (NRC) to calculate fuel behavior during power and cooling transients such as reactivity-initiated accidents (RIAs), anticipated transients without scram (ATWS), and loss-of-coolant-accidents (LOCAs) at burnup levels up to at least 62 GWd/MTU, the current limit for fuel burnup in the U.S. FRAPTRAN will be used to provide insight into fuel performance during transients and help guide experimental programs obtaining necessary data for regulatory criteria at high burnup. Development of the FRAPTRAN code is being done subsequent to, and building on, development of the FRAPCON-3 computer code, which has been released (Berna et al. 1997). FRAPCON-3 is used by the NRC for auditing licensing analyses of steady-state behavior at high burnup levels, and for other studies.

FRAPTRAN will serve as a research tool for analysis of fuel response to postulated design-basis accidents such as RIAs, ATWS, and LOCAs; understanding and interpretating experimental results; and guiding of planned experimental work. Examples of planned application of FRAPTRAN include defining transient performance limits, identifying data or models needed for understanding fuel performance, and in assessing fuel design changes such as new cladding alloys and mixed-oxide fuel. FRAPTRAN will be used to perform sensitivity analyses of the effects of parameters such as fuel-cladding gap size, rod internal gas pressure, and cladding ductility and strength on the response of a fuel rod to a postulated transient. Fuel rod responses of interest are cladding strain, location of ballooning, cladding oxidation, etc.

FRAPTRAN has already been used to support evaluations of the recent RIA experiments such as evaluating the effects of pulse width and cladding ductility on fuel response. Comparisons to the RIA data have shown that phenomena other than fuel thermal expansion are necessary to produce the observed hoop strains. In the area of guidance for planned and potential experimental programs, the code will also be used to evaluate the physical characteristics of the fuel rods that will be used by Argonne National Laboratory (ANL) for upcoming testing. Evaluations will include effects such as cold fuel-cladding gap size and cladding strength. It will also be used to evaluate the planned LOCA tests such as extent of cladding oxidation and ballooning of the test rods. FRAPTRAN has already been used to evaluate the possibility that test rigs in the Halden Boiling Water Reactor might simulate fuel response during a BWR ATWS (Cunningham and Scott 1999). This analysis has shown significant fuel sensitivities to the assumed oscillating power levels and coolant conditions.

FRAPTRAN is being developed from the FRAP-T6 computer code (Siefken et al. 1981). Since 1983, there have been few modifications to the code with the result that FRAP-T6 does not adequately predict fuel behavior at the current burnup levels. High-burnup modifications to FRAP-T6 to produce FRAPTRAN include model updates and additions to account for data and knowledge gained since FRAP-T6 was released; and general coding improvements to address known errors, ensure consistency across the coding, improve usability, and to delete coding and models that are no longer needed.

An assessment data base has been selected that emphasizes experiments that investigate the effects of burnup on fuel rod behavior during RIAs and LOCAs. In particular, these include the RIA tests conducted by the Japan Atomic Energy Research Institute (JAERI) (Fuketa et al. 2000) and the Institute for Protection and Nuclear Safety (IPSN) (Papin and Schmitz 1997). For assessing performance in predicting LOCA behavior, the LOCA experiments conducted in the National Research Universal (NRU) reactor will be used (Wilson et al. 1983).

Both FRAPTRAN and FRAP-T6/Version 21 have been run against the selected assessment cases to validate the performance of FRAPTRAN. The results show that FRAPTRAN provides a better comparison to the experimental data than FRAP-T6. The results also show that additional models would be needed in FRAPTRAN to reproduce the strains induced on the high burnup rods during the RIA experiments.

The current status of the FRAPTRAN effort is as follows: principal code development is completed, assessment cases have been run and evaluated, the code description and assessment reports have been written, and a peer review is in process. The code description and assessment reports will be finalized after the peer review is completed in FY-2001 and then the code will be issued. As material properties models are improved in the future, for example, from the work being done by ANL on high-burnup cladding, those models will be incorporated in FRAPTRAN.

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PWR Sump Blockage and Containment Coatings Service Level I Safety Concerns

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As a result of research findings related to resolving the BWR ECCS strainer blockage safety issue, RES was requested (Ref. 1) to determine if further action was needed for PWRs beyond the original resolution of Unresolved Safety Issue USI A-43 (Ref. 2) and also to investigate the performance of "qualified" containment protective coatings under post-LOCA conditions since such coatings have exhibited failure and formation of undesirable debris during the normal plant operational life cycle (Ref. 3). Generic Safety Issue - 191, "PWR Sump Blockage" is addressing the need for further action in PWRs and a coatings research program is underway at the Savannah Research Center (SRTC) designed to determine if "qualified' coatings could fail and failed coatings debris characteristics. The NRC staff and industry groups (e.g. NEI, WOG, protective coatings experts, et al) have been interacting over the past two years to ensure that available information/knowledge is brought to bear on these safety concerns.

A panel has been formed for this session to describe the knowledge base applicable to U.S. PWRs, bring forth significant findings, discuss the application of such findings to PWR emergency sump designs, and discuss how closure can be achieved without sacrificing needed safety margins while minimizing additional regulatory burden. The panelists will discuss: NRC's approach to assess debris accumulation on sump screens and ECCS pump performance, the application of the PIRT (Phenomena Identification and Ranking Table) approach to identify qualified coating failure phenomena and STRC research results obtained, overall industry follow and support of NRC's research activities, and plant-specific views on these safety issues. A short interactive session with attendees will conclude this session.

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2. SECY-85-349, Resolution of Unresolved Safety Issue A-43, "Containment Emergency Sump Performance," October 31, 1985.

3. NRC Generic Letter 98-04: "Potential for Degradation of the Emergence Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment'" July 14, 1998

DIGITAL INSTRUMENTATION AND CONTROL (I&C)

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PURPOSE

The purpose of the Draft NRC Research Plan for Digital I&C is to describe the proposed digital I&C research program in terms of its background, challenges and technical issues, and planned activities to meet the challenges. The plan will provide the basis for the research in this area for the next five years.

BACKGROUND AND CHALLENGES

Nuclear power plant licensees are currently replacing, and will continue to replace, analog I&C equipment with digital equipment. Two main reasons for analog-to-digital upgrades are (1) analog replacement parts are becoming more difficult to obtain and (2) digital I&C systems offer better performance and additional features compared to analog systems. While digital technology has the capability to improve both operational performance and safety, there are challenges to the introduction of this technology into nuclear power plants. These challenges include (1) constant changes in digital technology that requires the NRC to continually update its knowledge of the state-of-the-practice in digital system's design, testing and application, (2) the increased complexity of digital technology compared to analog technology, and (3) unique failure modes associated with digital technology. Failure to adequately address these challenges in other industries (e.g., aviation, medical, nuclear) has resulted in mishaps and near mishaps.

SUPPORT OF NRC GOALS

In response to the challenges of digital I&C systems, the NRC Office of Nuclear Regulatory Research (RES) is performing research to better understand digital technology and to update the tools used in assessing the safety of digital I&C applications in US nuclear power plants. The following list ways in which the digital I&C research program will support NRC regulatory activities.

1. Maintain safety by assuring those potential failure modes associated with digital technology are properly addressed by licensees.

2. Reduce undue licensee burden by proposing the modification, or removal, of regulatory criteria and actions that offer little safety benefit for the amount of effort required.

3. Reduce NRC resources needed to review topical reports and plant specific applications by providing more efficient methods and tools.

4. Increase public confidence by supplying appropriate technical information and criteria to the regulatory process in a timely manner.

As a short term goal, as needed by the Office of Nuclear Reactor Regulation (NRR), the digital I&C research program will develop the methods and tools needed to support improvements in the review of digital systems, while maintaining or improving the predictability of the review process.

In the long term, digital failure characteristics and probabilities need to be modeled sufficiently well so that digital systems can be effectively added to risked-based regulatory programs. Another long term goal is to develop new regulatory guidance for the review of emerging I&C technology in the digital areas.

TASKS WITHIN THE RESEARCH PROGRAM

In order to meet the goals discussed above, RES will engage in a series of tasks, grouped into the following four areas: (1) System Aspects of Digital Technology, (2) Software Quality Assurance, (3) Risk Assessment of Digital I&C Systems, and (4) Emerging I&C Technology and Applications. The following is a discussion of the tasks within each area.

<u>Systems Aspects of Digital Technology</u> - Systems aspects of digital I&C systems involve those factors, both internal and external, that impact the performance of the system. This plan discusses four types of system aspects that have the potential to impact plant safety and future regulatory decisions.

- ***Environmental Stressors*** include electromagnetic interference/radio-frequency interference (EMI/RFI), temperature, humidity, smoke, and lightning. Research efforts will provide appropriate acceptance criteria for the qualification of digital equipment against these stressors.

- ***Digital Requirement Specifications*** describe the functions expected from the digital I&C system, and they state how the digital system interfaces with other plant systems and components. Currently, it is difficult to review requirement specifications for correctness and completeness. Research efforts will provide the best methods and tools for the review of requirements specifications.

- ***Diagnostics and Fault-Tolerance*** are special features with many digital I&C systems that enable the system to detect internal problems and either avoid or handle the problem, or alert the operator to the problem. While these features could improve plant safety, research efforts will investigate both positive and negative safety impacts and determine the amount of credit that should be given to them in a review.

- ***Operating Systems*** control basic functions of a digital I&C system, including its communication functions, memory management, and processor scheduling. These systems are becoming larger and more complex, making the review of such systems difficult. Research efforts will identify the aspects of operating systems that may adversely impact safety.

<u>Software Quality Assurance</u> - Software quality assurance is a planned and systematic pattern of all actions necessary to provide adequate confidence that an item, or product, conforms to established technical requirements. While the NRC currently has a set of software quality assurance activities, these activities are resource-intensive for both the NRC and the industry. In addition, current software testing activities do not specify how they should be performed or how much testing should be conducted. This leads to an inconsistency in the amount of testing and testing methods for similar digital I&C systems. The digital I&C research tasks will address these issues in the following ways:

- ***Objective software engineering criteria*** provide a measurable acceptance level for software quality. Because software engineering is a young discipline, methods for evaluating software quality have relied on subjective judgment. However, research results from academia and other industries show potential software measures that could be used to establish minimum software quality acceptance levels. Research efforts will investigate the potential of using such measures for NRC regulatory purposes.

- ***Criteria for software testing*** are important to software quality assessments. Software test criteria should dictate the types of tests and the number/type of test cases. Research efforts will support software quality assessments by supplying software test criteria.

<u>Risk Assessment of Digital I&C Systems</u> -The NRC intends to increase the use of probabilistic risk assessment (PRA) technology in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data. Currently, I&C systems are not modeled in plant PRAs, however, a recent Accident Sequence Precursor database study demonstrates the prevalence and impact of I&C systems on plant safety (see Appendix A). As the NRC moves toward a risk-informed regulatory environment, the NRC will need the data, methods, and tools related to the risk assessment of digital I&C systems. The following tasks describe how the research program will address these needs.

- ***Performing analysis on digital I&C failure data*** provides several benefits to the NRC, including: (1) the ability to determine which digital failures have the largest impact on plant safety, (2) feedback on the effectiveness of NRC regulatory programs, and (3) support for the risk assessment of digital I&C systems. Research efforts will gather and assess digital failure data from domestic/foreign nuclear power plants and other industries having digital systems that are critical to safety. Particular attention will be paid to commercial off-the-shelf digital I&C equipment.

- ***Digital failure assessment methods*** are used by defense and aerospace industries to determine types of failures and their impact on overall safety. Understanding this information is particularly important as the NRC moves toward a risk-informed environment. To ensure the quality of digital failure assessments, research efforts will provide criteria outlining the proper use of failure assessment methods.

- ***Identifying the risk-importance of digital I&C systems*** will help the NRC determine the required level of regulatory review for digital upgrades and focus research efforts on those aspects of digital I&C systems having a significant impact on plant safety.

- ***Digital reliability assessment methods*** estimate the likelihood of a digital I&C failure that would adversely affect plant safety. Due to the complex nature of digital systems and the uniqueness of software failures, estimating the likelihood of digital failures is difficult. Several reliability assessment methods have been used by other industries and show potential for use in the nuclear industry. Research efforts will identify digital reliability assessment methods that are applicable to the nuclear industry and provide criteria for their proper use.

<u>Emerging I&C Technology and Applications</u> - New innovations in the area of digital I&C technology have the potential to help nuclear power plants in both operating efficiency and safety. NRC regulatory programs require knowledge about emerging technology and applications in order to make timely decisions. Research tasks associated with this area will provide the technical

information and criteria for regulatory decisions. The following are the emerging technologies and applications addressed by the research program; not including those that may appear in a few years.

- ***Predictive maintenance and on-line monitoring systems*** provide the automatic capability to determine system/component failure or the need for maintenance. Such systems will motivate changes to surveillance and maintenance practices at nuclear power plants that result in operational cost reductions. Research efforts will analyze the positive and negative safety impacts of this technology.

- ***Advanced instrumentation*** for measuring flow, temperature, pressure, neutron flux, and other plant variables hold the potential to improve upon plant efficiency, safety, or both. To make timely and informed regulatory decisions involving advanced instrumentation (e.g., power uprates), the NRC needs the technical bases surrounding this emerging technology.

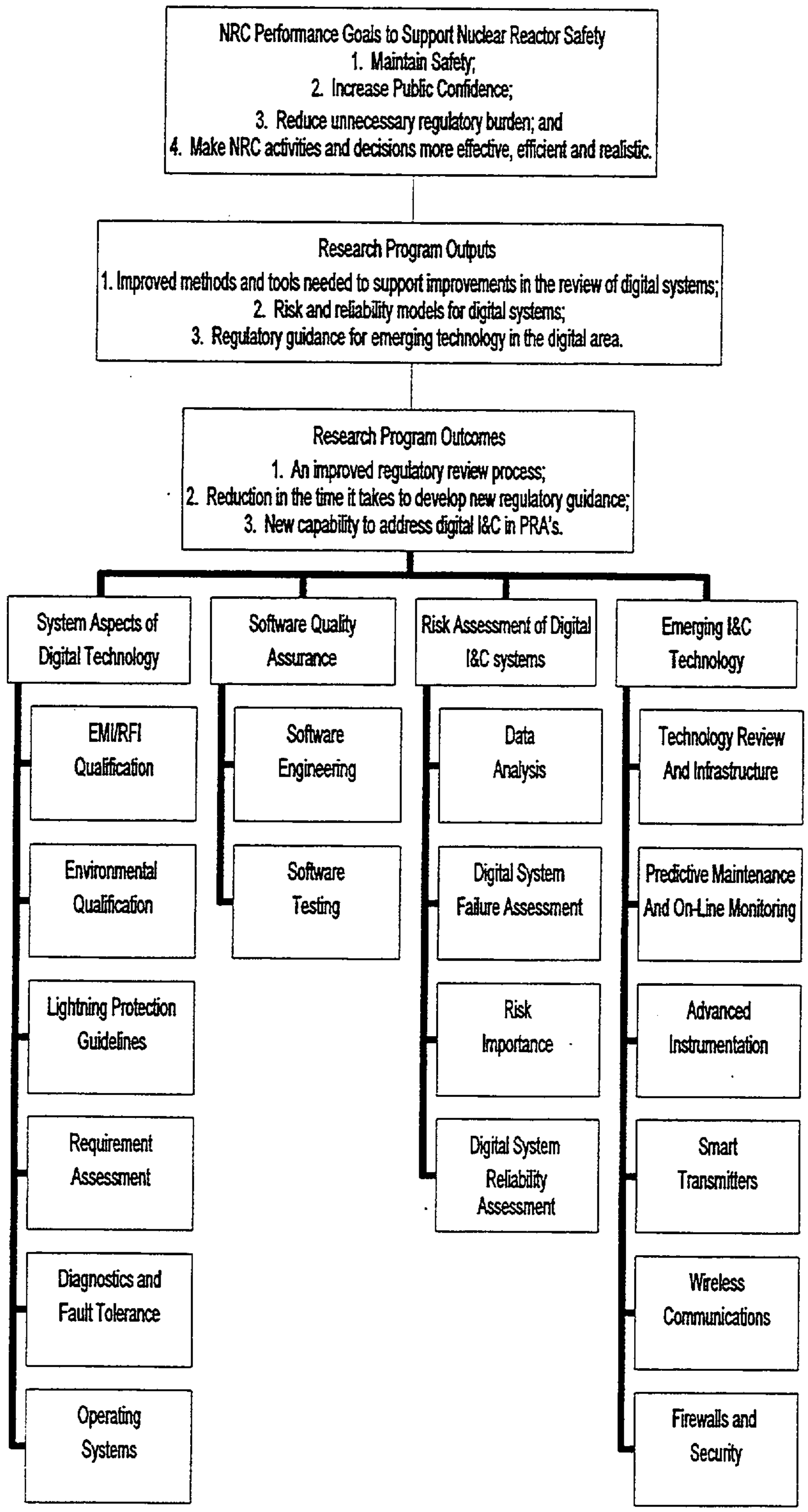
- ***Smart transmitters*** offer digital communication of data from the sensor to the control system. Some smart transmitters are also capable of providing compensating measures for instrument error or control functionality at the sensor. Research efforts will provide technical information to the NRC regulatory programs on this technology.

- ***Wireless communication*** is the transmission of plant data over radio-frequency networks. While this technology would eliminate some of the problems associated with cables, it also has its own inherent problems which will be identified by the digital I&C research program.

- ***Firewalls*** prevent hackers from accessing and corrupting/degrading the performance of computer systems. Research will be conducted to assess the potential for such corruption/degradation of computers in nuclear power plants. These efforts will identify what measures should be taken to prevent hacker access.

CONCLUSION

The digital I&C research program will meet the needs of NRC regulatory programs, as they relate to digital technology. The digital I&C research program will also include a task to continuously monitor the state-of-the-art in this area and develop new research projects to address any new safety concerns that may arise due to the implementation of emerging technologies. Digital I&C systems offer several benefits to NRC licensees, including increased operational efficiency and improved reliability. However, digital technology is complex and possesses failure modes not present in analog devices. In the short term, the digital I&C research program will provide methods and tools which support the introduction of digital technology without allowing new failure mechanisms into nuclear power plants. In the long term, the research program will provide timely information on emerging technologies and applications, and it will support the incorporation of digital systems into plant PRAs.



OVERVIEW OF THE DRAFT NRC RESEARCH PLAN
for
DIGITAL INSTRUMENTATION AND CONTROL (I&C)

Thermal Hydraulic and Severe Accident Analysis for Reactors and Spent Fuel

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This session offers a collection of papers spanning a range of technical issues related to analysis of design basis and severe accidents for disciplines ranging from two phase thermal hydraulics to multidimensional computational fluid dynamics and fission product aerosol transport and deposition. The common theme of this research is the purposeful application of improved methods to perform more realistic analysis for the goal of quantification of both safety margins and uncertainty. Improvements in regulatory effectiveness, risk informing of the regulations and unnecessary burden reduction all require the understanding of safety margins inherent to the NRCs current regulatory criteria and the quantification of uncertainty in safety analysis including phenomenological analysis in support of PRAs. It is through these activities that the NRC develops its greater understanding of reactor and fuel systems, and their associated response to accident conditions.

A proposed rule making on decommissioning plants has focused renewed attention on the risk of low probability, high consequence accidents caused by loss of cooling in spent fuel pools. In order to effectively analyze the cooling afforded by natural circulation of air through a spent fuel pool following a pool draindown accident, the NRC has performed detailed 3-D computational fluid dynamics (CFD) analysis which predict peak fuel temperatures and flow patterns.

In conjunction with regulatory activities associated with review of alternate repair criteria for steam generator tubes and the review of recent applications of the electrosleeve tube repair process, the NRC has closely evaluated the risk implications of steam generator tube performance during severe accidents. The NRC has recently initiated work to improve our understanding of the severe accident temperature and pressure conditions, and the uncertainty in their characterization as it applies to steam generator tubes as well as other RCS components.

The keystone of the NRCs thermal hydraulics program is the consolidation of improved modeling into the TRAC-M code and its graphical user interface (GUI) together with supporting experimental research that provides data for improved models. The incorporation of analytical capabilities from four separate codes into a modernized code with an extensible architecture will position the NRC to address operating reactor events and behavior as well the safety issues for new designs.

Recent regulatory activities have called for the re-analysis of the offsite radiological doses and health effects for accidents involving spent reactor fuel; in one case from a dry storage cask and in another from a spent fuel pool. Application of improved methods developed for reactor analysis have been applied to dry cask offsite dose analysis indicating the potential for significant burden reduction. Spent fuel pool accident offsite consequence analysis has been performed to address the risk reduction in time, as a result of radioactive decay.

Similar to the consolidation of thermal hydraulic modeling, the NRC is consolidating its different severe accident code models into a single vehicle, the MELCOR code. MELCOR, as the counterpart to the industry's MAAP code, provides the NRC with the capability to assess risk implications of regulatory issues, as well as the capability to perform analyses to provide the underlying technical foundation to the development of risk informed regulations.

Application of a CFD Code for Thermal Hydraulic Analysis of Spent Fuel Pool Accidents

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In support of the Nuclear Regulatory Commission (NRC) rulemaking activity related to decommissioning, the Office of Nuclear Reactor Regulation (NRR) is completing a study on spent fuel pool accident risks. The Office of Nuclear Regulatory Research (RES) is providing technical assistance to NRR in several areas. This report documents a thermal-hydraulic evaluation of spent fuel pool heatup after a low-probability complete loss of spent fuel coolant. Computational fluid dynamics (CFD) is used to predict fuel heatup and natural circulation flow paths throughout the spent fuel pool and the upper containment building. The predictions give insights into the phenomena governing the air cooling which provides most of the heat removal capacity during long-term cooling scenarios after a complete loss of coolant.

Spent fuel pool heatup predictions are typically made using codes tailored to the geometry and physics of spent fuel stored in a rack. Codes of this type include SHARP, SFUEL, and COBRA-SFS. A typical modeling approach assumes one-dimensional buoyancy-driven flows between idealized upper and lower control volumes which connect the fuel bundles together at the ends. Codes such as COBRA-SFS and SFUEL incorporate physical models for conduction, radiation, flow losses, clad oxidation chemistry, and other things. The flowfield assumptions, however, do not account for the large scale natural circulation flows in and around the containment and fuel racks. The assumption of a single well-mixed volume joining each of the bundles at the top and bottom of the racks assumes no pressure or temperature variations in this region. This idealized upper control volume provides the ultimate heat sink for these models.

Previous studies indicate that during steady-state conditions, the heat produced by the fuel is removed primarily through natural circulation flows. In addition, the largest source of uncertainty in these fuel heatup predictions is the natural circulation flow rate. The present predictions use CFD to predict the natural circulation flows that are simplified in typical spent fuel pool models. The focus of the predictions is on the three-dimensional natural circulation flowfield in and around the fuel pool, racks, and containment building. Physical models for radiation and clad chemistry are not incorporated. The predictions can be used to assess the flowfield assumptions used in other codes. The three-dimensional CFD predictions give valuable insights into the natural circulation air flow which is crucial to spent fuel pool cooling after a complete loss of spent fuel pool coolant.

The CFD model makes simplifying assumptions to represent the complex geometry of the racks and fuel. A porous region provides an equivalent flow resistance for the rack and bundles and aligns the flow in the vertical direction. A volumetric heat source adds the appropriate energy to the fluid in the active fuel regions. Predictions are obtained for steady-state conditions to determine the maximum fuel surface temperature for a given pool age and configuration. Models for radiation and clad chemistry, not included in these predictions, are considered important at elevated temperatures ($T > 600$ °C). Therefore, the current CFD predictions are more applicable at low temperatures, where these effects are minimal. At elevated temperatures, this limitation should be kept in mind.

The CFD model is sized to represent a typical BWR pool and containment building. The pool is filled to its capacity with 4200 fuel bundles of various ages in high-density racking. Predictions are made with fuel loads representative of a fuel pool 2, 3, 4, and 6 years after the reactor is shut down. Sensitivity studies are done on the ventilation rate, the outer wall heat transfer coefficient, the location of the hottest fuel, fuel burnup, the flow resistance within the racks, and heat conduction within the racks. Best estimate predictions of critical decay time (assuming burnup is 40 GWd/MTU) show that fuel temperatures remain below the temperature limits of 800 °C and 600 °C after 26 and 35 months of decay time, respectively. Predictions are made with the FLUENT CFD code. The finite volume mesh used for this model provides the resolution needed to resolve the important phenomena while remaining small enough to provide solutions with the available computer equipment. The overall size and computational complexity of the completed model preclude a grid independence study. In considering quantitative results, the limitations and modeling assumptions must be kept in mind.

Evaluation of Uncertainty in Steam Generator Tube Thermal Response During Severe Accidents

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As a result of issues raised by both the Office of Nuclear Reactor Regulation (NRR) and the Office of Nuclear Regulatory Research (RES) of the U.S. Nuclear Regulatory Commission (NRC) during the recent review of the electrosleeve steam generator tube repair process at Callaway and ANO steam generator analysis, a need to improve the understanding of the uncertainty of steam generator tube response during severe accidents has been identified. The issue of concern is related to the uncertainties on the time to failure of the reactor coolant pressure boundary components during a postulated severe accident. The focus is on modeling of high temperature and high-pressure thermal/hydraulic conditions within the primary system and the mechanical response of the structures and components that make up the primary system during these severe accident scenarios. The objective of the research is to develop an improved understanding of time-dependent thermal-hydraulic conditions in the hot leg, surge line, steam generator tubes, and other critical components such as steam generator manways, PORVs and Safety valves. To improve the understanding we will evaluate the uncertainties in the following areas:

- Accident Sequence Variations
- Plant Design Differences
- Inlet Plenum Mixing
- Tube to Tube Variations
- Core Melt Progression.

The basic approach is to build upon the analysis and understanding of severe-accident thermal-hydraulic conditions developed in NUREG-1570, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture," March 1998, and more recent analysis performed in support of the Callaway electrosleeve review. This work, in turn, had built upon previous investigations of reactor system integrity performed in conjunction with examination of unintentional reactor coolant system (RCS) depressurization for severe accidents. The SCDAP/RELAP5 code will continue to be used as the principal tool for analysis of the tube thermal hydraulic boundary conditions attendant to severe accidents. The code has received peer review for this specific application and its basic modeling approach is founded on the test program conducted at the Westinghouse 1/7th scale test facility under a cosponsored EPRI/NRC program to measure natural circulation flow during a severe accident. Even though the code has been demonstrated to be capable of calculating bulk flow conditions and circulatory flow, it is difficult to argue its inherent capability to calculate variations in fluid conditions from one tube to another. To address tube-to-tube variations, supplemental methods/approach will need to be adopted as discussed later.

<u>Accident Sequence Variations</u>: Past evaluations of steam generator tube integrity have focused on "high-dry" severe accident scenarios in which the RCS remains at high pressure, steam generator inventory is not maintained and, as a further challenge to the tubes, one or

more steam generators are depressurized. Station blackout events are symptomatically characteristic of this type of sequence. The new analysis will evaluate the effects of varying the sequences to include non-pressurizer loop and multiple loop depressurization. Analyses will be performed to address variations in reactor coolant pump seal leakage to include intermediate leakage rate behavior (past analyses have considered large leakage (175-250gpm) and low leakage rates). Supplemental analyses will also be performed to address the effects of SG tube leakage on bulk RCS and steam generator responses. Additional analysis, if determined necessary, will include effects of other slow leak-down scenarios caused by partially stuck open pressurizer relief valves, or temperature induced failure of small penetrations such as instrument taps.

Plant Design Differences: Earlier work addressed design specific factors by evaluating a variety of plant designs (e.g., Surry, Zion, ANO-2, Oconee and Calvert Cliffs). The orientation of the surge line is an important design specific factor that may affect analysis results. The vast majority of sensitivity studies, were performed, however, only for the Surry design. The new work will address uncertainties and sensitivities based on Zion plant specific analysis. This will widen the examination of uncertainties and specifically address a large class of plants. Additional plant variations may be done using the ANO plant and by developing comparisons to previous analysis done on the Surry, ANO, and Zion plants.

Inlet Plenum Mixing: Phenomenological uncertainty in the natural circulation calculation has centered on the issue of mixing in the steam generator inlet plenum, primarily because the mixing in the inlet plenum of hot outward flow from the hot leg with the cooler return flow of the tube bundle determines the fluid temperature for those tubes in the bundle carrying hotter forward flow. Additional uncertainty relates to heat transfer assumptions. The new analysis will involve a more rigorous treatment of uncertainties than was done in earlier work that included single and multiple parameter variations. Distributions will be developed for the individual mixing parameters and heat transfer coefficients and sampled using Latin Hypercube techniques. SCDAP/RELAP5 analysis will then be performed for sampled points to develop a probabilistically weighted picture of steam generator tube temperatures.

Tube-to-Tube Variations: The fluid and tube temperatures calculated for the tube bundle in SCDAP/RELAP5 analysis do not provide the discretization and resolution one might expect to see if over the thousands of individual tubes that carry hotter forward flow and cooler return flow. In order to estimate tube-to-tube variations in this flow situation, we plan to reexamine the experimental basis for the modeling, i.e., the 1/7th scale test data, to determine the appropriate variability for plant conditions. The use of more detailed Computational Fluid Dynamics (CFD) codes to predict inlet plenum mixing and tube-to-tube variations will also be used to address these issues. The use of CFD will require the code to be benchmarked against experimental data before application to plant analysis, however, CFD codes have greater inherent capabilities for solving this type of fluid flow problem.

Core Melt Progression: In addition to the phenomenological uncertainty attached to the thermal hydraulics of countercurrent natural circulation and mixing in the steam generator, there is also a general issue of uncertainty associated with severe accident progression or "core melt progression" as it is commonly known. Uncertainties in this arena relate to oxidation behavior and possible blockage and crust formation. We will reexamine the sensitivity to the rapid oxidation transient which is the dominant mechanism in heating of the reactor coolant system components.

USNRC Thermal-Hydraulics Program

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USNRC Office of Nuclear Regulatory Research has established an integrated approach to advance its capability to perform thermal-hydraulic reactor system analysis to audit vendor or licensee analyses of new or existing designs, to establish and revise regulatory requirements, to study operating events and to anticipate problems of potential significance. This approach is focused around the development of a modernized code that can be used for all transients of current designs. The existence of one code with an extensible architecture allows improvements to be made efficiently and will prevent the fragmentation of resources as well as the knowledge-base. Experimental programs are in progress with the specific aim of developing models to alleviate identified code deficiencies. USNRC staff in conjunction with external researchers will work jointly on this integrated program to provide the technological bases for regulatory decisions involving thermal-hydraulics.

The previous approach to maintaining a thermal-hydraulics system analysis capability relied on the existence of four separate codes, each with a dedicated purpose. However, the distinct differences in functionality have eroded over the years and presently, these codes are redundant in capability. The four codes were developed in the 1970's, so memory limitations and limited features of the available computer languages forced the use of cryptic programming styles. Because ingenuity was focused on overcoming these limitations, less was focused on code architecture. As a result, the codes are limited in their readability, extensibility, and ultimately maintainability. Code deficiencies and conservatism have been identified and require improvement but due to the architecture and budget reductions, making improvements to each of these codes is an inefficient prospect. To advance its current capability, USNRC is adopting a different approach by consolidating the capabilities of the suite of codes into one code to prevent the fragmentation of resources that occurs with four codes.

The base code for the consolidation is TRAC-P. The architecture of the code has been completely revamped to conform to the concept of modularity using Fortran90 language. This code is now referred to as TRAC-M to reflect the modernized architecture. The functionality of the predecessor codes are being incorporated into TRAC-M. To date, that of TRAC-B and RAMONA have been recovered and the consolidation of RELAP5 is in progress. Once completed, TRAC-M will have the ability to read both RELAP5 and TRAC-B input decks as well as legacy TRAC-P decks. At this stage, developmental assessment will be performed to select a set of constitutive relations that will allow TRAC-M to simulate the applications of the predecessor code with equivalent fidelity.

Capitalizing on the code architecture's enhanced extensibility, some work is being performed in parallel with the consolidation effort that focuses on making user requested improvements. A graphical user interface, SNAP (Symbolic Nuclear Analysis Package), will increase the ease of use of the code and help to minimize the user effect. The analysts will be provided with a graphically based input model generator as well as both two-dimensional (2-D) and three-dimensional (3-D) views onto which the data can be mapped and animated. SNAP will also provide runtime intervention so that the code can be run in a simulator-like mode. Other improvements include coupling to a 2-D and 3-D kinetics model, incorporation of an alternate matrix solver for use with large 3-D matrices will several 1-D connections, enabling an alternate less diffusive numerical scheme, and development of an Exterior Communications Interface (ECI), which facilitates coupling to processes or codes running outside of TRAC-M. The ECI is used to run the code in parallel, achieving high parallel efficiencies and a reduction in runtime.

Since the predecessor codes were known to have deficiencies in modeling some phenomena, experimental programs are underway to supplement the existing database with more detailed data using advances made in instrumentation. These models will be incorporated into the code when completed, which is estimated to range from the year 2001 to 2003. The programs include: Rod Bundle Heat Transfer Program to develop a mechanistic reflood model; Interfacial Area Transport to replace the static flow regime maps with a transport equation for interfacial area; Phase Separation at Tees to develop models over all flow regimes with data that is prototypic of reactor designs; and Subcooled Boiling at Low Pressure which will produce a model that is valid at low pressure.

USNRC staff in conjunction with external researchers will work jointly on this integrated program to provide the technological bases for regulatory decisions involving thermal-hydraulics.

Improved Radiological Consequence Assessment for Dry and Wet Storage of Spent Fuel

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The NRC has developed a more realistic analysis of offsite radiological doses for design basis evaluations of dry storage cask leakage. More specifically, the NRC estimated the radiological doses at a distance of 100 meters, due to cask leakage for normal and accident conditions, using a more realistic treatment of aerosol deposition in the cask. In accordance with the regulatory guidance in Interim Staff Guidance-5, past offsite dose assessments for dry storage cask accidents have assumed that radioactive material released from the fuel into the cask remained airborne in the cask, and therefore available for leakage, for the duration of the accident which is 30 days.

The NRC completed an in-house evaluation of the offsite doses using the RADTRAD (Radionuclide Transport and Removal and Dose Estimation) reactor accident analysis code together with values for parameters, such as cask volume and leak rate, from the Safety Analysis Report for the HI-STORM cask. The HI-STORM cask was chosen for this analysis, because it may be used at the first private fuel storage facility. The radioactive material releases from the fuel into the cask from Interim Staff Guidance-5 were used, consistent with the HI-STORM Safety Analysis Report. However, Interim Staff Guidance-5 does not credit deposition of aerosols in the cask. The primary objective of the evaluation was to provide more realistic quantification, with uncertainty bounds, of offsite doses associated with dry storage cask leakage by taking into account aerosol deposition in the cask.

For accident conditions, the evaluation showed that modeling gravitational settling of fission product aerosols in the cask reduces the offsite dose from 40 mrem to .1 mrem, lower by a factor of 400. Lower and upper bound estimates (i.e., 10^{th} and 90^{th} percentiles) for accident conditions are .03 mrem and .2 mrem, respectively. The uncertainty analysis for accident conditions considered the uncertainties in the diameter, density, and shape factor of the aerosols in the cask. For normal conditions, the evaluation showed that modeling aerosol deposition reduces the offsite dose for one leaking cask from .4 mrem to .001 mrem, lower by a factor of 400. For normal conditions, the dose reduction is limited by the small fraction of volatile fission products which can be in vapor form under normal operating temperatures and therefore not subject to aerosol deposition mechanisms. An integral evaluation of deposition using a mechanistic reactor accident analysis code such as MELCOR or VICTORIA may result in even larger reductions, as a result of considering additional aerosol deposition mechanisms beyond gravitational settling and more detailed modeling of the partitioning of the volatile fission products between condensed and vapor phases. Based on the NRC evaluation of offsite doses, it is concluded that more realistic treatment of aerosol deposition in a dry storage cask would result in a significant reduction in unnecessary regulatory burden.

As part of its effort to develop generic, risk-informed requirements for decommissioning, the NRC performed an in-house evaluation of the offsite radiological consequences of beyond-design-basis spent fuel pool accidents that concluded the short-term consequences (i.e., early fatalities) decreased by a factor of two when the fission product inventory decreased from that for 30 days to that for one year after final shutdown. Also, at one year after final shutdown, the short-term consequences may be decreased by up to a factor of 100 as a result of early evacuation. The long-term consequences (i.e., cancer fatalities and societal dose) were unaffected by the additional decay and early evacuation. The overall conclusion was that the consequences were generally comparable to those of reactor accidents.

The spent fuel pool accident consequence analysis also included sensitivity studies on ruthenium and fuel fines releases and plume spreading. These sensitivity studies concluded that, with the exception of the ruthenium release fraction, the parameters varied did not sufficiently impact the results, nor change the conclusion that the consequences were generally comparable to those of reactor accidents. Increasing the ruthenium release fraction from that for a non-volatile ($2x10^{-5}$) to that for a volatile (.75) resulted in a large increase in both short-term and long-term consequences due to ruthenium's high dose per curie inhaled. However, consequence increases from ruthenium were demonstrated to be largely offset by early evacuation. These sensitivity studies also found that using updated values for plume-spreading model parameters resulted in up to a 60% increase in long-term consequences. However, similar increases are expected when these updated values are used to calculate reactor accident consequences. Using updated values also resulted in up to a factor-of-15 decrease in short-term consequences.

Consolidation of Severe Accident Code Capabilities into MELCOR

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In 1957, one of the first assessments of the potential hazards associated with hypothesized severe accidents in nuclear power plants was described in the report titled WASH-740. It was recognized in WASH-740, as well as in the subsequent landmark WASH-1400 Reactor Safety Study (RSS) that severe accident progression was a complex assemblage of coupled physical phenomena that would require development of complex computer codes in order to render realistic and not overly-conservative estimates of severe accident progression and their consequences. An important realization of the RSS was that risk could be strongly influenced by low likelihood, high consequence (severe) accidents, but it was not until the TMI-2 event that severe accidents were elevated from the realm of the *hypothetical* to that of *reality*. The accident at TMI subsequently prompted two decades of vigorous NRC and industry research into the science of analyzing severe accidents in nuclear power plants.

From the beginning, analyses of severe accidents were conducted on two levels, or tiers, both involving the development of computer codes to predict the physical processes. One tier was a detailed physics tier wherein a narrow range of severe accident phenomena was modeled in high mechanistic detail. These models were often derived in conjunction with experimental programs focused on a particular severe accident phenomenon. The other tier was a system-level tier where the various models of the mechanistic tier were integrated to provide a self-consistent analysis of a complete plant accident. The system-level models often initially used simplified, parametric versions of the mechanistic tier models. However, in time, as the mechanistic tier models became validated and generally accepted (and as computing capabilities advanced) they have also been gradually incorporated, or consolidated, into the system-level codes. As a result of this consolidation process, today the system-level code models are nearly as detailed as those of the mechanistic tier codes. Today, the NRC system-level code with by far the broadest phenomenological scope is MELCOR.

Presently, the NRC is moving to complete the consolidation of severe accident knowledge and understanding into the MELCOR integrated accident analysis code in order to: 1) make the best use of limited research resources, 2) preserve the knowledge and understanding of severe accident phenomena and 3) enable easier access and application of this knowledge to current and future safety issues. Progress towards consolidating the knowledge in modeling containment phenomena (CONTAIN code capabilities) is described in this paper as well as current modeling improvements in the areas of core melt progression and the prediction of natural circulation effects in the reactor coolant system piping (SCDAP/RELAP-5 capabilities). Future applications of the MELCOR code are anticipated to include providing support to future level 2 PRA activities, risk-informing present and future regulations (eg. 10CFR50.44), and providing technical support to NRC decision-making processes.

Integrity of the Reactor Coolant Pressure Boundary

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Deregulation of the electric power industry in the United States has fundamentally changed the economic factors that govern the continued operation of nuclear power plants (NPPs). Before deregulation NPPs, which provide primarily baseload, were paid based on capacity. This compensation strategy changes fundamentally in a deregulated environment where NPPs must be cost competitive with other energy sources. Utility executives are therefore considering operational and repair scenarios that were unheard of as little as five years ago. These can include up-rating of the reactor, removal of flux suppression, and pursuing novel steam generator repair strategies as an alternative to simply plugging leakers. These actions, and the technologies used to justify them, can challenge conventional regulatory processes because new technologies are brought into application more quickly than has been the case in the past.

In this session we focus on two areas of current interest to illustrate these challenges. One area is the "Master Curve," a technology that provides a new way to describe the fracture toughness of the reactor pressure vessel. By moving away from the correlative / empirical methodologies in use today toward a more rigorous characterization of fracture behavior, the Master Curve offers the potential for greatly improved accuracy over current approaches. Nevertheless, acceptance of this approach has been slowed, at least in part due to a lack of an overall framework to apply Master Curve technology to a vessel integrity calculation, and due to an incomplete understanding of the synergistic relationships between various inputs to a reactor vessel integrity assessment and how these relate to (or influence) the level of conservatism in the end result.

The second area of focus includes new strategies for steam generator tube repair and inspection, as well as changes in the regulatory standards against which tube repairs are judged. Questions that arise include the reliability of inspector training, the uncertainties associated with certain inspection techniques, and the integrity of tube repair strategies under the non-design basis / severe accidents that need to be considered to be consistent with a risk informed framework.

This session features two presentations for each of these two topic areas. It is hoped that by reviewing both current research advances and current applications these papers will seed discussion that will provide insight regarding roadmaps for new technology implementation that can be successful in the future.

Research Perspectives on the Evaluation of Steam Generator Tube Integrity

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Steam generators have been the most troublesome of the major components in pressurized water reactors (PWRs). Tubes have been degraded by corrosion and wastage, pitting, denting, stress corrosion cracking, and intergranular attack. Industry efforts have been largely successful in managing degradation due to wastage and denting, but stress corrosion cracking and intergranular attack remain problems. Such cracking typically occurs in defect prone areas like crevices at the tube sheet and tube support plates where the local water chemistry can be more aggressive than that in the bulk or in places like the roll transition at the tube sheet where high residual stresses occur. To ensure the structural and leak integrity of degraded steam generators, one must be able to evaluate and characterize degraded tubes, evaluate structural integrity and leakage associated with degraded tubes, and characterize progress of degradation over the next inspection cycle.

These enabling technologies are the focus of an NRC research program on steam generator tube integrity at Argonne National Laboratory. This research is needed to permit the NRC to independently assess the adequacy of industry programs to ensure the integrity of steam generator tubes as degradation proceeds, new forms of degradation appear, and as new degradation management schemes are implemented.

The modes of SG degradation have changed from wastage in the early 70s to denting in the late 70s and 80s to stress corrosion cracking. These different modes of degradation have different impacts on integrity and require different kinds of NDE, material behavior, and structural integrity information.

The changes in degradation mode have lead to changes in inspection technology from simple single frequency bobbin coils and instruments to multiple frequency instruments and complex rotating and array probes used for current day inspections. While new probe designs such as the +Point clearly represent improvements in inspection technology, problems still remain. At present, the interpretation of NDE inspection data is somewhat subjective and depends strongly on the experience of the analyst. The capability to detect defects has far outstripped the capability to size the defects or otherwise characterize their effect on structural integrity. Improvements are needed in flaw sizing capability and in POD for flaws in areas of high background noise.

Personnel qualification and training also have enormous impacts on the effectiveness and reliability of inspections. The use of performance demonstration has lead to increased confidence in inspection reliability. However, it is not clear that the level of performance achieved by current requirements is consistent with industry and regulatory expectations. The round robin testing program being carried out under the NRC research program will help to quantify the inspection reliability of currently used methods of inspection. The potential impact of changing the performance demonstration requirements, e.g., increasing the passing grade from the current 80% for flaws 60%TW or deeper, can be considered in light of the results quantify the inspection reliability of currently used methods of inspection. The potential impact of changing the performance demonstration requirements, e.g., increasing the passing grade from the current 80% for flaws 60%TW or deeper, can be considered in light of the results currently being achieved.

Replacement steam generators using Alloy 690 tubing and incorporating a number of design improvements appear to be performing much better than the original designs using Alloy 600. However, laboratory studies suggest that Alloy 690 may not be highly resistant to cracking under some water chemistry conditions that may be of interest. Thus inspection requirements for replacement steam generators with Alloy 690 tubing need to be determined to assure early detection of any onset of degradation.

Much research has been done on the effect of flaws on the structural and leak integrity of steam generator tubing. However, much of this work has focused on the single planar flaw—a defect morphology that is not characteristic of much of the cracking that is currently being observed in steam generators. More typical crack morphologies show groups of cracks or segmented cracks. Assessing the integrity of such crack morphologies places additional burdens on NDE and the evaluation of structural integrity. By providing paths for current flow, the presence of ligaments can make cracks more difficult to detect and characterize. Depending on its size, the ligament can have a significant strengthening effect thus making the usual assumption of a planar bounding crack very conservative. On the other hand, if NDE procedures overestimate the size of the ligaments, calculations of structural capacity based on the NDE results could be nonconservative. Because the ligaments between adjoining cracks are relatively highly stressed, the failure behavior could be time (or rate) dependent even during design basis accidents like a MSLB or during condition monitoring tests to determine whether tubes meet the $3\Delta p$ criterion.

Although most prior work has focused on the potential for tube failure during design basis accidents like a MSLB, risk studies show that much of the risk due to steam generator tube failures is due to tube failures due to severe accidents during which tube temperatures can increase to 650–750°C. Under such conditions creep becomes an important failure mechanism for the tubes and the potential for increased leakage through flaws due to opening of existing throughwall flaws by creep deformation must be considered.

Technical Issues Stemming From Recent Steam Generator Experience at Arkansas 1 and Indian Point 2

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Steam generator tube degradation at Indian Point 2 and Arkansas 2 have led to instances where certain individual tubes failed to exhibit adequate structural margins against burst. This situation only came apparent at Indian Point 2 as a result of a tube failure event on February 15, 2000. At Arkansas 2, this situation was revealed during in-situ pressure testing conducted during inspection outages in January 1999 and again in November 1999. The staff has reviewed the circumstances of the incidents and has identified a number of technical issues which need to be addressed by industry to avoid similar such incidents in the future. These issues relate to inservice NDE inspection of the tubing, in-situ pressure testing, and operational assessment methodologies for demonstrating that adequate tube integrity will be maintained until the next scheduled inspection.

EPRI MATERIALS RELIABILITY PROGRAM: MASTER CURVE ACTIVITIES

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ABSTRACT

Recent nuclear industry attention has been focused on the direct use of measured fracture toughness properties in the assessment of RPV integrity. Specifically, efforts have been initiated to develop procedures for determining the material transition temperature based on measured fracture toughness testing (T_0) using the Master Curve approach. The direct determination of material fracture toughness, and a transition temperature associated with measured fracture toughness, represents a more precise measure of material resistance to crack initiation than earlier methods and should provide a more realistic assessment of RPV integrity. The EPRI Materials Reliability Project (MRP), through the Reactor Pressure Vessel Integrity Issue Task Group (RPV Integrity ITG) is an active participant in the coordination of U.S. industry activities supporting the application of the Master Curve approach for RPV integrity assessment. The primary goal of the Master Curve program is to resolve the various technical issues associated with application of the Master Curve approach for use in RPV integrity assessment. The Master Curve approach is a demonstrated technically superior method to assess RPV material condition. The objective of the MRP activity is to validate implementation of modern fracture toughness testing technology to permit direct measurement of vessel embrittlement. More specifically, the goal is to facilitate characterization of lower bound toughness in the transition region of irradiated steel using precracked Charpy specimens. The program consists of: proof of principal analysis; development of a physical basis for the master curve; facilitation of the development of testing and implementing standards and codes; development of plant specific submittal strategies; resolution of technical issues; evaluation of margins; and evaluation of the effect of use of the Master Curve on vessel risk. This paper provides a brief overview of the EPRI MRP program, the RPV Integrity ITG, and industry activities associated with development of the Master Curve approach for RPV integrity assessment. Specifically, this paper will provide details on the margins evaluations being performed to support application of the Master Curve.

NRC REVIEW OF THE TECHNICAL BASIS FOR USE OF THE MASTER CURVE IN EVALUATION OF REACTOR PRESSURE VESSEL INTEGRITY

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In the United States, the Nuclear Regulatory Commission (NRC) licenses utilities to operate nuclear reactors for a 40-year term. During this time, the fracture toughness of pressurized water reactors (PWRs) must be adequate to maintain vessel integrity during both routine operations, such as heat up and cool down, as well as during postulated accident (i.e. pressurized thermal shock, or PTS) events. The fracture toughness of reactor pressure vessel (RPV) steel provides a key input to both of these calculations. Currently, the ASME K_{IC} or K_{IR} curves, indexed to RT_{NDT}, are used to describe the fracture toughness of the RPV, and how fracture toughness varies with temperature. RT_{NDT} is determined as per ASME NB-2331 through a combination of Charpy V-notch and *NDT* testing. It is widely recognized that this RT_{NDT} -indexed K_{IC} / K_{IR} characterization of fracture toughness is conservative (i.e. under-represents the true fracture toughness) to varying amounts. The procedure is conservative because of the protocols of ASME NB-2331, and because only linear elastic fracture mechanics (LEFM) valid data were used to establish the position of the bounding K_{IC} and K_{IR} curves. The procedure is conservative to varying amounts because the tests used to establish RT_{NDT} (i.e, CVN and NDT tests) can only be correlated with fracture toughness, they do not represent fracture toughness itself.

Developments since the early 1970s set the scene for fundamental improvements to these correlative techniques. In 1980 Landes and Schaffer noticed a statistical "size" effect for specimens failing by transgranular cleavage. They demonstrated that larger specimens fail at lower toughness values, even when the severe size requirements of LEFM are satisfied. Beginning in 1984, Wallin and co-workers from VTT in Finland combined this "weakest link" size effect with micro-mechanical models of cleavage fracture. Wallin proposed a model that accounts successfully for specimen size effects, and provides a means to calculate statistical confidence bounds on cleavage fracture toughness data. These concepts, combined with Wallin's observation (made first in 1984, and reinforced in 1991) that all ferritic steels exhibit the same variation of cleavage fracture toughness with temperature, gave birth to the notion of a "master" transition curve for all ferritic steels.

ASTM recently passed a standard (E1921-97) that describes how to measure the Master Curve index temperature, T_o, based on limited replicate testing. E1921-97 also incorporates a modern understanding of elastic-*plastic* fracture mechanics, and so permits determination of T_o using specimens as small as precracked Charpys. Recently, ASME published Code Case N-629 that permits use of a Master Curve-based index temperature ($RT_{To} \equiv T_o$+35°F) as an alternative to traditional methods of positioning the ASME K_{IC} and K_{IR} curves. The potential for characterizing the entire fracture mode transition based on direct fracture toughness measurements using specimens already in nuclear RPV surveillance programs (ASTM E1921), and the ability to used this information to estimate a RT_{NDT}-like quantity (ASME CC N-629), has sparked considerable interest at some electric utilities to use Master Curve technology in support of both licensing and plant life extension activities.

In this document we summarize the technical basis for Master Curve technology, particularly as it applies to irradiated RPV steels and weldments. Additionally, we discuss application issues (establishment of margins to account for uncertainty, for example) that arise when the Master Curve technology for characterizing fracture toughness is applied to assess the fracture integrity of nuclear reactor pressure vessels.

Reactor Decommissioning

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Background: All reactors will eventually undergo decommissioning. After the removal of spent fuel and high activity internal components for disposal at high-level and low-level waste disposal facilities, large volumes of material will remain that may have surface or volume contamination. Clean-up technologies will be applied to remove significant surface contamination for disposal. The Nuclear Regulatory Commission's Radiological Criteria for License Termination states that

> "A site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from natural background radiation results in a TEDE [Total Effective Dose Equivalent] to an average member of the critical group that does not exceed 25 mrem (0.25 mSv) per year, including that from groundwater sources of drinking water, and that the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA)...." (Excerpt from 10 CFR 20.1402)

Regulatory decisions on whether the decommissioning criteria are being met will require knowledge about the distribution of residual contamination at a site at the time of license termination as well as the evolution of the distribution of that material for the period of regulatory interest, i.e. 1000 years. The decommissioning research program is developing the technical basis to support these decisions for reactors and non-reactor contaminated sites since the technical issues are similar once high activity waste has been removed.

Credible assessment for a period of 1000 years of the doses to people from residual contamination which may migrate through the biosphere is necessary to provide confidence in regulatory decisions about the release of contaminated sites for alternative use. Release from regulatory control will also make it possible that materials from such a released site could be used for other purposes at different sites (reuse).

NRC performance goals in the Nuclear Waste Safety Arena related to decommissioning include

- Maintain safety, protection of the environment, and the common defense and security.
- Make NRC activities and decisions more effective, efficient, and realistic.

In order to reach these goals research has been conducted in the past to develop analytical tools to simulate the long term effects of residual contamination from sites with very low levels of contamination. These tools continue to evolve and consist of conceptual and mathematical models of natural and man-made systems which are themselves interpretations and extrapolations of limited data collected to characterize those systems. The models include representations of the contaminant source term, driving forces such as infiltration and recharge events, hydrogeologic flow, engineered barrier performance, geochemical effects, biotic uptake, human exposure, and others. One aspect of current research is focused on developing a graded set of integrated models that will eventually be able to address problems with a degree of sophistication and realism that is appropriate to the level and distribution of the contamination and the complexity of the site. A second aspect of current research is methods and techniques for obtaining objective and defensible measurements of residual radioactivity.

Discussion: The current decommissioning research being discussed in this session focuses on two broad questions:

- What information is needed to make realistic estimates of radiation dose due to residual radioactivity at decommissioned sites?
- What tools are needed to effectively implement NRC requirements?

The topics to be presented address two aspects of these questions. Progress on improvements to the integrated model sets will be discussed in the context of reports on: (1) NRC development of probabilistic versions of the RESRAD and RESRAD-BUILD codes which were originally developed for the DOE by Argonne National Laboratory for use in site-specific dose calculations with regard to various decommissioning scenarios; (2) improvements to the DandD code developed by SANDIA National Laboratories to support screening analyses for the license termination of simple sites with minimal contamination; and (3) progress on SEDSS, also developed by SANDIA but designed to be a flexible platform that can accommodate the addition of more complex and realistic modules to handle the complexities of real sites and more complicated distributions of contamination. The second aspect to be addressed will be the optimization of data collection and interpretation in order to design effective and efficient surveys of contaminated sites where the contamination is distributed in three dimensions and not just on surfaces. The work being discussed includes work at ORISE on survey methodologies for volumetric contamination in inaccessible or complex geometries using current detection technology and complementary work at EML on the statistical design of the surveys and the potential contribution of state-of-the-art measurement techniques. This problem is of particular significance to decisions regarding the proposed unrestricted release of sites where materials on-site may be reused. Scenarios for such reuse must include accurate and cost effective determination of the location and amounts of contamination. A second perspective will be given to the session by an industry presentation on areas where the industry perceives a need for research to support decommissioning decisions.

In addition to the work on integrated models and survey methods the NRC research in this area presently includes projects addressing the uncertainties inherent in modeling flow in natural hydrogeologic systems, work directed toward understanding and modeling the geochemical processes that retard or enhance the movement of radionuclides in natural systems (including participation in the OECD/NEA Sorption Project), and studies of the long-term performance of engineered barriers and barrier materials (of potential value with regard to proposals for entombment). All of the process work is directed toward inclusion in SEDSS or a similar flexible platform that can implement process models when appropriate.

Future research will continue to refine process models and begin to address methodologies for monitoring in those cases where restricted release is the ultimate decision. A recent National Academies evaluation of DOE activities in clean-up and decommissioning identified institutional issues with regard to long term stewardship of sites with significant residual contamination. This can be anticipated to be a problem for non-DOE sites as well and effective monitoring plans and systems will be an associated issue that may require research.

NEEDED RESEARCH TO SUPPORT DECOMMISSIONING – AN INDUSTRY PERSPECTIVE

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Nuclear Energy Institute

Industry appreciates the opportunity to provide input to NRC research initiatives. In these times of reduced budgets, both Federal agencies and the private sector must optimize available resources. Each research dollar must be targeted to those proposals with the greatest potential to improve the safety and/or efficiency of nuclear technology companies.

In three important ways, the discipline imposed by limited resources, while sometimes painful, can be healthy and often yields high quality results. First, the approved research project will be focussed on solving a real problem of high priority to the organization. Second, the urge to re-invent the wheel will be controlled and research projects will be tailored to adapt and build on the huge body of excellent research available globally. Third, limited resources will encourage innovation and collaboration between government agencies and private sector resources.

NRC's Office of Regulatory Research has in fact accomplished just this outcome in the RESRAD revision project. Both industry and the agency recognized an important need for a dose assessment code for license termination of nuclear facilities with complex sites. The need was urgent as many facilities were in the process of developing decommissioning/license termination plans. Rather than start from scratch, NRC-RES decided to modify an existing code, RESRAD, to meet their specific needs. Finally, NRC-RES agreed to collaborate with EPRI to review the draft code. This type of approach has a greater probability of yielding a quality product within a realistic schedule and at a reasonable cost.

In looking to the future, industry has identified decommissioning issues, pertinent to Session 3A of this meeting, in the following areas that could benefit from additional research:

- In support of dose assessment
  - Sub-surface contaminated soil
  - Embedded pipe / volumetric contamination or inaccessible contamination in buildings
  - Parameter selection/justification for RESRAD

- In support of the GEIS supplement
  - In-situ concrete rubble disposal
    - Intruder/ exhumation scenario development
    - Isotope migration from concrete matrix
  - Enhanced Safestor (entombment)
    - Passive water barriers
    - Optimizing design
    - Realistic duration of engineered barriers

- In support of material clearance rulemaking
  - Harmonizing national & international dose assessment within concept of trivial risk
  - Realistic inventory of materials for clearance
  - Realistic industrial landfill disposal scenarios

- In support of transportation
  - Activity based equivalent for 1R/hr at 3 meters
  - Equivalent activity limit for LSA/SCO shipments at comparable risk

Each of these identified issues contains opportunities to provide clarity, ease implementation, reduce unnecessary burden, while maintaining or improving safety.

NEI understands that the agency has plans to address many of the issues identified. As the ultimate user of the final products, the industry stands ready to assist your efforts to focus the limited resources available on research that is targeted to provide the greatest return on investment.

NRC CODE DEVELOPMENT IN SUPPORT OF THE LICENSE TERMINATION RULE

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The Office of Research (RES) at the U.S. Nuclear Regulatory Commission (NRC) has been developing analytical tools to demonstrate compliance with the License Termination Rule. These tools include two versions of DandD, developed by Sandia National Laboratories, a revision RESRAD, developed by Argonne National Laboratory, and SEDSS, under development by Sandia National Laboratories. This paper will focus on DandD and SEDSS.

The DandD code implements the dose-assessment models developed in volume 1 of NUREG/CR-5512 to enable deterministic screening of sites for compliance with the rule. DandD (version 1) software was released in August of 1998, with the user's guide and parameter analysis documentation released in 1999. For compliance demonstration, the deterministic structure of version 1 required a combination of default parameter values resulting in a degree of excess conservatism. In August 1999, RES initiated development of a probabilistic version of DandD (version 2) that would not be encumbered by the restrictive default parameterization. Version 2 was released to the public in August of 2000 as a probabilistic tool for screening. Development is progressing on a revision for limited site-specific analysis.

SEDSS is a tool under development for site-specific analyses aimed primarily for those complex sites that are beyond the range of applicability of DandD and RESRAD. Those sites with existing ground-water contamination would be among this group of complex sites. SEDSS is designed to encompass a suite of models of varying complexity to enable multimedia dose analysis with models appropriate for the complexity of each individual exposure pathway. SEDSS also implements the NRC Decision Framework described in NUREG 1549.

DEVELOPMENT OF RESRAD PROBABILISTIC COMPUTER CODES FOR NRC LICENSE TERMINATION APPLICATIONS

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In 1999, the U.S. Nuclear Regulatory Commission (NRC) tasked Argonne National Laboratory to modify the existing RESRAD and RESRAD-BUILD codes to perform dose analysis for use with the NRC's license termination compliance process of the Standards Review Plan regarding site-specific applications. The RESRAD codes have been developed by Argonne to support the U.S. Department of Energy's (DOE's) cleanup efforts. Through more than a decade of application, the codes already have an established large user base in the nation and a rigorous QA support. The primary objectives of the NRC task are to: (1) extend the codes' capabilities to include probabilistic analysis, and (2) develop parameter distribution functions and perform probabilistic analysis with the codes. The codes also contain user-friendly features specially designed with graphic-user interface. In July 2000, the revised RESRAD (version 6.0) and RESRAD-BUILD (version 3.0), together with distribution parameters and other relevant information, have been developed and also are on distribution to the general public for beta testing and use.

Survey Methodologies for Volumetrically Contaminated Material: Surveying for Radionuclides in Inaccessible or Complex Geometry Materials

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Many times the materials planned for clearance cannot be surveyed directly due to the problem of inaccessible areas. This may include materials that have inaccessible surfaces or items that are buried, such as underground or embedded piping. The Oak Ridge Institute for Science and Education (ORISE) has been tasked with the development of survey and analytical methods to measure residual radioactivity in inaccessible areas to assure that applicable criteria have been met. Special emphasis has been placed on mixtures of clearance materials, such as embedded pipe in concrete, buried materials in soil, and heterogeneous mixtures of soil.

ORISE has considered the use of simulations to supplement the performance of survey methods for inaccessible materials. One example that has been implemented recently was to use MicroShield simulations of pipe geometries to convert radioactivity within the pipe to exposure rates. For example, for a given uniform distribution of Cs-137 activity on the interior pipe surface, the resulting gamma exposure rate at the desired measurement location can be determined using MicroShield. This computer model accounts for the source strength, geometry, and attenuation produced by the pipes and intervening soil thickness.

This simulation approach was implemented at a decommissioning site that contained an underground contaminated pipe. The 60 m long, 15-cm diameter pipe, was located approximately 1 meter below grade and contained varying levels of Cs-137. PVC pipes were installed perpendicular to the pipe for the purpose of assessing the gamma radiation being emitted from the pipe. The PVC pipes were open at ground level and were about 2.5 cm from the pipe at the subsurface level of the pipe. A buried PVC pipe was also installed in the ground in a non-impacted portion of the site to serve as a background measurement location.

Exposure rate measurements were then used to estimate the Cs-137 activity for a particular activity distribution. Due to size limitations at the measurement locations, NaI scintillation detectors were used to measure the gamma radiation, and calibration factors were established to convert the net count rate to exposure rate. The measured exposure rate can then be related to the modeled exposure rate to assess the quantity of Cs-137 in the pipe.

A number of other survey technologies will be discussed that relate to the survey of materials with inaccessible areas. These include the use of *in situ* gamma spectrometry measurements on equipment and materials, long range alpha detectors (LRADs) for measurement of inaccessible alpha contamination, and use of TLDs and small detectors for embedded piping.

ISSUES IN THE DESIGN OF SURVEYS
FOR VOLUMETRIC SOURCES OF RADIOACTIVITY

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The MARSSIM (NUREG-1575) has addressed the conduct of final status surveys for residual radioactivity in surface soil (top 15cm) and building surfaces (SS&BS). Extending the guidance to include volumetric sources raises a number of important issues. Some of these will apply to both subsurface soil volumes and to building debris, machinery and materials. Indeed, some of these issues have already arisen in SS&BS cases where compliance with an activity concentration limit rather than a dose limit is specified, but a MARSSIM survey design approach is desired.

Issue #1 MARSSIM surveys are designed to determine compliance with a release criterion (DCGL) corresponding to a specified dose limit. This has not been decided for the case of volumes of material that will be removed from the site, which may be subject to an activity (or detectability) limit. There is a generic problem with such limits in that the size of sample and/or method of measurement must also be specified before an adequate survey can be designed.

Issue #2 Volumes of material can be classified in a manner similar to that used for MARSSIM survey units, i.e. by the likelihood that concentrations in excess of the DCGL (or activity limit) will be encountered. However, there is must be a method for determining the appropriate size for a survey unit. In MARSSIM, survey unit size (for Class 1 and Class 2) is related to the size of the contaminated area assumed in the dosimetric model used to determine the DCGL.

Issue #3 How should a "volume of elevated of activity" be defined? This would logically be a volume of activity that contains a concentration of activity that would result in a dose in excess of the release criterion. If the release criterion is activity based, another basis for defining an "elevated volume" is needed.

Issue #4 In a MARSSIM integrated survey design, elevated areas can often be detected by scanning. "Elevated volumes" will be more difficult to detect if contamination is located deeper than the range or mean free path of the radiation emitted.

These and other issues that arise in the design of surveys of volumetric sources of radioactivity will be discussed along with some approaches that might help to resolve them.

Regulatory Effectiveness Assessment and Improvement

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Introduction/Background

The NRC has a strategic plan which was developed after a substantial effort that included publishing documents called Direction Setting Issues (DSIs). A significant part of the DSIs related to the concepts of risk-informed and performance-based approaches to regulation. The Commission provided extra emphasis to these concepts by issuing a White Paper in March, 1999 (SRM to SECY-98-144). The combination of the direction provided in the DSIs, the White Paper and the NRC Strategic Plan has motivated the Office of Nuclear Regulatory Research (RES) to create an area of activity that will provide the needed technical basis for accomplishing certain aspects of the Commission's performance goals to increase effectiveness, efficiency and realism and to reduce unnecessary regulatory burden.

Discussion of Current Activities

Effective regulatory requirements are those that deliver the expected outcomes relative to safety. Much of the NRC's current regulatory framework is prescriptive and was developed using traditional deterministic engineering approaches involving conservative assumptions. The framework provides significant safety margins but has come to be recognized as also involving significant and, in some cases, excessive regulatory burden on licensees and certificate holders. The Commission also recognized that regulatory improvements would enable NRC's own resources to be focused on the most safety-significant issues while providing flexibility in how licensees meet NRC requirements.

RES determined that one of the first steps toward improving the effectiveness of regulatory requirements should be to assess the effectiveness of a few important existing regulations. The assessments were based on a definition that the metric for effectiveness would be a comparison between the expectations and outcomes relative to the promulgated regulation. The rulemaking process provides a fairly clear depiction of why a regulation is required, how compliance can be achieved, and approximately how much cost would accrue to the licensed industry. The parameters associated with each of these factors represent the expectations side of the effectiveness assessment. As operational experience is accumulated by the licensed industry, operational data (including costs actually experienced by licensees) provides a basis for estimating the level of outcomes. RES is using publicly available information to develop comparisons in such a way that they represent consistent and meaningful evaluations.

The work on assessing the effectiveness of the Station Blackout rule has shown that a considerable effort was required to delineate the expectations and outcomes. It was recognized that effectiveness assessments for future regulations could be made more efficiently if the parameters that represent the set of expectations are more clearly made an integral part of the rulemaking process. Hence, developing guidelines for future regulatory activity was seen as a part of improving effectiveness.

The high-level guidelines for performance-based activities were developed, in part, to serve this need for regulatory improvement that will be classified as being performance-based. The guidelines, which use the definitions in the White Paper extensively, also provide the bridge to risk-informed and traditional regulatory activity. They are meant to be applied agency-wide to all the three arenas of NRC regulatory responsibility.

Additionally, RES has determined that the most efficient means for identifying candidates for reduction of unnecessary regulatory burden is to query licensees directly. In cooperation with the Office of Nuclear Reactor Regulation (NRR), RES sought such information directly in a public meeting from the company Commonwealth Edison, which operates a number of nuclear power plants. NRR played a very valuable role because they have the direct responsibility for regulatory oversight of nuclear power plants. NRC will be using the information obtained to develop the optimal technical basis for potential changes to regulatory requirements which would reduce unnecessary burden.

RES recognizes that the NRC must accomplish its goals in a dynamic environment. Chief among the changes visible over the horizon is deregulation of the electric power industry. The NRC cannot assume that the existing infrastructure of regulatory requirements will continue to provide the safety outcomes as huge changes occur in plant ownership and transmission systems. Although there does not appear to be any need for significant changes to the regulatory approaches, it is possible that the opportunities for regulatory improvement will arise from different sources as compared with the situation that existed just a few years ago.

Future Activities

RES plans to draw on the success of the effectiveness assessment of the Station Blackout rule and apply the methodology to the regulation on Anticipated Transients Without Scram, as well as other regulations. The high-level guidelines for performance-based activities have been recently submitted to the Commission. The ACRS has provided a favorable review of the guidelines, and if the Commission does the same, the implementations efforts on the guidelines will intensify. Meanwhile, RES will play an important role in anticipating future challenges, and exercise the foresight for developing needed tools and data necessary for sound decision-making on the part of the NRC.

Regulatory Effectiveness: What It Is and What It Shows for Station Blackout Rule

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The Nuclear Regulatory Commission's (NRC) Office of Nuclear Regulatory Research is reviewing the regulatory effectiveness of selected regulations, starting with station blackout (SBO) rule, to determine if the requirements are achieving the desired outcomes. This initiative is consistent with NRC performance goals to make NRC activities and decisions more effective, efficient, and realistic.

This paper provides an executive level summary of the "Final Report, Regulatory Effectiveness of the Station Blackout Rule," August 14, 2000, to show what regulatory effectiveness is and what it shows for the SBO rule. To assess the effectiveness of the SBO rule the regulatory expectations were compared to the outcomes. A set of baseline expectations was established from the SBO rule, and related regulatory documents in the areas of coping capability, risk reduction, emergency diesel generator reliability, and value-impact. The corresponding outcomes were developed based operating experience, and plant specific risk and reliability studies since the SBO rule was issued. Comparison of the SBO rule regulatory expectations to the outcomes concluded that although there are opportunities to improve the clarity of SBO related regulatory documents, the SBO rule is effective and the industry and the NRC costs to implement the SBO rule were reasonable considering the outcome.

As a lessons learned, to the extent that the NRC staff revises existing regulatory documents to be more risk-informed and performance-based, they may need to be modified to ensure consistent interpretation and use of terms, goals, criteria, and measurements. In addition, new regulations or the accompanying regulatory documents should include quantitative objectives to facilitate evaluation of its regulatory effectiveness.

HIGH-LEVEL GUIDELINES FOR PERFORMANCE-BASED ACTIVITIES

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This paper describes the development of high-level guidelines to identify and assess the viability of making components of the regulatory framework performance-based, consistent with direction from the Commission. The guidelines, their relationship to the risk-informed program, and the results of test applications of the guidelines are described. These guidelines can be applied to regulatory activities to identify and assess the use of performance-based regulatory approaches instead of prescriptive criteria to assure safe performance, and as such, should help to increase reliance on performance-based regulatory approaches throughout the agency.

The guidelines are intended to promote the use of a performance-based regulatory framework throughout the agency. In general, a performance-based regulatory approach focuses on results as the primary basis for regulatory decision-making and as such allows licensee flexibility in meeting a regulatory requirement. This in turn, can result in a more efficient and effective regulatory process.

Internal and external stakeholders have commented on the guidelines and their comments have been addressed in the development of the guidelines. Specifically, the staff has addressed concerns among some stakeholders that a performance-based regulatory framework would focus only on reductions in regulatory burden and that public health and safety would lose emphasis. The staff notes that a performance-based approach is intended to focus the regulatory framework on desired outcomes and would be applied in conjunction with the agency's defense-in-depth principles as articulated in the Commission's White Paper, "Risk-Informed and Performance-Based Regulation," SRM to SECY-98-144 (White Paper). The staff has used definitions from the White Paper for terminology such as "deterministic analyses," "risk insights," and "performance-based approach" in developing the guidelines. Consistent with the NRC's Strategic Plan and the White Paper, the guidelines are to be applied across the full spectrum of materials, processes, and facilities regulated by the NRC.

Application of the guidelines requires that the nature of the regulated activity and the safety issues be defined with specificity. To explore how such challenges can be met in practice, the staff selected two issues to test the guidelines. For each issue, an NRC panel was formed consisting of experts on the specific regulatory issue. The first issue is related to the ongoing effort to risk-inform 10 CFR 50.44 (Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors). Although the hypothetical regulatory change is thought to be plausible, it must be considered purely illustrative at this time while the alternatives that will be proposed for revisions to 10 CFR 50.44 are still under consideration. The second issue involves a recent change that was made to Subpart H (Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas) of 10 CFR Part 20. In this case, the guidelines were applied retrospectively for illustrative purposes. The results of tests clearly support the utility of the high-level guidelines.

Based on feasibility testing of the guidelines, the staff concludes that they can be used to effectively focus the regulatory framework to be more performance-based by:

(A) Identifying the components of the regulatory framework which can be made more performance-based. Note, the regulatory framework consists of the regulation and its supporting regulatory guides, standard review plans, technical specifications, NUREGs, and inspection guidance.

(B) Selecting or formulating performance parameters and associated performance criteria appropriate to the regulatory issue being addressed. For example, they facilitate identifying the level (i.e., component, train, system) at which performance criteria should be set.

Having established the feasibility of the guidelines, the staff plans to:

- Apply the guidelines in ongoing or future approved rulemakings, as appropriate.

- Apply the guidelines to ongoing regulatory efforts under Option 3 of SECY-98-300, "Options for Risk-Informed Revisions to 10 CFR Part 50."

- Apply the guidelines to suitable candidates identified as being not appropriate to be risk informed pursuant to the "Risk-Informed Regulation Implementation Plan" (SECY-00-0062, March 15, 2000).

- Develop a management directive to support agency-wide implementation of the guidelines in ongoing or future approved rulemakings and other regulatory activities as appropriate (e.g., the inspection process). Supporting guidance at the office level will occur through office letters;

- Develop a communications plan to promote broader awareness of performance-based approaches on the part of external stakeholders. Wider acceptance of the guidelines should lead to efficiencies and an overall increased level of performance-based activities.

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

11. ABSTRACT *(200 words or less)*

This contains summaries of papers to be presented at the Twenty-Eighth Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 23-25, 2000. 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.

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Monday 8:00 am	Plenary Session and Expert Panel	*Grand Ballrooms B,C,D&E*
12:00	MIT-Sponsored Luncheon	*Congressional Ballroom*
1:30 pm	Plenary Session and Expert Panel	*Grand Ballrooms B,C,D&E*
3:45 pm	Session 1	*Grand Ballrooms B&C*
	Session 2	*Grand Ballrooms D&E*
6:30 pm	National Labs Sponsored Poster Session	*Congressional Ballroom*
Tuesday 8:30 am	Plenary Session	*Congressional Ballroom*
9:00 am	Session 3	*Grand Ballrooms B&C*
	Session 4A	*Grand Ballrooms D&E*
10:30 am	Session 4B	
12:00	Elsevier Science- Sponsored Luncheon	*Congressional Ballroom*
1:30 pm	Session 5	*Grand Ballrooms B&C*
	Session 6	*Grand Ballrooms D&E*
Wednesday 8:30 am	Plenary Session	*Grand Ballrooms B,C,D&E*
11:30	Lunch - On Your Own	*See Hotel for Restaurant Specials*
1:00 pm	Session 7	*Grand Ballrooms B&C*
	Session 8	*Grand Ballrooms D&E*
3:30 pm	Rapporteur Panel	*Grand Ballrooms B,C,D&E*

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