NUREG/CP-0104

Transactions of the Seventeenth Water Reactor Safety Information Meeting

To Be Held at
Holiday Inn Crowne Plaza
Rockville, Maryland
October 23–25, 1989

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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PREFACE

This report contains summaries of papers on reactor safety research to be presented at the 17th Water Reactor Safety Information Meeting at the Holiday Inn Crowne Plaza in Rockville, Maryland, October 23-25, 1989. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, USNRC. Summaries of invited papers concerning nuclear safety issues from the electric utilities, the Electric Power Research Institute (EPRI), the nuclear industry, and from the governments and industry in Europe and Japan 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.

Speakers who did not submit summaries for inclusion in this report are indicated by an asterisk [*] in place of a page number in the Table of Contents.

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ASSESSMENT OF GENERIC ACCIDENT MANAGEMENT STRATEGIES CONSIDERED FOR NEAR TERM IMPLEMENTATION

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The U.S. Nuclear Regulatory Commission (NRC) and the industry are both participating in the identification of measures that can prevent the progression of a severe accident or mitigate its consequences. Information important for evaluating these accident management strategies for specific plants is expected to result from the ongoing Individual Plant Evaluation (IPE) program. However, NRC staff have identified a number of generic strategies which may not have to await the results of the IPE program and therefore can be considered for earlier implementation.

The NRC requested two of its contractors, Brookhaven National Laboratory (BNL) and Battelle Pacific Northwest Laboratories (PNL) to evaluate these strategies. The twenty one candidate strategies fall under three broad "global strategies:" 1) Conserving and Replenishing Limited Resources, 2) Use of Systems/Components in Innovative Applications, and 3) Defeating Interlocks and Component Protective Trips in Emergencies. Some strategies apply to BWRs or PWRs only, others apply to both types of plants.

This paper describes the evaluation of the strategies performed by Brookhaven National Laboratory. Battelle Pacific Northwest Laboratories will report separately on their evaluation.

Brookhaven National Laboratory assessed the proposed strategies by first detailing the objective of the strategy and listing the actions involved in the implementation. A description of the plant systems associated with the strategy was given. Next, the applicability of existing rules or plant procedures to a particular strategy was investigated. This was accomplished by a fairly detailed, but by no means exhaustive review of the emergency operating procedures of several plants, as well as utility and NRC reports related to accident management. This review indicated that many of the strategies were considered, at least in part, by existing procedures in particular plants. There appears to be considerable variation in the degree to which severe accident management has been addressed in the half-dozen plants for which emergency procedures were reviewed.

The proposed strategies were evaluated for their feasibility of implementation, their likely effectiveness in accomplishing their objective, and for any

*This work was performed under the auspices of the U.S. Nuclear Regulatory Commission. This paper does not necessarily express the opinion of the USNRC.

adverse effects that could be associated with their implementation. Hardware changes, other than very minor ones, were considered to be a negative influence.

Results of the evaluations are presented.

Responding to Emergencies: How Organization and Management Make a Difference

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There is an observable and definable process that occurs during the course of responding to an abnormal event at a nuclear power plant (NPP). Each of the elements that comprise that process involves collective action and, consequently, is influenced by the character and effectiveness of organizational and managerial arrangements. Factors which affect each element include overt ones like the allocation of authority and responsibility and the skill of the personnel, as well as covert factors like the methods used to resolve uncertainty. The purpose of this research project is to examine the process of response that occurs to an abnormal event at a NPP and where possible, to identify the organizational and managerial factors that influence that process.

The first task in this project involved the review and analysis of an extensive volume of documentation, primarily Nuclear Regulatory Commission (NRC) documents. Preliminary insights were developed for evaluating the dynamic behavior of organization and management factors during an accident situation. The evaluation of these factors was made relative to earlier work done at Brookhaven on the influence of organizational and management behavior on performance reliability during normal operations (FIN A-3956). The "Nuclear Organization and Management Analysis Concept" (NOMAC) (Haber, O'Brien, and Ryan, 1988) describes the functional and process-oriented structure of a NPP with a discussion of the work processes important for safety performance. In this work a NPP has been classified as a "machine-like bureaucracy" (Mintzberg, 1979) with its organization and management routinized in an extensive set of administrative, testing, maintenance, and operating procedures.

Given the extensive use of training and procedures under normal operating conditions, there has been an inclination to adopt the same organizational philosophy for dealing with accident and emergency conditions at a NPP. By their very nature, however, accidents and emergencies carry with them varying degrees of uncertainty and unexpected events. It is this necessity to deal with ambiguity that must be factored in to any discussion of the influence of organization and management on performance in the accident situation. The dilemmas posed for a machine bureaucracy in dealing with uncertainty, typically by ad hoc responses, are significant, and the conflict between conformance to procedures and the ability to improvise can result in dysfunctional behavior.

Based on the documentation reviewed during the first task of this project, it is possible to specify an observable and definable process for response to accident and emergency situations in a NPP. The process can be described by eight major elements; the initiating transient, information about plant behavior, diagnosing the problem, availability of emergency procedures, adequacy of emergency procedures, implementing emergency procedures, developing an ad hoc response, evaluating an ad hoc response and implementing an ad hoc response.

Importantly, the process to be described is an iterative one. Procedures and improvised responses are executed sequentially and the results of action are assessed and additional steps are taken if necessary.

Each of the eight elements in the process involves, to varying degrees, group decisions. Initially, there is a team of operators who are likely to make choices only after considering the situation among themselves. Once the Technical Support Center is activated, collaboration becomes more complicated and structured. Similarly, decision-making in the Emergency Operations Facility is often a team activity. Consequently, organizational, rather than individual or technological, factors will affect decision-making under accident and emergency conditions. To the extent that those factors can be controlled and channeled, organizational and management behaviors can improve their performance. Factors to be considered are overt behaviors such as communication flows, specificity of organizational goals, techniques for exercising authority, managerial attention and covert behaviors, most significantly, those involved in resolving uncertainty.

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FRAMEWORK FOR ACCIDENT MANAGEMENT*

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Accident management is an essential element of the Nuclear Regulatory Commission (NRC) Integration Plan for the closure of severe accident issues. This element will consolidate the results from other key elements; such as the Individual Plant Examination (IPE), the Containment Performance Improvement, and the Severe Accident Research Programs, in a form that can be used to enhance the safety programs for nuclear power plants. The NRC is currently conducting an Accident Management Program that is intended to aid in defining the scope and attributes of an accident management program for nuclear power plants. The fundamental objective of the Program is:

> "Each NRC licensee shall implement for each nuclear plant an 'Accident Management Plan' which provides a framework for evaluating information on severe accidents, including that developed through conduct of the Individual Plant Examinations (IPEs), for preparing and implementing severe accident operating procedures, and for training operators and managers in these procedures."

The accident management plan will ensure that a plant specific program is developed and implemented to promote the most effective use of available utility resources (people and hardware) to prevent and mitigate severe accidents. Hardware changes or other plant modifications to reduce the frequency of severe accidents are <u>not</u> a central aim of this program

To accomplish the outlined objectives, the NRC has developed an accident management framework that is comprised of five elements: (1) Accident Management Strategies, (2) Training, (3) Guidance and Computational Aids, (4) Instrumentation, and (5) Delineation of Decision Making Responsibilities. A process for the development of an accident management program has been identified using these NRC framework elements. A brief description of each of the six steps in this process follows:

Step 1 Assemble and Integrate Severe Accident Information

The existing, relevant severe accident information should be assembled and integrated into a data base to provide a foundation that will enable personnel involved in accident management to understand the characteristics of important severe accidents and the plant specific needs and capabilities for strategies to prevent or mitigate these accidents. The intent is to rely on information from existing sources, for example the Individual Plant Evaluation (IPE), and not to develop extensive new severe accident information.

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

Step 2 Determine General Accident Management Program Needs

The objective of this step is to determine the general accident management needs for each of the five NRC framework elements. General needs would be those that are associated with a broad spectrum of issues and are not specific to an individual strategy. For example, assessment of the general needs for the strategy framework element would address the capability of existing strategies to prevent or mitigate severe accidents, identify whether there is a need for strategies in addition to those already existing, and determine the necessary capabilities for any additional strategies.

Step 3 Identify Strategy Specific Development Needs

The objective of this step is to identify the specific needs for strategies that were identified in Step 2 as being needed for the prevention or mitigation of severe accidents. The process would: (1) provide a means for identifying and selecting potential strategies to meet the general needs identified, (2) develop preliminary guidance to carry out the strategy, (3) perform an evaluation to determine the capabilities of the selected strategies, and (4) select strategies for implementation based on a set of criteria that includes both feasibility and effectiveness.

Step 4 Develop and Implement Accident Management Program

The objective of this step is to implement the general and strategy specific needs from Steps 2 and 3 into a severe accident management program that is integrated and coordinated with the plants existing safety program. Included would be the development of any computational aids, guidance, procedures, engineered methods, instrumentation, delineation of decision making responsibilities, and training that were identified as necessary.

Step 5 Perform Program Validation

The objective of the fifth step is to provide a validation of the accident management program to ensure that it will provide the desired level of performance for the program. This objective is similar to the validation tasks identified for implementation of the symptom based Emergency Operating Procedures (EOPs).

Step 6 Incorporation of New Information

The objective of the final step is to provide a means of ensuring that new research information for severe accidents and utility experiences from operations, training, drills, and exercises can be incorporated into the accident management program. This step is important to ensure that the program is based on an up-to-date understanding of severe accidents and recent operating experiences.

The detailed processes necessary for each of the six steps are currently being identified. These processes provide the basis for development of guidance on the necessary and acceptable attributes of an accident management program.

EVALUATION OF STRATEGIES FOR SEVERE ACCIDENT PREVENTION AND MITIGATION[a]

Project Manager R. Tokarz

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The NRC is planning to establish regulatory oversight on severe accident management capability in the U. S. nuclear reactor industry. Accident management includes certain preparatory and recovery measures that can be taken by the plant operating and technical personnel to prevent or mitigate the consequences of a severe accident. Following an initiating event, accident management strategies include measures to 1) prevent core damage, 2) arrest the core damage if it begins and retain the core inside the vessel, 3) maintain containment integrity if the vessel is breached, and 4) minimize offsite releases. Objectives of the NRC Severe Accident Management Program are to assure that technically sound strategies are identified and guidance to implement these strategies is provided to utilities.

This paper will describe work performed to date by Pacific Northwest Laboratory (PNL) and Battelle Memorial Institute (BMI) relative to severe accident strategy evaluation, as well as work to be performed and expected results. Working with Brookhaven National Laboratory, PNL evaluated a series of NRC suggested accident management strategies. The evaluation of these strategies was divided between PNL and Brookhaven National Laboratory and a similar paper will be presented by Brookhaven regarding their strategy evaluation.

Work performed to date includes the following:

- a preliminary evaluation of the feasibility of developing selected emergency operating procedures for management of accidents that exceed a plant's design basis.
- evaluation of these strategies in terms of plant-specific procedures.
- evaluation of the feasibility and potential for risk reduction of similar procedures for both PWRs and BWRs where applicable.
- methods development research on means of identifying nuclear plant vulnerabilities to severe accidents and accommodating them with candidate strategies.

(a) Prepared for the U.S. Nuclear Regulatory Commission under Contract DE-AC06-76RLO 1830, NRC FIN 2930.

This work is being performed under the PNL, Accident Management Guidance project which consists of five tasks. Tasks 1 & 2 constitute the short-term portion of a larger set of NRC proposed strategies directed at a more complete evaluation of accident prevention and mitigation. The tasks are as follows:

- Task 1: Evaluate a set of Candidate Strategies for accident management, including development of guidelines for accident management procedures and evaluation of the procedures according to six specific criteria.

- Task 2: Review of Candidate Procedures, including review against existing plant-specific emergency operating procedures, probabilistic risk assessment studies (PRAs), and other available documents and evaluate the usefulness of candidate procedures for prevention and mitigation of accidents.

- Task 3: Identify and develop additional Accident Management Strategies, review the impact on existing PRAs, and evaluate the proposed strategies by the same criteria as was used for the NRC strategies.

This paper will stress the overall safety issues related to the research and emphasize the strategies that are applicable to major safety issues. The relationship of these research activities to other projects is discussed, as well as planning for future changes in the direction of work to be undertaken.

RECRITICALITY IN A BOILING WATER REACTOR FOLLOWING A CORE DAMAGE ACCIDENT[a]

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Principal Investigator R. Libby

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This paper will address the potential for a recriticality following core damage after a hypothetical loss-of-coolant accident in a boiling water reactor (BWR). The domain of conditions under which a recriticality is possible, as well as the consequences of a recriticality accident, will be discussed.

The core damage accident scenario is assumed to consist of the following events: 1) loss of coolant, 2) dry-out of core, 3) heat-up of fuel and control blade material, 4) melting and removal from core region of control blade material, and 5) eventual reflood of the core with water. The reflood could occur either before or after significant damage to the fuel occurs.

It is possible for a BWR to experience a recriticality following such a core damage accident. In the dry state, the core cannot be made critical, since criticality for dry uranium systems is possible only with fuel enrichments greater than 5 wt% ^{235}U. When the core is reflooded, however, criticality is possible. In the event that only unborated water is accessible (or is the first available water supply), it is important that reactor operators know the conditions under which recriticality is possible and the consequences of such a recriticality.

The fuel pins in a BWR have a fixed geometry that determines the core reactivity. In a core damage accident, however, fuel particles that are more reactive than the normal fuel pellet could be formed. The formation of both smaller particles (through fuel shattering) and larger particles (through melting and agglomeration) is possible. The reactivity is thus a function of fuel particle size, water-to-fuel ratio, and the concentration of boron in the reflood water.

Accident consequences will primarily be functions of the total energy release and the peak pressure pulse obtained during the event. These will, in turn, be functions of the system reactivity and the positive reactivity insertion rate. The system reactivity will depend on the fuel pellet (particle) size and moderation (water-to-fuel ratio), while the reactivity insertion rate will depend on the mechanical effects triggering the recriticality. These mechanical effects could include coolant reflood rates and/or the rate of the core collapse or fuel melting. Energy released through metal-water reactions will also contribute to the severity of the accident.

(a) Prepared for the U.S. Nuclear Regulatory Commission under Contract DE-AC06-76RLO 1830, NRC FIN 2930.

To bound the accident consequences, a worst-case scenario was developed which consists of a standing core without control blades. This configuration allows for the maximum possible reflood rate (hence reactivity insertion rate) and high initial reactivity. Although the standing core reactivity is slightly less than that for the optimum combination of fuel particle size and water-to-fuel ratio, the rate of reflood is at the highest rate possible. In addition, no fuel melting and dispersal scenarios are required to achieve this configuration. The only requirements are that the control blades melt and that the core is reflooded prior to onset of significant fuel damage.

Although a modest core collapse could increase the reactivity compared to an intact core, the reactivity insertion rate upon reflooding is less. Because of the restricted flow path caused by the collapse, the collapsed core debris would have much more resistance to the incoming flow of coolant than would a core consisting of standing fuel pins. This would result in a lower reflood rate and, hence, a lower reactivity insertion rate than is possible for a standing core. A substantial collapse would actually decrease the core reactivity, because a lower than optimal water-to-fuel ratio would result.

The paper will provide estimates of the energy release and peak power resulting from analysis of the bounding case. The power levels obtained will be compared to full power reactor operation. Potential accident management strategies such as lower coolant addition rates and alternate means for boration of emergency water will be discussed. Comments will also be made on future work relating to recriticality and the resulting energy releases from specific accident sequences.

Mitigation of Direct Containment Heating by Intentional Reactor Coolant System Depressurization

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An investigation of intentional depressurization of the reactor coolant system (RCS) as a means to mitigate direct containment heating (DCH) is presented. A DCH event could occur as the result of the dispersal of molten corium into the containment caused by a high pressure core melt ejection. The resulting containment pressure increase might be sufficient to cause containment failure. The probability of occurrence for a DCH event is a function of the driving pressure for the core melt ejection, the quantity of molten core ejected and the composition of the core melt. Although the latter two parameters are important, the RCS pressure is monitored and can be controlled to some extent by the plant operators during a severe accident. Possible strategies to mitigate DCH include (a) prevent core melting, (b) depressurize the RCS, or (c) take no action. Primary and secondary feed-and-bleed strategies are considered as basic options to prevent core damage. For a station blackout, where the ac power is lost, there is no safety injection. For a TMLB' transient, the additional failure of the auxiliary feedwater is assumed. For this transient, a primary or secondary feed-and-bleed strategy using existing systems is not possible because of unavailability of the safety injection and auxiliary feedwater. The operators can take action to depressurize the primary system using the pressurizer power-operated relief valves (PORVs), take no action, or find an emergency source of water independent of ac power to perform a feed-and-bleed operation. The last option may require some minor plant system modifications.

Analyses for depressurization of the Surry Nuclear Power Plant are being performed using the SCDAP/RELAP5 code for a hypothetical TMLB' sequence. Two cases with different times for the initiation of depressurization are considered. In the first case, the depressurization is initiated at the time of steam generator dryout. This case is called early depressurization. In the second case, the depressurization is initiated at the time of core uncovery. This case is called late depressurization. Both calculations show that the system can be depressurized to a level where DCH may be minimized. In the early depressurization, the surge line may also fail, leading to rapid depressurization of the primary. In the late depressurization, surge line failure is not predicted.

In Figure 1, the three depressurization strategies are compared, showing the effect on RCS pressure for each strategy. For early depressurization the RCS is rapidly depressurized to the accumulator setpoint. The accumulator flow into the RCS is calculated to be insufficient to

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

prevent core heatup.[1] The early depressurization calculations estimate that the RCS pressure can be red~~uced to less~~ than 1.5 MPa (220 psia) at the time of reactor pressure vessel breach (~10 hr). For the case of late depressurization we calculate that the accumulator inflow is sufficient to terminate core heatup and control cladding temperatures until the accumulators are empty. This ~3 hr delay in the core heatup provides the operators additional time to recover ac power and other plant systems that could provide means to establish long term core cooling. Calculations for no operator action[2] predict RCS depressurization, caused by surge line creep rupture, prior to the time at which the core heatup is calculated to resume in the late depressurization calculation. Surge line failure also results in accumulator injection, quenching, and reflooding of the core. However, this occurs only after significant core heatup (peak cladding temperature calculated to be ~2000 K) has taken place.

Modeling uncertainties in these calculations are large, and include (but are not limited to) the calculated core damage progression, inflow from the accumulators, and surge line creep rupture. The different strategies are evaluated based on engineering judgment. These strategies, in the order of preference, are: (a) perform feed-and-bleed operation if the ac power is recovered within a certain period of time; (b) establish an emergency source of feedwater independent of the ac power for a feed-and-bleed operation; (c) depressurize the primary system if an emergency source of feedwater is not available; and (d) take no action.

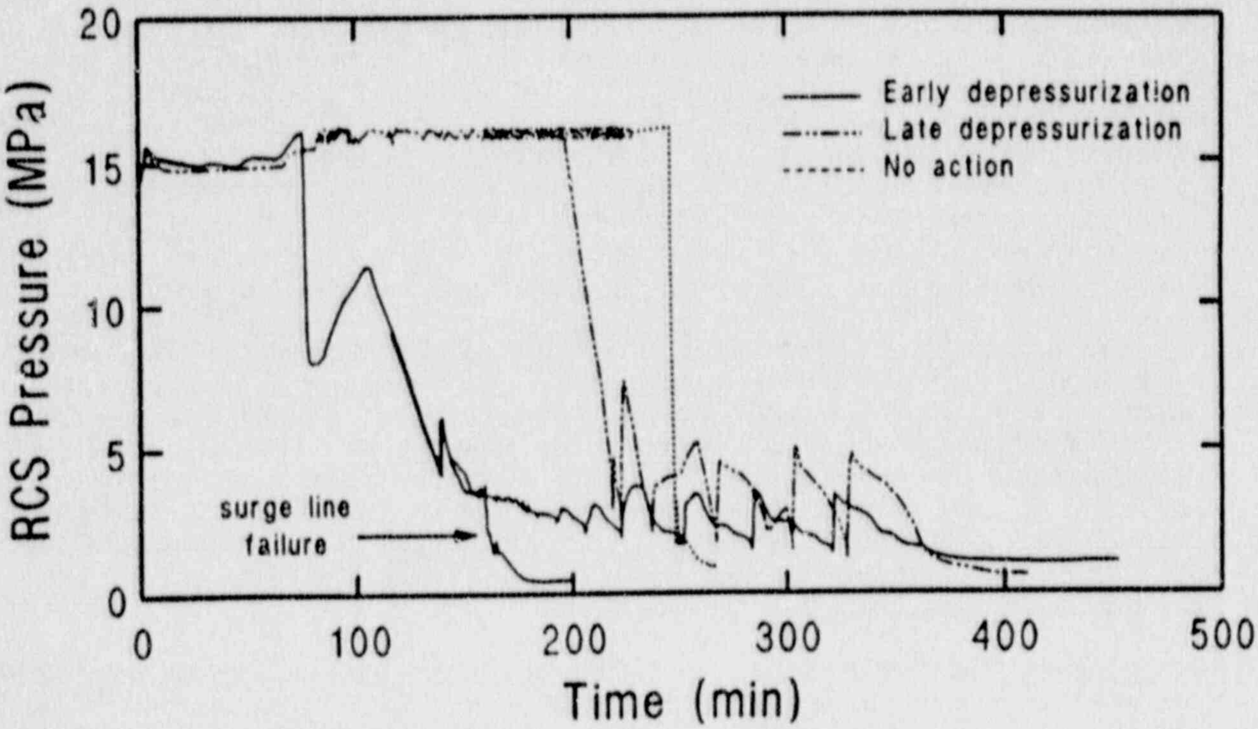

Figure 1. Calculated pressure for the three depressurization strategies.

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Reactivity Accidents:
A Reassessment of the Design-Basis Events

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The required safety analysis for light water reactors includes various reactivity events which must be shown to result in acceptable consequences in order for plants to be licensed. Although it is possible to hypothesize reactivity events that might lead to unacceptable consequences, in the past such unanalyzed events were either precluded by specific design features or required combinations of causative events which were judged too improbable to be of concern. The objective of this study was to develop as complete a list as possible of hypothetical reactivity events and then to screen these events in light of current knowledge to determine if any might require detailed analysis. The goal of the study was to either reconfirm or bring into question previous judgements on the adequacy of the reactivity events required for analysis. The screening was done by performing probabilistic and/or deterministic calculations and, where available using the research of others.

The focus of the study was influenced by the accident at Chernobyl. That event provided the incentive for this study to concentrate on accidents with relatively large reactivity insertions that could cause rapid fuel damage. In order to focus the study even more, the pressurized water reactor (PWR) analysis concentrated on a Westinghouse plant of four-loop design and the boiling water reactor (BWR) analysis on a BWR/4 design. However, reference is made to other designs throughout the report as the goal was to reach general conclusions that would be applicable to all PWRs or BWRs.

The primary results from the study are:

- Many sequences were studied in 16 broad categories. All but two of the sequences that have the potential to cause rapid fuel damage were estimated to have a frequency of occurrence too low to warrant further consideration at this time.

- Of the two sequences that have estimated frequencies in the range of interest for severe accidents, the most important is a refueling accident in a BWR. This is caused by the loading of fuel surrounding two or more positions where control blades have been removed and are inoperative. The NRC is studying this event and related sequences in a follow-on study so that it will be able to act in a timely manner on any corrective actions (such as changes in Technical Specifications) which may be deemed necessary.

- The other sequence with a significant estimated frequency of occurrence is the result of the flushing of boron during a BWR anticipated transient without scram (ATWS) because of an uncontrolled depressurization and injection of unborated water from low pressure cooling systems. Deterministic calculations are being done in a follow-up study in order to make a fully informed decision on the disposition of this sequence.

- Several sequences were identified during shutdown conditions that would not lead to rapid fuel damage but would lead to core melt. Although they have low estimated frequencies of occurrence, they may require further study to improve NRC's assessment of overall risk from shutdown.

- Several PWR sequences included the injection into the vessel of water with a relatively low boron concentration. These events were assumed in the study to lead to rapid fuel damage but no detailed calculations exist to verify this. Although the sequences have very low estimated frequencies of occurrence, calculations to better understand the system response, and determine whether rapid fuel damage is really possible, should be considered.

- To provide full closure to the issue of design-basis reactivity events, analyses being sponsored by the NRC as well as other nations need to be followed and the extrapolation of the results of this study to plants of different design (i.e. non-Westinghouse PWRs and non-BWR/4 BWRs) needs to be examined more fully.

SUMMARY

U. S. NUCLEAR INDUSTRY ACCIDENT MANAGEMENT GUIDELINES

BY

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INTRODUCTION

The Nuclear Management and Resources Council (NUMARC) serves as the United States nuclear power industry's principal mechanism for conveying industry views, concerns, and policies regarding industry-wide regulatory issues to the Nuclear Regulatory Commission (NRC) and other government agencies as appropriate. In particular, NUMARC is responsible for coordinating the combined efforts of utilities holding NRC operating licenses or construction permits for nuclear power plants on all regulatory aspects of operational and technical safety issues affecting the industry.

NUMARC and the Electric Power Research Institute (EPRI), in support of the NUMARC Severe Accident Working Group's (SAWG's) efforts with regard to accident management, has developed a framework for evaluation of plant-specific accident management capabilities. The purpose of this paper is to describe this framework and its objectives.

ACCIDENT MANAGEMENT OBJECTIVE

PROVIDE FOR SYSTEMATIC IMPLEMENTATION OF CERTAIN INSIGHTS AND RESULTS FROM AN INDIVIDUAL PLANT EXAMINATION (IPE) AND OTHER RELEVANT INFORMATION REGARDING SEVERE ACCIDENTS FOR THE PURPOSE OF ENHANCING A UTILITY'S CAPABILITIES DURING AN ACCIDENT TO TAKE MITIGATIVE ACTIONS IN AN EFFICIENT AND COST EFFECTIVE MANNER.

DEFINITIONS

Severe Accidents are those beyond the design basis that result in catastrophic fuel rod failure, core degradation, and fission product release into the reactor vessel, the containment or the environment.

Accident Management refers to actions taken during the course of an accident by the plant operating and technical staff to: (1) prevent the accident from progressing to core damage; (2) to terminate core damage if it begins; (3) to maintain containment integrity for as long as possible once the reactor system is breached; and (4) to minimize the effects of offsite releases.

Guidelines for Evaluating Accident Management Capabilities provides an industry-wide standard approach to plant-specific assessment of current accident management capabilities, and a listing and schedule for implementing plant-specific enhancements to that capability.

Plant-Specific Accident Management Program are those enhancement items identified and scheduled for implementation as an outcome of a utility-specific evaluation of accident management capabilities.

BACKGROUND

In order to implement an accident management program it is necessary to address: (1) personnel resources (organization, training, communications); (2) systems and equipment (restoration and repair, instrumentation, use of alternatives); and (3) information resources (procedures and guidance, technical information, process information).

While the SECY 89-012 partitioning of a severe accident into four phases does not provide a practical framework to organize and plan for plant-specific accident management, it does provide a useful independent perspective for the regulator to understand and view plant-specific accident management capabilities. Nevertheless, each of the four phases focuses on different success goals and responses. As such, it is worth discussing the relationship of each accident phase to the industry approach toward accident management.

NUMARC ACCIDENT MANAGEMENT GUIDANCE

Under the direction of the NUMARC SAWG, EPRI, with technical advice from the Institute of Nuclear Power Operations and Vendor Owners Groups, has developed generic "Guidelines for Evaluating Accident Management Capabilities". The intent of the guidelines is to focus industry use of certain IPE results and other severe accident insights, provide a framework for evaluating them in a plant-specific context and identify the type of possible changes that can be taken to enhance plant accident management capability. The five major steps outlined are:

Step 1: Identify categories of severe accident sequences for the plant.

Step 2: Review the severe accident sequence categories to identify potential enhancements to accident management capabilities.

Step 3: Organize and integrate candidate enhancements for accident management capability.

Step 4: Test candidate enhancements against specifics of sequences.

Step 5: Select and plan enhancements for implementation.

CONCLUSION

The integrated evaluation and application of insights from prior probabilistic risk assessments, plant-specific analysis, such as the IPE, and other programs provide a means of improving a plant's integrated capability to efficiently and cost-effectively handle severe accidents. The guidelines provide a systematic basis for assessing and applying those insights.

Heavy-Section Steel Technology Program - Fracture Issues

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ABSTRACT

Fracture prevention technology for steel reactor vessels in commercial nuclear power plants licensed by the Nuclear Regulatory Commission is assembled in national consensus Codes and Standards, the Code of Federal Regulation, and Regulatory Guides governing their design, construction, in-service inspection and evaluation. Further national consensus standards govern the testing required to generate the materials fracture toughness data base essential to the evaluation of fracture prevention margins. A primary mission of the Heavy-Section Steel Technology (HSST) Program at Oak Ridge National Laboratory (ORNL) has been to support the formulation and validation of the rules and evaluation procedures embodied in these Codes, Standards, Regulations and Regulatory Guides and generation of the required materials fracture toughness data. The HSST program continues to be heavily involved in the ongoing development of these regulatory documents.

Experience with the operation of commercial nuclear power plants, including data from reactor vessel surveillance programs, has identified areas in which the fracture prevention technology embodied in the present generation of national consensus Codes and Standards requires extension and/or refinement. Surveillance program data indicate that the Charpy upper-shelf fracture energy for a number of operating reactor vessels with high-copper welds will fall below the 50 ft lb. lower bound acceptance limit set by Appendix G to 10 CFR 50 before the plants reach the end of their current licensing period. This development generates a need for an extension of existing fracture prevention criteria, and the supporting fracture mechanics technology, to determine if, and under what circumstances, these plants could be permitted to operate beyond the currently defined licensing limits. Surveillance and research reactor data from the High-Flux Isotope Reactor (HFIR) at ORNL have indicated accelerated irradiation embrittlement under conditions which raise concerns relative to the long term structural integrity of some reactor vessel supports. Upcoming plant life-extension actions further emphasize the need to upgrade fracture prevention technology and the associated regulatory Codes and Standards to fully reflect current understanding of the long term effects of irradiation embrittlement. Understanding of the full range of fracture issues and limitations associated with potential pressurized-thermal-shock (PTS) events continues to evolve with respect to fracture modes, models, and data-range requirements. Development and validation of the fracture prevention technology advances required to meet these needs is the major thrust of the ongoing HSST program at ORNL.

Objectives for the current phase of the HSST program include (a) development and validation of fracture methodology for the prediction of mixed-mode crack propagation in low-toughness materials, (b) development and validation of innovative testing techniques for crack-arrest toughness testing at temperatures and driving forces representative of those encountered in potential PTS transients, (c) generation of crack-arrest data and interpretive analyses in support of an initiative to justify extension of the ASME Code K_{Ia} curve beyond its current 200 KSI$\sqrt{in}$. limit, (d) completion of a benchmark PTS test on a heavy-section steel vessel containing a low-upper-shelf weld and (e) providing continued support for the ongoing NRC evaluation of the effect of low-temperature low-flux irradiation embrittlement on the flaw tolerance of reactor vessel supports. Data and fracture prevention methodology derived from these program elements will contribute directly to resolution of fracture prevention issues associated with both the near term and extended life operation of nuclear reactor vessels. An example of this contriibution is to be found in results from prior PTS experiments which have included a demonstration of significant ductile tearing in advance of the initiation of cleavage fracture in a low-upper-shelf material under simulated PTS loading. A second example is the demonstration of crack-arrest toughness greatly in excess of the current ASME Code limit under temperature and toughness conditions similar to those associated with a PTS event in an irradiation embrittled reactor pressure vessel.

Close ties are maintained between the HSST Program and related programs both within the USA and abroad. Interaction with the sister Heavy-Section Steel Irradiation Program at ORNL is particularly closely coupled. Direct involvement of consultants and organizations with expertise and experience in the area of fracture prevention in steel structures is used to assure the HSST program benefits from a broad spectrum of informed input. Cooperative international information exchange programs, coordinated by the NRC, are in place with organizations in Europe and Japan. A major benefit from the international information exchange programs is the opportunity to participate in round-robin analyses of large scale fracture experiments and thereby obtain additional bases for the refinement and validation of fracture prevention technology advances. Cooperation with related international programs is further enhanced through HSST personnel memberships on appropriate Working Groups in the International Atomic Energy Agency and the Organization for Economic Cooperation and Development Committee on Safety of Nuclear Installations.

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*Research sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission under Interagency Agreement 1886-8011-9B with the U.S. Department of Energy under Contract DE-AC05-84OR21400 with Martin Marietta Energy Systems, Inc.

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Heavy-Section Steel Irradiation Program Summary

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ABSTRACT

Maintaining the integrity of the reactor pressure vessel (RPV) in a light-water-cooled nuclear power plant is crucial in preventing and controlling severe accidents which have the potential for major contamination release. The RPV is the only major safety-related component of the plant for which a duplicate or redundant backup system does not exist. It is therefore imperative to understand and be able to predict the capabilities and limitations of the integrity inherent in the RPV. In particular, it is vital to fully understand the degree of irradiation-induced degradation of the RPV's fracture resistance which occurs during service, since without that radiation damage it is virtually impossible to postulate a realistic scenario which would result in RPV failure.

For this reason, the Heavy-Section Steel Irradiation (HSSI) program has been established with its primary goal to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior, and in particular the fracture toughness properties, of typical pressure vessel steels as they relate to light-water reactor pressure-vessel integrity. The program includes the direct continuation of irradiation studies previously conducted within the Heavy-Section Steel Technology program augmented by enhanced examinations of the accompanying microstructural changes. Effects of specimen size, material chemistry, product form and microstructure, irradiation fluence, flux, temperature and spectrum, and post-irradiation annealing are being examined on a wide range of fracture properties including fracture toughness (K_{Ic} and J_{Ic}), crack arrest toughness (K_{Ia}), ductile tearing resistance (dJ/da), Charpy-V-notch (CVN) impact energy, dropweight NDT, and tensile properties. Observations of radiation-induced microstructural changes using the atom probe field ion microscope and the high resolution transmission electron microscope provide a firmer basis for extrapolating the measured changes in fracture properties to wider ranges of irradiation conditions. The principal materials examined within the HSSI Program are high-copper welds since their post-irradiation properties are most frequently limiting in the continued safe operation of commercial RPVs. In addition, a limited effort will focus on stainless steel weld overlay cladding, typical of that used on the inner surface of RPVs, since its post-irradiation fracture properties have the potential for strongly affecting the extonsion of small surface flaws during overcooling transients.

The current investigations of high-copper weldments fall into two major categories: first, the amount of the irradiation-induced shifts and any accompanying changes in shape of both the ASME K_{Ic} and K_{IR} curves in welds which maintain relatively high resistance to ductile fracture, and second, the determination of the overall degree of post-irradiation embrittlement for low upper-shelf (LUS) welds in which the resistance to ductile tearing falls to particularly low levels following irradiation. The Fifth and Sixth Irradiation Series, addressing the shifts in the K_{Ic} and K_{IR} curves, respectively, are both examining two identical high-copper, high upper-shelf weldments. The results of the Fifth Series, which is virtually complete, have shown that the shape of the irradiated K_{Ic} curve can change such that its actual shift is slightly greater than that predicted by small specimens. Testing of the irradiated crack arrest specimens from the Sixth Series, exposed under the same conditions as the fracture toughness specimens in the Fifth, will be completed this year. The Tenth Irradiation Series will begin an extensive evaluation of irradiation-induced embrittlement in LUS welds by examining the beltline fabrication weld from the Midland Unit No. 1 RPV which was abandoned prior to start up. The submerged arc weld, made by Babcock and Wilcox using Linde 80 flux and copper-coated welding wire, is typical of many LUS welds actually in service. The series will include an extensive initial unirradiated characterization of chemistry and fracture properties and a full range of post-irradiation fracture tests on specimens up to at least 2 inches thick in both the transition and upper shelf regimes, including an assessment of irradiation-rate and saturation effects. Further examinations of LUS material are planned for subsequent irradiation series including detailed evaluations of the shifts in their K_{Ic} and K_{IR} curves and the degree of recovery and rate of re-embrittlement of their fracture toughness associated with post-irradiation annealing.

Results from the HSSI studies will be integrated to aid in resolving major regulatory issues facing the U.S. Nuclear Regulatory Commission which involve RPV irradiation embrittlement such as pressurized-thermal shock, operating pressure-temperature limits, low-temperature overpressurization, and the specialized problems associated with low upper-shelf welds. Taken together the results of these studies also provide guidance and bases for evaluating both the aging behavior and the potential for plant life extension of light-water reactor pressure vessels.

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USE OF J/J_M-R CURVES IN ASSESSING THE FRACTURE BEHAVIOR OF LOW UPPER SHELF TOUGHNESS MATERIALS

By

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SUMMARY

The objectives of this investigation were to; (1) present experimental results and development of experimental analyses which attempt to define the limits of the J/J_M controlled region of crack extension in bend type laboratory fracture mechanics specimens; (2) evaluate the fracture behavior of low upper shelf toughness materials per the NRC-proposed $J_{0.1}$ criterion, and (3) evaluate the effects of constraint on the micromechanisms of fracture in a low upper shelf toughness material. Current limitations on J-controlled crack growth as described in ASTM E1152-87 are highly restrictive (crack extension is required to be less than 10% of the initial specimen remaining ligament). This is particularly limiting in fracture analyses of nuclear plant structures which require development of J-R curves with large amounts of crack extension for use in instability analyses. A recent proposal by the NRC, involving use of the value of J at 0.1 inches of crack extension as a measure of a material's fracture resistance, still involves extension of the J-R curve beyond the ASTM E1152 limits for 1/2T and 1T laboratory specimens.

The experimental test matrix included J-R curve tests on 1/2T, 1T and 2T compact [C(T)] and three-point-bend [SE(B)] specimens of several medium to high strength steels (ASTM A710, A516, A106, A302 Gr B, and A533B steels and a high yield strength, 3%Ni steel). The J-R curve tests were conducted in accordance with ASTM E1152-87 with the exception that crack extensions on the order of 60% of the initial remaining ligament were achieved. The data were analyzed both in terms of the deformation theory J integral and the modified J integral (J_M). For the A302 Gr B and the 3%Ni steel, the effects of the micromechanisms of the fracture process on the R-curve behavior were examined.

Results of the J-R curve tests indicate experimental support for extension of the E1152 crack growth validity limits to approximately 30% of the initial remaining ligament. An analysis, based on the Paris-Hutchinson omega criterion was shown to define a triangular region of J-R curve validity, consistent with omega = 1. In the majority of the cases examined, this analysis indicates loss of the controlling singularity between 20% and 30% crack growth. A graphical analysis for determining the acceptability of a flawed low upper shelf toughness vessel, based on the J and Tearing Modulus at 0.1 inches of crack extension ($J_{0.1}$ and $T_{0.1}$) was shown to simplify structural instability analyses.

POWER REACTOR EMBRITTLEMENT DATA BASE*

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SUMMARY

Regulatory and research evaluations of embrittlement prediction models and of vessel integrity under load can be greatly expedited by the use of a well-designed computerized embrittlement data base. The Power Reactor Embrittlement Data Base (PR-EDB) is a comprehensive collection of data from surveillance reports and other published reports of commercial nuclear reactors. The uses of the data base require that as many different data as available are collected from as many sources as possible with complete references and that subsets of relevant data can be easily retrieved and processed.

The objectives of this NRC-sponsored program are:

1. to compile and to verify the quality of the PR-EDB;

2. to provide user-friendly software to access and process the data;

3. to validate or confirm embrittlement prediction models; and

4. to interact with standards organizations to provide the technical bases for voluntary consensus standards that can be used in regulatory guides, standard review plans, and codes.

To achieve these goals, the data base architecture was designed after much discussion and planning with prospective users, namely, material scientists and members of the research staff.

The current compilation of the PR-EDB (Version 1) consists of 92 reactors, 137 different base material points, and 82 different weld data points. Menu-driven software programs have been written to facilitate maintenance, processing, and

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evaluations of the data. The time required to process and evaluate different types of data in the PR-EDB has been drastically reduced from previous data bases.

EPRI, reactor vendors, and utilities are in the process of providing back-up quality assurance checks of PR-EDB and will be supplementing the data base with additional data and documentation. Also EPRI, the BWR-OG Supplemental Material Surveillance Committee, and General Electric have agreed to release the boiling water reactor data for insertion into the PR-EDB.

A vast amount of test reactor data from the MPC data base, the HSST program, the PSF experiment, the NRL/MEA experiments, and the KFA experiments has been collected and will be integrated with PR-EDB to form a larger Embrittlement Data Base (EDB). This data base will be used to explore and test a variety of models for irradiation damage mechanism and prediction.

The computerized PR-EDB and associated software have been successfully applied to the following three evaluations:

1. Analysis of the A302B and A533B Standard Reference Materials in Surveillance Capsules of Commercial Power Reactors;

2. Comparison of Surveillance Data in PR-EDB with "Trend Curves" in Regulatory Guide 1.99, Revision 2; and

3. Comparison of the increase in the brittle-to-ductile transition temperature and the reduction in the Charpy-V upper shelf energy for specimens fabricated in the LT orientation with those in the TL orientation. The data contain only pairs with the same fluence in the same capsule.

This effort and coordination with the NRC research staff, EPRI, and industry has led to the adoption of PR-EDB as the basis for an industry-wide data base by the EPRI Reactor Vessel Embrittlement Management Project.

Resolution of the Gundremingen Reactor Vessel Anomaly by Accelerated Irradiation Test of Vessel Trepan Material

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Initial mechanical properties tests of beltline material trepanned from the decommissioned KRB-A pressure vessel and archive material irradiated in the UBR test reactor presented a major anomaly in relative radiation embrittlement sensitivity. Poor correspondence of material behavior in test vs. power reactor environments was described for the weak test orientation (ASTM L-C), but a good correspondence was observed for the strong orientation (ASTM C-L). Possible explanations for the unusual directionality include a heretofore unobserved fluence-rate-effect.

Grossly dissimilar radiation sensitivities between strong and weak material test orientations in terms of Charpy-V (C_v) 41-J transition temperature elevation, have not been observed previously in surveillance program irradiations or test reactor irradiations of pressure vessel steels. Differences in relative irradiation effect to the C_v upper shelf energy level have been observed on comparing data for L-C vs. C-L or TL vs. LT test orientations. In general, the preirradiation difference is reflected in the postirradiation upper shelf energy values wherein that orientation having the greater value initially undergoes the larger decrease by irradiation. NRC Regulatory Guide 1.99 reflects this in its use of percentage change, rather than absolute change, in its prediction formulation. In the present case, however, the trepan findings suggest that the orientation having the lower value initially also underwent the greater radiation-induced embrittlement, especially in upper shelf energy reduction.

The KRB-A vessel was made of several ring forgings joined by circumferential welding. The forging material is 20NiMoCr26 which is similar to ASTM A 336 steel. The composition of the trepans included 0.71 Mn, 0.013 P, 0.75 Ni, 0.38 Cr, 0.62 Mo and 0.16 Cu (wt-% values) and matched very closely, the composition of the archive material obtained from the General Electric Company by the NRC. To resolve the cited anomaly directly, C_v specimens from an essentially-nil fluence region of the vessel were irradiated together with archive material at 279°C in the UBR test reactor. Data for both orientations are compared to reveal the source of the vessel trepan vs. archive material differences.

The significance to NRC Regulatory Guide 1.99 of the determination and the impact on reactor vessel surveillance programs and PLEX planning are discussed.

EMBRITTLEMENT OF THE SHIPPINGPORT REACTOR SHIELD TANK

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Surveillance specimens from the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory showed an unexpectedly high degree of embrittlement relative to the data obtained on similar materials in Materials Testing Reactors (MTRs). The results suggest a possible negative flux effect and raise the issue of embrittlement of the pressure vessel support structures of commercial light water reactors. To help resolve this issue, a program was initiated to characterize the irradiation embrittlement of the neutron shield tank (NST) from the decommissioned Shippingport reactor. The Shippingport NST operated at 55°C (130°F) and was fabricated from A212 Grade B steel, similar to the vessel material in HFIR. The inner wall of the NST was exposed to a total maximum fluence of ~6×10^{17} n/cm^2 (E>1 MeV) over a life of 9.25 effective full power years. This corresponds to a fast flux of 2.1×10^9 n/cm^2·s and is comparable to the conditions for the HFIR surveillance specimens.

The sampling effort was sponsored jointly by the NRC and the DOE Plant Life Extension Program (PLEX) at Sandia National Laboratory. The actual sampling was performed by a group from Pacific Northwest Laboratory. Several samples of the base metal and weldments were obtained from the inner wall of the NST along with the corresponding samples from the very slightly irradiated outer wall. The irradiation embrittlement has been characterized by Charpy-impact and tensile tests as well as hardness measurements. Specimens were obtained in LT and TL orientations, and from three regions across the thickness of the NST wall. Material from the outer wall, which was protected by ~3 ft. of water and hence had a fluence ~10–6 less than the inner wall, was used to obtain the unirradiated baseline data.

The results indicate that irradiation increases the 15 ft·lb Charpy transition temperature (CTT) by ~25°C (45°F) and decreases the upper shelf energy. The shift in CTT is not as severe as that observed in the HFIR surveillance specimens and is consistent with that expected from the MTR data base. However, the actual value of CTT is high, and the toughness at service temperature is low, even when compared with the HFIR data. The increase in yield stress is ~50 MPa, which is comparable to the HFIR data. The results also indicate a lower impact strength and higher transition temperature for the TL orientation than that for the LT orientation.

Some effects of the location across the thickness of the wall are also observed for the LT specimens; CTT is slightly greater for the specimens from the inner region of the wall.

Annealing studies indicate a complete recovery of irradiation embrittlement after a 2-h anneal at 400°C. The transition curve for the annealed inner wall specimens is virtually identical to that for the as-received outer wall. Annealing had no effect on the transition curves for the outer wall.

Metallographic characterization indicates a hardness profile across the thickness of the NST plate; the hardness of the inner and outer regions of the wall are 10% higher than the wall center. The influence of hardness on the Charpy-impact and tensile data are discussed. The results for weld specimens from the inner and outer walls are also presented.

FUTURE PLANS FOR NRC THERMAL-HYDRAULIC RESEARCH

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Office of Nuclear Regulatory Research

Water Reactor Safety Information Meeting
October, 1989

NRC has been developing thermal-hydraulic Systems Codes for 15+ years. The codes have now reached an acceptable level of accuracy and maturity for current generation LWRs. Further development is not likely to produce major changes in our understanding of plant performance or the consequences of initiating events. Thus, NRC's objective is to bring the current thermal-hydraulic code development program to a successful conclusion while, at the same time, maintaining capability for thermal-hydraulic analysis at the minimum level necessary to apply developed codes to reactor issues.

Within our overall plan are several specific elements that are designed to achieve this overall objective. NRC intends to maintain a cadre of experts (contractor and in-house) to perform thermal-hydraulic analysis of LWRs such as those required for operating reactor events (e.g., Davis-Besse, LaSalle), licensing issues, and generic research (e.g., front-end of severe accident sequences).

NRC also plans to maintain code development/research at the minimum level necessary to ensure codes are acceptable for advanced LWR and CANDU analyses. It is also planned to continually review new information to see if it invalidates our current understanding of code accuracy.

It is anticipated that NRC will retain involvement in international thermal-hydraulic activities provided the resource commitment is minimized and there is substantial benefit to NRC.

Finally NRC expects to establish and maintain low-cost, experimental capability at universities through construction and operation of scaled loops representing major U.S. Reactor types.

Implementation of these plans will provide the NRC with an expanded applications capability to systematically assess reactor behavior in operating LWRs, advanced LWRs, CANDU and PIUS reactors.

The UK Contribution to Improvements in TRAC and RELAP5

by

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and

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Summary

The re-organization of the UK electricity industry continues to bring many changes not only to the CEGB but also to the UK Atomic Energy Authority. These changes are partially reflected by the new affiliations of the two authors. However, in spite of the changes, there has been continuity within the UK with regard to the ICAP collaboration, which is viewed as an important contributor to confidence and progress in the analysis of transient thermal-hydraulic behaviour. Work has continued on both code assessment and code improvement and studies have been performed with existing and new versions of RELAP5 and TRAC PF1. As explained in a paper to this meeting in 1988, it is intended that RELAP5 will be used in the UK for the formal assessment of licensing calculations concerning small break LOCA and pressurized transients in the Sizewell B plant, while TRAC PF1 will be used for the assessment of large break LOCA calculations. The licensing calculations themselves will be performed using Westinghouse codes.

The present paper focusses on the work that has been performed by UK staff on the development of the new versions, namely RELAP5/MOD3 and TRAC PF1/MOD2. As stated in the 1988 paper, the UK contribution to the ICAP Code Improvement Plan is in three areas. The first and largest of these, is the problem of post-CHF heat transfer and quenching in large break LOCA. The UK effort has therefore been directed at the TRAC code for this case. The second area is that of interphase drag under wet wall conditions, which is of particular interest in small breaks and intact circuit faults, where it is very important to provide an accurate calculation of the mixture level in the core and in the steam-generators. The third area is the implementation of an improved offtake model in order to make better predictions of the flow and quality from a junction or a break in a horizontal pipe, such as the PWR hot leg, where stratified conditions might exist. The interphase drag and offtake model improvements have both been directed at the RELAP5 code. Although the UK developments have been aimed at TRAC or RELAP in accordance with our planned usage of the codes, the broad intention was that the improvements would be adopted for both codes.

The present paper includes an outline of a new reflood model that has been developed for TRAC PF1/MOD2. This consists of a reflood hydraulics model, which provides much closer correspondence in terms of physical behaviour to that observed in separate effect reflood experiments. In particular, a much less oscillatory flow of entrained liquid is predicted. Also heat transfer correlations are selected which correspond to those measured both in transient reflood experiments and in steady-state hot patch experiments. Another development, which is implemented as an option, is an analytic model for quench progression, which is based on axial conduction theory and on experimental data from a wide range of reflood and other quench experiments. This model should provide a more accurate prediction of quench behaviour without requiring the extremely fine numerical mesh to represent the cladding temperatures in the quench front region.

The paper also contains a description of a model to represent the external thermocouples, which has also been implemented in TRAC PF1/MOD2. This should allow the direct use of the important LOFT large break data in code validation, without having to make separate (and controversial) judgements on the effects of the external thermocouples in particular transients.

The new interphase drag model which has been implemented in RELAP5/MOD3 is described and an outline is given of the significant amount of assessment that has been carried out on this model over the last year. This shows a clear improvement, but with some remaining differences from experimental data obtained during fast transients. The offtake model, which has also been implemented in RELAP5/MOD3, has previously been described and assessed in the context of a special version of RELAP5/MOD2, so there is little discussion of this work in the present paper.

Summary of ROSA-IV/LSTF First Phase Test Program and Station Blackout (TMLB') Test Results

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This paper summarizes major test results obtained at the ROSA-IV Large Scale Test Facility (LSTF) during the first phase of the test program. The results from a station blackout (TMLB') test conducted at the end of this phase are described in some detail.

The LSTF is an integral test facility being operated by the Japan Atomic Energy Research Institute for simulation of PWR thermal-hydraulic responses during small-break loss-of-coolant accidents (SBLOCAs) and operational/abnormal transients. It is a 1/48 volumetrically-scaled, full-height, full-pressure simulator of a Westinghouse-type 4-loop (3423 MWt) PWR. Testing on this facility started in May 1985, and the first phase test program ended in July 1988, conducting forty-two test runs. The second phase program started in March 1989, after replacement of the simulated core fuel-rod assembly.

The primary objective of the first-phase program was to define the fundamental plant responses during SBLOCAs and transients. Most of the tests were performed with assumed component/operator failure(s) which tended to simplify the system thermal-hydraulic responses. These assumptions included failure of the high pressure injection system (HPI) and auxiliary feedwater (AFW) system, as well as no operator actions taken until core heatups began. Attempts were made in several SBLOCA test, after core heatups were detected, to simulate the plant recovery procedures as well as candidate accident management measures for prevention of high-pressure core melt situation.

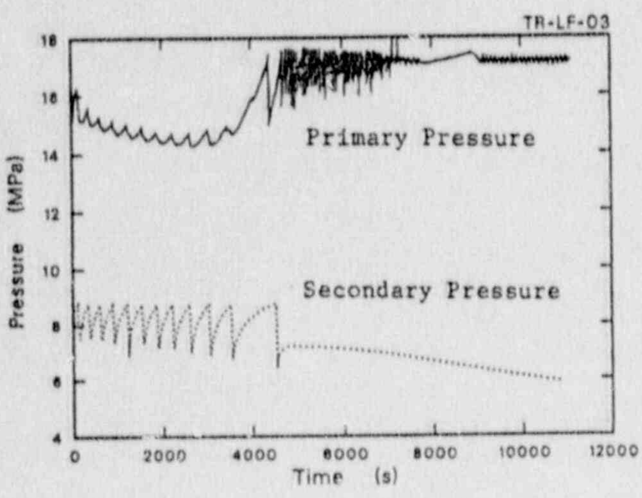
The forty-two first-phase tests included twenty-nine SBLOCA tests, three abnormal transient tests and ten steady-state natural circulation tests. The SBLOCA tests encompassed eleven different break locations including: cold leg (leg pipe top, bottom and side), hot leg (top, bottom and side) cross-over leg, pressurizer top, steam generator U-tube, vessel upper and lower heads. The test results indicated significant break-location dependency of the break flow rate.

The cold-leg break tests focused on the steam generator liquid holdup effects on the core liquid level depression during loop seal clearing. One of these tests (Run SB-CL-18) has been chosen by the OECD-NEA Committee on the Safety of Nuclear Installations (CSNI) for its International Standard Problem No. 26 (ISP-26). This ISP is open to participation from non-OECD countries. The calculation submission deadline is the end of December 1989.

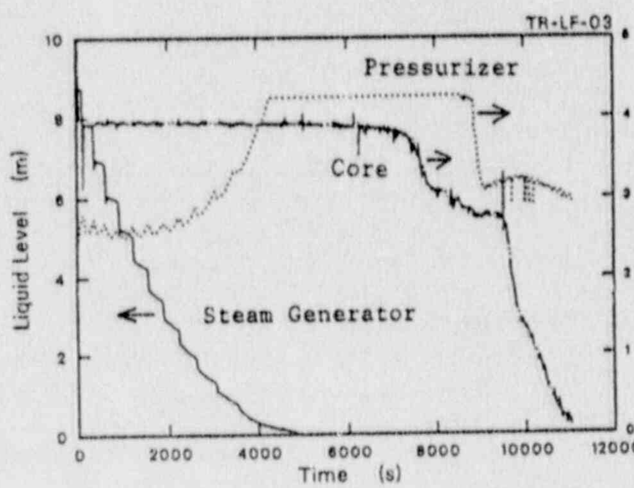
The steady-state natural circulation tests studied the loop flow rate dependency on core power and coolant inventory, SG heat transfer under degraded secondary cooling conditions, primary coolant inventory distribution including counter-current flow limiting (CCFL) phenomena at the hot leg and U-tube inlet, and parallel-channel non-uniform behavior of U-tubes.

The transient test conducted at the end of first-phase program, Run TR-LF-03, simulated the well-known TMLB' sequence. Since the system response for this scenario is sensitive to ambient heat losses and fluid leakage, the LSTF facility scale, largest among the currently-operating full-pressure PWR simulators, was a strong advantage in simulating this scenario.

This test was initiated by closing the feedwater and steam valves, initiating the core power decay, and initiating the pump coastdown, all at the same time. The transient responses during this test are summarized in Fig. 1. Since the emergency a.c. power and turbine-driven AFW were assumed unavailable, the core decay power, transferred to SGs by single-phase natural circulation, boiled off the SG secondary inventory through the SG safety valves that opened cyclically. The primary pressure increase, due to SG heat transfer degradation, lifted the pressurizer safety valve intermittently after 4200 s, causing coolant inventory loss as well as liquid holdup in the pressurizer. The natural circulation stopped as soon as the hot leg fluid saturated, at 7800 s, as vapor plugs formed at the U-tube tops. The vessel liquid level continued to drop, and core uncovering started eventually at 9600 s. The test was terminated at 11000 s when the core maximum temperature reached to 920 K.



Primary and secondary pressures



Primary and secondary liquid levels

Fig. 1 Transient Responses During Station Blackout (TMLB') Test, Run TR-LF-03

EPRI RESEARCH ON WATER HAMMER PREVENTION, MITIGATION,AND ACCOMMODATION

by

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D.A.Van Duyne, Stone & Webster
H. H. Safwat, Bechtel

Water hammer events occurring in nuclear power plants, occasionally result in equipment damage which can affect plant operation. To eliminate, or significantly reduce the impact of such events, EPRI has established a research program to investigate physical phenomena, design methods, and procedures aimed at prevention, mitigation, and accommodation of water hammer.

The initial tasks of this project focussed on compilation of nuclear plant waterhammer events, and root cause analyses examining the individual systems which have proven to be most susceptible to waterhammer.

Seven specific mechanisms which can lead to severe waterhammer have been identified. Four of these result from condensation of steam, whereas the others involve steam propelled water slugs, filling of voided lines, and rapid valve actuation. Examples of initiating events resulting in some of these mechanisms, which have occurred in plant systems are discussed, along with corrective measures taken to prevent them.

Finally, a brief description of available codes, and their performance in analyzing some specific problems relevant to waterhammer are discussed.

MIST Program: Risk Dominant Transient Testing

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Abstract

The Multi-Loop Integral System Test (MIST) Facility is a scaled physical model of a Babcock & Wilcox lowered loop, nuclear steam system. MIST was part of a multi-phase program to address small-break loss-of-coolant accidents (SBLOCAs) specific to the Babcock & Wilcox-designed plants. Data from MIST are used to benchmark the adequacy of system codes, such as RELAP5 and TRAC, for predicting abnormal plant transients.

Toward the end of the test program, a series of tests were performed to explore operating procedures for mitigating various accident conditions and investigate possible alternative strategies. This included three tests referred to as Risk Dominant Transients, in which a small-break loss-of-coolant accident was accompanied by the lack of particular auxiliary equipment or control systems that would normally be employed to mitigate the accident condition.

Two of these tests examined SBLOCA transients without the benefit of the high-pressure injection (HPI) system. The first of these utilized standard abnormal transient operating guideline (ATOG) control schemes, and the second employed a more aggressive steam generator pressure control strategy to cool the plant. The third Risk Dominant Transient was a station blackout test. This transient is characterized as an SBLOCA accompanied by the loss of all AC power, which disables some emergency equipment, reduces the capacity of others, and makes the operator reliant on station batteries to provide power to the plant's critical instruments. The results and observations from these tests are summarized in the publication.

UMCP MIST COUNTERPART TEST

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Two Small Break LOCA transients are compared to illustrate a scaling methodology for reduced pressure integral facilities. Mapping test 3004 is conducted in the MIST full pressure, full height facility. The counter-part test MIS0317 is scaled and performed in the reduced height, reduced pressure UMCP facility. Inventory is used as the chronological scale and pressure, normalized with the initial and system saturation pressures, is used as characteristic parameter to describe the system behavior.

The appropriately normalized results conclusively demonstrate that: a) the same phenomena are observed in the two facilities; b) the sequence of events is analogous and c) the trends described by the normalized pressure versus inventory traces are in good quantitative agreement. Each energy transport mode traversed by the two facilities is compared and the phenomena present are described in detail. The differences between the high and reduced pressure tests are outlined.

The findings clearly indicate that pressure and height can be scaled for transient where limited boundary conditions are applied (Auxiliary Feed Water only) and where the break is subcooled. A statement on sensitivity to the initial conditions is also included to define the limitations of the quantitative results.

NDE RELIABILITY AND ADVANCED NDE TECHNOLOGY VALIDATION[a]

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SUMMARY

This paper reports on progress for three programs: (1) Evaluation and Improvement in Nondestructive Examination Reliability for Inservice Inspection of Light Water Reactors (LWR) (NDE Reliability Program), (2) Field Validation, Acceptance, and Training for Advanced NDE Technology, and (3) Evaluation of Computer-based NDE Techniques and Regional Support of Inspection Activities. Activities for each of the programs is summarized below.

The NDE Reliability Program objectives are to quantify the reliability of inservice inspection techniques for LWR primary system components through independent research and establish means for obtaining improvements in the reliability of inservice inspections. The areas of significant progress will be described concerning ASME Code activities, re-analysis of the PISC-II data, the equipment interaction matrix study, new inspection criteria, and PISC-III.

One of the emphases of the past five years on this program has been the development of requirements for NDE/ISI qualifications for personnel, equipment, and procedures. This work has achieved a major milestone since the requirements in Section XI Appendices VII and VIII have passed the formal acceptance process of ASME Code. Some work remains in that supplements for several inspection applications are still under development.

A re-analysis of the PISC-II round robin test data was made to apply commonly employed statistical analysis procedures to the data and to relate the data to parameters commonly used in the USA. The analysis showed some important trends and confirmed the importance of performing an effective, near surface examination.

The equipment interaction matrix is a study to develop a technical basis for permissible variation in equipment operational parameters for ultrasonic test systems. The work this past year focused on the development of worst case flaws that would cause the maximum adverse variation in equipment response. Once these were found, they were used to perform sensitivity studies to determine the amount of variation they caused in system response for a given variation in this parameter. In general, for bandwidth variations of up to 10%, the output was always limited to less that 10%. For narrow band systems,

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(a)Work supported by the U.S. Nuclear Regulatory Commission under Contract DE-AC06-76RLO 1830; Dr. J. Muscara, NRC Program Manager, FIN B2289

it was found that for small changes in center frequency, the system response could change radically--up to 40%. New requirements are in preparation for narrow band center frequency requirements.

The new inspection criteria task is developing methodologies and criteria for improved ISI (type, extent, frequency) to meet goals of failure probability, radiation releases, or core melt probabilities. Work on the use of probabilistic risk analysis (PRA) has been conducted on eight nuclear plants, and results have shown a significant trend in the data indicating that PRAs can be used to look generically at the ranking of reactor systems and to set a prioritization for their reliability insofar as they affect the stated goals. This work is occurring in conjunction with participation in an ASME Task Force on Risk-Based Inspection Guidelines.

Work continues in addressing the inspectability of coarse grained materials, and a topical report is being written. A cooperative program in conjunction with the Center for NDE at Iowa State University, under EPRI funding, is being conducted to develop engineering tools for requirements of surface specifications for ASME Code. Currently, Code has no UT surface requirements, and the goal of this work is to develop requirements to insure that the surface conditions do not preclude an effective examination from being conducted.

Participation continues in the PISC-III Program to relate the work of this international program to conditions and practices in the USA.

The objectives of the second program are to develop field procedures for the AE and SAFT-UT techniques, perform field validation testing of these techniques, provide training in the techniques for NRC headquarters and regional staff, and work with the ASME Code for the use of these advanced technologies. The focus of the work has been to use the SAFT technology to undergo blind testing at the EPRI NDE Center. This has involved tests with the technology to detect and characterize defects in thick section plates and passing the IGSCC sizing performance demonstration test. Plans are in place to conduct further blind tests at the NDE Center and in the PISC-III Program.

The AE technology is undergoing evaluation in an on-reactor test to monitor a recirculation inlet nozzle-to-safe end weld on the Limmerick Unit 1 Nuclear Power Station. This is a test to evaluate a condition detected by ultrasonic examination after replacement of the recirculation system piping and is being monitored to assess, in an on-line continuous manner, the condition of the component. Testing is currently in progress, and to date, the results have shown no significant trends. A code case on the AE technology is going through the formal process in Section XI.

The final program's objective is to evaluate the reliability and accuracy of interpretation of results from computer-based ultrasonic inservice inspection systems, and to develop guidelines for NRC staff to monitor and evaluate the effectiveness of inservice inspections conducted on nuclear power reactors. This program started in the last quarter of FY89, and the extent of the program was to prepare a work plan for presentation to and approval from a technical advisory group of NRC staff.

IMPROVED EDDY-CURRENT INSPECTION FOR STEAM GENERATOR TUBING'

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A comparison was made of different types of probes and their performance under different steam generator test conditions. The probe types include the differential bobbin probe, the absolute bobbin probe, the pancake probe and the reflection probe. The generator test conditions include tube supports, copper deposits, magnetite deposits, denting, wastage, pitting, cracking and IGA. The study was based mostly on computed values, with the limited number of test specimens available used to verify the computed results. The instrument readings were computed for a complete matrix of the different test conditions, and then the test conditions determined as a function of the readings in a least-squares technique. A comparison was made in the errors in fit and instrument drift for the different probe types.

Measurements and computations were also made of the effect of sampling rate on the accuracy of the defect depth measurement for the bobbin probe. For simple defect signals, a relatively coarse sample rate is sufficient to determine the defect depth. The frequency response of the signal produced as the coil is scanned past the defect determines the desired response of the instrument amplifiers.

The computations of the change in instrument reading due to the defects have led to an "inversion" technique in which the defect properties can be computed from the instrument readings. This has been done both experimentally and analytically for each of these probe types. Experimental measurements have been made for the different probes, including the application of the reflection probe to IGA standards. These reflection probe measurements were made on the tube outer surface with a very sharply focused probe to determine material properties, and from the bore with our standard inspection probe to determine how well this condition can be measured in the field.

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HYDROGEN WATER CHEMISTRY FOR BWRS: A REVIEW OF THE EPRI DEVELOPMENT PROGRAM

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ABSTRACT

Boiling water reactors (BWRs) in the United States have experienced extensive intergranular stress corrosion cracking (IGSCC) in their austenitic stainless steel reactor coolant system piping, resulting in serious adverse impacts on plant availability, O&M costs, and personnel radiation exposures. A major research program to provide remedies for BWR pipe cracking was cofunded by EPRI, GE, and the BWR Owners Group for IGSCC Research between 1979 and 1988. Results from this program show that the likelihood of IGSCC depends on reactor water chemistry (particularly on the concentrations of ionic impurities and oxidizing radiolysis products) as well as on material condition and the level of tensile stress. Tests have demonstrated that the concentration of oxidizing radiolysis products in the recirculating reactor water of a BWR can be reduced substantially by injecting hydrogen into the feedwater. Recent plant data show that the use of hydrogen injection can reduce the rate of IGSCC during power operation to insignificant levels if the concentration of ionic impurities in the reactor water is kept sufficiently low. This approach to the control of BWR pipe cracking is called hydrogen water chemistry (HWC). This paper presents a brief summary of the results of EPRI's HWC development program from 1982 to the present. In addition, ongoing and future work to investigate the feasibility of adaptating HWC to protect the BWR vessel and major internals components from stress corrosion cracking damage is described.

CAST STAINLESS STEEL AGING: MECHANISMS AND PREDICTIONS

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A program is being conducted to investigate the significance of low-temperature embrittlement of cast duplex stainless steels under LWR operating conditions and to evaluate possible remedies to the embrittlement problem for existing and future plants. The scope of the investigation includes the following goals: (1) establish the mechanism of embrittlement and validate the simulation of in-reactor degradation by accelerated aging, (2) evaluate the effects of key compositional and metallurgical variables on embrittlement, and (3) obtain fracture toughness data to predict the degree of toughness loss suffered by cast stainless steel components during normal and extended service life of reactors.

Microstructural and mechanical properties data are obtained on 25 experimental heats (static-cast keel blocks and slabs) and six commercial heats (centrifugally cast pipes and static-cast pump impeller and pump casing ring) as well as reactor-aged material of CF-3, CF-8, and CF-8M grades of cast stainless steel. The ferrite contents of the cast materials range between 3 and 30%. The ferrite morphology for the castings containing >5% ferrite is either lacy or acicular. The centrifugally cast pipe material has equiaxed or radially oriented columnar grains, while the static cast keel blocks, slabs, and the pump impeller have a mixed grain structure.

Charpy-impact, tensile, and J-R curve tests were conducted on several experimental and commercial heats of cast stainless steel that were thermally aged up to 30,000 h at temperatures between 290 and 450°C (~555 and 840°F). The results indicate that aging at these temperatures leads to an increase in tensile strength, a decrease in impact energy, J_{IC}, and tearing modulus of the material, and the ductile-to-brittle transition curve shifts to higher temperatures. In general, the low-carbon CF-3 grades of cast stainless steels are the most resistant and molybdenum-containing high-carbon CF-8M grades are the most susceptible to low-temperature embrittlement. The effects of material variables on the embrittlement of cast stainless steels are evaluated. The ferrite morphology has a strong effect on the degree or extent of embrittlement, while material composition influences the kinetics of embrittlement. The kinetics of embrittlement can vary significantly with small changes in the constituent elements of the cast material.

The mechanisms of embrittlement of cast duplex stainless steel have also been established. Embrittlement is caused by brittle fracture associated with either cleavage of ferrite or separation of ferrite/austenite phase boundaries. The formation of α' phase by spinodal decomposition of the ferrite, provides the strengthening mechanism to raise the local tensile stress above the critical value for cleavage and, thus, promotes brittle fracture. Precipitation and/or growth of phase-boundary carbides or nitrides leads to a brittle failure by phase boundary separation and also facilitates cleavage of the ferrite by particle cracking. Therefore, the degree of brittle fracture and, hence, the degree of embrittlement of a specific heat of cast stainless steel depends strongly on the amount and spacing of the ferrite in the duplex structure. Cast materials that are sensitive to embrittlement either have a semi-continuous ferrite morphology or provide an easy fracture path via phase-boundary separation. For some materials although a portion of the material may fail in a brittle fashion, the surrounding austenite provides ductility and toughness, e.g., cast materials with low ferrite content or the low-carbon grades of cast stainless steels.

The kinetics of embrittlement are controlled by three processes, viz., spinodal decomposition, precipitation and growth of phase boundary carbides, and precipitation of G phase in ferrite. The kinetics of embrittlement for a specific cast stainless steel depend on the relative contributions of carbide and G-phase precipitation; the activation energies can range between 65 and 230 kJ/mole. The influence of material variables on the kinetics of the three processes is discussed.

The embrittlement of cast stainless steels can be recovered by a short-time anneal for 1 h at 550°C and water quenching. However, preliminary data show that the recovery annealed material reembrittles in a relatively short time.

Mechanical property data have been analyzed to develop the procedure and correlations for predicting the kinetics and extent of embrittlement of reactor components from known material parameters. The method and examples of estimating the impact strength and fracture toughness of cast components during reactor service are described. The lower bound values of impact energy and fracture toughness for cast stainless steels at LWR operating temperatures are defined.

Environmentally Assisted Cracking in Light Water Reactors

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Environmentally assisted degradation problems that have been considered during this year include (1) stress corrosion cracking (SCC) of austenitic stainless steels (SS), (2) fatigue of Type 316NG SS, and (3) SCC of ferritic steels used in reactor piping, pressure vessels, and steam generators.

Fracture-mechanics crack-growth-rate tests on compact tension (1TCT) specimens of Types 304, 316NG, and 347 SS were performed to determine the effects of water chemistry on the SCC. Because operating reactors have made significant improvements in coolant purity over the past several years, recent attention has focussed on quantifying the effects of low levels of organic impurities that are typically harder to detect and remove with conventional water cleanup systems. The tests show that some organic substances detected in reactor coolant systems at low concentrations actually inhibit SCC of sensitized Type 304 SS under simulated BWR water chemistries. The mechanism for this beneficial effect has not been definitely established, but since the decrease in the crack growth rates is not accompanied by a corresponding decrease in electrochemical potential (ECP) of the steel, it appears that the organic substances block sites on the specimen surface where cathodic reduction of dissolved oxygen occurs; a process that couples with anodic dissolution at the crack tip in a slip-dissolution mechanism of crack advance.

The data base on SCC of Types 304 and 316NG SS developed at Argonne has been correlated in terms of existing models for the crack tip strain rate and compared with the comprehensive model developed by Ford and Andresen that attempts to account for the effects of impurity concentration (i.e., water conductivity), ECP, degree of sensitization, and loading history. Theoretical and empirical models for the crack tip strain rate provide a better correlation of crack growth rates with loading history than more conventional parameters such as the stress-intensity factor K or the change in stress intensity ΔK under cyclic loading conditions. However, the existing models significantly underestimate the effect of load ratio R at high R values. The trends of the crack-growth-rate data developed at Argonne are in qualitative agreement with the model of Ford and Andresen, but the model seems to overestimate the benefits of low coolant impurity levels and low levels of sensitization of the steel, as measured by the electrochemical potentiokinetic reactivation (EPR) technique.

Fatigue tests on Type 316NG SS were conducted in air and in a simulated BWR environment at frequencies of 0.5, 0.05, and 0.005 Hz. Baseline tests were performed in air under strain control during which the stroke histories were monitored. Comparison tests were then conducted under variable stroke control that simulated the stroke histories in the strain-control tests. This was not done on a cycle-by-cycle basis but instead, the average stroke amplitude over a range of cycles was matched (e.g., for cycles 0–100, 100–

1,000, 1,000-10,000, etc.). The fatigue lives of the specimens under strain and stroke control were in excellent agreement. A similar procedure was used in tests performed in the simulated BWR environment. Even at a relatively high frequency of 0.5 Hz the fatigue lives in the environment were about one half of those in air, and the reduction in life increased as the frequency decreased.

SCC tests were also conducted on 1TCT specimens prepared from a plate of A533-Gr B pressure vessel steel containing 0.018% sulfur. In addition to conventional specimens, composite 1TCT specimens of A533-Gr B/Inconel-182/Inconel-600 were fabricated by overlaying the ferritic steel with In-182 weld metal and then electron-beam welding In-600 to the In-182. The specimen geometry is such that the crack will proceed from the In-182 into the ferritic steel. A533-Gr B specimens and specimens of this material that were plated with either nickel or gold to reduce contact between the surface of low-alloy steel and the environment are also being tested. Comparison of the crack growth rates from the bare and plated specimens under identical water chemistry and loading conditions will provide insight into whether electron transfer through the oxide film on the bulk surface of the ferritic steel is important in the overall SCC process. Similarly, the composite specimens were plated with nickel to simulate crack growth in a clad ferritic steel vessel, where only the crack surface of low-alloy steel is exposed to the environment. The results will be compared with data obtained from bare low-alloy steel specimens to establish whether data from the latter can be used to characterize the SCC behavior of a clad ferritic vessel. In addition, the composite specimens will be used to determine whether the threshold stress intensity factor for SCC in the ferritic material is influenced by the nature of the starter crack, i.e. a SCC crack in the In-182 or a fatigue precrack in the conventional specimens. These tests are being performed in high-purity water with 0.2-0.3 ppm dissolved oxygen at 289°C at a load ratio of 0.95 and a frequency of 0.08 Hz.

IMPROVING CHECK VALVE RELIABILITY
THROUGH RESEARCH REGARDING DEGRADATION OF INTERNALS

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Since the multiple check valve failures that occurred in 1985 at U.S. nuclear power plants, a significant amount of research has been done toward predicting the performance and degradation of check valves in operation. Ideally, check valves should be sized to provide full disc lift under normal flow conditions, and should be located sufficiently away from upstream flow disturbances to achieve long-term, trouble-free operation. In practice, however, a vast majority of the check valves at power plants do not fulfill these ideal requirements. In spite of this apparent misapplication, only a small percentage of these valves suffer from accelerated degradation.

This paper will present the results from research undertaken to quantify the effects of a number of types of upstream flow disturbances on the degradation of swing check valve internals. Under the Phase I research program sponsored by NRC/Small Business Innovation Research (SBIR), over 2,000 tests were run on instrumented swing check valves. The effects of upstream disturbance proximity, flow conditions, and valve geometry were systematically varied according to a predefined matrix. Disc motion and impact force against the backstop were measured in the tests. These measurements form the basis of predictive fatigue and wear models being developed under Phase II of a continuing research program. The theoretical wear models are being refined by comparison against controlled wear tests using aluminum hinge pins to accelerate wear. Bending stress caused by disc stud impact against the stop is also being measured by strain gages to refine the fatigue prediction model.

A summary of overall goals, results to date, and comparisons against available plant data will be presented in this paper.

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EVALUATION OF CHECK VALVE MONITORING METHODS*

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Summary

Check valves are used extensively in nuclear plant safety systems and balance of plant (BOP) systems. Catastrophic failures of swing check valves have occurred in several nuclear plants and have resulted in water hammer, overpressurization of low-pressure systems and damage to flow system components. These failures have largely been attributed to severe degradation of internal parts (i.e. hinge pins, hinge arms, discs, and disc nut pins) resulting from instability (flutter) of check valve discs under normal plant operating conditions. Present surveillance requirements for check valves have been inadequate for timely detection and trending of such degradation.

The NRC has had a continuing strong interest in resolving check valve problems. In support of the NRC's Nuclear Plant Aging Research (NPAR) Program, ORNL has been carrying out an aging assessment of check valves. A primary objective of these studies is to identify, evaluate, and recommend methods for detecting check valve degradation and incipient failure.

Several developmental or commercially available check valve diagnostic monitoring methods developed by others are being evaluated by ORNL, including those using acoustic emission, ultrasonic, and radiographic techniques. Two other potential monitoring techniques identified by ORNL are also being studied; one based on fluid pressure noise and one based on magnetic flux. The evaluations in each case have focussed on identifying capabilities to provide diagnostic information useful in determining check valve aging and service wear effects (degradation) and/or indication of undesirable operating modes (e.g., disc instability).

A description of each monitoring method is provided in this paper including examples of test data acquired under controlled laboratory conditions. In some cases, field test data acquired in-situ are also presented. The advantages and disadvantages of each method are compared and suggested areas in need of further development are identified.

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RESULTS OF GATE VALVE FLOW INTERRUPTION TESTS IN THE RWCU LINE ENVIRONMENT

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Recent testing sponsored by the U.S. Nuclear Regulatory Commission (NRC) showed that for at least some valves installed in nuclear applications, the equations used by industry to size the valves do not conservatively calculate the thrust needed to close the valves under design basis loadings. The tests also showed that the results of in situ valve testing at lower loadings cannot be extrapolated to design basis loadings. The testing was conducted by researchers from the Idaho National Engineering Laboratory (INEL) to provide to the NRC a portion of the technical data base for the NRC effort regarding Generic Issue 87 (GI-87) "Failure of HPCI Steam Line Without Isolation." The test program also provides information applicable to Generic Issue II.E.6.1, "Insitu Testing of Valves" and a related document, IE Bulletin 85-03, "Motor Operated Valve Common Mode Failures During Plant Transient due to Improper Switch Settings."

Of the three BWR process lines covered under GI-87, an unisolated break in the RWCU supply line was selected for the first phase of testing because such a break would have the greatest safety impact. The high pressure coolant injection (HPCI) steam supply line and the reactor core isolation cooling (RCIC) steam supply line will be addressed in subsequent research efforts. All three GI-87 process lines have common features. All communicate with the primary system, pass through containment, and have normally open isolation valves.

In this initial test program, two representative RWCU isolation valves were subjected to the hydraulic qualification tests described in ANSI B16.41, the nuclear valves qualification standard, and then to full flow RWCU pipe break flow interruption tests. In all, fourteen flow interruption tests were performed, ten on Valve A and four on Valve B. In the Valve A tests, the parametric study included varying both the degree of inlet water subcooling and the pressure. Break flows were maintained throughout the 30-second valve closure. The four Valve B tests were all performed at a normal BWR 10°F subcooling, and the inlet pressure only was varied. The test loop and valves were instrumented to determine the valve response to flow, including a load cell installed in the valve stems to measure thrust.

Test results show that for the valves tested, the variables used by industry for determining valve thrust are not conservative. The results further indicate that internal valve design differences can result in large response differences and that prototypical testing may be necessary to determine actual valve performance.

Valve Testing for UK PWR Safety Applications

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Extensive programmes of testing and development have been established by the Central Electricity Generating Board (CEGB) to support the design, construction and operation of Sizewell 'B', the UK's first PWR.

A Blowdown Rig for the Assessment of Valve Operability - (BRAVO) has been constructed at the CEGB Marchwood Engineering Laboratory to reproduce PWR Pressuriser fluid conditions for the full scale testing of Pressuriser Relief System (PRS) valves. A full size tandem pair of Pilot Operated Safety Relief Valves (POSRVs) is being tested under the full range of pressuriser fluid conditions. Tests to date have produced important data on the performance of the valve in its Cold Overpressure protection mode of operation and on methods for the in-service testing of the valve.

Also, a full size pressuriser safety valve has been tested under full PRS fluid conditions to develop a methodology for the pre-service testing of the Sizewell valves. Further work will be carried out to develop procedures for the in-service testing of the valve.

In the Main Steam Safety Valve test programme carried out at the Siemens-KWU Test Facilities, a single MSSV from three potential suppliers was tested under full secondary system conditions. The test results have been analysed and are reflected in the CEGB's arrangements for the pre-service and in-service testing of the Sizewell MSSVs.

Valves required to interrupt pipebreak flow must be qualified for this duty by testing or a combination of testing and analysis. To obtain guidance on the performance of such tests gate and globe valves have been subjected to simulated pipebreaks under PWR primary circuit conditions. In the light of problems encountered with gate valve closure under these conditions, further tests are currently being carried out on the BRAVO facility on a gate valve, in preparation for the full scale flow interruption qualification testing of the Sizewell main steam isolation valve on the Siemens-KWU Test facility in 1990.

The results obtained to date have influenced the selection of hardware for Sizewell, supported the licensing process and provided an input to the specification of in-service testing procedures.

OVERVIEW AND STATUS OF EPRI MOV TEST PROGRAM

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An MOV Test Program is being planned by EPRI in coordination with other MOV-related activities in the nuclear power industry. The objective of the EPRI MOV Test Program is to obtain data on MOV performance and extrapolation of performance, for a range of MOV types and applications typical in nuclear power plants. The ultimate intent of the program is to develop a methodology and supporting data to assist utilities in their responses to recent requirements (NRC Generic letter) that the operability of MOV's be evaluated under design basis conditions. The results of the EPRI Program will allow utilities to effectively plan and interpret MOV tests, and to simplify and minimize the required in-plant testing.

Phase 1 of the EPRI MOV Test Program is currently underway. This phase has focused on determining the data needs of the industry and on identifying the extent to which existing MOV data (from in-plant tests or from other programs) satisfy these needs. This step permits EPRI to concentrate the additional testing and data evaluation in Phase 2 where it is most needed. The Phase 1 work includes:

- An analytical/experience based evaluation of the key factors affecting MOV performance and performance extrapolation.
- An evaluation of the range of types and applications of MOVs in nuclear power plants.
- A review of selected available MOV test data to determine the extent to which these data cover the key factors and the range of MOVs.

The product of the technical evaluations in Phase 1 is as follows:

- Definition of data needs for method validation.
- Plan for tests and evaluations to be conducted in Phase 2.

SUMMARY

TITLE: " A Systems Engineered Approach Towards Improved Safety and Performance of Motor Operated Valves"

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ORGANIZATION: Bechtel-KWU Alliance

A global systems approach, which considers all parameters that potentially affect valve and operator performance, is necessary to improve confidence that motor operated valves will maintain integrity and perform their intended function at design basis conditions.

These parameters can only be properly evaluated through a comprehensive process which integrates engineering, analysis, in-situ and full flow testing, maintenance and subsequent trending measurement.

This paper describes how this process has been successfully used in Europe and how it applies to the U.S. with improved performance and reliability of motor operated valves. The testing, experiences and lessons learned are critical in satisfying the NRC generic letter.

The process can be visualized as two interactive paths; the verification of the ability to function (design verification) and the verification of readiness to function (actual condition of the MOV). Neither of the two paths alone is capable of verifying performance of a MOV during design basis conditions. However, applied in a combined approach, one can predict the MOV performance.

The design verification is accomplished through an engineering evaluation supplemented by computer programs utilizing data from extensive full flow tests, separate effects component testing, material research and testing and confirmatory finite element analysis. The engineering evaluation also considers parameters such as clearance, tolerance and deflection behavior. The types of testing, components tested, test facilities, and results will be discussed. Recommended testing of U.S. components important in answering the Generic Letter will also be provided. Simplified test devices have been shown to be extremely beneficial in the extrapolation of full scale testing.

A computer analysis is used to streamline calculations for the required torque to seat, to unseat and stem travel under design basis conditions. Analysis is also performed for valve parts in the load path and pressure boundary parts. The technical considerations for determining the load predictions and evaluation of components will be described.

However, the design conditions can not practically be simulated in the plant, especially accident pressure differentials and degraded voltage. In addition, the actual plant valves are perhaps in an unknown condition of wear and maintenance.

Therefore, the need exists for a readiness to function verification. The analysis described above is repeated for conditions which can be achieved easily in the plant during insitu testing. This means that the analysis maybe performed for conditions other than design. The analysis calculates the required torque for seating, unseating and stem travel under the in-situ conditions.

The results of this analysis are compared to the results measured during insitu (diagnostic) testing. The number of variables are reduced allowing the comparison of measured values to calculated values with more confidence. The valve/actuator is then assessed for the effects of wear and maintenance condition to achieve a readiness to function status.

A "state of the art" diagnostic testing must be used in order to obtain sufficient definition for diagnosis. The preferred method is a three-phase inline active power measurement combined with a switch setting measurement from the motor control center (MCC), called the leading measurement. In the case of insufficient data for the MOV, a one time baseline measurement at the MOV is required using strain gauge measurement, etc. However, future trending measurements will be performed from the MCC to reduce man-rem dose and compared to previous signatures. The diagnostic system is designed to obtain all necessary data with only one stroking cycle of the valve thereby reducing seat wear.

Valve and actuator improvements as well as philosophies regarding valve application, limit switch seating, diagnostic testing, and in-service testing will be discussed. The merits of simple component test devices, such as the simulator for bench testing each actuator for simulated valve loading will also be provided.

A comprehensive approach, based on the principles described herein, will be presented to satisfy the generic letter.

GOOD PRACTICES FOR EFFECTIVE MAINTENANCE TO MANAGE AGING AT NPPs[(a)]

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INTRODUCTION

The Nuclear Regulatory Commission's Office of Nuclear Regulatory Research is sponsoring this effort to study maintenance as it relates to aging, under the Nuclear Plant Aging Research (NPAR). Maintenance is the primary means of combating and correcting the effects of aging degradation at nuclear power plants. Maintenance effectiveness directly affects the safety of nuclear power plants. Several recent plant events have shown that improper maintenance or a lack of maintenance can be a significant contributory cause of plant incidents (e.g., transients at Rancho Seco and Davis Besse, and the Salem anticipated transient without scram event). In these, and other events, safety related plant equipment functions have been impaired by poor maintenance practices or a lack of maintenance on the specific equipment or in ancillary equipment that affects the ability of the safety related equipment to function. In some cases, the ancillary equipment has been nonsafety-related equipment, or balance-of-plant equipment that has not received adequate maintenance attention. In addition to the potential for causing safety significant plant transients, poor or lacking maintenance may allow the licensing basis of the plant to be eroded without detection.

In some cases, time-related aging degradation of equipment that has not been detected, corrected or managed by the maintenance program has been a significant contributory factor.

The NPAR program has developed significant information on aging degradation, detection, mitigation, and correction practices for safety-significant structures, systems and components that can be factored into maintenance programs and their effectiveness.

SCOPE

Maintenance to manage aging is focusing on the following:

- a systematic approach to maintenance for aging management,
- the measurement of maintenance program effectiveness in aging management,
- the conduct of maintenance tasks for detecting, mitigating and correcting aging degradation,
- the qualification and training of maintenance personnel in aging modes, effects, and consequences,

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(a) Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under U.S. DOE Contract DE-AC06-76RLO 1830.

- the control of maintenance work to ensure the effective management and control of aging, and

- the maintenance facilities and equipment needed to manage aging degradation.

Specifically, lessons learned regarding the maintenance of motor-operated valves (MOVs) from Oak Ridge National Laboratory's research will be evaluated for wider applications to maintenance programs. The selection of MOVs provides a basis for comparison with one of the other industry programs being investigated. In addition, the hardware studies at PNL (i.e., on diesel generators and snubbers) and other NPAR laboratories will provide input to this effort.

The current major inputs from industry sources outside the commercial reactor field are:

- Japanese nuclear plant maintenance practices,

- U.S. Air Force B-52 Bomber maintenance practices,

- U.S. Navy submarine maintenance practices, and

- U.S. commercial airline maintenance practices.

These investigations of other industries seek tested solution to problems which could be useful to the U.S. nuclear power industry.

AGING RISK ASSESSMENT METHODOLOGY: DEMONSTRATION STUDY ON A PWR AUXILIARY FEEDWATER SYSTEM

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The problem of analyzing age-dependent plant risk necessitates rethinking a basic principle of PRA; the assumption that all equipment failure probabilities are constant. This paper presents a methodology to detect and quantify increasing failure rates. The methodology is applied to a 16 year old PWR auxiliary feedwater system. Based on plant-specific maintenance data and the accompanying PRA, this study concludes that statistically significant age-dependent trends exist and that the implied core damage frequency is increasing, and identifies which equipment items are the major age-dependent contributors to core damage frequency. A useful product of the work is the development of a step-by-step procedure for aging risk assessment.

A number of assumptions were made in order to accomplish this work. The assumptions involve the interpretation of raw data, the analysis of data, and use of data analysis results in PRA models. Regarding the data itself, it was assumed that the component maintenance records obtained for use in this study were complete and that the date the equipment item was returned to service was an acceptable surrogate for the date of failure. For the analysis, it was assumed that component failures obey a nonhomogeneous Poisson process, with hazard function $\lambda(t) = e^{\alpha + \beta t}$. Diagnostic checks indicate that the data are consistent with this model. It was further assumed that the beginning of the observation coincided with installation, and that the PRA is adequately complete as modeled, for all but the auxiliary feedwater system.

Results of the data analysis are presented in Table 1. Note that significant increasing rates of failure were exhibited by pump discharge header check valves for the back-leakage mode, turbine-driven pumps for failure to run, individual header motor-operated valves for failure to open and for feedwater pump steam-binding. The significance level is the probability of observing the data, or any more extreme data, if the data were being generated by a constant failure rate process. The very low numbers show evidence of increasing rate of failure; for instance, both the check valves and turbine driven pumps have significance levels less than 1%.

Using failure probability estimates for the equipment listed in Table 1 as inputs to PRA models of the AFW system, the implied, age-dependent core damage frequency was calculated. Results indicate a total increase of one order of magnitude in ten years by propagating the mean values of the basic events throughout the PRA. Nearly a two order of magnitude increase is calculated by sampling the distributions for the individual basic events. Table 2 shows the age-dependent results for employing a broad and narrow definition of failure, respectively. For the period 8 to 10 years, Figure 1 indicates the contributors to the increase in risk by shaded regions beneath the total time-dependent risk curve, based on the broad definition of failure. The check valves dominate this period, but steam-binding and turbine driven pumps are rapidly increasing.

Overall, the work has shown that age-dependent behavior can be inferred from minimal data sources, and that the implications of risk can be computed from this data. These capabilities provide a starting point from which to develop age-dependent PRA for aging management of nuclear power plants. The primary purpose of such aging analysis is not for extrapolation of risk into the distant future, but to help determine near-future contributors to risk and to add insight to decisions for corrective actions. This use of aging risk analysis is consistent with the primary use of PRA in general.

TABLE 1.

Component Group	Failure Mode	Significance Level (%)	Hazard Doubling Time (Years)
Pump discharge header check valves	Back leakage	<0.02	1
Turbine driven pumps	Fail to run	0.2	2
Individual header motor operated valves	Failure to open	5.7	4
Feedwater pump	Steam-binding	2.6	1

TABLE 2. TOTAL TIME-DEPENDENT PLANT CORE DAMAGE FREQUENCY

	Broad Failure Definition		Narrow Failure Definition	
Year	Propagating Means	Propagating Uncertainties	Propagating Means	Propagating Uncertainties
0	6.6E-04	6.5E-04	6.9-04	7.2E-04
5	7.4E-04	7.4E-04	6.8-04	6.8E-04
7	1.5E-03	1.4E-03	6.9-04	6.8E-04
8	2.6E-03	2.7E-03	7.1-04	8.7E-04
10	6.6E-03	3.3E-02	2.8-03	2.9E-02

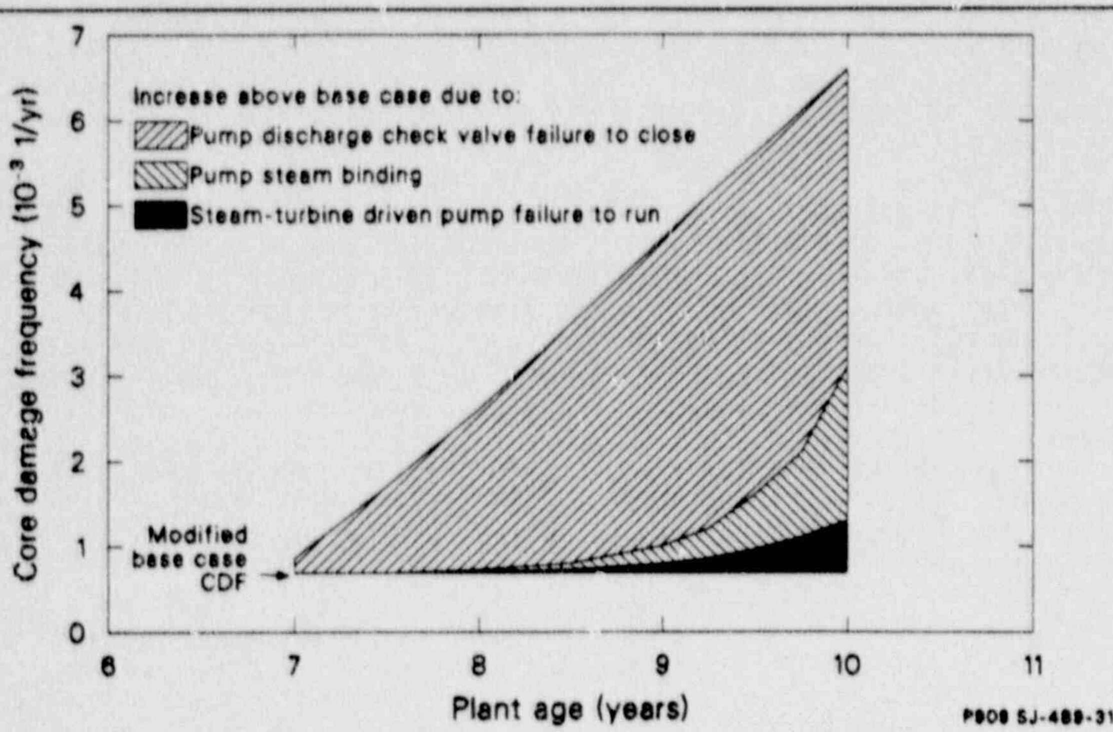

Figure 1

SAND89-1755A*

Loss-of-Coolant Accident (LOCA) Testing of Aged Cables for Nuclear Plant Life Extension

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SUMMARY

Sandia National Laboratories is currently conducting long-term aging research on representative samples of nuclear power plant Class 1E cables. The objectives of this program are to determine the suitability of these cables for extended life (beyond 40 year design basis) and to assess various cable condition monitoring (CM) techniques for predicting remaining cable life. Twelve different cable products have been aged for long times at relatively mild exposure conditions with various CM techniques employed during the aging process.

Three separate test chambers were used for aging groups of cables to nominal equivalent lifetimes of 20, 40, and 60 years. The aging included a simultaneous thermal and radiation exposure at about 100°C and 10 krad/hr for 3, 6, and 9 months for the three chambers, giving total doses of 20, 40, and 60 Mrads. During aging, small cable specimens, insulation specimens, and jacket specimens were removed at 1-month intervals. These specimens will be used for future elongation and dielectric withstand measurements.

Following the aging process, the cables in each chamber were exposed to a sequential accident profile consisting of 110 Mrad of high dose rate gamma irradiation followed by a simulated design basis loss-of-coolant accident (LOCA) steam exposure (except the 20-year chamber which is still being aged). The LOCA conditions followed the IEEE 323-1974, Appendix A recommended profile (including superheated steam conditions) for the first 4 days. The remainder of the profile was 121°C for 6 days.

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*The Long-Term Cable Aging Program is supported by the United States Nuclear Regulatory Commission and performed at Sandia National Laboratories, which is operated for the U.S. Department of Energy under contract number DE-AC04-76DP00789.

During the LOCA tests, the cables were energized to a nominal voltage of 110 Vdc. Individual cable insulation resistances (IRs) were monitored automatically throughout the LOCA test at varying scan intervals from 10 seconds during the transient portions of the tests to 5 minutes during long steady portions. Measurements on each individual cable were also performed periodically using an IR test apparatus that was used during aging. Based on these measurements, selected plots of IR as a function of time and temperature for various cables will be presented. Several cables had low IRs during the test and some of these caused 1 A fuses to blow. A number of the cables performed well during both the 40- and 60-year LOCA tests, indicating a good potential for life extension for many popular cable products.

Life Assessment Procedures for Major LWR Components[a]

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Summary

The Aging Assessment and Mitigation Project is a part of the USNRC Nuclear Plant Aging Research Program. The main objective of the project is to develop an understanding of the aging degradation of the major light water reactor (LWR) structures and components and to develop procedures for predicting the useful life of these components so that the impact of aging on the safe operation of nuclear power plants can be evaluated and addressed. This paper describes the current status of the project and presents life assessment procedures for pressurized water reactor (PWR) steam generator tubes and LWR cast stainless steel components.

The major effort of the project consists of integrating, evaluating, and updating the technical information relevant to aging and license renewal from current or completed NRC and industry research programs. The project has five steps: (1) identify and prioritize the major reactor components, (2) identify degradation sites, mechanisms, stressors, and potential failure modes of each component and then evaluate the current inservice inspection methods, (3) assess advanced inspection, surveillance, and monitoring methods and evaluate mitigating methods to reduce aging damage, (4) develop residual life assessment models and procedures, and (5) support the development of technical criteria for license renewal. The first two steps have been completed, which include qualitative aging assessment of twenty major components. A thorough assessment of both advanced fatigue monitoring and advanced material evaluation methods is being conducted as part of Step 3. Various life assessment procedures are being developed for five major components as part of Step 4: PWR reactor pressure vessel, metal containments, reinforced concrete containments, steam generator tubes, and cast stainless steel components. The project results will be used to develop technical guidelines for making license renewal decisions.

A general life assessment procedure for many of the major components is as follows: (a) evaluate the damage state of the component at the beginning of the operating period, (b) estimate the additional damage expected during the operating period, (c) evaluate component integrity at the end of the operating period to ensure that acceptable safety margins exist, and (d) establish an inservice inspection program.

The heat-exchanger tubes are the life-limiting components of a PWR steam generator. Statistical modelling is used to predict the rate of heat-exchanger tube degradation because of the large number of tubes in each generator and because of the periodic assessment of tube condition

a. Work sponsored by the United States Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under DOE Contract No. DE-AC07-76ID01570.

that is routinely made for PWR steam generators by eddy-current testing. Where possible, mechanistic understanding of the degradation processes is used to estimate the dependence of degradation rates on stressors such as stress, temperature, and chemical environment. This allows statistical data on degradation in one steam generator location to be used in estimating degradation rates in locations where the effects have not yet been observed and assists in the evaluation of changes in operating conditions that change the stressors.

The life assessment procedure presented in this report uses a Weibull distribution to describe the times to failure of steam generator tubes for failures caused by most of the corrosion and stress corrosion cracking mechanisms. The Weibull distribution was chosen in preference to other possibilities because it is easy to handle mathematically, and, in test cases, was found to accurately describe the measured data and provide useful predictions of future behavior. With the aid of Weibull distribution, one can project data from the failure of relatively few tubes to estimate future tube failure rates and to predict when tube failures from a given mechanism will be so numerous as to affect the operation of the steam generator.

The three most widespread types of degradation mechanisms affecting U.S. PWR plants are (1) pure-water stress corrosion cracking (PWSCC) on the primary side, (2) intergranular attack (IGA) and intergranular stress corrosion cracking (IGSCC) on the secondary side, and (3) fretting wear and thinning. Other corrosion mechanisms, such as wastage, pitting, and denting have occurred in many older steam generators but have been avoided in most newer generators by changes in operating procedures. Fatigue failure of a U-bend has occurred at one plant and the failure is considered significant because of its location and nature.

The life assessment procedure for steam generator tubes specifies the types of information regarding the construction and operating history of the generator that should be gathered and explains how this information should be used to identify which modes of degradation are most probable. Methods for evaluating inspection data are discussed, and the procedure for performing a combined statistical analysis for several concurrent types of degradation is presented.

The life assessment procedure for cast stainless steel components addresses the degradation caused by thermal embrittlement and fatigue. Thermal embrittlement affects both Charpy V-notch impact energy and fracture toughness. The procedure is organized into eight steps. The first three steps involve the review of design, fabrication, construction, inservice inspection, and operating history records. The fourth step involves a fracture-mechanics evaluation to determine if any existing or undetected defects could potentially impair the structural integrity of the component during the next operating period. In the fifth step, the material condition at the end of next operating period is estimated using an analytical model proposed in the paper, microstructural data, and/or measured properties. In the sixth step, the results of the fourth and fifth steps are combined to evaluate the structural integrity of the component. In the seventh step the required actions (none, repair, replace, or shut down) are chosen. The eighth step involves establishing the plan for next inservice inspection.

COMPREHENSIVE AGING ASSESSMENT OF CIRCUIT BREAKERS AND RELAYS FOR NUCLEAR PLANT AGING RESEARCH (NPAR) PROGRAM PHASE II

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SUMMARY

The Phase I study- NUREG/CR-4715,BNL-NUREG-52017 AN,RV, "AN AGING ASSESSMENT OF RELAYS AND CIRCUIT BREAKERS AND SYSTEM INTERACTIONS," identified relays and circuit breakers used in nuclear plants, failure mechanisms of these relays and circuit breakers, that some failure mechanisms are age related, and that failure of safety systems is possible from relays and circuit breaker failures if adequate maintenance and testing are not performed. It also proposed some potential inspection, surveillance and condition monitoring methods to detect significant aging methods prior to loss of safety function. The objectives of Phase II are to: (1) identify and characterize aging and service wear effects of circuit breakers and relays which, if unchecked, could impair plant safety; (2) identify and develop methods of inspection, surveillance, and condition monitoring, and of evaluating residual life of circuit breakers and relays, which will assure timely detection of significant aging effects prior to loss of safety function; (3) evaluate the effectiveness of storage, maintenance, repair and replacement practices in mitigating the rate and extent of degradation in circuit breakers and relays caused by aging and service wear.

The relays and circuit breakers which are being reviewed are:

- **Protective Relays** - they protect plant power systems from effects of electrical overloads, faults and transients
- **Auxiliary Relays** - actuated by protective relays for high current applications
- **Control Relays** - used in nuclear protective system logic
- **Timing Relays** - delays operating function until initiating condition has existed for a selected time
- **Electronic Relays** - solid state device used in protective or control relay applications
- **Molded Case Circuit Breakers** - 480 volt and below; they are the most prevalent, used to supply individual circuits and feeders for low voltage AC and DC distribution
- **Metal Clad 480 volt Circuit Breakers** - used in the power supply to 480 volt distribution buss as well as to feed

individual circuits for major safety related equipment such as medium sized motors

o **Metal Clad 4KV Circuit Breakers** - these are housed in large metal cabinets and are used for main circuit breakers for large safety related equipment and the emergency power busses

The research effort has reviewed and is verifying improved inspection, surveillance and condition monitoring (ISM) methods. Testing of naturally aged and degraded circuit breakers and relays is being performed. Eighteen ISM methods are being evaluated. They include: Vibration/Acoustics, Surge Current Comparison, Pick-up/Drop-out voltage, Inrush/Holding current, Current Signature Analysis, Operability, Set point Drift, Timing tests, Magnetic Flux Monitoring, Contact/coil Resistance, High Potential Test, Insulation Resistance, Infrared Thermal Scanning, Infrared Pyrometry, On-contact Temperature measurement, Ion Detection, Lubrication, Visual inspection and Root cause Failure Analysis. Some techniques are intrusive, requiring disturbing of leads, etc.; the following are the most encouraging in that they are non-intrusive, can be used with all types of circuit breakers and relays, data can be obtained rapidly, trended and computer analyzed:

o **Vibration/Acoustics** - preliminary tests have shown that circuit breakers and relays have distinct and repeatable vibration signatures, and aged and refurbished devices experience differences in amplitude and frequency

o **Infrared Thermal Scanning** - this technique has shown differences in thermal signatures in aged devices

Additional tasks include (1) in-situ testing at plants to confirm analyses and laboratory results and to provide estimates of cost effectiveness and practicability of application; (2) evaluation of the role of maintenance in mitigating age effects and (3) application and /or development of methodology which will provide a service life prediction and define surveillance and inspection intervals.

The research results will be utilized to identify inspection, surveillance and condition monitoring needs, define inspection intervals, provide service life parameter limits and service life prediction methodology. Recommendations will be made to modify or develop regulatory guides and may provide additional information to supplement existing NRC bulletin on counterfeit and refurbished circuit breakers.

BEYOND DESIGN BASIS ACCIDENTS IN SPENT FUEL POOLS GENERIC ISSUE 82

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INTRODUCTION

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool (SFP) and storage racks is to cool the spent fuel assemblies and maintain them in a subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks. It was concluded, in WASH-1400, that the risks from spent fuel storage were orders of magnitude below those involving the reactor core because of the simplicity of the spent fuel storage pool design.

The reasons for the re-examination of SFP accidents are twofold. First, spent fuel is being stored instead of reprocessed. This has led to the expansion of onsite fuel storage by means of high-density storage racks, which results in a larger inventory of fission products in the pool, a greater heat load on the pool cooling system, and less distance between adjacent fuel assemblies. Second, some laboratory studies have provided evidence of the possibility of fire propagation between assemblies in an air cooled environment. In addition, in recent years, increasing knowledge in the geosciences has led to a better understanding that, although still highly unlikely, it is more likely that nuclear power plants in the Eastern United States (i.e., east of the Rocky Mountains) could be subjected to earthquake ground motion greater than for which the plants were designed.

SUMMARY OF TECHNICAL FINDINGS

Assuming that the water is drained, or boiled off from the SFP, the fuel rods will heat up until the buoyancy-driven air flow is sufficient to prevent further heatup. If the decay heat level is high enough to heat the fuel rod cladding to about 900 °C (1650 °F) the oxidation becomes self-sustaining, resulting in a Zircaloy cladding fire. Propagation of the Zircaloy cladding fire to older adjacent assemblies is likely if the decay heat level in an older adjacent assembly is high enough to heat that assembly to within 100 to 200 °C (200 to 400 °F) of the self-sustaining oxidation temperature. Although propagation of a Zircaloy cladding fire to one to two year old fuel by only thermal radiation can occur, the older fuel would have to be next to the hottest assemblies.

The conditional probability of a Zircaloy cladding fire given a complete loss of water was found to be 1.0 for PWRs and 0.25 for BWRs. The PWR value is based on the use of high-density storage racks and the BWR value is selected based on the use of high-density directional storage racks, with the channel box in place. The use of open frame racks or cylindrical racks with large inlet holes would result in a reduction in risk. The cooling time to preclude a Zircaloy cladding fire could be reduced to less than 20 days, for a conditional probability of 0.05 of a Zircaloy fire for both fuel types.

The risk from the storage of spent fuel in the SFP at light water reactors is dominated by the beyond design basis earthquake accident scenario. The seismic capacities, or fragility, of two older SFPs indicate that the high confidence of low probability of failure (HCLPF) is about three times the safe shutdown earthquake (SSE) design level. The HCLPF values are estimated to be in the 0.5 to 0.65 g range. The structural capacity of the elevated BWR pool is lower than that for the PWR pool located at the ground level, however the lower conditional probability of a Zircaloy fire for the BWR fuel assembly design offsets the higher seismic failure frequency. The probability of a Zircaloy cladding fire, resulting from the loss of water from the SFP, is estimated to have a mean value of $2x10^{-6}$ per reactor year for either the PWR or the BWR SFP. The seismic event contributes over 90% of the PWR spent fuel damage probability, and nearly 95% for the BWR.

The source term for the SFP accident is not the same as the source term associated with core damage accidents. The consequences of a SFP accident which results in the complete loss of water are dominated by the long lived isotopes, such as cesium and strontium. The health consequences are dominated by the risk of latent cancer fatalities due to long term exposures.

SUMMARY OF VALUE/IMPACT STUDY

The best estimate values are based on a population density of 340 people per square mile within a 50 mile radius from the site and result from the release of radionuclides from the last fuel discharge, 90 days after being discharged and an accident frequency of $2x10^{-6}$ per reactor year. The best estimate of the consequences of a SFP accident which results in spent fuel damage to approximately one-third of an equivalent reactor core is $8x10^{6}$ person-rem. This total dose translates to a public health risk from a SFP accident of 480 person-rem over an average remaining lifetime of 30 years. The best estimate offsite property damage cost is $4,000 million (1988 $s). The best estimate of the onsite costs for a SFP accident is $1,180 million (1988 $s), including five years of replacement power to replace the damaged SFP. Based on an average remaining lifetime of 30 years, an accident frequency of $2x10^{-6}$ per reactor year and a 5% discount rate, the present value of the offsite property damage is estimated to be $124,300 and the present value of the onsite property damage is estimated to be $32,400.

The value/impact and cost-benefit evaluations for the proposed alternatives for Generic Issue 82 do not indicate that cost effective options are available to mitigate the risk of beyond design basis accidents in SFPs. The option to use low density storage racks for recently discharged fuel has a best estimate value/impact ratio of $32,000 per averted person-rem based on a reduction in spent fuel damage frequency of $2x10^{-6}$ per reactor year. The use of a post-accident spray system, with an assumed decontamination factor of 45, to mitigate the consequences of a SFP accident has a best estimate value/impact ratio of $3,300 per averted person rem.

CONCLUSIONS AND RECOMMENDATIONS

The backfit criteria (10 CFR 50.109) that (1) a substantial increase in the overall protection of the public health and safety is achieved, and (2) the direct and indirect costs of implementation are justified are not met, and the "No Action" alternative is recommended for the resolution of GI-82.

The risk and consequences of a SFP accident appear to meet the Safety Goal Policy Statement objectives. They would also meet the proposed $1x10^{-6}$ per reactor year large-release frequency guideline, at least pending definition of a "large release" by the Commission. Therefore the recommended resolution, the "No Action" alternative, is justified.

Although these studies conclude that most of the SFP risk is derived from beyond design basis earthquakes, this risk is no greater than the risk from core damage accidents due to seismic events beyond the safe-shutdown earthquake. Therefore, reducing the risk from SFPs due to events beyond the safe-shutdown earthquake would still leave a comparable risk due to core damage accidents. Because of the large inherent safety margins in the design and construction of the SFP, the "No Action" alternative is justified.

Generic Safety Issue 106
Piping and the Use of Highly Combustible Gases in Vital Areas

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General Design Criterion 3, "Fire Protection," in Appendix A to 10CFR Part 50 states that "Structures, systems and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effects of fires and explosions." With respect to this criterion, it is noted that combustible gases such as hydrogen, propane, acetylene, and methane are used during normal operation of nuclear power plants as well as in plant laboratories. All of these gases should be considered with respect to their onsite storage requirements as well as their distribution and use in plant buildings. Hydrogen is the most prevalent combustible gas used in nuclear power plants. and is of principal interest.

Hydrogen has been used for many years as a coolant for electric generators in BWRs and PWRs and as an additive in the volume control tank in the chemical volume and control system of PWRs to control oxygen in the reactor coolant system during plant operation. In addition, a number of BWRs have recently installed or will use hydrogen-water chemistry (HWC) systems to reduce oxygen concentration in the reactor coolant system during plant operation and, hence, reduce the potential for intergranular stress corrosion cracking. This HWC application for BWRs involves the use of large quantities of hydrogen and oxygen. Gaseous hydrogen storage facilities may have a number of fixed tanks containing hydrogen at high pressures (e.g. up to about 2400 psig) and ambient temperature or involve the use of large, transportable tanks containing high pressure hydrogen at ambient temperature. The storage facility has a pressure control station to reduce the hydrogen pressure to less than about 200 psig prior to distribution to plant buildings. such as the auxiliary building of PWRs and the turbine building of PWRs and BWRs via small diameter, field-run piping. Liquid hydrogen storage is in vacuum-jacketed vessels at low temperature and at pressures up to about 150 psig. The hydrogen is vaporized in ambient temperature vaporizers prior to distribution to the plant buildings.

The storage area should be located at a distance from safety-related buildings and air intakes since it contains large quantities of hydrogen. However, plants are known for which the storage area is in close proximity to safety-related buildings or structures. The piping to electric generators typically would not be near safety-related equipment. However, again plants are known which have safety equipment such as diesel generators, batteries, switchgear, and motor control centers in the turbine building. The auxiliary building in PWRs which contains the volume control tank is a safety-related structure, which houses most of the components of the safety-related systems of the plant. Leaks or breaks in that portion of the hydrogen distribution system within the PWR auxiliary building or PWR or BWR turbine building could result in the accumulation of an explosive mixture of air and hydrogen and represent a threat to plant safety because of the potential loss of safety-related equipment. Hydrogen gas also represents, to a lesser degree, a potential threat to safety-related equipment because of its presence in PWR waste gas systems, distribution to the containment recombiner system in some PWRs, the augmented off gas system in BWRs, small bottle supplies in reactor/auxiliary buildings and in turbine buildings, and from station batteries.

The safety issue represented by hydrogen, propane, acetylene, methane, and other combustible gases is identified as Generic Safety Issue (GSI) 106, "Piping and the Use of Highly Combustible Gases in Vital Areas." It was identified in NUREG-0705 and is related to GSI 136 which is concerned with the new uses of large quantities of hydrogen and oxygen in BWRs for the HWC system. The staff approval of EPRI topical report NP-5283-SR-A in July 1987 was the resolution of GSI 136. This EPRI report provides guidance for the design, construction, and operation of permanent hydrogen systems for the hydrogen water chemistry applications. In the acceptance letter, the staff referred to recent potentially hazardous events involving hydrogen supplies for generator cooling and the volume control tank for PWRs and suggested that EPRI modify the report to include guidance for these applications. This modification was not made.

The hydrogen system is not a safety-related system. Hence, the hydrogen distribution systems and design features to ensure safe operation of the plant have not normally been included in the Final Safety Analysis Report (FSAR). At this time, the only staff requirements in the Standard Review Plan (SRP) that are directly pertinent to GSI 106 are in Revision 2 to SRP 9.5.1, dated July 1981. The work on Issue 106 will provide (a) PWR plant and BWR plant information including information gathering on propane, acetylene, methane, and any other identified combustible gases but excluding the new hydrogen water chemistry installations and use covered by the EPRI report, (b) recommendations for safety features, such as in the EPRI report, to reduce or mitigate the consequences of hydrogen system leaks or breaks beginning with the yard hydrogen tank farm through the end use in reactor/auxiliary/turbine buildings of PWRs and BWRs (including hydrogen from waste gas, recombiner use, portable bottles and station batteries), and (c) cost-benefit analyses needed for the resolution of Issue 106.

GENERIC ISSUE 57
EFFECTS OF FIRE PROTECTION SYSTEM ACTUATION
ON SAFETY-RELATED EQUIPMENT

By

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SUMMARY

Generic Issue 57 is concerned with fire protection system (FPS) actuations which have resulted in adverse interactions with safety-related equipment at operating nuclear power plants. Operational experience showed that safety-related equipment subjected to fire suppressants could be rendered inoperable and even cause a fire. This operational experience also indicated numerous spurious actuations of the FPS initiated by operator testing errors or by maintenance activities (e.g., welding), steam, or high humidity in the vicinity of FPS detectors.

On June 22, 1983, IE information Notice 83-41 was issued to alert licensees and provide examples of recent experiences in which actuation of fire protection systems caused damage or inoperability of systems important to safety. The IE Notice indicated that the plant Fire Hazards Analysis required by Appendix R to 10 CFR 50 and by the related NRR Branch Technical Position (BTP 9.5.1) requires not only consideration of the consequences of a postulated fire, but also consideration of the effects of fire-fighting activities.

On June 8, 1988 a prioritization of this issue was performed and a MEDIUM priority ranking was assigned. Consistent with existing procedures, a quick review was performed to determine if a resolution was possible without the need for expending large amounts of time and other resources. A fire risk scoping study by Sandia National Laboratories (SNL) completed at about that time (NUREG/CR-5088) indicated that the frequency and consequences of inadvertent FPS actuations are considerably higher than those found earlier in the prioritization study for this issue. Our review of NUREG/CR-5088 and related information, including more recent operational experience, prompted a limited study to reevaluate the risk associated with FPS actuation. As a result of this scoping study, performed with technical assistance by SNL, it was determined that this issue could be assigned a HIGH priority ranking.

FPS actuations which result in adverse interaction with plant systems important to safety reduce the availability of such systems needed to achieve safe plant shutdown or to mitigate a postulated accident. This concern is accentuated when common cause initiators and common mode failures of safety-related equipment are considered. Examples of common cause initiators include earthquakes, smoke intrusion into multiple fire

zones, and fire suppressant intrusion into multiple fire zones affecting several safety-related systems. Examples of common mode failures of safety-related systems and/or auxiliary systems supporting safety-related systems include electrical shorts in instrument cabinets and electrical power distribution centers, CO_2 ingress into the fresh air intake of emergency diesel-generator sets, CO_2 induced thermal stresses and cracking station battery casings, with loss of offsite power during an earthquake. It should be noted that a number of common cause initiators and common mode failures are not mutually exclusive and they may be part of a single event sequence.

The results of the scoping study performed by SNL which took into account the above considerations of common cause failure initiators, show that the risk associated with this issue could be considerably higher than that estimated in the original prioritization. The frequency of inadvertent FPS actuations was calculated to be 0.12 events/R-Y based on the reported operational experience alone. In a group of ten basic root causes of inadvertent FPS actuations, including common cause initiators such as a fire in one area inside the plant with its attendant smoke propagation in other areas of the plant; a fire outside the plant, with smoke ingress into the plant; and an earthquake-induced damage/spurious signals to FPS, the core damage frequency (CDF) contributions range from about 1.0E-6 to 2.0E-4 events/R-Y.

Based on the findings of our augmented risk analysis, a decision to proceed with a comprehensive program using fire zone methodology to develop more realistic estimates of areas of vulnerability is under consideration.

RESOLUTION OF GENERIC ISSUE 115, "ENHANCEMENT OF THE RELIABILITY OF WESTINGHOUSE SOLID STATE PROTECTION SYSTEM"[a]

by

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SUMMARY

This paper presents the regulatory resolution of Generic Issue 115, "Enhancement of the Reliability of Westinghouse Solid State Protection System." Generic Issue (GI) 115 addresses the reliability impact of the Westinghouse (W) Solid State Protection System (SSPS) reactor trip function on the frequency of Anticipated Transients Without Scram (ATWS) events at W plants. The regulatory resolution of GI 115 involved a technical evaluation, a regulatory evaluation based on the technical findings, the formulation of a decision rationale, and the recommendation for resolution.

The technical evaluation required a generic cost-benefit analysis of the costs and risk reduction (benefit) of alternative enhancements to the W SSPS reactor trip function. A generic model represented the SSPS reactor trip contribution to W ATWS core damage sequences and risk (person-REMs exposure due to an ATWS core damage release). Base case risk, options risk, and options costs were evaluated. Six options were considered, ranging from vendor-recommended upgrades of SSPS UV cards, redundant and diverse SSPS reactor trip breakers, to a redundant and diverse reactor trip logic system.

As part of the evaluation, the operational experience with two key components of the W SSPS received particular attention: the undervoltage (UV) driver cards and the reactor trip breakers (RTBs). Improvements to the SSPS UV driver cards have been recommended by W, but not implemented at all W plants. The RTBs used in W plants are W-supplied DS-416 and DB-50 series breakers, appropriately modified for use in the Reactor Protection System (RPS). There has been no appreciable change in the failure rates of these breakers since the ATWS rulemaking analysis in 1983.

Cost-benefit results were calculated using point estimates and Monte Carlo simulation with uncertainty distributions. With the exception of one option, which would result in an increase in risk if implemented, all other options would provide a reduction in risk corresponding to a range of

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reduction in core damage frequency (CDF) by 1.0E-7 to 2.1E-6 events/reactor-year and a cost benefit ratio generally higher than the $1,000/person-REM nominal screening value of backfit considerations.

Based on the technical evaluation, a regulatory analysis was performed and a decision rationale developed for the resolution of GI 115. The regulatory analysis concluded from the findings of the risk and cost-benefit analyses performed that enhancements to the W SSPS were not warranted in accordance with the backfit rule, 10 CFR Part 50.109(a)(3) and that no new regulatory requirements were necessary to resolve GI 115.

Based on the technical evaluation of this issue in general, and the six options in particular, the following insights were presented for consideration:

- Decrease the RTB surveillance test frequency in conjunction with the addition of an automatic trip function to the contactors supplying the field current to the Rod Control Motor-Generator (MG) Sets and/or the MG Sets' output breakers. These changes, if implemented, would contribute toward (1) reducing the regulatory burden on the affected licensees and applicants, and (2) extending the life of the RTBs as well as providing a diverse and redundant interruption of power to the control rods, thus improving, or at least maintaining, the current reliability of the reactor trip function.

- The licensees proposing to adopt an approach such as the above should be allowed to do so assuming that the recommendations contained in W Technical Bulletin NSID-TB-85-16 have been implemented. These recommendations, developed in accordance with the requirements of 10 CFR Part 21, have already been implemented at several W plants.

- Incorporate the above insights in the design of the advanced light water reactor (ALWR) plant proposed by the Electric Power Research Institute (EPRI). Incorporation of these design features at this early stage of the ALWR design would be more efficiently implemented than in a backfit setting.

A distribution of the technical analysis report (NUREG/CR-5197) and the regulatory analysis report (NUREG-1341) has been made to include all W licensees. The insights of NUREG-1341 could form the basis of industry initiatives for design and/or Technical Specification changes.

SUMMARY

RESOLUTION OF GSI B-56
EMERGENCY DIESEL GENERATOR RELIABILITY

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The need for an emergency diesel generator (EDG) reliability program has been established by 10 CFR Part 50, Section 50.63, "Loss of All Alternating Current Power," which requires that licensees assess their station blackout coping and recovery capability. EDGs are the principal emergency ac power sources for avoiding a station blackout. Regulatory Guide 1.155,"Station Blackout," identifies a need for (1) a nuclear unit EDG reliability level of at least 0.95, and (2) an EDG reliability program to monitor and maintain the required EDG reliability levels. NUMARC-8700,"Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," also provides guidance on such needs.

The resolution of GSI B-56,"Diesel Reliability" will be accomplished by issuing Regulatory Guide 1.9,Rev. 3, "Selection, Design, Qualification, Testing, and Reliability of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Plants." This revision will integrate into a single regulatory guide pertinent guidance previously addressed in R.G. 1.9, Rev. 2, R.G. 1.108, and Generic Letter 84-15. R.G. 1.9 has been expanded to define the principal elements of an EDG reliability program for monitoring and maintaining EDG reliability levels selected for SBO. In addition, alert levels and corrective actions have been defined to detect a deteriorating situation for all EDGs assigned to a particular nuclear unit, as well as an individual "problem" EDG.

RG 1.9, Rev. 3 (Proposed) was issued FOR COMMENT in November 1988 and the last set of comments was received in May 1989. There were 14 respondees comprised of: 8 utilities, NUMARC, EPRI, ASME, IEEE, IMO DeLaval and one individual. The staff revised RG 1.9, Rev. 3 based on comments received and through a series of information exchange meetings held with NUMARC's B-56 Task Force. NUMARC has revised Appendix D of NUMARC 8700 to provide a greater level of guidance regarding industry-wide practices related to maintaining EDG reliability levels. These two references are therefore complementary with respect to EDG reliability and will be implemented through current industry practices and compliance with the station blackout rule.

GENERIC ISSUE 113
DYNAMIC QUALIFICATION AND TESTING OF LARGE BORE HYDRAULIC SNUBBERS

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Summary

The reliability of large bore hydraulic snubbers (LBHSs) is a present safety concern in nuclear power plants. LBHSs are used primarily to restrain large components such as steam generators and reactor coolant pumps during seismic events. The United States Nuclear Regulatory Commission (USNRC) developed Generic Issue 113 (GI-113), "Dynamic Qualification and Testing of Large Bore Hydraulic Snubbers," with the objective of evaluating the reliability of these snubbers in operating commercial nuclear power plants. For the purposes of this research, LBHSs are defined as those hydraulic snubbers with rated load capacities equal to or greater than 50 Kips.

Initially there were no test machines capable of testing LBHSs for functionality, so they were exempted from surveillance testing prior to 1980. When LBHSs were later tested, numerous deficiencies, many of which would render the LBHSs inoperable, were found.

This paper describes the INEL/USNRC LBHS research program whose objective is to assess the need to improve the reliability of LBHSs and to determine what, if any, requirements should be implemented to achieve necessary improvements. The program has assessed the current state of LBHSs; including types, approximate numbers and locations of LBHSs in plants; operational, maintenance and current in-service inspection and test requrements; and the availability of state-of-the-art testing equipment. An attempt was made to determine the level of environmental and dynamic qualification. Operating experience data, including information contained in the Nuclear Plant Reliability Data System and Licensee Event Reports, were summarized to determine the modes of snubber failure.

The above information, along with the recommendations and conclusions generated in the final stages of the program, are currently planned to be published as a NUREG/CR in late 1989.

IMPROVING THE RELIABILITY OF SERVICE-WATER SYSTEMS AT NUCLEAR POWER PLANTS

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Fouling and clogging caused by sedimentation, corrosion, and the buildup of biological organisms is a persistent problem whenever raw water from a river, lake, or ocean is used as a heat exchange medium. The fouling caused by raw water cooling of service-water systems at nuclear power plants is a particular concern because of the potential for affecting safety-related heat exchangers and components, including online and redundant backup units. Open-cycle service-water systems, as discussed here, provide cooling to reactor support systems required during shutdown and emergency conditions and relate to those heat exchangers and components cooled directly by raw water.

The Nuclear Regulatory Commission (NRC) staff has been aware of fouling problems in service water cooling systems in operating plants for about ten years. Since 1980, fouling of service water systems caused by biofouling (fouling by bivalves such as Asiatic clams, blue mussels and American oysters; and by other biological organisms), sedimentation and corrosion products was discovered in a number of operating plants. Responses from licensees to NRC Bulletin 81-03 indicated that more than half of the operating plants active at the time were considered to have a high potential for biofouling. Responses also indicated that activities of licensees and applicants for biofouling surveillance and control ranged widely and were judged in many instances inadequate to ensure safety system reliability. Not all of the facilities with high potential for biofouling had adopted effective control programs. By 1982, several reports of serious fouling events in open-cycle service water systems had been discovered. These events resulted in plant shutdowns, reduced power operation for repairs and modifications, and degraded modes of operation. In view of these operating experiences on service water systems, the NRC in early 1983 established Generic Issue 51, "Improving the Reliability of Open-Cycle Service Water Systems." To resolve this issue, the NRC initiated a research program at the Pacific Northwest Laboratory (PNL), to study the conditions that allow fouling and to compare alternative surveillance and control programs to minimize service water system fouling. The PNL study is completed in early 1989, and following is a summary of the findings:

Bivalve fouling occurs because environmental conditions within the service-water system allow bivalves to settle, attach and grow. Sedimentation usually results whenever flow velocities fall below 3 fps. Galvanic corrosion, concentration cell corrosion, and microbiologically influenced corrosion (MIC) all occur in open-cycle water systems, and concentration cell corrosion and MIC are strongly linked to sedimentation. All types of fouling can occur simultaneously or in close sequence. Fouling of one type will enhance the potential for other types of fouling to occur by increasing surface roughness, decreasing flow area, and changing flow velocities.

There is no single solution to biological, sediment, and corrosion fouling. An effective surveillance and control program must satisfy certain criteria to address major areas of the service-water system and the major fouling types. A comprehensive program will keep fouling to a level that will not jeopardize safe operation. Three fouling program alternatives were developed.

Value/impact (or cost/benefit) studies were performed on these three alternatives. The alternative chosen to be most cost effective is a baseline fouling program. It consists of two principal elements of a control program, continuous chlorination (for example during bivalve spawning seasons) and periodic flushing and flow testing of redundant and infrequently used cooling loops, to minimize flow blockage that would result from biofouling and sediment/corrosion product buildup. In addition, a surveillance program is recommended to regularly inspect the intake structure for macroscopic biological fouling organisms, sediment, and corrosion. If this baseline fouling program is followed, the problems associated with fouling caused by biofouling, sedimentation and corrosion products would be greatly reduced.

The NRC recommendation for the resolution of Generic Issue 51 is the implementation of the baseline fouling program as described above. This baseline fouling program is to be included as part of a proposed generic letter titled "Service Water System Problems Affecting Safety-Related Equipment." This proposed generic letter is an integrated approach to address fouling and other additional concerns that the NRC staff has regarding service water system problems affecting safety-related equipment.

PRELIMINARY OBSERVATIONS OF UPCOMING PHASE II GATE VALVE FLOW INTERRUPTION TESTS

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A current research program sponsored by the U.S. Nuclear Regulatory Commission (NRC) and conducted by researchers from the Idaho National Engineering Laboratory (INEL) is testing the ability of full-scale flexible wedge gate valves to close under design basis flow and pressure loadings. The purpose of this program is to provide technical information for the USNRC effort regarding Generic Issue 87, "Failure Of The HPCI Steamline Without Isolation." The test program also provides information applicable to Generic Issue II.E.6.1 "Insitu testing of valves" and its related documents IE Bulletin 85-03, "motor operated valve common mode failures during plant transients due to improper switch setting" and the generic letter the USNRC staff is considering that could expand the application of IE Bulletin 85-03 to all safety related valves, including those that might be mispositioned.

Phase I testing was completed in June 1988, and results are being analyzed. The objective of Phase II of the program is to expand the technical data base in determining whether isolation valves in high energy BWR piping systems will close against high flows in the event of a pipe break outside containment. Generic Issue 87 includes those BWR process lines that communicate with the primary system, pass through containment, and contain normally open isolation valves. Three process lines fall under this description: (1) the high pressure coolant injection (HPCI) steam supply line, (2) the reactor core isolation cooling (RCIC) steam supply line, and (3) the reactor water cleanup (RWCU) supply line. Of the three, an unisolated break in the RWCU supply line was determined to have the greatest safety impact and the subject of the Phase 1 test program. The Phase II test program will be configured to answer questions raised by results of the Phase I testing on the RWCU valves and will include steam flow interruption testing representative of the HPCI system.

The Phase II test program will include three 10-in. valve assemblies representative of valves installed in the HPCI system and three 6-in. valves representative of valves installed in the RWCU system. Each valve assembly will be subjected to the hydraulic qualification test specified in ANSI B16.41, the nuclear valve qualification standard, for their pressure class and then to three flow interruption tests. The first flow interruption test for each valve will be conducted at normal BWR primary conditions for steam or water, as applicable. The second and third flow interruption tests will be parametric studies based on valve response in the first tests. Full pipe break flow will be maintained throughout the valve closure. The valve and test loop will be instrumented to monitor flow, static and dynamic pressure, valve disc position, motor operator torque, current, voltage, and valve stem thrust.

Results from Phase I testing showed that for the valves tested, significantly more thrust was required to close the valves than would have been calculated in the original motor operator sizing. Phase II testing will increase the number of valve designs tested under both steam and water environments and attempt to isolate the parameters that affect valve closure requirements. Phase II of the test program will be conducted in August and September 1989.

INSITU TESTING OF MOTOR-OPERATED VALVES IN NUCLEAR POWER PLANTS

17th Water Reactor Safety Meeting
Rockville, Maryland
October 23-25, 1989

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This paper presents a perspective of the status of in situ testing of motor operated valves in nuclear power plants. The objectives of in situ testing are discussed. A short history of in situ testing of motor-operated valves in nuclear plant applications is offered. Several Points are discussed concerning NRC generic letter 82-10 on in situ testing of motor-operated valves. Recent developments regarding in situ testing are discussed followed by a perspective on needed research and development.

Summary

OVERVIEW OF NRC's HUMAN FACTORS REGULATORY RESEARCH PROGRAM

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The purpose of human factors research at the NRC is to provide the technical basis for supporting regulatory actions taken to ensure nuclear safety. Human factors research is a multidisciplinary endeavor relying heavily on the behavioral sciences and involving a variety of engineering disciplines. The objectives of human factors research are to (1) broaden our understanding of human performance and identify the causes of human errors related to safe operations in the commercial nuclear industry (2) accurately measure human performance for the purpose of identifying methods for enhancing safer operations and precluding critical errors and (3) develop technical bases for nuclear regulatory requirements, recommendations, and guidance.

Personnel performance contributes to about half the significant events each year at nuclear power plants and to a larger percentage of events at nonreactor facilities. An understanding of the factors shaping human performance can focus regulatory attention and guide regulatory actions pertaining to licensee personnel. To understand personnel error, research to characterize and measure human capabilities and limitations is needed. The Human Factors Regulatory Research Program Plan provides the framework for researching the many factors that shape human performance such as cognitive processes, training, qualifications, organization, supervision, procedures, performance aids, and interfaces between humans and systems. The human factors research supports regulatory decisions affecting operators, maintenance personnel, technicians, and managers within the nuclear industry and incorporates human reliability analyses into probabilistic risk assessments.

Detailed information on the causes of human errors is needed to support regulatory actions. Planned human factors research will provide the methods and causal data required to establish the technical basis for actions that depend on an understanding of human performance during operations and maintenance for both nuclear power plants and materials licensees.

Another aspect of human factors research is to integrate both human and hardware reliability in NRC licensing, inspection, and regulatory decision making. As the knowledge gained from the study of human capabilities and limitations, human performance analysis, and reliability assessment is integrated, the technical basis will exist for evaluating human performance issues and proposed regulatory initiatives.

The human factors research program is divided into distinct and interrelated program activities: (1) Personnel Performance measurement, (2) Personnel Subsystem, (3) Human-System Interface, (4) Organization and Management, and (5) a group of Reliability Assessment activities. The purpose of the Personnel Performance Measurement activity is to improve the Agency's understanding of the factors influencing personnel performance and the effects on the safety of nuclear operations and maintenance by developing improvements to methods for collecting and managing personnel performance data. Personnel Subsystem research will broaden the understanding of such factors as staffing, qualifications, and training that influence human performance in the nuclear system and will develop the technical basis for regulatory guidance to reduce any adverse impact of these influences on nuclear safety. Research in the Human-System Interface activity will provide the technical basis for ensuring that the interface between the system and the human user supports safe operations and maintenance. Organization and Management research will result in the development of tools for evaluating organization and management issues within the nuclear industry. And finally, the Reliability Assessment group of activities includes multidisciplinary research that will integrate human and hardware considerations for evaluating reliability and risk in NRC licensing, inspection, and regulatory decisions.

IMPROVING SAFETY THROUGH AN INTEGRATED APPROACH FOR ADVANCED CONTROL ROOM DEVELOPMENT

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SUMMARY

With the fast development of computer technology, the potential exists for improving operational safety of nuclear plants by using advanced operator tools in the control room. This development may take place either as a gradual addition of new equipment or replacement of old equipment in existing control rooms, or as concepts for new control room designs. In both cases, the overall goal is to ensure that safety, and also efficiency, are improved.

Knowing the potential, and also the limitations of the human, the operator may in principle be assisted by Computerised Operator Support Systems (COSSs) in performing a variety of tasks. This has resulted in the gradual introduction of systems for alarm handling, failure detection, disturbance diagnosis, procedural advice and others, often based on process modelling techniques or expert system technology. The overall effect on safety from adding a large number of COSS's in the control room could, however, be negative due to the increased complexity of the control room. To ensure a maximum benefit from the new technology, a careful integration of the various systems must take place, resulting in a well coordinated interface between the operator and the process.

To investigate how to introduce modern technology for maximum benefit to plant safety and efficiency, the OECD Halden Reactor Project has started the development of an Integrated Surveillance And Control System (ISACS). This activity is intended to be relevant both for retrofitting into existing control rooms and for development of new control room concepts. The basis for the activity is the experience at Halden in developing specific COSSs, and the activity around the experimental control room HAMMLAB where detailed validations of operator tools have been performed for a number of years. The first goal in the ISACS project is to have a first, limited prototype in operation at the end of 1990.

When the concept for ISACS was developed, an analysis was made of the tasks performed by the operator in various operational situations, and the potential use of computerised support systems. The goal for the work was to arrive at an optimum division of tasks between the operator and COSSs. In handling of a specific situation, whether optimisation of normal operation or coping with a disturbance or accident, the operator

always goes through a sequence of tasks. Starting with the identification of the process state, he considers alternative strategies for coping with the situation, and finally implements the preferred strategy. If one looks at where COSSs may assist the operator, they can be of use in all these tasks. As an example, disturbances may be detected at an early stage by use of process models running in parallel to the process, they may be diagnosed by knowledge based systems, efficient strategies may be tested by faster than real time simulation and actions may be implemented by use of a computerised procedure system. Operator support systems performing the tasks described above have been developed or are at present being developed at the Halden Project, constituting a basis for integration within ISACS.

The need for an integrated approach when designing the operator interface is obvious. First of all, the addition of COSSs creates new information, adding to what already exists. Knowing that conventional control rooms today create an overflow of information, especially in disturbances and accidents, the need increases for prioritising information to be presented. Also, ISACS will create new, high level information based on systematic analysis of input from the process and COSSs. The analytical redundancy introduced by having several systems analysing the same task using different techniques improves the reliability of the conclusions drawn and recommendations made. This high-level reasoning requires a module in the integrated system that keeps an overview of the process and the tasks performed by the specific COSSs. In ISACS, the "Intelligent Coordinator" is introduced, partly to cover this task. Secondly, the same man-machine interface, consisting of an overview display, lower level displays and modules for plant control will be used for different purposes depending on the plant state. A standard for the man-machine interface, where the same principles are used in normal operation and in accident handling, will ensure that the operator more easily receives the information and performs the right actions.

The paper will describe in more detail the structure of ISACS and the modules to be included in the first prototype. The planned approach to validate the system, to ensure that the goal of improving safety is reached, will also be presented.

HUMAN FACTORS SURVEY OF ADVANCED INSTRUMENTATION AND CONTROLS

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The nuclear power industry has used analog instrumentation and controls (I&C) in their control rooms and technical support centers since the first nuclear power plant went on-line in the late 1950's. Even today the industry, as a whole, has been slow to implement advanced/digital I&C. The utilization of digital I&C appears, however, to be the wave of the future because most of the analog components and systems are becoming obsolete and no longer available. These advanced systems will also probably be utilized in the life extension of nuclear plants. It has been demonstrated in other industries that digital I&C provides almost error-free performance that is three-to-four orders of magnitude better than analog components performing the same function. With the increase in sophistication in the operation of modern nuclear power plants that is needed to handle the multiple (and sometimes conflicting) goals of efficiency, reliability, economic operation, and safety, the nuclear industry will be driven to the use of advanced I&C.

Oak Ridge National Laboratory (ORNL) is currently performing a research project for the Nuclear Regulatory Commission's (NRC) Office of Nuclear Regulatory Research. The purpose of the project is to provide the technical basis for the development of regulatory criteria to evaluate the safety implications of human factors associated with advanced I&C in nuclear power plants. During the first part of this project a survey of the United States (U.S.) and Canadian utilities and vendors was conducted. The survey was oriented towards determining the human factors issues related to the current, planned, and potential future uses of digital systems in control rooms and technical support centers.

The survey was administered at all of the U.S. nuclear vendors and five utilities who have begun to use advanced I&C. Human factors/digital system issues were also discussed with one utility and vendor in Canada. Groups of persons interviewed at each facility included human factors personnel, control room operators, software developers, I&C engineers, and trainers/instructors.

The survey instrument consisted of open-ended questions which were constructed through an iterative process and pilot-tested at a number of national laboratories. The instrument was divided into six main areas: computer-generated displays (CGD), controls, expert systems, organizational support, training, and related topics.

The survey was conducted by a team of three scientists. The U.S. facilities were visited for one day each; the Canadian sites for a day-and-a-half. Personnel at each utility/vendor were interviewed either individually or in groups of two-to-five. The amount of time spent with particular people varied between one-half and three hours. The survey instrument was used to guide the course of the interviews, but the discussions themselves were semi-structured and took form as they proceeded. Only those items which were applicable to either a specific facility or particular group of people were discussed.

Human factors issues identified during the survey include the following:

1. Operator acceptance and trust of advanced I&C/blind reliance on computer output.

2. Impact of CGDs on the operator's mental model.

3. Role change in the control room from operator to supervisor/demographical, selection, and qualification requirements for operators in the control room of the future.

4. Need for an advanced I&C guideline equivalent to NUREG-0700 (i.e., interface of the human, displays, and controls)/lack of consistency among human-advanced I&C interfaces.

5. Need for a dynamic allocation of functions and tasks between the human and advanced I&C/what tasks are appropriate for the human in an advanced I&C control room/locus of control between the human and the advanced I&C.

6. Information overload - what is the threshold where intelligent operator aids are needed?

7. Adequacy of existing training programs, techniques, methods, and tools for training on advanced I&C.

8. The operations staff must be involved during the entire life-cycle of the instrumentation/control system.

9. CGDs for multiple users.

10. User friendliness of CGDs and human-computer interfaces.

11. Effect of advanced I&C on operator performance/job efficiency.

12. Organizational climate of the nuclear utility.

13. Task analysis in knowledge acquisition for expert systems.

14. Systems integration/how does one manage human factors information?

15. Display of maintenance information in the control room (the operator needs to know the status of plant equipment).

The human factors issues were prioritized in regards to their importance by representatives from both ORNL and NRC. The highest rated items were those dealing with operator acceptance, role change, NUREG-0700 equivalence, information overload, and training programs.

The Effects of Local Control Station Design Variation on Plant Risk

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The existence of human engineering deficiencies at local control stations (LCSs) was addressed in a study (NUREG/CR-3696) conducted by the Pacific Northwest Laboratory (PNL). PNL concluded that "the existence of these human factors deficiencies at safety significant LCSs increases the potential for operator errors that could be detrimental to plant and public safety." However, PNL did no specific analysis to evaluate the effects of LCS design variations on human performance, on plant risk, or on the cost benefit feasibility of upgrading LCSs. The purpose of the present investigation was to conduct such an analysis. The specific objectives of the research were (1) to further define important local control stations, human factors related LCS design variations, and typical human engineering deficiencies (HEDs) at LCSs; (2) to determine the effect of LCS design variations on human performance, i.e., on risk-significant human errors (HEs); (3) to determine the effect of LCS-induced human performance variation on plant risk as measured by core melt frequency (CMF); and (4) to determine whether LCS improvements (upgrades in LCS design to mitigate HEDs) are feasible in a scoping-type value-impact analysis.

The analytical approach was centered around the application of a Probabilistic Risk Assessment (PRA) methodology. This method was selected because of the requirement to relate LCS design variations to plant risk in quantifiable terms. The PRA, however, does not provide an immediate and direct method of assessing the impact of LCS variations on human performance. Thus, the PRA was linked with other human performance analysis methods.

Using the PNL study as a basis, BNL further defined LCS HEDs through two data sources: (1) The NRC Office of Nuclear Reactor Regulation's Emergency Operating Procedure Inspection Program of 22 nuclear power plants (NPPs), and (2) BNL LCS on-site surveys of four NPPs. Once the data was collected, the Oconee PRA (NSAC-60) was selected for the analysis because it unambiguously modeled several human activities in the operation of the plant's safe shutdown facility and auxiliary shutdown panel.

Having identified a suitable PRA, the specific design characteristics of its local control stations were obtained. Oconee LCSs were systematically varied along human factors dimensions by "downgrading" them to incorporate typical HEDs not already present in the design and "upgrading" them to generally improve the man-machine interface. The downgrade results in an increased CMF. Thus, a range of LCS designs (to envelope the full span of those observed) were developed around the "base case" modeled in the PRA. A total of nine LCS design configurations were considered. LCSs were varied along two general dimensions judged to embody most of the significant human factors parameters: Functional Centralization (FC) and Panel Design (PD). FC related to the way in which safety functions handled by LCSs were distributed throughout the plant. The FC level was defined as high, medium, and low based on the number of local panels required to execute safe shutdown functions. PD reflected the human engineering characteristics of individual panels along criteria such as those provided in NUREG-0700. Again, low, medium, and high design levels were defined.

To determine the effect of these LCS design variations on human performance, a panel of appropriate experts was assembled and the Success Likelihood Index/Multi-Attribute Utility Decomposition (SLIM-MAUD) method (NUREG/CR-3518) was utilized to derive revised HEPs for each LCS design configuration. A set of unique HEPs for each of the nine LCS design configurations was then entered into the PRA and the CMF was calculated. Using the change in CMF as a measure of benefit and limited data on the costs of upgrading local control panels, a scoping-level, value-impact analysis was performed (NUREG/CR-3568).

The results can be summarized as follows. There was an overall effect of LCS variations on human performance. The transition from the worst LCS configuration to the best resulted in an absolute reduction or improvement of 0.82 in mean HEP (reduction by a factor of 20). The transition from low to high levels of FC was associated with a 0.46 (86%) reduction in mean HEP. The majority of the effect was accounted for in the transition from the low to medium levels. The Panel Design dimension also had an effect on human performance although not as large as functional centralization. Upgrading from a low to high panel design resulted in a 0.29 (69%) reduction in mean HEP.

The overall effect of LCS variations on plant risk was sizeable. The transition from overall worst to overall best LCS configuration was associated with a decrease in total CMF of 4.82E-4 events/RY. This is a decrease of 77% in CMF. One must realize, however, that both configurations represent extremes of LCS design (good and bad) and hence, most likely do not represent any current plant. The Functional Centralization dimension had a large effect on plant risk. Most of the effect was due to the upgrade from low to medium FC. The results for the Panel Design dimension were similar to those for FC.

There was a degree of uncertainty in the analysis of plant risk, the value-impact calculations, as well as in the generalization of these findings to other LCSs and NPPs. However, within these constraints, the value-impact analysis indicated that the LCS upgrades that were clearly significant were upgrades in panel design only. Almost all FC upgrades were classified as insignificant. Upgrades along both PD and FC dimensions simultaneously were split between borderline and insignificant.

Thus, this analysis indicates that while upgrades in FC had greater risk significance than PD upgrades, their cost was much greater and resulted in generally low Value-Impact Ratios (VIRs). Relative to changes in FC, upgrades in PD are inexpensive, and since PD upgrades were also associated with notable risk reduction, significantly high VIRs were achieved. Combination FC and PD upgrades were mainly influenced by the costs of FC changes which, as discussed, were more expensive and led to lower VIRs.

EMPIRICAL EVALUATION OF THE EFFECTS OF POWER PLANT LIGHTING ON HUMAN PERFORMANCE

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The illumination in the control rooms of many operating nuclear plants falls below the levels specified in the NUREG-0700 guidelines. However, a close examination of the research literature upon which these guidelines were based revealed that the underlying human performance data were acquired with laboratory conditions and with tasks very different from those typically found in control rooms. In a pilot study, empirical methods were developed and empirical data were gathered regarding the levels of illumination sufficient for performing tasks analogous to those performed in control rooms.

In a computerized laboratory test-bed, the speed and accuracy of performance on the following tasks were measured under a wide range of illumination conditions: scanning a bank of edgewise meters to determine which meter displayed a needle position different from the others, scanning the meters to determine which meter displayed a pre-specified value, examining hardcopy X-Y plots to discern the value of a function along one coordinate given the value of the other coordinate, and reading hard-copy sample procedures in order to detect mis-spellings. In a power plant control room simulator, data were likewise collected in a meter reading task under a range of illumination levels. The results obtained from both settings suggested that adequate performance in control room tasks can be achieved at illumination levels well below those recommended in NUREG-0700.

The present report will summarize the results of a series of follow-on studies which attempted to confirm and expand the results obtained in the previous pilot study. The objectives of this follow-on work were to attempt to replicate the results of the pilot study using more subjects and more illumination levels, to examine the effects of glare, fatigue, aging, and their possible interactions with illumination, and to validate laboratory findings by examining performance during realistic operational scenarios in a control room simulator under different lighting conditions. A series of four studies was conducted:

- A laboratory study of meter reading performance with illumination and glare being varied, using subjects who spanned a wide age range.

- A laboratory study using other tasks that are characteristic of power plant operations and maintenance activities. The tasks of interest included a meter calibration task, hard-copy reading task, and plant drawings interpretation task. Illumination and contrast were manipulated with subjects spanning a wide range of ages.

- A study in a nuclear power plant control room simulator in which brief, but realistic, operational scenarios were presented under various illumination conditions.

ARD

- A laboratory study using the aforementioned meter reading task, to focus on the interacting effects of illumination, glare and fatigue.

It is anticipated that the results of all of these studies will be available for presentation in October, 1989. Preliminary results suggest the following conclusions:

- The primary findings of the pilot study were replicated. Again, performance deteriorated only at illumination levels well below those recommended by NUREG-0700.

- Glare had a more debilitating effect on performance under some illumination levels than others.

- The performance of older subjects decreased to a greater extent than that of younger subjects with decreasing illumination and increasing glare, but the illumination levels at which these effects occurred for the older subjects were, nonetheless, below the levels recommended by NUREG-0700.

- To some extent, it was possible to predict subjects' performance on tasks analogous to those performed in power plants by their performance on standard tests of visual acuity and contrast sensitivity.

- Subjective ratings suggested that preferred levels of illumination were higher than the levels at which actual performance deteriorated.

The results of this series of studies may expand the options that plants have in meeting their lighting requirements.

An Approach Toward Estimating the Safety Significance of Normal and Abnormal Operating Procedures in Nuclear Power Plants

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The Nuclear Regulatory Commission's TMI Action Plan calls for a long-term plan to upgrade operating procedures in nuclear power plants. The scope of Generic Issue Human Factors 4.4, which stems from this requirement, includes the recommendation of improvements in nuclear power plant normal and abnormal operating procedures (NOPs and AOPs) and the implementation of appropriate regulatory action. This paper will describe the objectives, methodologies, and results of a Battelle-conducted value impact assessment to determine the costs and benefits of having the NRC implement regulatory action that would specify requirements for the preparation of acceptable NOPs and AOPs by the Commission's nuclear power plant licensees.

The results of this value impact assessment are expressed in terms of ten cost/benefit attributes that can be affected by the NRC regulatory action. Five of these attributes require the calculation of change in public risk that could be expected to result from the action which, in this case, required determining the safety significance of NOPs and AOPs. In order to estimate this safety significance, a multi-step methodology was created that relies on an existing Probabilistic Risk Assessment (PRA) to provide a quantitative framework for modeling the role of operating procedures. The particular PRA used was chosen for its relatively extensive modeling of human error. An extensive Licensee Event Report (LER) search and the exercise of expert judgment also played important roles in this methodology. The purpose of this methodology is to determine what impact the improvement of NOPs and AOPs would have on public health and safety.

The first phase of the work was designed to determine the contribution of operating procedures of current quality to the occurrence of procedure-related operational errors during normal and abnormal conditions. The PRA was reviewed to identify transient initiating events that might be affected by improved NOPs and AOPs. The PRA was reviewed also to identify potential human errors and operator recovery events modeled to occur during normal and abnormal operating conditions and that might be affected (i.e., reduced in frequency) if operating procedures were improved. The expert judgment of personnel experienced in the use of procedures in nuclear power plants was used to determine whether NOPs or AOPs would actually be used in the particular operator recovery actions identified. By these means five transient initiating events and six operator actions were identified. Then all accident sequences and cut-sets contained in the reference PRA which involve any of the potentially affected transient initiating events or potentially affected operator actions were identified. This step produced a list of all core-melt accident sequences and cut-sets in the PRA that may potentially be affected by improvements to operating procedures.

Next, Licensee Event Reports (LERs) in the Sequence Coding and Search System (SCSS) database were reviewed in order to estimate the current "baseline" contribution of procedure-related errors to the frequency of the transient initiating events identified previously. This produced an estimate of the portion of those frequencies that could be affected by a procedure improvement program. Likewise, expert judgment was consulted to evaluate the role of written procedures in the operator actions/errors identified previously which, in turn, produced that part of those events' given probabilities that could be affected by improved procedures. Based on these results, the affected portion of each of the base-case affected parameter values was calculated. These values represented the portion of the original values given in the PRA which is contributed by procedure-related errors and that could possibly be reduced by improving procedures.

The second phase of this benefits-determination part of the project was devoted to estimating the extent to which improved operating procedures would tend to reduce the contribution of procedure-related errors to the occurrence of the eleven subject transient-initiating events and operator actions. A survey of expert opinion was used to accomplish this. Ten people with extensive experience in using operating procedures were asked to estimate whether and to what extent improved NOPs and AOPs would affect the occurrence of the eleven events. The survey resulted in a percentage reduction in the affected portion of each of the eleven affected parameter values due to improved procedures. These resulting, slightly lower, adjusted-case affected parameter frequencies were then substituted in the original PRA cut-sets in place of the original base-case affected parameter values. The adjusted-case affected core-melt frequency was then calculated to be slightly lower than the original base-case frequency due to these potential improvements in procedures. To complete this phase, this lower adjusted-case affected core-melt frequency was used in the calculation of the estimated reduction in public risk due to improved operating procedures. This reduction in public risk was, in turn, used in the calculation of the five benefits-related attributes of the value impact assessment.

Ascertaining the costs of regulatory action aimed at improving NOPs and AOPs was accomplished by evaluating the costs of a proposed NRC program involving the NRC and its licensees directed at upgrading NOPs and AOPs. This proposed upgrade program consists of five parts. The NRC would first review and synthesize recent information on operating procedures and produce a manual which would set forth guidelines for the composition of high quality procedures. The Commission would then convene an NRC/Industry working group to jointly formulate procedure upgrade program policy. Each individual plant would then plan its own upgrade program based on these policies. As part of this planning, plant personnel would identify those procedures that are safety-related and would, therefore, require particular emphasis in the upgrade program. Each plant would then execute its own plant-unique upgrade program. The NRC would complete the overall program by auditing each plant's new or upgraded procedures. The costs of this program and the potential benefits to be derived from the improved procedures comprise the value impact assessment of the potential NRC regulatory action.

Procedure Violations in U.S. Nuclear Power Plants

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In response to the Chernobyl accident, the U.S. Nuclear Regulatory Commission (NRC) sponsored a study to distinguish procedure violations from procedure-related errors, and to assess the extent, nature, and consequences of procedure violations in U.S. nuclear power plants. The issue of procedure violations is of interest because, in addition to the design flaws of the RBMK reactor, procedure violations, or intentional departures from procedural guidance, by plant engineering and operations staff were among the root causes of the Chernobyl event.

A review of the limited relevant literature indicated that there is reason to believe that the failure to follow procedures has been a root cause of previous incidents in U.S. nuclear power plants. However, the available literature does not report whether these failures to follow procedures were intentional or accidental nor what the causal mechanisms were that led to the failures. And because any steps that might be taken to prevent significant events resulting from the failure to follow procedures would depend on this type of information, one of the first tasks of this project was to develop a taxonomy of behaviors that could be characterized as failures to follow procedures for use in reviewing incident reports.

Although finer distinctions among types of behavior are possible, for the purposes of the present project, three classes of behavior were defined. These are (1) Level A violations, in which a worker performs actions that deviate from the procedure, is aware of the actions, and is aware that his or her actions deviate from the procedure; (2) Level B violations, in which the worker performs actions that deviate from the procedure, it cannot be determined from reading the incident report that the worker was aware of his or her actions, but it seems likely that a worker with minimum qualifications and experience, exercising a minimal degree of care, would not have performed the actions; and (3) Level C violations, in which the worker performs actions that deviate from the procedure, is unaware of his or her actions (slips), or is unaware that his or her actions deviate from the procedure's intent (mistake).

Armed with these definitions, reports of significant events involving the failure to follow procedures that occurred during the years of 1983 to July, 1988 were gathered and the relevant information they contained was coded by human factors specialists and then reviewed by NRC reactor operator licensing examiners at the Pacific Northwest Laboratory. The types of reports collected included Licensee Event Reports (N=600), inspection reports (N=665), augmented inspection team reports (N=40), and other NRC-generated documents. The information coded from each report included the plant involved in the incident, the region, mode and power level during the incident, reactor type, severity, cause of the violation where possible, the activity/procedure involved, personnel involved, and the results of the violation.

Preliminary analyses of these data indicated that the large majority of failures to follow procedures described in the incident reports fell into the Level C category. Causal factors that were mentioned in the reports included insufficient detail in the procedures, ambiguity in how the information was presented, and inaccuracies in the procedures. The analyses also suggested that Level A and Level B violations had occurred at a rate of about one violation every two years per reactor, and no trends in the frequency of violations were apparent.

Preliminary findings also suggested that the majority of failures to follow procedures at all levels involved licensed and non-licensed operators, but maintenance and health physics technicians, contractor personnel, and security personnel were also represented in a substantial number of the incidents. The larger number of operations personnel involved in the incidents is likely a result of the fact that operations tasks are more highly proceduralized and under greater scrutiny than other plant functions, rather than implying that operations personnel are more prone to violate procedures.

Consequences associated with the failures to follow procedures varied in seriousness. Many of the failures to follow procedures resulted in violations of technical specification requirements or in no consequences for worker or public health and safety. However, some of the incidents resulted in unplanned worker exposures to radiation, safety system actuations, and damage to plant equipment.

Other findings and a discussion of the conclusions to be drawn from the data await the completion of the project.

Operator Alertness and Performance on 8-Hour and 12-Hour Work Shifts

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Human reliability becomes an acute problem when people are required to work in continuous, 24-hour per day operations, particularly when the work tasks involve sustained vigilance and monotonous procedures. Rotating shiftwork schedules requiring sustained, monotonous vigilance tasks are commonplace in utility power plant operations. The most important problem is that the physiological characteristics of the human body make it difficult for operators to sustain peak alertness and performance when working night shifts. The body's endogenous circadian rhythms dictate a natural period of low alertness between 1 a.m. and 6 a.m., which corresponds to the habitual human sleep period. As a result, human vigilance and decision-making ability is often severely compromised during night work. This fundamental problem is compounded by difficulties associated with the frequent day-night inversion of the operator's sleep/activity cycle required by rotating work schedules. Consequently, operators typically experience difficulty sleeping during the daytime and are relatively sleep-deprived when working the first few night shifts. A final factor to be considered is fatigue and loss of alertness, which may be associated with extended hours of work.

Attention has recently been directed to the alertness and performance problems of rotational shiftworkers in the nuclear power industry. Growing awareness of elevated rates of human errors and accidents on night shifts and reports of operations personnel falling asleep on the job have contributed heavily heightened interest in this subject. The industry is now carefully considering the effects of different shift rotation systems, including evaluation of the most recent of industry trends in shift scheduling - schedules that include 12 hour work shifts. Surveys show that within the past 5 years about 20 percent of commercially operational nuclear power plants have instituted schedules that use only 12 hour shifts, or schedules using a combination of 8-hour and 12-hour shifts. Many more plants routinely use 12-hour work shifts during plant outages and refueling operations.

In respose to this growing trend, the NRC has funded research which is a first attempt to compare alertness, operator performance, and sleep-wake patterns in subjects working simulated 8-hour and 12-hour shifts at the Human Alertness Research Center (HARC), located at the Institute of Circadien Physiology in Boston, MA. This paper describes a study that addresses important questions about alertness and performance in rotational shiftwork and nightwork in general, and in extended 12-hour work shifts in particular.

Research and Simulation Facilities of HARC

The HARC is a unique research facility that was established to study the determinants of human alertness and performance in persons performing simulated work tasks in round-the-clock operations. The NRC study is designed to control for many confounding factors that influence a shiftworker's alertness and performance, such as environmental lighting, ambient temperature, quality and quantity of sleep when off duty, diet, caffeine and alcohol intake, workload and work task design. It is nearly impossible to control for such factors in the workplace (although applied research in actual work situations will be an important validation step for later investigations). The HARC facility provides a simulated control room where factors such as environment and work task can be controlled. The HARC control room is complete with control panels, a Foxboro process control simulator, and computer test stations.

Experimental Design and Methodology

Subjects work in teams of two, alternating work tasks that require vigilance over system variables, responding to auditory and silent alarm signals, record keeping, and cognitive and decision making task to take corrective actions. Delays in recognizing and responding to alarms are computed and recorded throughout the work shift for later analysis of work task efficiency.

At hourly intervals during the work shift, each research subject leaves the command center to complete a 10-15 minute computer-based performance routine that has been developed at the Walter Reed Army Research Institute. The Walter Reed performance assessment battery includes subtests for logical reasoning, reaction time, numerical column addition, serial addition/subtraction, and time estimation. Each subject also completes hourly rating scales for subjective alertness.

Throughout the simulated work shift, physiological alertness is monitored continuously by recording electroencephalogram (EEG - brain waves), electrooculogram (EOG - eye movements), and electromyogram (EMG - muscle tone). This is accomplished using a portable recording device called the Medilog, which stores the data for later visual and computer analyses. At two hour intervals each subject's physiological alterness level is assessed using the Multiple Sleep Latency Test (MSLT), which is a standardized research protocol involving repeated opportunities to lie down and fall asleep in a dark room. EEG is monitored continuously and the subject is awakened shortly after sleep onset in order to to prevent the accumulation of brief restorative sleep periods during the work shift.

One of two simulated duty shifts are worked: (1) Six 8-hour evening shifts (3 p.m. to 11 p.m.), three days off, then six night shifts (11 p.m. to 7 a.m.) or (2) Four 12-hour night shifts (7 p.m. to 7 a.m.), three days off then four additional 12-hour night shifts (7 p.m. to 7 a.m.). Every subject sleeps in the laboratory for two nights of habituation immediately before the first simulated work shift in either protocol.

When not "on duty" in the control room, subjects reside in one of two attached and self-contained two-bedroom residential apartments with complete kitchen facilities. They are restricted to bed for an 8-hour sleep period beginning approximately one hour after the end of each work shift. Physiological variables for determining sleep-waking state are recorded continuously during this sleep period. Electrodes and recording devices are removed, then replaced before the next scheduled work shift. Caffeine, alcohol, and psychoactive drugs are disallowed throughout the study. Subjects leave HARC for the three days between shifts, but keep detailed logs of sleep-wake activities and caffeine and alcohol consumption.

This paper will describe in greater detail the design of the study, measurement techniques for alertness and sleep, work routine, work task performance measures, and cognitive performance test protocols. It will review the role of circadian factors in human alertness and performance, and discuss previous research findings in this area. It will discuss other variables that are known to influence human alertness in the workplace, such as caffeine, alcohol, and working environment. The physiological basis for shiftworker sleep problems will be explained in the context of the ongoing research project at HARC. Finally, the paper presents previous research on shiftwork and fatigue which may be relevant to a comparison of 8-hour and 12-hour shifts.

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Aging Assessment of Auxiliary Feedwater Systems*

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A Phase I study of Auxiliary Feedwater (AFW) Systems has been conducted at Oak Ridge National Laboratory (ORNL) as a part of the Nuclear Regulatory Commission (NRC) Nuclear Plant Aging Research Program. This study has largely focused on two areas:

1. A review of historical failure data for AFW System components.

2. The detailed review of the AFW System design and operating practices at a plant owned by a cooperating utility.

Historical failure data from the Institute of Nuclear Power Operations' Nuclear Plant Reliability Data System, S.M. Stoller's Nuclear Power Experience, and the NRC's Sequence Coding and Search System were combined to form a single AFW System failure database. In developing the combined database, the data from the three sources was supplemented by characterization of several aspects of each of the reported failure events, including consistent assignment of component type, method of failure discovery, and significance of the failure, from a system perspective.

The review of the failure database has identified that AFW pump drivers, including turbine, motor, and diesel drivers, have been historically the principal sources of AFW System degradation. Turbine drivers, in particular, have been a dominant single source of degradation for the AFW System. Failure of valve operators was also a significant source of system degradation. There were more failure records that were attributable to valve operators than any other component type (although the overall system degradation due to valve operator failures was less than that associated with pump drivers).

The failure data review also indicated that a significant number of AFW System failures were detected during demand conditions, as opposed to being detected by programmatic monitoring or routine observation. Of those failures detected during demand events, a large percentage was due to failures associated with various instrumentation and control circuit components.

The detailed review of AFW System design and operating practices at the plant owned by a cooperating utility was conducted primarily in order to assess the extent to which identifiable failure modes could be expected to be determined by current programmatic monitoring practices. During the review of monitoring procedures, an estimate of the number of test-related actuations of AFW System components was also made in order to provide an insight into the extent to which testing itself was involved in service wear.

The results of the plant-specific review indicated that there were a number of failure conditions which would not be detectable by current monitoring practices. The conditions found by the review which could exist and not be detected were largely related to one or both of two categories:

1. Inability to perform as required under design basis conditions.

2. Failure of various instrumentation and control related features.

The plant-specific review also indicated that the monitoring practices that are used to meet the surveillance requirements imposed by Tech Specs may result in an excessive imposition of test-related wear on certain components.

The Phase One study resulted in the following recommendations:

1. Conduct a detailed review of AFW turbines and their controls.

2. Conduct a Phase II AFW System study, the primary goal of which should be to develop a recommended structure for surveillance requirements and practices. This structure should provide periodic verification of operability of various components that are either not being tested or are being inadequately tested currently, as well as avoid excessive testing of certain components.

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INSTRUMENT AIR SYSTEM - AGING IMPACT ON SYSTEM AVAILABILITY[1]

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SUMMARY

As part of ongoing efforts to understand and manage the effects of aging in nuclear power plants, an aging assessment was performed for the Instrument Air (IA) system, a system that has been the subject of much scrutiny in recent years. Despite its non-safety classification, instrument air has been a factor in a number of potentially serious events. This report presents the results of the assessment and discusses the impact of instrument air system aging on system availability and plant safety. This work was performed for the U.S. Nuclear Regulatory Commission (NRC) as part of the Nuclear Plant Aging Research (NPAR) program.

To perform the complex task of analyzing an entire system, the Aging and Life Extension Assessment Program (ALEAP) System Level Plan was developed by Brookhaven National Laboratory and applied successfully in previous system aging studies. The work presented herein was performed using two parallel work paths, as described in the ALEAP plant. One path used deterministic techniques to assess the impact of aging on compressed air system performance, while the second path used probabilistic methods. Results from both paths then were used to characterize aging in the instrument air system.

The findings from this study have formed a technical basis for understanding the effects of aging in compressed air systems. The major conclusions from this work are highlighted in the following paragraphs. Some of the conclusions have application beyond the bounds of the instrument and service air systems.

- This study has identified aging trends in component failure rates, component relative importances, and system unavailability that could have an increasing impact on system availability and consequently affect plant safety in later years.

- Compressors, air system valves, and air dryers were found to make up the majority of failures. The increase in failures found in passive components such as piping, aftercoolers/moisture separators, and receivers was greater over time, but these still constituted only a small percentage of overall failures.

- The effectiveness and quantity of preventive maintenance devoted to a component significantly affected the amount of failures experienced.

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[1] Work done under the auspices of the U.S. Nuclear Regulatory Commission.

- Individual plant maintenance records for instrument and service air systems were found to be the most comprehensive source of data for performing aging analyses.

- As a continuously operating system, with minimal control room instrumentation due to its non-safety classification, most air system problems are detected by local monitoring and indication, walkdowns and inspection, and preventive maintenance inspection/surveillance.

- Review of compressed air system designs and studies using a PRA-based system model revealed that the redundancy of key components (compressors, dryers, IA/SA crossconnect valve) was an important factor in system availability. Overall design configuration had an impact on the pervasiveness of air system problems.

- Total loss of air events are uncommon. The majority of events resulted in degraded operation (low IA pressure, IA quality out of limits). Procedures and testing for the response of personnel and equipment to these conditions should be developed.

- Human error was a significant cause of failures in critical components such as compressors and dryers, as well as at the system and intersystem level. Training should be augmented in two key areas: 1) operation and maintenance of critical air system components, and 2) importance of instrument air to other plant systems particularly safety systems.

- The systems outside of instrument air that were most often affected by IA problems are containment isolation, main feedwater/main steam, auxiliary feedwater, and the BWR scram system. The most commonly affected components were AOV's and SOV's.

- The probabilistic work entailed the development of a computer program (PRAAGE-IA) using a PRA-based IA system model to perform time-dependent PRA calculations. Time-dependent failure rates were developed from the data base findings and input to the program to calculate system unavailability and component importances for various ages. Results from the probabilistic work showed that when the time-dependent effects of aging are accounted for, two significant system effects are seen: 1) system unavailability increases moderately with age, and 2) component relative importances change with age. During early operation, leakage in both IA/SA piping and support system piping was the most important contributor to system unavailability. However, during later years aging can cause compressors and air dryers/filters to become increasingly important.

The findings presented in this report form a sound technical basis for understanding and managing the effects of aging in IA systems. Future work will include improvements to current maintenance, monitoring, training, off-normal response procedures, and surveillance practices to mitigate aging degradation.

AGING EFFECTS IN COMPONENT FAILURE AND DOWNTIME DATA AND IMPACTS ON PLANT RISK

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This work consists of two phases. The objective of the first phase of the work is to develop and apply data analyses approaches to evaluate aging effects in component failure and downtime data. The objective of the second phase of the work is to develop and apply approaches to determine the core melt frequency impacts and plant risk impacts from component aging effects.

In the first phase of the work, data analyses procedures and software are being developed to audit plant records and NPRDS data to determine if aging effects are exhibited. The procedures which are being developed are more powerful than previous techniques in that component time line histories can be segmented for maximum aging resolutions, partial histories can be analyzed, specific causes can be examined, and component data can be optimally aggregated to increase the power to identify aging effects. Software are being developed to automate the analyses in an intelligent manner. The procedures have been applied to selected component failure and downtime records to not only demonstrate the process but to determine if aging effects are indeed exhibited in the data. The results to date indicate even more strongly that significant aging effects are exhibited in a wide number of components and plants, event though maintenance and testing are being carried out.

In the second phase of the work, procedures and software are being developed to allow PRAs such as the NUREG 1150 PRAs to be used for aging evaluations. The procedures allow aging effects to be quantified and to be prioritized with regard to their impacts on core melt frequency, accident sequence frequency, and public health risk. The procedures separate the evaluations of component effects of aging from the evaluations of the resulting risk impacts. This allows general types of aging effects to be considered and allows efficient calculation of the risk impacts. The demonstrations which have been carried out show how aging effects can be prioritized and how means of controlling the impacts can be identified.

THE USE OF NPAR RESULTS IN PLANT INSPECTION ACTIVITIES[1]

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SUMMARY

The Nuclear Plant Aging Research (NPAR) Program is a hardware oriented research program which has produced a large data base of equipment and system operating, maintenance, and testing information. A review of the NRC Inspection Program and discussions with NRC inspection personnel have revealed several areas where NPAR research results would be valuable to the inspector. This paper describes the NPAR information which can enhance inspection activities, and provides alternatives for making these pertinent research results available to the inspectors.

The NRC Inspection Program emphasis is on evaluating the performance of licensees by focusing on requirements and standards associated with administrative, managerial, engineering, and operational aspects of licensee activities. The Program recognizes that licensees may satisfy NRC requirements differently, and therefore expresses inspection guidance in the form of performance objectives and evaluation criteria. For the resident and regional inspectors, procedures have been written covering various subject areas, such as operations, maintenance, and surveillance. Some of these procedures contain guidance related to aging degradation.

Associated with each NPAR study is the need to determine the role of inspection, maintenance, and monitoring in counteracting aging and service wear effects. The role of maintenance in managing aging is an important area where NRC emphasis has been applied. A review by the NRC of maintenance performed at several plants resulted in the conclusion that "Most utilities do not perform condition monitoring due to inadequate knowledge of degradation mechanisms and the relationship between measurable parameters and predicted functional capability." The output from NPAR in this area could assist the inspector in determining the extent of licensee inadequacies where appropriate.

To obtain further delineation of the NRC inspector needs, presentations were made to the resident inspectors at three of the five regions. These discussions provided the inspectors with a summary of the results from the NPAR Program. Their comments, supplemented by a written questionnaire, indicated that NPAR results can be of use to the inspector when provided in a format directed to their activities.

The types of information generated by NPAR which were found to be relevant to inspection needs include the following:

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[1] Work done under the auspices of the U.S. Nuclear Regulatory Commission.

- functional indicators - NPAR reports identify parameters which can be monitored or measured to detect aging degradation. The inspector can apply these results to enhance visual inspections (walkdowns) and to evaluate licensee programs for assuring equipment and system operability.

- failure modes, causes, effects - operating experience data evaluated in NPAR studies can alert the inspector to prevalent system and equipment failure mechanisms. The potential for failure rate changes with plant age is an NPAR output of interest to the inspector in evaluating preventive maintenance resources.

- stresses which cause degradation - an inspector can benefit from knowing the environmental and operational stresses which cause aging degradation.

- maintenance recommendations - the inspector is required to evaluate aspects of a licensee's maintenance program for a number of different inspections, including special team inspections. NPAR reports contain a review of current maintenance practices, a summary of vendor-recommended maintenance, and recommendations for preventive and corrective maintenance which can be used to detect and mitigate the effects of aging.

- inspection prioritization - based on the failure rate determined through a detailed operating experience review and a model developed through Probabilistic Risk Assessment (PRA) techniques, NPAR system reports present the relationship between age degradation and plant risk. These results can be applied in the Inspection Program for redirecting inspection resources as the plant ages.

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NPAR APPROACH TO MANAGING AGING IN NUCLEAR POWER PLANTS[a]

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Over the past five years, the Nuclear Plant Aging Research program (NPAR) has been devoted to developing technical understanding of the time dependent processes that, through deterioration of components, systems, or structures (C/S/S), can reduce safety margins in nuclear power plants. A major and necessary element of the program involves the application of this basic knowledge in defining functional approaches to managing aging by anticipating and mitigating important deterioration processes.

Fundamental understanding and characterization of aging processes are being accomplished through NPAR-sponsored research projects, review and analysis of aging related information, integration of NPAR results with those from industry and other aging studies, and interfacing of all of these with the existing body of codes, standards and regulatory instruments that convey aging-related guidance to NPP licensees. Products of these efforts are applied to structuring and providing aging-related technical recommendations in forms that are useful in:

1. developing and implementing good aging management practices,

2. developing regulatory guidance and requirements for understanding and managing aging during normal plant operations and in support of license renewal, and

3. planning and implementing other regulatory actions and initiatives in which aging-related concerns have a bearing on scope or priority.

In developing good practices for managing aging, it is necessary to 1) identify the C/S/S in which aging is an important concern; 2) develop an understanding of the active aging processes and their relationships to specific materials, environments, and stressors; and 3) specify and prioritize monitoring and mitigation programs.

Selection of components is based upon both deterministic and probabilistic considerations. The key overall selection criterion is whether or not aging could lead to a departure from an acceptable safety envelope.

Understanding aging involves developing and applying both empirical and mechanistic relationships at an appropriate level of detail to selected C/S/S. The variables that govern these relationships reflect C/S/S design and composition (i.e., materials of construction and their condition); environments to which C/S/S are exposed, e.g., elevated temperature, corrosive media, ionizing radiation, and the collection of active stressors,

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(a) Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under U.S. DOE Contract DE-AC06-76RLO 1830.

e.g., cyclic loading, high stress, thermal gradients, high electrical potentials, to which C/S/S are subject.

Given an acceptable understanding of the mechanisms and kinetics of aging in important C/S/S, the final and key step in assuring that aging does not compromise safety is to implement effective practices for monitoring and mitigating aging. These practices, implemented principally through maintenance programs, involve the use of appropriate observational methods in testing, inspection, surveillance, and condition monitoring and preventive and corrective maintenance of the type and frequency needed to avert C/S/S failures that could threaten plant safety.

The products of NPAR studies, in general, deliberately address each of the key elements discussed above. In addition, good practices for managing aging are being compiled in manuals and documents specifically devoted to that subject. The principal compendium of good aging management practices catalogues all currently available information on understanding and managing aging in a format that mirrors that of the Standard Review Plan (NUREG 0800) and Regulatory Guide 1.70 (Standard Format and Content of Safety Analysis Reports). This systems-oriented format is commonly used and well understood by both licensees and regulators. This approach also provides a needed measure of assurance that aging concerns will be comprehensively addressed throughout the plant. A manual of good practices for understanding and managing aging is being developed. This will be a living document that evolves as the state-of-the-art of the understanding of nuclear plant aging expands.

NDE Research At NASA Langley Research Center

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The Nondestructive Measurement Science Branch at NASA Langley is the Agency's lead Center for NDE research. The focus of the laboratory is to improve the science base for NDE, evolve a more quantitative, interpretable technology to insure safety and reliability, and transfer that technology to the commercial sector.

The task is broad and requires a multidiscipline of professional researchers to achieve its goals. Involved in the current program are about 50 people, split evenly between civil service and nonpersonnel service. There are approximately 20 Ph.D.'s in the program with the following disciplines: physicists, chemists, materials scientists, computer scientists, and electrical engineers.

To address the broad needs of the Agency, the program has developed expertise in many areas, some of which are in ultrasonics, nonlinear acoustics, nano and microstructure characterization, thermal NDE, x-ray tomography, optical fiber sensors, magnetic probing, process monitoring sensors, and image/signal processing.

Our laboratory has recently dedicated its new 20,000 square foot research facility bringing our lab space to 30,000 square feet. The new building is designed to permit each office to have computer access to each lab so that experimentation can be monitored throughout the building. The new facility includes a high bay for the x-ray CAT scanner, a revolutionary new concept in materials measurement. The CAT scanner is called QUEST, for quantitative experimental stress tomography lab. This system combines for the first time a microfocus x-ray source and detector with a fatigue load frame. Three dimensional imaging of density/geometry of the tested sample is thus possible during tension/compression loading.

This system provides the first 3-D view of crack initiation, crack growth, phase transformation, bonded surface failure, creep--all with a density sensitivity of 0.1% and a resolution of about 25 microns (detectability of about 1 micron).

In ultrasonics, we are developing practical measurement technologies that overcome limitations of the traditional approaches. For example, new transducers are being developed that are phase insensitive thus permitting accurate measurements in complex nonplaner geometries and in anisotropic materials such as composites. New signal processing concepts have led to minimum error deconvolution capabilities to insure that the measurements represent clear quantitative information about the part measured without bias from the measurement technique or system. Image reconstruction from such quantitative data bases permits multiparameter assessment of materials thus increasing reliability and interpretability of the data.

In thermal NDE, we have developed noncontacting, remote capability of determining quantitative diffusivity--a measurement linked directly to basic material properties. Through computational modeling, the thermal data can be inverted to determine the internal properties of the measured structure. Thus, we are in effect, developing in thermal NDE a technology similar to that of x-ray tomography or MRI imaging.

Examples of these advances will be discussed in the oral paper focusing on ultrasonics, thermal NDE and smart structures.

A Method to Integrate Human Factors Expertise into the PRA Process

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Background

Thirty years ago the U.S. military and U.S. aviation industry, and more recently, in response to the U.S. Three Mile Island and U.S.S.R. Chernobyl accidents, the U.S. commercial nuclear power industry, acknowledged that human error, as an immediate cause and as an influencer in the form of maintainability, test, or surveillance programs, is a primary contributor to high-reliability systems unreliability and risk. A 1985 U.S. Nuclear Regulatory Commission (USNRC) study of Licensee Event Reports (LERs) suggests that upwards of 65% of commercial nuclear system failures involve human error.

Human Performance Assessment in PRA

Until very recently, PRAs adhered to a very engineering-centered approach with primary attention given to the hardware component of the commercial nuclear power plant. The human component was treated in only a peripheral manner usually by a systems engineer with little or no training in human factors. More specifically, when the human component was addressed, the analyst took a very narrow perspective on human factors, that is, he analyzed performance primarily cognizant of the individual operator and the single piece of equipment directly in front of him. Additionally, analysis of human performance was usually limited to tasks performed after onset of an abnormal situation, focusing primarily on errors of omission which might exacerbate the situation, and on the likelihood that recovery actions might be undertaken to return the system to a normal operating state. Little or no consideration was given to human errors of commission, especially cognitive errors, or to human errors preceding the onset of the abnormality such as maintenance and testing, or how workspace layout and habitability factors influence human performance. Particularly conspicuous was a lack of attention to person-person factors such as group processes, direct supervision, management policy, organizational, and external environment characteristics, which collectively have been determined to be both significant proximal and distal (latent, benign) causal factors in accidents of public notice within the commercial nuclear industry during the past 10 years. Despite these oversights, it has been concluded by those who perform PRAs, that human error is a dominate factor in plant risk, although the analyst has not been able to estimate, with any real degree of certainty, the magnitude and nature of human error, or its associated causal factors.

USNRC Integration and Applications Research

In response to the above recognition of need, the USNRC has initiated research directed toward the integration of human factors and hardware engineering expertise throughout the PRA process. Success in this area is perceived as vital to the overall credibility of future PRAs. That is, no matter how adequate state of knowledge single task and sequence evaluation methods are, if they are not fully exploited by qualified human factors and human reliability specialists, realistic assessments of human and hardware performance separately, and in combination, will not be achieved.

Also, in response to the prevailing state of PRA, the USNRC has initiated research directed toward methods for systematically applying PRA results as a technical base to: (1) retrofit/redesign equipment-centered and personnel-centered components of commercial nuclear power plants, (2) prioritize and resolve industry-wide safety issues involving human behavior, (3) establish risk-based performance measures for monitoring the effectiveness of operations, maintainability, test and surveillance programs, and (4) establish criteria for selecting, training and licensing, and subsequently evaluating the performance of plant personnel.

This paper presents a method for integrating broad-based human factors expertise into the probabilistic risk assessment (PRA)/human reliability analysis (HRA) process in a standardized manner to achieve results which: (1) provide more realistic estimates of the impact of human performance on nuclear power safety, (2) can be fully audited, (3) provide a firm technical base for equipment-centered and personnel-centered retrofit/redesign of plants enabling them to meet internally and externally imposed safety standards, and (4) yield human and hardware data capable of supporting inquiries into human performance issues which transcend the individual plant.

The TALENT Concept

Research on a concept for achieving the goals of integrating human factors expertise fully into the PRA process, and systematically applying PRA results to regulatory decision making, is in progress. The concept is currently known as TALENT, i.e., Task Analysis Linked Evaluation Technique. TALENT focuses on bringing together source data, quantification methods, and broad-based human factors expertise. TALENT is based on the premise that humans in complex high-reliability systems are driven to action or inaction, especially during periods of high stress, to a greater degree by latent and active cognitive, behavioral, and social factors, than they are by equipment-centered factors. TALENT is eclectic in that it takes advantage, wherever possible, of products from the other topic areas of the USNRC reliability assessment research program, products from other reliability assessment research programs within the U.S. such as EPRI, and products of international commercial nuclear communities such as the United Kingdom Atomic Energy Directorate, and the Commission of European Communities Nuclear Directorate. It also takes advantage of human factors products from non-nuclear research programs.

Top-down function, task, and related timeline or link analyses, forming the core of TALENT, are constructed from detailed reviews of system documentation, operating experience data, interviews with system personnel, and observations of system operations and maintenance activities. They describe, in detail, the what, why, when, where, and by whom of both human and hardware initiatives and/or responses required by the system during startup, normal operation, accident mitigation, accident recovery, and shut-down. They provide the PRA team with a credible technical basis for: (1) identifying safety related tasks and task sequences needing probabilistic assessments, (2) specifying endogenous and exogenous factors believed to influence human performance on each task action and task action sequence, (3) scaling the nature of that influence from positive to negative, and (4) contributing inputs to logic or plant models used as a general framework for estimating overall system unreliability and risk.

The TALENT implementation procedures take full advantage of earlier guidelines published by the NRC for conducting PRA studies, such as NUREG/CRS-2300, 2728 and 2815, as well as guidelines developed inside and outside the NRC to integrate human reliability analysis (HRA) and PRA such as a TEEM and SHARP respectively.

The Cognitive Environment Simulation as a Tool for Modeling Human Performance and Reliability

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ABSTRACT

Various studies have shown that intention errors, or 'cognitive error,' are a major contributor to the risk of disaster. Intention formation refers to the cognitive processes by which an agent decides on what actions are appropriate to carry out (information gathering, situation assessment, diagnosis, response selection). Cognitive errors in the coordination of abnormal operations are critical to risk and safety because, when an erroneous intention to act is formed, the problem solving agent not only will omit correct acts (omission error), but will also carry out other acts *that are correct given the perceived situation,* but are incorrect given the actual situation. This means that intention errors lead to a kind of common mode failure.

Understanding, measuring, predicting and correcting cognitive errors depends on the answers to the question--what are difficult problems? The answer to this question defines what are risky situations from the point of view of what incidents will the human-technical system manage safely and what incidents will the human-technical system manage poorly and evolve towards negative outcomes. Difficult problems are related to the ability to carry out the cognitive activities involved in intention formation: what information must be monitored and gathered? what knowledge must be activated and utilized to determine the state of the process and appropriate responses?

We have made progress in the development of such measuring devices through an NRC sponsored research program on cognitive modeling of operator performance. The approach is based on the *demand-resource match* view of human error. In this approach the difficulty of a problem depends on both the nature of the problem itself and on the resources (e.g., knowledge, plans) available to solve the problem. One can test the difficulty posed by a domain incident, given some set of resources by running the incident through a cognitive simulation that carries out the cognitive activities of a limited resource problem solver in a dynamic, uncertain, risky and highly doctrinal (pre-planned routines and procedures) world. The cognitive simulation that we have developed to do this in NPP accidents is called the Cognitive Environment Simulation (CES).

We will illustrate the power of this approach by comparing the behavior of operators in variants on a simulated accident to the behavior of CES in the same accidents.

SUMMARY

ORGANIZATIONAL LEARNING IN COMMERCIAL NUCLEAR POWER PLANT SAFETY: AN EMPIRICAL ANALYSIS

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The need for knowledge in organizations that manage and run high risk technologies is very high. When first introduced, bridges, natural gas lines, and commercial air travel all appeared to be dangerous. Today, these innovations are accepted and commonplace. By recognizing and dealing with the problems in these systems, the number and severity of incidents have been reduced and society is better able to live with the consequences of the residual danger. To the extent that the members of high risk organizations develop useful knowledge about the problems they confront, these organizations will be more effective. Useful knowledge in this context is knowledge about the relationship between the organization's actions and various safety outcomes.

The acquisition of useful knowledge is referred to as organizational learning. The theoretical roots of this concept are well established in the academic literature and in practice, especially in manufacturing industries. This paper focuses on organizational problem solving and learning as it relates to the safe and efficient management of commercial nuclear power plants. The authors are co-investigators on a larger team working under contract with the Nuclear Regulatory Commission to develop a logical framework that enables systematic examination of potential linkages between management and organizational factors and safety in nuclear power plant performance. Management and organizational factors that facilitate or impede organizational learning are only a part of the larger study, but are the major focus of this paper. In this paper, the theoretical roots of the concept of organizational learning are discussed, relationships to measures of safety and efficiency of commercial nuclear power plants are hypothesized, and empirical findings which provide partial tests of the hypotheses are discussed. This line of research appears promising; implications for further research, regulatory application, and nuclear power plant management are described.

In the research reported here, the outcome variables focused upon are plant efficiency and safety. Efficiency measures used are critical hours and outage rate. The safety measures are scrams, significant events, forced outage rate, safety system

actuations, and safety system failures. Results showed that the efficiency measures are significantly positively correlated with each other. The safety measures also tend to be significantly positively correlated with each other, with the exception of safety system failures, which appear to be a different dimension of safety. Efficiency measures and safety measures were not correlated with each other, thus suggesting that efficiency and safety are distinct performance outcomes, and one does not come at the expense of the other.

Organizational problem solving and learning capacity in nuclear power plants is hypothesized to be influenced by the ability and willingness to recognize problems and having the resources to correct perceived deficiencies. In this research, problem recognition is operationalized in terms of number of reported major violations and licensee event report data. Problem recognition is not enough though, for without adequate resources the insights gained from knowledge cannot be translated into action. For purposes of this research, resources were operationalized in terms of return on assets, which is a profitability measure which controls for size, and debt to equity ratio, which reflects the soundness of the utility's overall financial position and its access to debt markets to fund capital investments that may be needed for safety. Problem solving and learning also imply a time dimension, for it is only over time that one could expect to observe the results of effective learning. Consequently empirical tests conducted in this study have been designed to look at time lags in the relationships between problem solving and performance results.

Hypotheses were developed and tested concerning the effects of utility financial resources, major violations and licensee event reports on subsequent safety and efficiency measures. The analytical technique employed was regression using polynomial distributed lags. Results suggest that both financial resources and organization problem solving/learning have significant effects on the outcome variables when time is properly taken into account.

The results of this study require further corroboration, for there were limitations to the measures employed and restrictions on the amount of data available. The limitations are described in the paper. However, the results are promising, and suggest that further study of factors which promote problem solving capacity and learning may provide important insights into the management and organization of nuclear power plants to promote plant safety and efficiency.

GENERALIZING HUMAN ERROR RATES: A TAXONOMIC APPROACH

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It is well established that human error plays a major role in malfunctioning of complex, technological systems and in accidents associated with their operation. Estimates of the rate of human error in the nuclear industry range from 20-65% of all system failures. In response to this, the Nuclear Regulatory Commission has developed a variety of techniques for estimating human error probabilities for nuclear power plant personnel. Most of these techniques require the specification of the range of human error probabilities for various tasks. Unfortunately, very little objective performance data on error probabilities exist for nuclear environments. Thus, when human reliability estimates are required, for example in computer simulation modeling of system reliability, only subjective estimates (usually based on experts' best guesses) can be provided.

The objective of the current research is to provide guidelines for the selection of human error probabilities based on actual performance data taken in other complex environments and applying them to nuclear settings. This involves identifying tasks in non-nuclear power plant environments that are similar to those in nuclear settings, uncovering human error data bases for these non-nuclear tasks, and establishing error rate anchors for the tasks in nuclear settings based on the non-nuclear tasks.

A key feature of this research is the application of a comprehensive taxonomic approach to nuclear and non-nuclear tasks to evaluate their similarities and differences, thus providing a basis for generalizing human error estimates across tasks. In recent years significant developments have occurred in classifying and describing tasks. Initial goals of the current research are to: (1) identify alternative taxonomic schemes that can be applied to tasks, and (2) describe nuclear tasks in terms of these schemes.

Three standardized taxonomic schemes (Ability Requirements Approach, Generalized Information-Processing Approach, Task Characteristics Approach) are identified, modified, and evaluated for their suitability in comparing nuclear and non-nuclear power plant tasks. An agenda for future research and its relevance to nuclear power plant safety is also discussed.

SAFETY SYSTEM FUNCTION TREND INDICATOR: THEORY AND TEST APPLICATION*

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The purpose of this paper is to summarize research conducted on the development and validation of quantitative indicators of safety performance. This work, performed under the "Risk-Based Performance Indicator (R:PI) Project," FIN A-3295, for the Office of Research (RES), is considered part of NRC's Performance Indicator Program which is being coordinated through the Office for the Analysis and Evaluation of Operational Data (AEOD).

For the past several years, similar programs have been undertaken by the nuclear industry, specifically by INPO. It is understood that identification of plants with degrading or unacceptable performance is a cornerstone of a safety philosophy which governs the necessary condition for safety improvement in nuclear power plants.

Work performed at Brookhaven National Laboratory (BNL) concentrated in two specific areas:

1. the determination of various options available for the construction of risk/ reliability based indicators taking into account the extent to which data availability, model complexity, and man-power requirements are needed to implement the indicators, and

2. the evaluation/validation of these indicators in terms of their abilities for:

   a. detecting degraded performance (trends),
   b. detecting unacceptable performance (alerts), and
   c. identifying the various performance contributors.

Several reports have been generated to address the aforenoted tasks. The program originally focussed on risk-based indicators at high levels of safety indices[1] (e.g., core-damage frequency, functional unavailabilities, and sequence monitoring). The program was then redirected towards a more amenable goal, safety system unavailability indicators, mainly due to the lack of PRA models and plant data. In that regard, BNL published a technical report that introduced the concept of cycle-based indicators and also described various

*Work performed under the auspices of the U.S. Nuclear Regulatory Commission.

alternatives of monitoring safety system unavailabilities[2]. Further simplification of these indicators was requested by NRC to facilitate their applications to all plants in a timely manner. This resulted in the development of Safety System Function Trend (SSFT) indicators which minimize the need for detailed system model as well as component history. The theoretical bases for these indicators were developed through various simulation studies to determine the ease of detecting a trend and/or unacceptable performance[3]. These indicators, along with several other indicators, (including Option 2 INPO indicators) were then generated and compared using plant data as a part of a test application[4]. The SSFT indicators, specifically, were constructed for a total of eight plants, consisting of two systems per plant. Emphasis was placed on examining relative changes, as well as the indicator's actual level. Both the trend and actual indicator level were found to be important in identifying plants with potential problems. It should be noted that the level of a SSFT indicator does not necessarily correspond to the average system unavailabilities estimated conventionally by PRAs. This is mainly due to the dynamic nature of these indicators as well as the simplified models used for their construction.

For the remainder of FY89, we are concentrating on the implementation issues of these indicators, such as utilization of NPRDS data, and the determination of alert levels. The long-range risk-based indicator technology will be studied within FY90, which includes plant risk monitoring through utilization of PRA models (such as NUREG-1150 or IPEs).

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RISK-BASED CONFIGURATION CONTROL SYSTEM: ANALYSIS AND APPROACHES*

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This paper presents an evaluation of the configuration risks associated with the operation of a nuclear power plant and the approaches to control these risks using risk-based configuration control considerations. In that context, the actual and maximum potential configuration risks at a plant are analyzed and the alternative types criteria for a risk-based configuration control systems are described.

The risk-based configuration calculations which are studied here focus on the core-melt frequency impacts from given plant configurations. By calculating the core-melt frequency for given configurations, the configurations which cause large core-melt frequency increases can be identified and controlled. The duration time in which the configuration can exist can then be limited or the core-melt frequency level associated with the configuration can be reduced by various actions. Furthermore, maintenances and tests can be scheduled to avoid the configurations which cause large core-melt frequency increases. Present technical specifications do not control many of these configurations which can cause large core-melt frequency increases but instead focus on many risk-unimportant allowed outage times. Hence, risk-based configuration management can be effectively used to reduce core-melt frequency associated risks at a plant and at the same time can provide flexibility in plant operation.

The alternative strategies for controlling the core-melt frequency and other risk contributions include:

1. Controlling the increased risk level which is associated with the configuration.

2. Controlling the individual configuration risk which is associated with a given duration of a configuration.

3. Controlling the time period configuration risk from configurations which occur in a time period.

The characteristics of these strategies and the advantages/disadvantages of each of them will be discussed. Criteria associated with these strategies will also be discussed.

*Work performed under the auspices of the U.S. Nuclear Regulatory Commission.

Configuration risks are finally analyzed in order to evaluate the alternative strategies for control. The characteristics of configuration risks analyzed are: the risk significant configurations that can occur during plant operation, the risk-level associated with such occurrences, the expected frequency of occurrence of such configurations and the risks allowed in current tech specs from these configurations. Also, the historical occurrences of configuration risks at a plant site are studied using plant-specific component outage data from the Nuclear Plant Reliability Data System (NPRDS). Based on these evaluations, approaches and the criteria for effectively detecting and controlling configuration risks will be discussed.

DEVELOPMENT AND INTEGRATION OF PROGRAMMATIC PERFORMANCE INDICATORS

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This paper describes the results of an evaluation of maintenance-related programmatic performance indicators, and summarizes the direction being taken in a new project to integrate indirect performance indicators for nuclear power plants. Programmatic indicators allow NRC to monitor, at a distance, trends in functional activities before a significant impact appears on safety.

Evaluation of Maintenance Indicators

Previously presented work[1] described the selection of candidate performance indicators associated with maintenance for continued analysis. This evaluation focused on two aspects of the selected indicators:---

(1) an evaluation of the state of maintenance programs in the narrative text of SALP reports versus the frequencies of inadvertent ESF actuations from test & maintenance errors; and

(2) an evaluation of alternative methods for analyzing the thermal performance of plants as an integral indicator of maintenance program effectiveness.

The reevaluation of the inadvertent ESF actuations indicator confirmed its ability to present evidence of less-than-ideal maintenance programs. The comparison of the numbers and trends of events with SALP assessments indicates that the plants having the higher frequencies are those plants considered to exhibit a lack of root-cause analysis, an inadequate equipment performance trending system and a lack of effective management involvement. Further analysis examined the corrective actions taken by the plants to the ESF actuations as described in the full-text LERs. The plants exhibiting the highest frequencies of events had a tendency to report "discipline" or "counselling" as their corrective actions rather than any modification to the systems (hardware or people). Believing that discipline or counselling comprise an adequate response to events is *prima facie* evidence that no true root-cause analytical process exists. A pattern of discipline-only corrective actions corresponds to a mind-set that describes accident causation as "pilot error."

The evaluation of thermal performance has focused on a measure called Daily Power Loss (DPL), defined to be the average shortfall of power sent out below the maximum dependable capacity, calculated on a daily basis. Data associated with this measure are currently reported to NRC in the docketed monthly operating reports. An example of this DPL measure is shown in Figure 1, which includes annotations of the causes of losses; these are almost entirely caused by balance-of-plant (BOP) equipment failures. The frequency and duration of the DPL parameter represent broadly the reliability of the BOP equipment, which, in turn, is a product of the maintenance programs. Other causes of losses include tech. spec. forced outages. Further work is being performed on developing an indicator associated with DPL based on systems analytical methods such as fast Fourier transforms; such methods can provide unique perspectives on indicators such as DPL.

Integration of Indirect Indicators

Work has started on the creation of two frameworks to provide a basis for integrating current NRC indirect performance indicators. This integration will show the relative functional areas being monitored, and what, overall, is the position of the plant based on the individual indicator measures. These frameworks will represent both hardware and people factors in the overall system, including the formal and informal influences of humans on safety performance.

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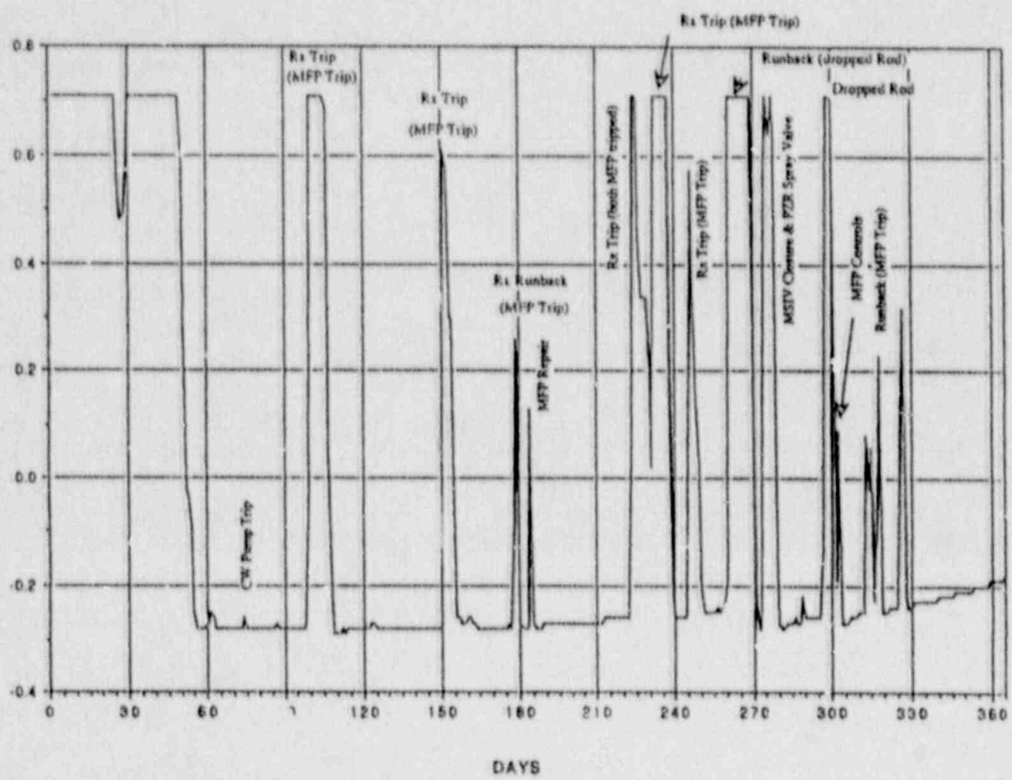

Figure 1, Daily Power Loss Plot

Extending the Evaluation of MAPPS: Results of User Participation

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SUMMARY

An evaluation of the user friendliness and usefulness of the Maintenance Personnel Performance Simulation model (MAPPS) was conducted at three nuclear facilities. The same protocol, comprised of five parts, was administered to each of the participants. During the first part of the evaluation process, power plant personnel were presented with an orientation and training session. Both the orientation and training were evaluated at the end of the data collection effort. Using a procedure selected by the facility as a guide, a task analysis was then performed in order to elicit plant specific knowledge useful in setting up the MAPPs input deck. For the third part of the evaluation, plant personnel participated in manual preparation of the MAPPs input parameters setup deck and in preparation of the actual data file. Next, personnel then participated in limited debugging of logic in the input deck and subsequent running of the MAPPS model. They also helped in defining parameters for a number of sensitivity runs which were successfully conducted. Lastly, simulation output was reviewed and some preliminary interpretation of the results was performed. Constructive criticisms and commentary were elicited through use of face to face interviews and through the application of a standardized survey. Acceptance of the model was positive with users indicating that MAPPS was able to give users a better understanding of common maintenance related tasks. Many reasons given for making the MAPPS simulation available in a personal computer (PC) environment and increasing the diagnostics available to first time users of the system.

Orientation and training at the facilities proceeded in rapid order due, in part, to volunteers being comfortable with computers. Currently, MAPPS requires a mainframe and Crosstalk(TM) software to establish a link to the IBM 370 mainframe. After this linkup is achieved, personnel must define task and subtask parameters on-line.

Personnel training on the MAPPS simulation encompassed achieving computer link up, performing task analysis, familiarity with the MAPPS taxonomy for human activity, instructions on how to code branching and decision making activities by crews and briefings on various model features such as the modeling of crew recovery actions. Participants demonstrated a good grasp of MAPPS basics and had a strong facility for estimating working environment temperatures, task time availability, essentiality of tasks, and branching within task sequences all of which are estimated in preparing MAPPS runs. They were also briefed on the various model defaults for basic parameter values.

MAPPS simulation output includes success proportion, the number of crew attempts to perform a task, detected and undetected crew errors, false alarms (crew misperceptions that they had committed errors), and the time required to complete the task. This output is available for up to one hundred iterations of a particular task sequence. If more than one hundred iterations are desired by users, then the outputs may be chained together. Often, users will use the same sequence and employ a slightly different configuration of parameters. By so doing, they can gauge the sensitivity of tasks to changes in task related factors. When performing sensitivity runs the number of iterations, as well as ability level of crews, time since the task was last performed and environmental factors may all be modified by persons wishing to run the model.

Personnel participating in the MAPPS evaluation were able to quickly grasp the model essentials and were interested in extending their knowledge of tasks by performing a number of sensitivity runs. For example, trending analysis at two facilities was performed for the task success proportion where tasks had been performed under varying conditions of exposure times, temperatures, and crew ability levels.

During the evaluation, suggestions were made to vary the format of the output, allow for easy editing during data entry, to avoid the need for a mainframe computer, and to allow for friendlier diagnostics. Transfer of the MAPPS simulation to a PC environment will address many of the needs highlighted by these comments and is anticipated to be accomplished prior to the end of FY 1990.

SEVERE ACCIDENT ZIRCALOY OXIDATION/HYDROGEN GENERATION BEHAVIOR NOTED FROM IN-PILE TEST DATA[a]

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SUMMARY

The primary source of hydrogen (H_2) during severe LWR accidents is from oxidation of Zircaloy (Zry) cladding by steam. In a PWR like TMI-2, complete Zry reaction would produce ≈ 1000 kg H_2; a comparable BWR would yield nearly 2000 kg H_2. Release of such large quantities of H_2 to containment structures with an air atmosphere could produce destructive deflagrations or explosions, with pressures in excess of design values. An understanding of the processes affecting H_2 production are thus of importance to LWR risk assessment.

In this paper Zry-oxidation/H_2-generation data from several in-pile severe-fuel damage (SFD) test programs are presented and compared for common findings. The experiments evaluated include the partial-length (≈0.9 m) 32-rod bundle tests performed in the Power Burst Facility (PBF), the full-length high-temperature (FLHT) tests performed in the National Research Universal (NRU) reactor at Chalk River (Canada), and the smaller (0.5 m rod length) BWR DF-4 test conducted in the Annular Core Research Reactor (ACRR). Although these tests were conducted over a wide range of experiment conditions, a number of common findings are observed which have a significant impact on the in-vessel H_2 source term for severe accidents. The principal issues assessed concern the effects of Zry melting and bundle reconfiguration on H_2 generation.

With respect to Zry melting, a comparison of on-line H_2 and cladding thermocouple data for the PBF, BWR DF-4, and NRU-FLHT tests indicate that the major portion of hydrogen release occurred after melt temperatures were reached. Extensive metallurgraphy was performed for the PBF and DF-4 tests, indicating that Zry-bearing melt continued to oxidize during and following melt relocation. Arguments for cutoff or significantly deminished H_2 generation upon Zry melting and relocation are not supported by these test data.

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a. This work was sponsored by the U.S. Nuclear Regulatory Commission in cooperation with an international partnership which includes Belgium, Canada, Federal Republic of Germany, Finland, Italy, Japan, Netherlands, Republic of Korea, Spain, Sweden, Switzerland, United Kingdom, American Institute of Taiwan, and the Electric Power Research Institute.

For BWR canned fuel assemblies, it has been postulated (by the IDCOR program) that Zry melting and debris relocation will lead to a completely blocked BWR fuel assembly and flow diversion to peripheral bundles. As a result steam access and hydrogen production are terminated at melt relocation, using the IDCOR-MAAP-BWR code. The validity of this hypothesis hinges on two key assumptions, total flow area blockage and an intact BWR channel box. The DF-4 test was specifically designed to address meltdown behavior of BWR structural and control components. Results of the BWR DF-4 test indicate early channel box failure due to attack by control blade melt debris, specifically eutectic interaction between stainless-steel melt (the cladding material of the B_4C control blade) and Zry channel box. Metallurgical examination revealed that all but the lower 10-percent of the channel box had been destroyed by eutectic melt interaction.

The DF-4 thermocouple data also indicate that Zry-oxidation induced temperature escalation continued well after initiation of Zircaloy melting and relocation, with continued steam access to the degraded bundle throughout the test. These findings are corroborated by post-test metallurgical observations of residual open flow area and a high degree of oxidation of once-molten/relocated Zry debris. Only partial flow area blockages were also noted for the PBF and NRU test bundles. Neither the DF-4, or any of the PBF and NRU tests, have indicated complete flow area blockage required for termination of steam access and continued H_2 generation in degraded test bundles.

In summary, in-pile test data indicate: (a) a continued high rate of oxidation during and after Zircaloy melting and relocation; (b) only partial flow area blockages; and (c) destruction of the BWR channel box by Zr-Fe eutectic melt interaction which allows for continued steam access and H_2 generation in degraded fuel bundles. Observation from these tests do not indicate inherent limitations on H_2 generation by core degradation, other than that due to steam supply conditions.

HYDROGEN MIXING EXPERIMENTS IN THE HDR-FACILITY

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During severe accidents in light-water reactors, substantial amount of hydrogen can be generated by the metal-water reaction during core heat-up and by core-uncovery as well as by virtue of core-concrete interactions after vessel lower head failure. This hydrogen is released into the containment. A key issue within this context relates to the global and local hydrogen distribution and associated mixing phenomena in multi-compartment geometry in order to plan for proper mitigation measures.

During Phase II of the HDR Safety Program, a preliminary H_2-distribution experiment, T 31.5, was performed in December 1987 in the aftermath of the International Standard Problem (ISP) 23 for the purpose of obtaining a first set of experimental data for long-term gas transport behavior in a large-scale, multi-compartment facility in the presence of steam under natural convection conditions. This preliminary experiment was needed for reliably planning the major H_2-distribution test series E 11.1 through E 11.5 to be performed between May and August 1989.

Once a realistic temperature gradient was established in the containment as initial and boundary condition resulting from the large break LOCA-experiment of ISP 23 it was decided to initiate the steam injection 20 min after blowdown begin for 15 min. Immediately after shut-off of the steam injection, the H_2/He-mixture (He: 85 vol %; H_2: 15 vol %) was injected for 12 min. into the HDR-containment.

The presentation gives a consistent and detailed overview of experimental data of measured temperatures, hydrogen concentrations and velocities in different regions of the HDR-facility.

The paper presents major experimental results in form of the transient built-up and decay of gas concentration as well as the time evolution of other important quantities influencing the gas transport. With these informations as background, reasons for the gas concentration redistribution at later times in the transient will be provided.

These experimental data will be supplemented by a summary of the comparison between measurements and pre-test predictions with different codes by various institutions. Possible reasons for some observed deviations will be identified. Emerging modeling approaches by different codes for the complex HDR-experiments will be summarized.

The measured results presented for T 31.5 evolve from a typical representative HDR large break LOCA-scenario. Other scenarios, especially those for small break scenarios may lead to quite different patterns. This is the subject of the H_2-distribution test group E 11.1 - E 11.5 in the first half of 198[illegible]

In fact, prior to the major Test Group E11, a scoping experiment E 11.0 was performed to examine the HDR-containment response for a slow steam heatup process, experimental results of which will be presented because of their importance with respect to blindly predict the E 11.2 and E 11.4 experiment, chosen as PHDR-Benchmark Exercises. The experimental data of E 11.0 indicate extreme stratification phenomena as well as a thorough heatup of all components as well as the annular gap between steel shell and secondary concrete containment. These phenomena are quite different from those observed in all previous HDR-experiments but show similarities with effects observed during the previous DEMONA-experiments and FIPLOC-verification tests in the Battelle model containment. The presentation concludes with the descriptions of the actual, revised experimental procedures used for E 11.2 and E 11.4.

Multi-compartment Hydrogen Deflagration Experiments in the Battelle-Frankfurt Model Containment

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Introduction:

Hydrogen combustion (deflagration, detonation) in an LWR containment during a severe accident is an important concern in reactor safety. So far, deflagration phenomena have been experimentally investigated mainly in single-volume facilities; these resulted in moderate pressure build-ups. Therefore, recent research has been directed predominantly at a possible transition from deflagration to detonation, from which relevant pressure peaks are expected.

The present paper, however, deals with experimental investigations on deflagration phenomena in multi-compartment containment geometries, which turned out to yield much higher pressure build-ups than had been expected on the basis of the findings of single-compartment experiments.

Objectives:

Identify and quantify risk-relevant hydrogen deflagration phenomena which may occur in a compartmented LWR containment in the course of a severe accident.

Provide an experimental data base for further development and validation of multi-compartment H_2 combustion codes.

Investigate in particular: Acceleration of a flame front propagating through a chain of compartments and interconnecting vent openings, and its effect on pressure increase. Influence of transverse venting. Effect of H_2 concentration gradients along the flame path. Interaction of combustion and gas distribution.

Experimental Facility and Instrumentation:

The experiments are being performed in the 640-m^3 Battelle Model Containment using a chain-type four-to-five-compartment configuration with a spark igniter at the dead end of the compartment chain. Before ignition, one to three upstream compartments (41 m^3 each) are filled with a hydrogen-air mixture, the remaining downstream compartments being filled with air of 1 bar. The maximum admissible pressure differential acting on the partition walls of the model containment amounts to 1.5 bar (0.15 MPa), the maximum admissible overpressure for the outer containment shell is 4.9 bar.

The quantities measured are: initial H_2 concentration; transient pressure, temperature, flow velocity in vent openings, and flame front location (IR detectors).

Test Matrix:

A two-compartment deflagration pre-test with 14 % initial H_2 concentration, which was performed in December 1988, resulted in excessive overloads and severe damage to test facility and instrumentation. Twelve additional tests with one to three compartments filled with max. 11 vol. % hydrogen were successfully performed in April and May 1989. Another set of at least four tests is scheduled for the end of 1990.

Supporting Model Calculations:

At the beginning of the experiments, no code was available to reliably predict multi-compartment deflagration effects. On the basis of the experience made in the December pre-test, the H_2 combustion models of the CONTAIN 1.10 code were modified to recalculate this particular pre-test and to predict the course of the first deflagration experiment of the new test series and the resulting pressure build-up. In the further course of the test series, the code modifications and the plant-specific parameters were improved from experiment to experiment, resulting in an accuracy of the peak pressure prediction of < 0.1 bar.

Major Findings and Conclusions:

- Hydrogen combustion phenomena in a multi-compartment geometry differ significantly from those in a single-compartment geometry.

- In the vent openings between the compartments (blockage ratios between 65 and 93 %) a jet is formed which ignites the hydrogen in the downstream compartments almost simultaneously (flame propagation velocities of several hundred meters per second), thus reducing the venting effect of the downstream openings and yielding high local overpressures.

- This "jet-ignition" effect was observed even in cases of moderate initial pressure build-ups in the upstream compartment (> 10 kPa) and/or of low H_2 concentrations (> 5 vol. %) in the downstream compartment.

- Jet ignition is strongly dependent on the geometry: It is very effective (as described above) for jets entering the downstream compartment in longitudinal direction, and less effective for transverse jets.

- As a reaction on the rapid pressurization of the downstream compartment following jet ignition, the upstream compartment pressure increases due to backflow and further combustion of residual hydrogen.

- A bypass opening between the first compartment and the dome compartment of the containment limits the pressurization of the first compartment.

- The described phenomena were observed in the present Battelle model containment geometry. Thorough analysis of these phenomena and subsequent development of improved deflagration codes will be necessary to transfer the experimental findings to real containment geometries and to assess possible consequences for reactor safety.

TMI-2 Reactor Vessel Metallurgical Sampling Project

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A project is underway to obtain metallurgical samples from the bottom of the TMI-2 reactor vessel. The purpose of this effort is to determine the metallurgical effects that the melted TMI-2 core material had on the bottom of the vessel and the instrument penetrations located there. The in vessel work is scheduled to begin in November of 1989.

The current general plan and sequence for taking samples from the bottom of the TMI-2 vessel is described below. The reactor vessel sampling is being undertaken with the general ground rule that we must not damage the reactor vessel pressure boundary; it must be able to retain water with a modest amount of pressure in the reactor coolant system. Further, the core baffle plates read in excess of 2000 R/hr and are to be stored in the reactor vessel, thus water is needed in the reactor vessel to shield personnel operating the long handled tools to obtain the reactor vessel samples.

Sampling is currently planned to be performed at both in-core nozzle locations and at open areas away from the nozzles. All work is to be performed through an 18 inch wide slot in a shielded work platform. The slot is about 40 feet above the sampling area and all tooling will be required to operate remotely through this opening.

The first step in the reactor vessel sampling effort will be to sever the in-core nozzles a few inches above the vessel inside surface. The nozzles are about 12 inches high, 2 inches in diameter and are made of Inconel 600. It is estimated that anywhere between 10 and 20 nozzles will be severed at this point. This number will include both those severed for nozzle penetration samples and those removed to permit ease of access in the bottom head area. The nozzle cutting tool is an hydraulic driven abrasive saw.

The removed portion of the nozzles will be retained in small special handling and storage containers. This will preserve the nozzles' individual identities so that metallurgical studies can be correlated back to the observed core debris in the bottom head.

The first reactor vessel samples will be taken from locations without in-core nozzle penetrations. The samples will be removed with a Metal Disintegration Machine (MDM) cutting device (electrical discharge cutting process). This device is being developed by PCI Energy Services. The MDM head has two U-shaped graphite electrodes that cut into the vessel surface at an approximate 45 degree angle so as to cut a triangular sample about three inches on a side and approximately 6-1/2 inches long. This will locally remove about one half of the vessel total wall thickness.

The MDM device is positioned in the vessel by means of a manual tool positioner that is mounted in the center of the reactor vessel on top of the work platform. The manual tool positioner consists of several straight pipe sections and was developed by GPUN for use with the plasma arc cutting torch. MPR has developed a hinged arm that attaches to the manual tool positioner and articulates outward from the center of the hemispherical lower head of the reactor vessel. A lead screw is used to allow the arm to extend/retract about 16 inches so as to position the MDM cutting head to the desired location for taking the reactor vessel samples. The MDM head must be positioned against the vessel with a load of about 600 lbs at angles up to 45 degrees.

While the positioning of the MDM cutting head is done manually from within the containment building, the controls for the MDM cutting process are located in the command center outside of containment. We estimate, at this time, that the sample cutting time will take approximately 10 to 15 hours, depending on the cutting process parameters and assuming that we do not run into any unforeseen problems. Once the samples are cut they will be placed in individual shielded containers similar to those used to hold the in-core nozzles.

The next set of vessel samples will be taken at in-core nozzle locations. Prior to cutting a sample that contains an in-core nozzle, a number of preparatory steps must be done as discussed below.

Each in-core penetration in the reactor vessel bottom head (there are 52) consists of a 3/4 inch-schedule 160 Inconel pipe and a two inch diameter in-core nozzle. The in-core pipes penetrate through close tolerance holes in the reactor vessel and are welded to the nozzles and the vessel on the inside surface of the lower head. Removing a vessel sample at these nozzle locations will remove the seal and retaining nozzle-to-pipe weld. To create a replacement seal, an in-core seal plug will be installed prior to the sa ing operation. The seal plug will be installed inside the 3/4 inch pipe, about 1 inch below where the sample will be taken. The plug is used to plastically expand the 3/4 inch Inconel pipe radially outward such that it seals against the vessel bore hole. This seal has been shown in tests to be leak tight at pressures in excess of 1500 psig and has a load retention limit of about 30,000 lbs. To install these plugs, the in-core instrument strings will be retracted from the seal table several feet to make room for the plug. Next, loose debris will be cleaned out of the pipe hole and a measurement will be made of the inside pipe diameter.

We basically have only 30 calendar days to do this entire operation. The goal is to remove between 8 and 20 samples, and if past history at TMI-2 is any guide, there are going to be some long days as well as some trying and frustrating moments. One thing that TMI-2 has taught us is to be prepared for a few surprises.

Bottom Head Failure Program Plan

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Earlier this year the NRC staff presented a Revised Severe Accident Research Program Plan (SECY-89-123) to the Commission and initiated work on that plan. Two of the near-term issues in that plan involve failure of the bottom head of the reactor pressure vessel. These two issues are (1) Depressurization and DCH and (2) BWR Mark I Containment Shell Meltthrough. For the former, the timing of bottom head failure is important in relation to the window for intentional depressurization or the occurrence of a failure elsewhere in the reactor coolant system as a result of natural circulation. And the mode of vessel failure, should it occur, is important because small penetrations and large openings would have greatly different effects on high pressure melt ejection and subsequent direct containment heating. For the latter, which is expected to be a low pressure event because of automatic depressurization, the timing and mode of bottom head failure will determine the rate of flow of material out of the vessel and across the concrete floor. Debris flow rate, in turn, has a big effect on heat transfer and attack on the metal structures. In both cases the timing and mode of failure will have a definite relationship to debris conditions (quantity, composition, superheat) at the time of failure. These conditions have a particularly strong impact on subsequent events.

Some work on bottom head failure has already been done in connection with the analyses performed for the NUREG-1150 risk studies and the TMI-2 accident evaluation. An indication of what work has been done and the general results is given below.

ORNL has developed models for several competing failure mechanisms for BWRs. The modes of failure considered are (a) failure of the penetration welds at the vessel wall, (b) failure of the instrument tube side wall, and (c) failure of the vessel wall itself. In cases where a pressure differential exists, creep-rupture of the metal is calculated; for other cases failure occurs at the melting point. Analyses to date indicate that both penetration failure modes occur shortly after bottom head dryout and within about ten minutes of each other whereas gross failure of the bottom head is not predicted to occur until 3-1/2 hours later.

INEL has performed analytical and experimental work directly related to bottom head failure in connection with several programs. The TMI-2 evaluation included (a) analytical studies on potential early failure of instrument penetrations from ablation by a molten jet, (b) finite-element studies of lower head heatup and mechanical response, (c) SCDAP/RELAP5 studies to determine coupling between system thermal hydraulics and debris behavior, and (d) TMI-2 core and vessel examinations. In addition, INEL has made creep rupture measurements, established master rupture curves, studied the influence of weldments on failure, and examined the influence of heat-affected zones. Although the timing and nature of bottom head failure will depend upon plant design and accident conditions, results from their TMI-2 analyses indicate that global creep rupture for PWRs is less likely than penetration failure.

SNL has conducted a number of analyses and experimental activities to examine the failure of LWR vessels. (a) Jet impingement experiments have been conducted to examine erosion of steel by high temperature melts. Metallic melts were found to be much more erosive than ceramics. (b) Flow of molten materials into penetrations has been analyzed with a code called PLUGM. Analyses suggest that the melt would not have enough superheat to cause failure in the small diameter PWR guidetubes. (c) Ejection of instrument tubes after weldment failure has been analyzed and experiments have been performed to validate the models. Under some conditions, differential thermal expansion will prevent tube ejection. (d) Global rupture of the bottom head has been analyzed with a model by Pilch that considers transient temperature and stress distributions and calculates creep rupture using a Larson-Miller parameter. For a Grand Gulf (BWR) analysis, binding of the instrument guide tubes prevented ejection, and global failure occurred 37 to 50 minutes after imposition of the vessel temperature increase.

In addition to the government-sponsored work mentioned above, EPRI and FAI performed studies on vessel failure for the Industry Degraded Core Rulemaking Program (IDCOR). EPRI examined the failure of a PWR vessel bottom head without penetrations, as found in some Combustion Engineering reactors. They found that a corium pool induced creep rupture in tens of minutes whereas a corium stream or jet causes plastic strain failure in tens of seconds. FAI examined the failure of BWR and PWR vessels with bottom head penetrations using the MAAP code. In both BWRs and PWRs, they found that penetration failures occurred in tens of seconds to several minutes, and therefore no credit was claimed for coolability within the lower plenum.

To give more attention to this subject as called for by the revised Severe Accident Research Plan, two things are being done. First, work previously done is being reviewed carefully to develop an overall picture and to determine the reliability of assumptions used in those studies. Second, new work is being planned for FY90 to try to complete a reasonable understanding of the failure process. The review and planning are being done in close cooperation with the ACRS. Results of this exercise will be presented in this paper.

The 1886 Charleston Earthquake--An Overview of Geological Studies

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The strongest historic earthquake in the southeastern United States occurred in 1886 near Charleston, S.C. The northeast-trending meizoseismal zone (encompassing Modified Mercalli intensity-X effects) was about 35 km wide and 50 km long, and the estimated body-wave magnitude (m_b) was between 6.6 and 7.1. The 300-year historic record documents persistent weak seismic activity near Charleston and suggests that potential may remain for recurrence of a strong earthquake. This possibility is reinforced by the discovery near Charleston of surficial features interpreted to represent a class of earthquake-induced paleo-liquefaction features known as sand blows. A few of these features can be ascribed to the 1886 earthquake, but most have strongly developed soil profiles, and some contain wood and carbonaceous clasts yielding radiocarbon ages long predating the historic earthquake near Charleston.

In South Carolina, Holocene liquefaction features are rare in generally compact, hard-to-liquefy sediments more than 250,000 years old. In contrast, they are abundant in loose, easily liquefiable, late middle to late Pleistocene sediments of the Charleston area. These features occur, in much smaller numbers, in sediments of similar age away from Charleston along the South Carolina coast southward to Bluffton, S.C., and northward to Southport, N.C. The ages and abundance of these features in the Charleston area suggest that Holocene seismicity has been much stronger there than in other parts of coastal South Carolina.

Radiocarbon ages from wood and charcoal clasts in prehistoric sand blows provide our most detailed chronology for the late Holocene earthquake history of the Charleston area. These data (and historic records) indicate that liquefaction-producing earthquakes occurred in the coastal South Carolina region around 100 YBP, 600 YBP, 1200 YBP, 1700 YBP, and 3200 YBP. Therefore, the apparent average recurrence interval has been about 550 years during the last 2,000 years. Although regular age distributions in small samples can be produced by chance, the documented sequence presently suggests periodic recurrence for earthquakes strong enough to produce liquefaction.

Other types of liquefaction features, also interpreted to be earthquake induced, are found in the Charleston area within the vicinity of the 1886 meizoseismal zone. These include V-shaped sand intrusions and ground shattered by sand intrusions. These types of features do not contain carbonaceous materials that will provide precise ages for their formation, but the degree of soil profile development upon them suggests prehistoric Holocene ages. Thus it can be concluded, on the basis of several types of liquefaction evidence, that multiple liquefaction-producing earthquakes have taken place near Charleston during the Holocene.

Recent seismic activity in the Charleston-Summerville area is occurring at depths of 3 to 15 km. These depths indicate that the ultimate source of seismicity lies well below the Coastal Plain cover. The origins of earthquakes within this basement complex beneath the Charleston area remain obscure despite numerous geologic, geophysical, and seismic studies during the past decade. A number of possible faults in the basement and deep Coastal Plain sediments have been located from geological investigations and reflection surveys, but none of these represents a major fault compatible with modern seismicity or with the intensity distribution of the 1886 earthquake. Several probable early Mesozoic basins and associated border faults have been identified, but conclusive evidence linking them to modern seismicity in the area is absent. The only basement feature that lacks obvious analog elsewhere in the Coastal Plain is a circular structure 31 mi in diameter; this structure may be a Paleozoic impact crater or caldera structure.

Neither historic nor prehistoric surface faulting has been documented in the Charleston area. The shallowest known fault is a small, high-angle reverse fault cored near Mount Holly, S.C., where middle Eocene Santee Limestone was thrust about 1.5 ft over basal Cooper Group (upper Eocene). Eocene units of the Cooper Group, especially the Parkers Ferry Formation, typically consist of plastic calcareous clay. The plasticity of this unit, which underlies all of the Charleston area, may retard or prevent the propagation of faulting upward above the level of the Santee Limestone and restrict near-surface tectonic expression to warps rather than faults.

A scenario of near-surface warping is compatible with the geometry of the Eocene-Oligocene unconformity within the Cooper Group. The geometry of this unconformity, defined by shallow subsurface drilling, reveals three domes and two intervening troughs, but no discrete faults. One domal feature, near Fort Bull along Ashley River Road south of the Ashley River, is overlain by late Pleistocene marine sediments at elevations higher than expected in the Charleston area. The anomalous elevation of this unit suggests that this dome may have been upwarped about 5 ft within the last 80,000 years. The area is aseismic at present, but it lies adjacent to, and directly along strike with, the northwest-southeast-trending Ashley River seismogenic zone, which terminates at its southeast end near Middleton Place. The other two domes, located near Mount Holly and Bonneau Ferry, are overlain by early to middle Pleistocene sediments that do not readily reveal evidence of recent upwarping. The northeast border of the Mount Holly dome, however, lies parallel to and just southwest of the West Branch of the Cooper River. The alignment of the river's course just beyond the edge of the dome suggests relatively recent tectonic activity in this area. These geomorphic observations link two of the domes to geologically recent tectonism in the Charleston area, but it remains undemonstrated whether or not these features are directly linked to deep-lying seismogenic faulted basement blocks.

PALEOLIQUEFACTION INVESTIGATIONS ALONG THE ATLANTIC SEABOARD IMPLICATIONS FOR LONG-TERM SEISMIC HAZARD

by

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The Charleston S.C. earthquake of August 31, 1886 (MM intensity X, estimated body wave magnitude 6.6-7.1) stands out as the largest seismic event to occur along the Atlantic Seaboard during historical times. In the late 1960's and early 1970's, it was generally accepted within the seismological and engineering community that seismicity occurring in the Charleston, S.C. area was related to a tectonic structure or structures unique to the epicentral area of the 1886 earthquake. Thus, for purposes of deterministic seismic design considerations for critical facilities, the occurrence of a similar large earthquake outside the Charleston, S.C. area was not generally considered a credible event. However, over the past two decades, extensive multidisciplinary investigations have failed to conclusively identify the cause or seismogenic source of the 1886 Charleston earthquake. Consequently, some investigators have taken the position that seismic events, similar to the 1886 earthquake, may occur along the Atlantic Seaboard outside the Charleston, S.C. area.

Within the past five years, investigations by the USGS, the University of South Carolina, and Ebasco Services Incorporated have documented the existence of liquefaction features caused by the 1886 Charleston earthquake and at least three large prehistoric earthquakes of magnitudes similar to the 1886 event. These studies and statistical considerations of the frequency vs. magnitude relationship of historical seismicity suggest that in the Charleston area the recurrence interval for large events similar to the 1886 earthquake is on the order of about one to two thousand years. In light of the very long return period documented in the Charleston area, the absence of large earthquakes elsewhere in the eastern U.S. during the limited historical record does not in and of itself preclude the possible future occurrence of similar rare earthquakes in other eastern U.S. locales.

For the past three years the United States Nuclear Regulatory Commission has supported a search for seismically induced paleoliquefaction features along the Atlantic Seaboard. These investigations are designed to determine in a systematic fashion whether or not seismically induced prehistoric liquefaction features, such as those associated with the 1886 Charleston, South Carolina earthquake, are present elsewhere in young sediments of the Atlantic Coastal Plain. The discovery of similar liquefaction features in other areas along the Atlantic Seaboard could indicate that large potentially damaging earthquakes have not been restricted to the Charleston area in the recent geologic past. Conversely, if at the conclusion of this systematic search no evidence of similar paleoliquefaction features are discovered outside the Coastal Plain of South Carolina, then the uniqueness of this area in the context of eastern United States seismicity would tend to be confirmed.

Initial phases of this study focused on developing a comprehensive control data set based on the evaluation of liquefaction sites and features located in the Charleston, S.C. area. A total of over 100 probable liquefaction sites were identified on the basis of the detailed review of historical accounts of the 1886 earthquake, and results of more recent studies conducted by investigators from the USGS, the University of South Carolina, and the authors. After identification of liquefaction sites and features in the Charleston, S.C. area, studies then centered on characterizing their geologic, stratigraphic, and hydrologic setting and identifying criteria by which similar locales could be identified elsewhere in the Atlantic Coastal Plain. These studies also included the development of recognition criteria which could be used to distinguish seismically induced liquefaction features from pseudoliquefaction features (i.e. other features which look similar but are not seismic in origin).

Based on these results, Ebasco is presently conducting a systematic search for similar seismically induced paleoliquefaction features in other parts of the Atlantic Coastal Plain. To date the search has focused on late Quaternary beach and near shore deposits in Virginia, North Carolina, South Carolina, and Georgia. These deposits are most similar to the units where the great majority of liquefaction features have been identified in the Charleston area. In addition, limited studies have also been conducted along the James River in Central Virginia and near Wilmington, Delaware (locales of moderate seismicity in the 1800's). To date, reconnaissance investigations have been completed at over 600 potential sites. More detailed investigations have been completed at over 250 of these same field sites, where conditions were determined to be especially conducive to the development of liquefaction features under low to moderate levels of seismic ground motion. Numerous psuedoliquefaction features have been observed. However, to date, conclusive evidence of seismically induced liquefaction has been found only in South Carolina.

Of the liquefaction features identified within South Carolina, many lie as much as 100 to 150 kilometers from Charleston, well outside the epicentral area of the 1886 event. Further, based on the extensive soil profiles which have developed on these outlying liquefaction features they are probably not associated with the 1886 Charleston earthquake. Cross cutting relationships observed at several of these outlying liquefaction sites clearly document the occurrence of at least two pre-1886 seismic events. The largest of the outlying liquefaction features appear to be associated with the "older" liquefaction episode. While smaller than the largest liquefaction features previously documented in the Charleston area, they are comparable in size to many liquefaction features observed in the epicentral area of the 1886 earthquake. Detailed studies are presently under way at several newly discovered outlying sites in South Carolina to provide insight as to whether paleoliquefaction sites located well outside the epicentral region of the 1886 earthquake are due to liquefaction associated with a larger pre-1886 Charleston earthquake, or liquefaction associated with earthquakes originating in locales outside the Charleston, S.C. area.

PALEOSEISMIC HISTORY OF THE MEERS FAULT, SOUTHWESTERN OKLAHOMA AND ITS IMPLICATIONS TO EVALUATIONS OF EARTHQUAKE HAZARDS IN THE CENTRAL AND EASTERN UNTIED STATES

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The Meers fault is part of the northwest-trending Frontal Wichita fault system, which extends more than 700 km across south-central Oklahoma and the Texas Panhandle. This system of faults separates a series of crustal uplifts on the southwest from very deep sedimentary basins to the northeast. This structural trend coincides with the Southern Oklahoma aulacogen, a fault-bounded trough that formed in the Middle Cambrian and extends into the craton from the southern margin of North America.

The Meers fault is one of the few faults in the United States east of the Rocky Mountains that is known to exhibit evidence of late Quaternary tectonic displacement. Southwest-facing fault scarps extend along a 26-km segment of the fault. Lineaments along the southeastern projection of this trace suggest the total length of Quaternary faulting could be as much as 37 km. Despite the evidence for recent faulting, southwestern Oklahoma has had very little historical earthquake activity. There have been no macroseismic events associated with the fault and there is no pattern of microseismicity that would suggest it is a capable fault.

Detailed geologic mapping and trenching at four localities along the northwestern part of the active trace indicate left-lateral oblique slip (down-to-the-southwest) on a steeply northeast-dipping to nearly vertical fault. The ratio of lateral to vertical slip is between 1:1 and 3:1. The summarized displacement data from these sites are tabulated below.

Preliminary analysis of 19 radiocarbon dates from the trenches indicate there have been at least two (possibly three) surface faulting events on the Meers fault during approximately the past 5000 years. The most recent event occurred about 1500 years ago. Based on the inferred rupture dimensions, these paleoseismic events were probably associated with earthquakes in the magnitude range of M_w 6¾ to M_w 7¼.

Analysis of faulted alluvial terraces along Canyon Creek suggests the recent episode of surface faulting was preceded by a long period of quiescence. Estimated terrace ages suggest that the quiescence lasted for at least several tens of thousands of years and may have lasted for hundreds of thousands of years.

Identification of earthquake sources and assessment of earthquake hazards in the central and eastern United States is complicated by the generally poor correlation between historical seismicity and geologic structure. This is certainly still the case for small to moderate magnitude events, but a pattern may be present for the larger events. Four late Precambrian or Paleozoic aulacogens extend into the craton from the Paleozoic southern margin of North America. From west to east these are: the Delaware aulacogen in west Texas; the Southern Oklahoma aulacogen in southern Oklahoma; the Reelfoot aulacogen north of the Mississippi Embayment; and the Mount Rogers Aulacogen in South Carolina and North Carolina. If one includes the paleoseismic events on the Meers fault, which is along the Southern Oklahoma aulacogen, major earthquakes are spatially associated with three of the four aulacogen. The New Madrid 1811-1812 earthquake sequence was along the Reelfoot aulacogen and the 1886 Charleston, South Carolina earthquake was spatially associated with the Mount Rogers aulacogen. The evidence for large paleoseismic earthquakes on the Meers fault and the spatial association of the large historical earthquakes to other late Precambrian or early Paleozoic aulacogen suggest the late Cenozoic strain deformation may be localized along these major zones of crustal weakness. Although identification of specific source structures for major historical earthquakes is still problematical, this pattern suggests that the occurrence of large earthquakes is not random. Expansion of the data set for large earthquakes in the central and eastern U. S. by examining geologic evidence for paleoseismic events is contributing to a better understanding of the nature and causes of major earthquakes. Ultimately, this will result in more reliable assessments of the potential seismic hazards in the central and eastern United States.

SUMMARY OF DISPLACEMENT DATA

	VERTICAL (m)			LATERAL: VERTICAL	CUMULATIVE NET SLIP (m)	NUMBER OF EVENTS	NET SLIP PER EVENT (m/event)
	Total Separation	Brittle	Ductile				
SITE 1	3.6±0.3	1.8	1.8±0.3	a)2.5:1 b)1.3:1*1	a)9.7±2.0 b)5.9±0.5*1	2(?)	a)4.8(?) b)3.0
SITE 2	2.4±0.4	1.8±0.2	0.6±0.4	1.3:1	3.9±1.4	2	2.0±0.8
SITE 4							
Upper Channel	2.7±1.0	0.9±0.2	1.8±1.0	1.3:1	4.4±1.2	2	2.3±1.2
Lower Channel	2.7±0.7	1.0±0.2	1.7±0.7	1.4:1			
SITE 5							
Browns Cr. al.	4.5±0.5	0.2 - 0.3*2	4.2±0.5	----	----	----	----
Porter Hill al.	4.4±0.9	0.6 - 1.0*2	3.6±1.0	----	----	----	----

*1. Assumes aspect ratio same as sites 2 and 4.
*2. From Luza et al. (1987).

THE POTENTIAL FOR GREAT EARTHQUAKES IN THE CASCADIA SUBDUCTION ZONE, COASTAL PACIFIC NORTHWEST--EVALUATION OF GEOLOGIC METHODS OF ASSESSMENT

by

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SUMMARY

A fundamental question in earthquake hazards research in the Pacific Northwest is the potential for future great earthquakes on the Cascadia subduction zone in Oregon and Washington. If plate-interface earthquakes have recurred in the late Holocene, determination of their magnitude, extent, and age is critical for resolution of safety issues in the region. A number of different geologic field studies aimed at resolving these issues are in progress in the coastal areas of the Pacific Northwest. Because the record of paleoseismic events is fragmentary, a variety of different methods are being used to piece together the late Quaternary tectonic and paleoseismic history of the region. Work in progress falls into two groups--paleoseismology studies of the middle and late Holocene (<5 ka) coastal record, and studies of cumulative late Quaternary tectonic deformation as expressed by marine and fluvial terraces and river valley morphology.

Holocene studies are focused primarily on identifying small (<2 m), sudden changes in relative sea level in coastal sedimentary sequences. Much debate has centered on whether observed changes in these sediments are abrupt or unique enough to be due to coseismic movements or whether they were produced by non-tectonic processes. The interbedded peats and muds so typical of late Holocene estuarine sequences in the Pacific Northwest are not unique to tectonically active coasts, and thus they cannot be assumed to have been caused by coseismic subsidence. For this reason, the most convincing studies have focused on evidence suggesting sudden, significant rises in relative sea level caused by coseismic subsidence of the land surface. In Oregon, Peterson and Darienzo found no evidence of deposition within marshes during either major historic floods or large storm surges. Delicate plant macrofossils found by Atwater at the upper contact of buried marsh surfaces in southwestern Washington are some of the best evidence for sudden, coseismic subsidence. In the estuarine muds located directly above these contacts, Hemphill-Haley found transported shelf diatoms, apparently carried by tsunamis. Detailed field and modeling studies of sand layers capping some of the buried marsh surfaces by Reinhart and Bourgeois should show whether the large, landward-directed surges of sandy water that accompanied some of the subsidence events were locally-generated tsunamis. In more protected environments, microfossil studies of modern diatom and foraminifera assemblages in Oregon marshes by Nelson, Jennings, and Kashima have shown that similar assemblages in cores can be used to distinguish gradual from sudden changes in relative sea level. These methods will help decipher the character (relative amount of pre-, co-, and post-seismic subsidence and uplift) of individual subsidence events.

Initially, coastal sediment studies hinged on the assumption that ^{14}C dating of estuarine sediments would be used to determine the age of coseismic subsidence events and to correlate these events from site to site along the length of the subduction zone. However, work by Atwater and others in Washington and our comparative dating program in central Oregon shows that conventional ^{14}C ages of buried marshes correlated using field relationships vary widely depending on the type of organic material analyzed and methods of sample preparation. Our work suggests that accelerator mass spectrometry ^{14}C analysis of carefully selected and extensively pretreated samples may be accurate enough to correlate events that are only 600-800 years apart. Tree-ring studies, such as those by Yamaguchi, offer the best hope of accurately dating the last one or two subsidence events, but this method can be applied to only those few sites with well-preserved fossil cedar trees.

Studies of the late Quaternary record are also being used to identify regional patterns of deformation, and these patterns can be compared with similar data from historically active subduction zones to help define the earthquake hazard in Cascadia. For example, new mapping and chronologic data obtained by Muhs, Kelsey, and others from a sequence of marine terraces in south-central Oregon showed that uplift rates were typical of rates from other active subduction zones. In the same region McInelly and Kelsey used detailed terrace and fault mapping to highlight the important role that local structures play in the deformation of marine terraces in southern Oregon. Active folds and associated flexural-slip faults in this area show that the southern part of the Cascadia subduction zone is dominated by deformation on local structures, rather than regional deformation from great plate-interface earthquakes.

In central Oregon, studies of fluvial terraces by Personius have also provided information on the styles and rates of forearc deformation. Many subduction zone forearcs have tectonically active fold and thrust belts, but ^{14}C and thermoluminescence dating of fluvial terraces along the Siletz, Siuslaw, Smith, and Umpqua rivers suggest slow, regional uplift and a general lack of active folds or faults in the Oregon Coast Range.

Another aspect of regional assessment of deformation is Rhea's study of river and drainage basin morphology throughout the Oregon Coast Range. While this analysis revealed numerous anomalies in stream gradient and valley geometry, further integration of geomorphic, lithologic, and structural data is needed to help identify patterns of regional differential deformation.

Field efforts of the last few years have produced convincing evidence of coseismic changes in land level along the Washington and Oregon coasts. However, at any single site the number, age, character, and magnitude of the accompanying earthquakes remains to be documented. It is still unclear whether coseismic events were responses to local faulting or folding or to regional deformation during great plate-interface earthquakes. Studies of coastal sediments suggest that regional seismic events (>100-km-long ruptures) are plausible in northern Oregon and southwestern Washington, but the sedimentary record in southern Oregon is best interpreted as the product of local events. These interpretations point to segmentation of the Cascadia subduction zone.

Although most studies have focused on the Holocene, regional neotectonic studies provide the critical tectonic framework needed for interpreting the fragmentary record of Holocene seismic events. Thus, when combined with seismological and geophysical studies and comparisons with other subduction zones, these regional studies may provide the most convincing evidence for segmentation of the Cascadia subduction zone. A credible segmentation scenario using both Holocene and Pleistocene data will be the basis for future estimates of magnitude and extent of probable plate-interface earthquakes in the region--critical information for the assessment of the safety of nuclear facilities.

THE NOVEMBER 25, 1988 SAGUENAY EARTHQUAKE IN QUEBEC PROVINCE AND ITS IMPLICATIONS FOR SEISMIC HAZARD ESTIMATES

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The 25 November 1988, Ms=6.0, Saguenay earthquake is the largest in eastern North America since the 1935 Temiskaming earthquake, also in southeastern Canada. Ongoing studies of this important event have already produced a number of surprising results concerning earthquake hazard estimates in eastern North America. Preliminary observations at this point include: 1. The epicentral region is characterized by low historic seismicity; 2. The steeply-dipping rupture is confined to the lower crust (25-30 km deep) and yet it seems to be geometrically correlated with fracture controlled topographic lineaments; 3. Several faults were activated in the aftershock sequence, possibly including one at a shallow angle; 3. Attenuation of high-frequency seismic waves over hundreds of kilometers is very low and is lower along than across the structural grain of the Appalachians; 4. Ground failure, including slope failures and liquefaction-related effects, were induced over a broad area, substantially larger than expected from the magnitude of the earthquake. Some of these points are discussed in more detail below.

The 1988 Saguenay earthquake was centered in the Laurentide Mts-Saguenay River region, 70 km west of the Charlevoix seismic zone. In light of the historic record and more than a decade of monitoring by a regional seismic network, this region had not been recognized as a significant source of seismicity; hazard maps were drawn accordingly. This earthquake was unexpected and underscores the limitation of the 150 to 300-year long historic record in identifying all possible sources of potentially damaging earthquakes in eastern North America. Thus, seismic hazard estimates based on a source distribution that reflects only the historic pattern of seismicity may underestimate the hazard in other regions of low historic seismicity.

Not only is the 1988 Saguenay epicentral region characterized by low background seismicity, but the aftershock sequence is also thinly populated, when compared to other sequences in eastern North America associated with main shocks of similar size, such as the 1982 Miramichi sequence in New Brunswick. Nevertheless, aftershocks recorded by a temporary network operated for about a month, when combined with data on the main shock, provide sufficient data to tentatively resolve the rupturing fault: a northwest striking fault dipping steeply to the northeast with reverse and left-lateral motion (North et al, 1989). The rupture appears to be confined between 25 and 30 km in depth. Aftershocks suggest another planar feature parallel to and about 10 km southwest of the main rupture. Both of these steeply-dipping features are limited down-dip by a plane dipping shallowly to the west and defined by a broad scatter of epicenters, possibly a third fault activated in the sequence. These three hypothetical faults fit one or the other of the nodal planes of the focal mechanism for the main shock. Topographic lineaments in the 1988 epicentral region show a pronounced peak in azimuthal distribution in a direction subparallel to the inferred northwest striking rupture plane. Moreover, a major lineament, at least as long as the 1988 hypocenter is deep, is spatially correlated with the surface extrapolation of that plane. Considering the recently discovered correlation of the 1983 Goodnow (Mb=5.2) and the 1985 Ardsley (Mb=4.0) ruptures with fracture-controlled lineaments, the possible correlation of the Saguenay rupture with a structure that has a surface expression seems plausible and merits close attention. Such correlation, however, may remain hypothetical because of the unusually large depth of the 1988 hypocenter.

Numerous studies have shown that seismic wave attenuation is lower in eastern than in western North America. Earthquakes of comparable magnitudes and distances tend to generate higher ground motion in the East than in the West. Data from the Saguenay earthquake not only support this general conclusion, but they suggest that the difference in attenuation between the East and the West may be even more pronounced than previously thought. Both peak and spectral ground motion amplitudes from this event were significantly higher than predicted from ground motions of previous eastern earthquakes. The higher than expected ground motion at high frequencies and at large distances may be due in part to the large depth of the 1988 source (28km). Both intensity data and theoretical modeling of seismic wave propagation suggest that ground motion amplitudes of events in the lower crust relative to amplitudes of events in the shallow crust are lower at short distances, comparable at the hypocentral depth, and higher at larger distances. In plausible configurations of source depth and velocity structure, peak acceleration values for a deep crustal earthquake may decay very slowly out to epicentral distances of 100-200 km.

The Saguenay data have also provided the first instrumental evidence of attenuation anisotropy. Wave propagation is significantly more efficient along the northeast structural grain of the Appalachians than across this grain. A similar conclusion was reached from the shape of isoseismal contours which tend to be elongated along structural trends of the Appalachians. The Saguenay earthquake is the first M≥6 eastern event to be recorded on scale at distances ranging from few tens to one thousand kilometers. How representative this event is in terms of ground motion will be resolved only with new data. Before then, it seems prudent to consider this earthquake representative where it increases previous hazard evaluations.

The Saguenay earthquake induced liquefaction and related ground failures at least 25 km away from its epicenter (i.e., 37 km from its hypocenter). Based on western data, this earthquake which is at least 25 kilometers from the nearest surficial deposits, would not generally be expected to cause liquefaction (Youd and Perkins, 1987). This phenomenon is likely to be due in part to lower attenuation and, consequently, higher ground motions from similar earthquakes in the East than in the West, as discussed above. In northeastern North America, there is a broad distribution of materials that are likely to be susceptible to liquefaction. These materials include glacial and post-glacial deposits, typically lacustrine, marine, and fluvial sands, silts and clays, which have not been overridden by retreating ice sheets and have high porosities. These poorly consolidated sediments often rest on competent, high velocity bedrock that allows seismic energy to reach the sediments with little attenuation. The above factors result in high liquefaction potential and may significantly contribute to seismic hazards in eastern North America, particularly in the glaciated Northeast.

The Saguenay earthquake triggered both liquefaction and slope failures. Similar effects are expected from M≥6 earthquakes in the same area during the Holocene. Thus, paleoseismic techniques may be useful in reconstructing the distribution of large prehistoric earthquakes improving seismic hazard estimates in the Laurentide-Saguenay-Quebec region. Detailed descriptions of deformation structures at sites of documented liquefaction provide information regarding the types of earthquake-induced structures that form in glaciolacustrine and glaciofluvial deposits. Furthermore, comparisons of site conditions and liquefaction features that formed at different distances from the source may help to develop a scaling relationship for estimating the size of past earthquakes. Analysis of the physical characteristics of deposits as a function of their environment of deposition will be useful in the development of such a scaling relationships as well as in the identification of indicator deposits for future paleo-earthquake investigations.

A National Seismographic Network for Assessing Seismic Hazards

by

Robert P. Massé and Andrew J. Murphy

Network developed by National Earthquake Information Center,
U.S. Geological Survey

To access the seismic hazard of a region and to establish the design and construction criteria for critical facilities such as nuclear power plants, detailed information is required on the frequency of occurrence, geographical distribution, magnitude, and energy spectra of earthquakes. Also important is information on the frequency-dependent attenuation of seismic waves. This information can all be obtained from data recorded by networks of seismograph stations.

In the past, the frequency bandwidth, dynamic range, calibration, and recording systems of seismograph stations have all been deficient to some degree. Advances in technology now make it possible to record very high quality seismic data over a large frequency band and with a large dynamic range. A new seismograph network for the United States which takes advantage of advances in technology is currently under development. This network is the United States National Seismograph Network (USNSN).

The USNSN is a cooperative effort between the National Earthquake Information Center (NEIC) of the U.S. Geological Survey and the Nuclear Regulatory Commission. The USNSN will be installed and operated by the NEIC.

The network will consist of approximately 150 seismograph stations distributed across the lower 48 states and across Alaska, Hawaii, Puerto Rico, and the Virgin Islands. The design goal for the network is the on-scale recording by at least five well-distributed stations of any event of magnitude 2.5 or larger in the continental United States, Hawaii, and Puerto Rico, and of any event of magnitude 3.5 or larger in Alaska.

Each station of the network will record three-component information with a very high dynamic range (210 dB). The frequency response will be broadband with primary emphasis on the range 20 Hz to 100 seconds. Data from each USNSN station will be transmitted in real time via satellite to the NEIC in Golden, Colorado. Calibration of the stations can be accomplished automatically or remotely for each channel. The satellite link has sufficient capacity and cost effectiveness to make feasible the transmission of regional network data in addition to the USNSN data.

The rapid access to all USNSN data will be provided by the NEIC. This will be accomplished both via a dial-up capability to the event waveform data base and by satellite transmission in a broadcast mode. All earthquake data will also be distributed on compact disk with read only memory (CD-ROM) to all institutions having an interest in the seismic data.

Overview of NUREG-1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants"

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One principal supporting element to the NRC staff's severe accident closure process is the reassessment of the risks of such accidents using the technology developed through the 1980s. This reassessment constitutes a snapshot of the risk of severe accidents at five commercial U.S. nuclear power plants - Surry, Sequoyah, Zion, Peach Bottom, and Grand Gulf. NUREG-1150 summarizes the results of the PRAs and gives perspectives on how the results may be used by the NRC staff in carrying out its regulatory responsibilities.

NUREG-1150 was first issued in draft form in January 1987 for public comment. In response, 55 comments were received totaling over 800 pages. Also, comments were received from three organized peer review committees: the Kout's Committee reviewed the uncertainty methods; the Kastenberg Committee and a special committee of the American Nuclear Society reviewed NUREG-1150 as a whole. All the comments received have been placed in the NRC Public Document Room. Appendix D provides a summary of these comments and NRC responses. The current (second) version of NUREG-1150 (Second Draft for Peer Review), dated June 1989, reflects improvements made as a result of the comments on the previous version of NUREG-1150 and the continuing development of the technology.

The objectives of NUREG-1150 are:

- To provide a current assessment of severe accident risks,

- To summarize the perspectives gained in performing the risk analyses, and

- To provide PRA models and results that can support ongoing prioritization of potential safety issues.

The analyses discussed in the report include the analysis of the frequency of severe accidents, the performance of containment and other mitigative structures in such accidents, and the offsite consequences of these accidents. The accident frequency analyses consider initiating events at full power operation. For two plants, both internal events (e.g. random failure of plant equipment, operator errors) and external events (e.g. earthquakes, fires) have been considered as initiating events; for the other three plants, only internal initiating events have been studied.

Significant improvements have been made in NUREG-1150 since the first draft was issued, especially in several areas. Expert judgment was used to fill in gaps in the understanding of severe accidents; the process for obtaining these judgments was improved over that used in the draft version. Also, the display of the results was improved.

In the NUREG-1150 report, the results of the five plant analyses are briefly described. The details of the analyses are reported in supplementary reports. NUREG-1150 highlights results and discusses perspectives that are to be gained from the results and uses of the analyses as a resource in the regulatory process. Principal results include:

- Core damage frequency (with and without external events),

- Principal contributors to the mean core damage frequency,

- Probability (conditional and absolute) of early containment failure,

- Source term results, and

- Several measures of public risk, including a comparison with the Reactor Safety Study and safety goals.

It is important to consider what NUREG-1150 is and is not:

- It is a snapshot in time of risk at five commercial nuclear power plants reflecting plant design and operational information as of March 1988,

- It is a quantitative and qualitative resource document,

- It is not the sole basis for making regulatory decisions, and

- It is not an estimate of the risks at all commercial nuclear power plants.

The second draft of NUREG-1150 is being reviewed by a special committee formed by the NRC. The purpose of the review is to provide the NRC with a technical peer review of the adequacy of the methods, insights, analyses, and conclusions. In particular, the committee will provide its comments and views on the following: the adequacy with which NUREG-1150 addresses the comments of the Kastenberg Committee; the adequacy of the description of uncertainties in both the front end (event frequency) and back end (severe accident phenomena); the extent to which PRA should focus on the low probability tails of the risk distributions; the degree to which the methods in NUREG-1150 can be used as standard methods in PRA; and recommendations. The review began in July 1989 and is expected to take roughly one year to complete. During this time, regular briefings by the NRC staff and its contractors will take place. Following completion of the peer review, the risk analyses will be updated as necessary, and final documents prepared.

SAND88-2253C (Summary)

USE OF EXPERT JUDGMENT IN NUREG-1150*

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I. INTRODUCTION

The expert judgment process used in NUREG-1150, "Severe Accident Risks: An Assessment for Five U. S. Nuclear Plants," is an advance over those processes developed in previous probabilistic risk assessments. The new process was used to obtain expert judgment on issues expected to be the main contributors to the potential risk of five nuclear plants. The use of expert judgment helped to incorporate both the experience and the research results obtained since the Reactor Safety Study (USNRC, 1975). The new process also enabled NUREG-1150 to include a comprehensive uncertainty analysis, an analysis that had been lacking in the earlier Reactor Safety Study.

This process for gathering expert judgment was developed in response to criticisms applied to the previous Reactor Safety Study by the Lewis Committee (Lewis, 1978) and to review comments of the Kouts Committee (1987) on the draft NUREG-1150. The process is based on accepted decision analysis techniques and findings from the numerous studies involving the quantification of judgment. The result, a formal process for eliciting and documenting expert judgment for risk assessment, is one of the major contributions of NUREG-1150.

Definition, Quality, and Use of Expert Judgment

Expert judgments are expressions of opinion, based on knowledge and experience, that experts make in responding to technical problems. Specifically, the judgments represent the expert's state of knowledge at the time of response to the technical question (Keeney and von Winterfeldt, 1988). Expert judgment is not restricted to the experts' answer but includes the experts' mental processes (definitions, assumptions, and algorithms) for arriving at answers.

The quality of expressed expert judgments can vary according to the manner in which they are gathered. Guidance on how judgments should be gathered, such as with minimal bias, comes from a large body of experiments on human cognition and communication. Much of this research has been stimulated by three fields--psychology, decision analysis, and artificial intelligence. The quality of expert judgment is usually evaluated in terms of the methods used to gather these judgments. For example, state-of-the-art elicitation methods should avoid any systematic bias and include consistency checks with feedback to the experts to allow for revisions of expressed judgments.

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Need of Expert Judgment in Risk Assessment

The use of expert judgment in risk assessment is often implicit. However, risk assessment frequently needs explicit expert judgment as a source of data, particularly if one or more of the following situations exist:

1. No other data for answering the technical problem or issue are available.
2. High variability characterizes the data.
3. Experts question the applicability of the data.
4. Existing data needs to be supplemented, interpreted, or incorporated with model or code calculations.
5. Analysts need to determine the state of knowledge about what is currently known, what is not known, and what is worth learning.

II. EXPERT JUDGMENT PROCESS

The expert judgment process used in NUREG-1150 includes steps recommended by state-of-the-art probability encoding studies. These steps are:

1. Selection of issues and experts;
2. Elicitation training;
3. Presentation of issues to the experts;
4. Preparation of issue analyses by the experts;
5. Discussion of issue analyses and elicitation of expert's judgments, and
6. Recomposition and aggregation of expert's judgments.

III. CONCLUSION AND RECOMMENDATIONS

As with all scientific and statistical analyses, expert judgment can be misinterpreted and misused. It is important to understand the proper use of expert judgment to reduce the likelihood of improper use.

Expert judgment should not be held as equivalent to technical calculations or models based on physical and mathematical laws or extensive data. But, expert elicitations are useful for those problems where recognized methods or sufficient data from solving the problems do not exist. They provide quantitative representations of the state of knowledge of such problems at the present time. These representations provide a basis for decision making and communications throughout the industrial, scientific, and regulatory communities interested in these issues. Interested parties (e.g., utility groups, regulatory agencies) who disagree with or are concerned by the results of expert assessments may become motivated to do experimentation or analysis on particular issues. If these efforts result in improved understanding of the issues and solutions that differ from the expert's assessment, then this is a contribution or success of the expert judgment process, not a failure. The expert assessments serve as the best representation of an issue until such time as the state-of-knowledge changes, and the expert assessment may be a catalyst for improved knowledge on the issue. The documentation of the experts' rationale and models allows reviewing and upgrading the results, if future experiments or analyses produce new information.

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

IMAGE EVALUATION TEST TARGET (MT-3)

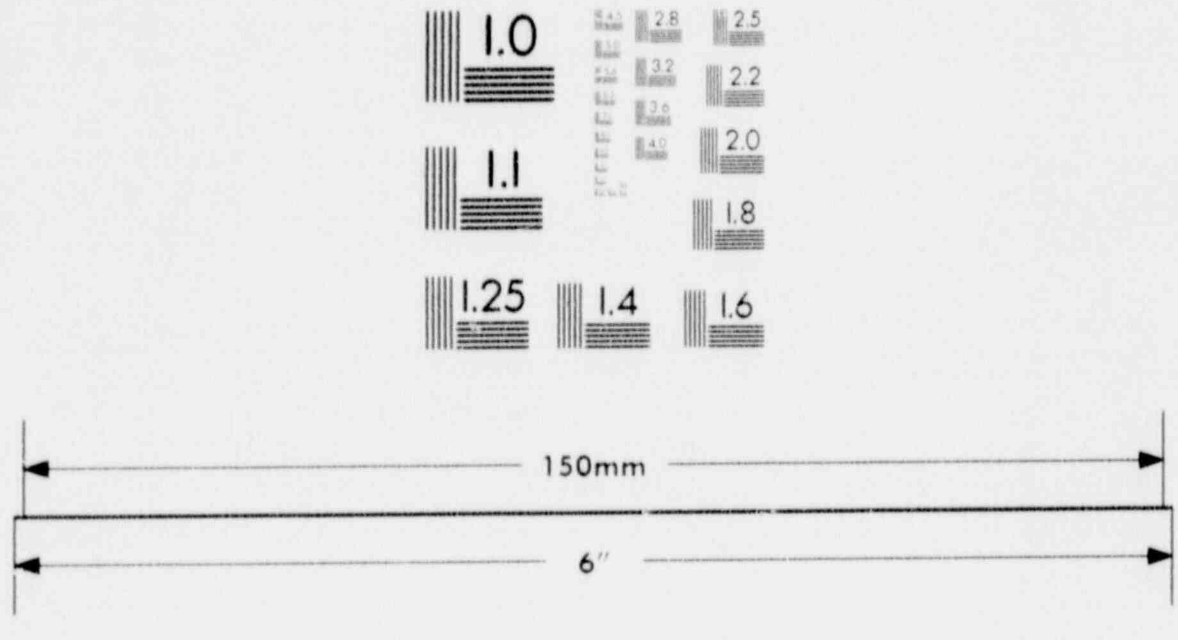

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IMAGE EVALUATION TEST TARGET (MT-3)

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External Event Analysis Methods for NUREG-1150*

by

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Abstract

The U.S. Nuclear Regulatory Commission is sponsoring probabilistic risk assessments of six operating commercial nuclear power plants as part of a major update of the understanding of risk as provided by the original WASH-1400 risk assessments. In contrast to the WASH-1400 studies, at least two of the NUREG-1150 risk assessments will include an analysis of risks due to earthquakes, fires, floods, etc., which are collectively known as "eternal events". This paper summarizes the methods to be used in the external event analysis for NUREG-1150 and the results obtained to date.

The two plants for which external events are being considered are Surry and Peach Bottom, a PWR and BWR respectively. The external event analyses (through core damage frequency calculations) were completed in June 1989, with final documentation available in September.

In keeping with the philosophy of the internal events analyses for NUREG-1150, which are intended to be "smart" PRAs making full use of all insights gained during the past ten years developments in risk assessment methodologies, the corresponding external event analysis will also be performed by newly-developed simplified methods. These methods have been under development at Sandia National Labs under the sponsorship of the NRC's Division of Risk Assessment as part of their Dependent Failure Methodology Development Program. The first application of these new methods was to the seismic analysis of six power plants as part of the NRC-NRR's program for the resolution of USI A-45, Adequacy of Decay Heat Removal Systems. Extension of these methods to fire, flood, etc., has been continuing during the past two years.

In contrast to most past external event analyses, wherein rudimentary systems models were developed reflecting each external event under consideration, the simplified NUREG-1150 analyses are based on the availability of the full internal event PRA systems models (event trees and fault trees) and make use of extensive computer-aided screening to reduce them to sequence cut sets important to each external event. This provides two major advantages in that consistency and scrutability with respect to the internal event analysis is achieved, and the full gamut of random and test/maintenance unavailabilities are automatically included, while only those probabilistically important survive the

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*This work is supported by the United States Nuclear Regulatory Commission and performed at Sandia National Laboratories, which is operated for the US Department of Energy under Contract Number DE-AC04-76DP00789.

screening process. Thus, full benefit of the internal event analysis is obtained by performing the internal and external event analyses sequentially.

The external event analysis begins with a review of the FSAR, related design documents and the systems descriptions in the internal events PRA. Important components are located on general arrangement drawings. The fire study Appendix R submittals are the basis for the initial identification of area fire and flood boundaries and barriers. Shortly thereafter a plant visit of 2-3 days duration is made, involving an integrated team of 6-8 specialists in the various external events and at least one systems analyst from the internal events PRA team.

The seismic risk assessment is the critical path item due to the time it takes to assemble the structural drawings and models. A best estimate structural dynamic response calculation is made coupling design beam-element models with a realistic model of the underlying soil column using the CLASSI soil-structure interaction code. The result is distributions for floor slab accelerations, and estimates of variability and correlations. Component fragilities are obtained either from a generic data base or derived on a plant specific basis as needed. Dual probabilistic screening methods are used to determine important cut sets while allowing for explicit incorporation of correlations. The seismic hazard itself is obtained by extrapolation from the results of the NRC sponsored Eastern Seismic Hazard Characterization Program. These results are directly modified to include local soil amplifications as needed.

The fire and internal flooding analysis tasks proceed in a parallel fashion. Fire initiator frequencies are obtained from an updated historical data set developed at SNL. Partitioning of building fire frequencies (for which data are available) down to sub-area frequencies is based on cable loading, electrical cabinet distributions and transient combustible estimates based on walkdown observations and a transient combustible data base developed at Sandia. Component damage temperatures (rather than auto-ignition temperatures) are based on SNL fire tests. The COMPBRN III code is used to predict component temperatures in fire areas where growth and separation are important considerations. Vital area analysis using the SETS and COMCAN codes provide sequence cut sets for quantification, including barrier failure and random failures as appropriate. A fire detection/suppression histogram developed at SNL is used to incorporate fire fighting timing into the analysis.

Similar approaches are used for internal and external flood, tornado, winds, etc. A major economy is achieved by analyzing fire and flood together and seismic, wind and tornado together due to the commonality of the analysis processes. For example, it is a minor task to extend the seismic fragility derivations to be applicable to wind fragilities. Similar economies arise in the screening steps for fire and flood.

UTILITY USE OF PRA
BY
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Northeast Utilities (NU) operates four nuclear power plants in Connecticut: Connecticut Yankee at Haddam Neck and Millstone Units 1, 2 and 3 in Waterford. Differences in design and age add to the complexity of operating the four units: Millstone Unit 1 is a G.E./Ebasco BWR, Millstone Unit 2 is a C.E./Bechtel PWR, and Connecticut Yankee and Millstone Unit 3 are Westinghouse/S&W PWRs with commercial operation dates separated by 18 years.

NU's first involvement with PRA dates back to the late 1970's when limited PRAs of decay heat removal systems of Millstone Unit 1 and of Connecticut Yankee were performed. This work provided significant insights into systems limitations and opportunities for improvements, thus inviting PRA application on a larger scale. In particular, the success of limited scope PRA in identifying systems limitations left unanswered the question of whether limitations existed in other systems which may be more severe in terms of overall plant safety. This developed within NU the recognition of the strength of PRA as a system engineering and decision making tool, and of the need for full plant PRA. The accident at Three Mile Island provided the final impetus for a comprehensive PRA program. This program, initiated in the early 1980's, includes the in-house performance of full PRAs for the four units, their maintenance as living PRA's, and their use in most facets of nuclear power plant engineering and operations. PRA activities are currently carried out by a dedicated staff of 13 engineers supported by other disciplines as needed. This program is supported and complemented by an internal framework for action in response to PRA insights.

The importance of developing such a framework cannot be overestimated, because this is the process by which needs and uses of the living PRA program are specified. Without specific commitments to and from engineering and operations, PRA role, characteristics and uses are not specified and the PRA becomes ineffective, providing only a snapshot in time of plant risk. If, conversely, the PRA model is tied to the dynamics of engineering and operations, it becomes a critical decision making tool and, in turn, is continuously improved with time. In fact, its use in critical plant decisions enhances the model, as PRA recommendations affecting plant operations receive great scrutiny by the people responsible for the areas most affected by them. Thus, modeling assumptions are routinely questioned and tested in the operational environment, and operator actions are tested on the plant specific simulator. Also, plant specific data bases are continuously refined. Thus, the organizational framework surrounding the PRA program is paramount to the quality, vitality and effectiveness of the PRA program.

NU's organizational framework supporting the PRA program includes:

- o A Corporate Policy Statement endorsing safety goals for core melt, individual and population risk and committing to corrective action commensurate to the significance of PRA findings.
- o A routine and prompt notification procedure to maintain senior

management appraised of status of the risk assessment for each unit.

- A procedure requiring that all proposed design changes be reviewed at the conceptual stage for PRA impact. This allows us to prevent narrowly focused backfits which, inadvertently, achieve a specific engineering objective at the expenses of safety in some other plant area.
- A procedure requiring that all plant changes requiring a 10CFR50.59 safety evaluation be PRA evaluated. This is done to assure that changes which could be acceptable based on current regulation, but increase risk, will not be implemented; and to assure that changes which are unreviewed safety questions, but reduce risk, are proposed.
- PRA "Key Assumption Documents" to communicate critical assumptions and prevent changes without previous review and concurrence.
- A project prioritization scheme to evaluate and prioritize all proposed projects (ISAP). NU was the original utility participant in the Integrated Safety Assessment Program (ISAP), for which Millstone Unit 1 and Connecticut Yankee were pilot plants. ISAP provides means to address through a comprehensive ranking methodology projects and issues from all sources. These projects and issues are evaluated according to public risk impact (PRA), reliability, economics, personnel productivity, personnel safety and external impact. Based on weighting factors for the above attributes, all proposed projects and issues are then ranked to establish a priority for implementation. In many instances, projects that have little value or low benefit-to-cost ratios are dropped altogether. Experience with the first two rounds of evaluation and project prioritization indicates that there is far more agreement than disagreement between the utility and regulators. Because of the success with ISAP on Millstone Unit 1 and Connecticut Yankee, NU is extending ISAP to Millstone Units 2 and 3.

In addition to the activities identified above, PRA input has also been used in support of day-to-day plant operations including:

- Prioritization of projects for Quality Assurance review during scheduled outages.
- Prioritization of Emergency Operating Procedures for operator training.
- Licensing support, including justification for continued operation and Technical Specification changes.
- Providing recommendations regarding preventive maintenance and surveillance frequencies for critical equipment.

Our framework of PRA application has therefore evolved from the early application for dispositioning PRA identified plant vulnerabilities only, to the current broad application to all issues for which public risk is a component. Our corporate policy commitment to reduction of Core Melt Frequency (CMF) of our plants is an obstacle to operational strategies which may increase CMF. As an example, in our ongoing transition to longer fuel cycles, we may commit to midcycle shutdowns to test PRA identified critical safety systems if compensatory actions are not identified which will allow no reduction in CMF. An additional use of our PRA program is in fulfillment of NU's IPE commitments.

"Main Results of the German Risk Study, Phase B"

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An overview is given on the investigations and main results of the German Risk Study, Phase B. As reference for the analysis Biblis B, a representative German nuclear power plant PWR of the 1300 MW-electric class was selected.

The investigations performed in Phase B only refer to plant internal analysis, i.e., accident sequences leading to plant damage states (PRA level 1), core melt accidents, containment performance and fission product release (PRA level 2) have been dealt with. Thereby, in addition to loss of coolant and transient initiating events, also contributions resulting from plant internal fire and flooding and from external events have been considered.

Referring to accident sequences not coped with by the design safety systems (accident sequences beyond safety design limits), special attention was given to the analysis of plant internal accident management measures which still can be applied to prevent core degradation or at least to mitigate its consequences. Particularly, alternative emergency measures for depressurization of the reactor coolant system and recovery of emergency core cooling, so called bleed and feed measures, have been analyzed in detail.

With regard to core melt accidents, the course of a core melt event, i.e., in-vessel core melt and ex-vessel molten core concrete interaction, as well as containment performance under various loads, have been analyzed taking into account more recent results of safety research. Thereby, especially processes and phencmena which could lead to an early failure of the containment structure have been considered.

RISK SENSITIVITY DUE TO THE CONSEQUENCE PARAMETER UNCERTAINTIES*

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As a part of the NUREG-1150 analysis, consequences were calculated using the MACCS computer code [1] for postulated releases following a severe accident at five selected power plants. Source terms associated with these accidents reflected uncertainties in the physical processes, and were calculated in a parametric form with inputs originated by a limited Latin Hypercube Sampling (LHS) Monte-Carlo technique [2]. Ranges and distributions for the relevant parameters were obtained from expert elicitation. However, source term characteristics for each type of accident had no uncertainty associated and were based on previous analyses or results of computer simulations.

During the reanalysis of the PWR plants (Surry and Zion) [3], a Steam Generator Tube Rupture (SGTR) with Secondary Valve stuck open was identified in the Level 1 assessment to be significant to the overall plant risk. An ad-hoc Level 2 analysis of this sequence showed a potential for large releases and provided some information about characteristics of the accident. Warning time, initial time of release, elevation and energy of release were left to engineering judgment, warning time being the most uncertain parameter. A warning interval of five hours was agreed upon by the analysts.

A sensitivity study was performed with respect to all the release characteristics of this accident. Distributions were sampled using the LHS technique and fifty sets of consequences were calculated by MACCS on a unique high source term from the NUREG-1150 analysis for the Zion plant.

Site specific meteorology, population and land characteristics were used, with the Zion assumption of total evacuation of 1.1 m/s following a 2.3 hour delay from operator warning [4].

A regression analysis of the results showed a strong dependency of all conditional consequences to the accident warning interval. Overall risk calculations for the Zion plant, however, do not appear to be very sensitive to the uncertainties, due to a relatively low core damage frequency, with the exception, perhaps of the risk of early health effects.

*This work was performed under the auspices of the U.S. Nuclear Regulatory Commission. This paper does not necessarily express the opinion of the USNRC.

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SAND89-1631A

PERFORMANCE OF CONTAINMENT PENETRATIONS UNDER SEVERE ACCIDENT LOADINGS[1]

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Summary

The objective of the present paper is to provide a complete summary of efforts to date to better understand the behavior of containment penetrations when subjected to severe accident conditions. The research activities discussed herein are a part of the Containment Integrity Programs, which are managed by Sandia National Laboratories for the U.S. Nuclear Regulatory Commission. Because much of the information has been widely published, a detailed presentation will be provided only for those programs that have either been recently completed or are ongoing.

The overall goal of the Containment Integrity Programs is to develop test validated methods to predict the ultimate pressure capacity, at elevated temperatures, of light water reactor (LWR) containment structures. In order to accomplish this goal, a series of scale model containment tests have been conducted including a 1:8-scale steel model and a 1:6-scale reinforced concrete model. Because of the reduced scale and limited number of tests, the model tests could not include all of the various penetration designs. Thus, several separate test programs have been conducted in which various features of containment penetrations were tested in order to determine their leakage behavior when subjected to severe accident conditions. Past containment penetration research programs have included testing of typical compression seals and gaskets, electrical penetration assemblies (EPAs), and a personnel airlock.

Recently, a series of tests were conducted to determine the leakage behavior of inflatable seals when subjected to postulated severe accident combinations of containment pressure and temperature. Also, there is an ongoing research activity to determine the capacity of bellows that are used at some containment penetrations. Further testing of the pressure-unseating equipment hatch in the 1/6-scale reinforced containment model is also underway. These three research programs will form the major portion of this paper.

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1. This work was supported by the U.S. Nuclear Regulatory Commission and performed at Sandia National Laboratories, which is operated by the U.S. Department of Energy under contract number DE-AC04-76DP00789.

Inflatable seals are used to prevent leakage around the perimeter of personnel and escape lock doors in approximately 10% of U.S. containments. The test program included the two primary seal designs currently in use in nuclear containments. For each seal design, a pair of unaged and a pair of aged seals were subjected to a series of leakage tests; thus, a total of four series of inflatable seals tests were conducted. During each test series, leakage tests were performed first at room temperature and then finally at elevated temperature. For each leakage test, the seals were first inflated to the desired level and then the chamber (containment) pressure was increased until significant leakage (≥10,000 std. ft^3 per day) began. The test results indicate that, regardless of seal design or applied aging, significant leakage will not begin until the containment pressure exceeds the normal operating seal pressure. A relatively simple, empirically based, analytical model will be presented to predict the containment pressure, for a given seal pressure and temperature, at which significant leakage can be expected.

Bellows are employed at most process piping penetrations of steel containments in order to minimize the piping loads applied to the containment shell while maintaining the containment pressure boundary. Bellows are also used at the penetration of the vent lines into the suppression chamber in Mk-I containments. Currently, plans are underway to conduct a series of tests in which representative bellows are subjected to postulated severe accident loadings. During the tests, various combinations of internal and external pressure, axial compression, elongation, and lateral deformation will be applied to typical bellows geometries. The magnitudes of deformations to be applied will be determined from global shell analyses of typical containments at the position of the most critical bellows. The goal of these tests is to develop methods to predict the pressure and deformation conditions that will likely cause a tear in the bellows, which would result in a large leak path past the containment.

A series of tests are underway on the pressure-unseating equipment hatch in the 1:6-scale reinforced concrete model. The tests are designed to provide information on the effects of bolt preload, seal aging, temperature, and type of gasket material on the containment pressure at which leakage can be expected. An analytical model has been developed to estimate the containment pressure and temperature at which onset of leakage can be expected and the rate of leakage for containment pressures beyond those required to initiate leakage. The test results will be used to improve and validate this analytical model.

Finally, the application of the penetration research programs in the overall analysis methodology for predicting the performance of LWR containment will be described.

SIZEWELL 'B' CONTAINMENT MODEL TEST

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INTRODUCTION

The United Kingdom's Central Electricity Generating Board (CEGB) has tested a 1:10th-scale model of the containment building of Sizewell B to determine its ultimate pressure carrying capability. Sizewell B is a pressurized water reactor that is housed in a prestressed-concrete containment. The design pressure used for the containment and the model is 0.345 MPa. The containment structure is based on a Bechtel design--making it very similar to some of the prestressed containments in the United States. The containment model was tested to structural failure to demonstrate its pressure reserve and provide data to benchmark computer analyses.

A total of 712 sensors were employed to monitor and record the structural behavior of the model during the hydrostatic tests. The data will be used to validate computer codes used for the design and ultimate load analyses of full-scale containment structures.

The Nuclear Regulatory Commission (NRC) is participating in this program to further their understanding of containment performance. Previous containment experimentation has been conducted at Sandia National Laboratories for the NRC, and has included the testing of five steel containments and a 1:6-scale reinforced-concrete containment building. Sandia personnel, acting as the NRC's technical agent, have been participating on the peer review group for the Sizewell B model testing program.

OBJECTIVE

The data generated during the test will be used to validate the analytical techniques used in the U.K. to assess the ultimate capability of containment structures. The data also contributes to understanding of the failure mechanisms of similar structures.

FEATURES OF THE MODEL

The model was scaled from the full-size prototype, Sizewell 'B', and is approximately 1:10 scale. The model comprises a prestressed cylindrical shell and hemispherical dome with a flat reinforced concrete base.

The model used a rubber bladder as the pressure boundary during the hydrostatic tests. A steel liner was not included in the model structure.

Some of the major features included in the 1:10-scale model are:

- a flat basemat that is 0.42 meters thick,
- a 0.13 m thick cylinder wall,
- a hemispherical dome that is 0.10 m thick,
- 1 equipment hatch,
- 2 personnel air lock penetrations,
- 3 buttresses used to anchor the hoop tendons, and
- 1 piping cluster.

TESTING

The low pressure testing of the model included pressurizing the containment structure 4 times to 1.15 times its design pressure (1.15 x 0.345 MPa) for a peak pressure of 0.397 MPa at the dome apex. Testing concluded with an overpressurization of the model, which reached .77 MPa at the dome apex before a failure of the basemat occurred.

During the low pressure tests the model behaved essentially elastic and with essentially no signs of permanent set in the structure.

The ultimate load test began with a slow steady pressurization of the model while the sensors on the containment model were scanned. The majority of the model behaved elastically until a pressure of about 0.5 to 0.6 MPa was reached. As the pressure was increased, the model began to show signs of nonlinear behavior as reported by the sensors; however, very little visual evidence, such as extensive cracking, was obvious. (The pressures stated are the applied pressure or conceptually the pressure at the dome apex. Due to the weight of the water a hydrostatic pressure also exists that varies linearly from zero at the dome apex to approximately 0.065 MPa at the top of the basemat.)

At approximately 0.77 MPa (pressure at the dome apex), the basemat began to bulge significantly and no further increase in pressure could be applied. As a result of the basemat bulging, the model tipped to one side, exposing a large portion of the bottom of the basemat. Spalled concrete was evident over a region just inside the bearing plates for the meridional tendons and also towards the center of the basemat.

The full paper will further describe these tests and present data gathered during the test.

SAND89-1998A1

ASSESSMENT OF EFFECTS OF STRUCTURAL RESPONSE ON PLANT RISK

PRELIMINARY RESULTS FOR PEACH BOTTOM ATOMIC POWER STATION

by

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Summary

The U.S. Nuclear Regulatory Commission has been sponsoring tests at Los Alamos National Laboratory on the dynamic response of Seismic Category I reinforced concrete shear wall structures. As test results accumulated, it became clear that there was a significant difference between as-calculated and measured shear wall stiffnesses and frequencies, and that these differences existed both in static and dynamic tests. For very low level tests, measured frequencies were found to range between 50% and 80% of the computed values. During simulated earthquake tests, measured frequencies were found to further decrease as the earthquake level increased. The observed differences between calculated and measured stiffnesses and frequencies represents a potentially important issue relative to the seismic design and safety of nuclear power plants, for the following reasons:

a. In the typical PWR and BWR power plant, most of the safety injection systems and piping, the emergency on-site power systems and the control room itself are located in shear wall Seismic Category I structures, and these structures have (calculated) fixed base frequencies typically in the 5 Hz to 20 Hz range.

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b. Based on the LANL tests, and depending on the level and frequency content of the earthquake time history and the local soil conditions, these structures could have effective frequency reductions of up to 50 percent, i.e., in the range of 2.5 Hz to 10 Hz.

c. Most broad band strong motion recorded earthquake accelerograms have the majority of their energy in the 2 Hz to 8 Hz range. Thus if the structure has an effective frequency in this band, the excitation of the structure would be greater than was considered in the design of the structure. Both the loads experienced by the structural members and in-structure floor acceleration spectra would be increased.

d. Since the (calculated) in-structure floor response spectra which are used to design and qualify safety equipment have been based on calculated structural stiffnesses and frequencies, it is possible that certain safety equipment could experience greater seismic loads than were specified for qualification.

Thus, this "frequency difference" issue has potentially important implications with respect to the safety of power plants during seismic events.

In order to assess the importance of this "frequency difference" issue the U.S. Nuclear Regulatory Commission has funded Sandia National Laboratories to re-evaluate several existing seismic PRAs by modeling and incorporating the effects of the frequency reductions. This report presents the results for the initial re-evaluation of the seismic risk at the Peach Bottom Atomic Power Station.

Re-evaluation of the original PRA (performed as part of the NUREG 1150 program) was performed with a preliminary model incorporating both static and dynamic reduction in stiffness for the critical concrete structures at Peach Bottom. As a result of this evaluation, the mean core damage frequency was found to increase from the original value of 7.66E-05 to 1.24E-04 per year, a 60% increase. This increase resulted primarily from two structures (the crib house and cooling towers) whose natural frequencies were reduced from 13 to 20 Hz down into the amplified acceleration region of ground motion input.

In addition, an evaluation of "design-like" structural dynamic calculations with and without the stiffness reductions was made. It was found that, in many cases, in-structure floor spectra were significantly amplified due to the stiffness reduction, and that spectral peaks were also shifted (downward) significantly. In addition, calculated net wall loads and moments were increased by up to 20%. These results have significant implications for the design and safety analysis of nuclear power plants.

THE HIGH LEVEL VIBRATION TEST PROGRAM

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SUMMARY

As part of cooperative agreements between the United States and Japan, tests have been performed on the seismic vibration table at the Tadotsu Engineering Laboratory of Nuclear Power Engineering Test Center (NUPEC) in Japan. The objective of the test program was to use the NUPEC vibration table to drive large diameter nuclear power piping to substantial plastic strain with an earthquake excitation and to compare the results with state-of-the-art analysis of the problem. The test model was subjected to a maximum acceleration well beyond what nuclear power plants are designed to withstand.

The High Level Vibration Test (HLVT) model was constructed by modifying the 1/2.5 scale model of one loop of a PWR primary coolant system which was previously tested by NUPEC as part of their seismic proving test program. The upper and middle steam generator shell supports of the model, which simulated the actual plant condition, were removed and the steam generator shell was truncated. Furthermore, the four lower support columns for the steam generator were replaced by a pin-type support. These modifications shifted the natural frequency of the test model into the frequency range where the vibration table has maximum exciting capacity and provided assurance that inelastic response could be achieved.

A total of 300 channels of instrumentation, including accelerations, displacement and more than 200 channels of strain, were provided to measure the inelastic vibration behavior of the test model, especially the elastic-plastic strain distribution in the piping.

A modified earthquake excitation was applied and the excitation level was increased carefully to minimize the cumulative fatigue damage due to the intermediate level excitations. Since the piping was pressurized, and the high level earthquake excitation was repeated several times, it was possible to investigate the effects of ratchetting and fatigue as well.

Elastic and inelastic seismic response behavior of the test model was measured in a number of test runs with an increasing excitation input level up to the limit of the vibration table. In the maximum input condition, large dynamic plastic strains were obtained in the piping. Crack initiation was detected following the second maximum excitation run. Crack growth was carefully monitored during the next two additional maximum excitation runs. The final test resulted in a maximum crack depth of approximately 94% of the wall thickness.

The HLVT program has enhanced understanding of the behavior of piping systems under severe earthquake loading. As in other tests to failure of piping components, it has demonstrated significant seismic margin in nuclear power plant piping. The test provided extensive data which are being used to evaluate elastic and inelastic dynamic analysis techniques. Furthermore, it provided unique test data to understand fatigue crack initiation and growth under seismic loading conditions. Efforts are continuing to evaluate the test results and to perform more refined post-test analysis. A blind post-test prediction program involving engineers not previously involved with the program is also being performed.

* Research has been performed under the auspices of the U.S. Nuclear Regulatory Commission.

RELAY TESTING AT BNL

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SUMMARY

INTRODUCTION

Brookhaven National Laboratory (BNL) has conducted a seismic test program on relays. The testing has been performed at Wyle Laboratories. The purpose of the test program was to investigate the influence of various parameters (e.g. input frequency, electrical mode, adjustments) on the seismic capacity levels of relays. A total of forty six specimens of nineteen popular relay models have been selected for testing.

TESTING METHODS

The bulk of the program consisted of single axis, single frequency sine dwell tests in the frequency range of 1-50 Hz at an interval of 2.5 Hz. The input level was progressively increased or decreased until the failure threshold was established in all three orthogonal directions, for three electrical modes (i.e. operating, nonoperating and transition), and two contact conditions (i.e. normally open and normally closed). In some tests, the specimens did not fail within the vibration capacity of the shake table. A contact chatter duration of 2 ms or greater has been used as the failure criterion. Limited single frequency testing was performed to determine the influence of adjustment of spring tension and contact gap with hinged armature relays. The effect of end play was investigated by raising and lowering the disk of rotary disk relays.

Subsequent to the single frequency tests, multifrequency testing was performed on a limited number of specimens with the random input. The specimens were excited in one, two and three directions in separate tests, and simultaneously, in case of multiaxis testing. Spectral shapes have been matched, as much as possible within the shake table limitation, with the respective single frequency capacity levels so that the conversion factors relating the single frequency test inputs to the multifrequency test response spectra can be computed. Three specimens of each of four models of the same relay type were tested biaxially following the shape of the response spectrum recommended in ANSI/IEEE Std C37.98. This test explored the similarity concept for relays. The multifrequency and the adjustment tests were performed only for the worst electrical condition and input direction determined from the single frequency testing.

TEST RESULTS

The single frequency test results provide the seismic capacity of each specimen at various frequencies. The results indicate that most relays are strongly influenced by the input frequencies and directions, electrical modes and contact states. The ratio of the capacity level test response spectrum (TRS) and the sine dwell input of a relay model varied from 2.1 to 4.5 in the frequency range of 5-30 Hz. The average value was 2.3 in the frequency range of 5-15 Hz. Test results of the four models of the same relay type did not demonstrate close dynamic similarity of these models. Importantly, for two models, the operating mode controlled the relay capacity levels.

CONCLUSIONS

The test results depict characteristics of the relays. The correlation between the capacity level multifrequency TRS and the corresponding single frequency input obtained from the test program will be an effective tool to predict the capacity level TRS for earlier vintage relays for which only single frequency test data exist. Further testing is recommended to confirm the effect of successive short duration contact chatters in tripping a device.

* Research has been performed under the auspices of the U.S. Nuclear Regulatory Commission.

SEISMIC MARGIN ASSESSMENT OF HATCH NUCLEAR POWER PLANT

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INTRODUCTION: The Electric Power Research Institute (EPRI) is sponsoring a Seismic Margin Assessment (SMA) of Plant Hatch Unit 1 for the purpose of assessing the practicality of the recently developed seismic margin methodology. Plant Hatch is a General Electric boiling water reactor (BWR) - four with a Mark 1 Containment located on a soil site.

The Nuclear Regulatory Commission (NRC) and EPRI have developed similar approaches to performing an SMA. The NRC performed a trial SMA of the Maine Yankee Atomic Power Station and EPRI performed a trial SMA of the Catawba Unit 2 Power Plant; both are PWR's on rock sites. The NRC has reviewed the EPRI SMA methodology and finds it an acceptable approach. Both the NRC and EPRI have joined together to evaluate the SMA methodology for a BWR plant on a soil site. Georgia Power Company (GPC) agreed to use Plant Hatch Unit 1 as this trial BWR. The NRC is participating in the program in an independent review capacity under the direction of D. J. Guzy of the NRC Research Branch with an NRC peer review group. The NRC is also funding a separate study utilizing the event tree/fault tree modeling of front line and support systems. The EPRI SMA methodology applies the success path modeling approach.

The primary purpose of the SMA is to demonstrate sufficient margin over the Design Basis Earthquake (DBE) and identify the most seismically vulnerable components in the prescribed paths for safe shutdown. GPC/SCS is also performing portions of the USI A-46 review of Plant Hatch Unit 1 in conjunction with the SMA since both programs have so many common steps. SMA provides a practical approach to address the safety significance of a higher seismic hazard than the plant was originally designed to withstand. Work is also proceeding at EPRI to condition the EPRI SMA methodology for use in the resolution of the Severe Accident Policy for seismic.

GENERAL DISCUSSION: The fourth approach given in the EPRI methodology was used to select the Hatch Seismic Margin Earthquake (SME). The SME has a peak ground acceleration (PGA) of 0.3g and a spectra shape based on NUREG/CR-0098 median centered spectrum. The Hatch DBE is a Housner type spectrum with a PGA of 0.15g. If the probabilistic seismic hazard curves had been available at the start of the assessment they would have been used in the selection of the SME. However, a comparison of the recently published EPRI seismic hazard curves for Plant Hatch shows that the chosen SME spectrum envelopes, at all frequencies, the median uniform hazard spectrum at 10^{-5} probability of exceedance.

Since Hatch is a soil site, liquefaction potential due the SME was evaluated with a resulting high-confidence-low-probability-of-failure (HCLPF) level of 0.28g. The associated settlement was found not to be detrimental. Critical slopes were also evaluated and no serious instabilities were found due to the SME.

A new soil-structure interaction (SSI) analysis was performed to accurately predict the median centered structural and equipment response due to the SME. New soil profiles were developed and varied to account for uncertainty. The original building seismic models were revised as necessary to improve the accuracy of the predicted responses.

Two independent functional paths which can achieve and maintain a safe shutdown condition following an SME were chosen. These success paths, along with their associated components, fulfill the requirements of both SMA and USI A-46. For Plant Hatch either program would produce almost identical lists. The final list consisted of 344 items.

An evaluation of relay chatter was performed for all relays required for the success path components by either evaluating the effects of relay chatter or establishing the seismic adequacy of the relay. The evaluation included 4358 combinations of relay/component actions for approximately 1000 relays. Approximately 92 percent were screened out as chatter acceptable, operator action required, or actuator is immune to chatter. Five percent have been screened out using the relay generic equipment ruggedness spectra (GERS) and one percent was screened out using existing seismic qualification reports. The remaining two percent are still under evaluation.

The seismic capability walkdown was performed by two Seismic Review Teams (SRT). The teams used the SQUG GIP walkdown forms with an additional sheet added to address other items such as the relay walkdown and seismic interaction for flooding. A total of 344 items were evaluated during three separate walkdowns. Two hundred sixty three (263) items have been seismically verified at a HCLPF level of at least 0.3g. Modifications are required to obtain a HCLPF level of 0.3g for 62 items while 19 items still require a walkdown or further evaluation.

SUMMARY: There are several observations that can be made concerning the Plant Hatch SMA. The evaluation of Plant Hatch soil conditions (i.e., soil liquefaction, slope stability, soil profiles, and the SSI analysis) and the relay evaluation are very significant parts of the SMA effort. The SSI analysis accounted for most of the margin over the plant DBE. This occurred due to the excessive conservatism in the original plant SSI. A majority of the walkdown effort involved the evaluation and inspection of equipment anchorage. The anchorage concerns that were identified were due primarily to poor installation. Combining portions of USI A-46 with SMA was a very cost effective way to perform both programs at Plant Hatch. It is concluded that the EPRI SMA methodology provides a practical, cost effective method to identify seismic margin up to the limits of the earthquake experience data base.

Fuel-Coolant Interactions and Vapor Explosions

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An explosion involves the rapid conversion of energy from one form to another. Before the explosion is initiated, the energy must be stored in a form that exists for some time without significant dissipation of available energy or conversion to other forms of energy, i.e., a metastable state. A vapor explosion is such a process in which hot liquid (fuel) transfers its internal energy to a colder, more volatile liquid (coolant); in doing so the coolant vaporizes at high pressures and expands, doing work on it surroundings.

To be more precise the vapor explosion can be considered as a subset of a fuel-coolant interaction in which the timescale for heat transfer between the liquids is smaller than the timescale for pressure wave propagation and expansion in a local region of the fuel-coolant mixture. Therefore, the rise in pressure locally forms a shock wave, which spatially propagates with a velocity which is greater than the characteristic speed of sound in the mixture ahead of the shock front (Mach No. > 1). A significant fraction of the thermal energy initially stored in the fuel could be transferred to the coolant as the fuel is fragmented. The key feature of the vapor explosion is that the shock wave propagation through the mixture directly contributes to the rapid fuel fragmentation and associated heat transfer to the coolant; i.e. analogous to shock heating in a chemical detonation.

A "non-vapor explosion" is a fuel coolant interaction which does not exhibit these shock wave characteristics. Thus fuel fragmentation is not necessarily linked to shock wave propagation and the rapid boiling phenomena does not spatially propagate on a timescale equal to pressure wave propagation. A large amount of coolant vapor may be produced in this process and the fuel may still become finely fragmented, producing significant quantities of hydrogen if the fuel is metallic. Yet the character of the fuel-coolant interaction is not explosive. One should note that analogous to a deflagration such as event might still be destructive under certain conditions.

A recent comprehensive review of this physical process (1) details the current state of knowledge. In this paper and presentation our focus is the risk significance of FCI's and a case study of its containment loading implications beyond alpha-mode failure.

A comprehensive risk assessment effort in WASH-1400 was the first to estimate the likelihood of containment failure by a vapor explosion. For the vapor explosion process it was determined that the containment could be threatened by three possible damage mechanisms: (1) dynamic liquid phase pressures on structure, (2) static overpressurization of the containment by steam production, and (3) a solid missile generated from the impact of a liquid slug accelerated by the vapor explosion. Rapid hydrogen was not considered due to large estimates of in-vessel oxidation. Based on analyses it was determined that one of the major concerns from the FCI was a direct failure of the containment caused by missile generation from a vapor explosion (designated "alpha-mode" failure). This might occur when a vapor explosion occurs in the lower plenum of the reactor vessel and the surrounding water and/or fuel are accelerated as a slug to impact the reactor vessel head generating a solid missile. WASH-1400 estimated the conditional probability alpha-mode failure (given a complete core melt) to be 10^{-2}/reactor-yr. Since the accident at Three Mile Island, a number of investigations have reexamined this phenomenon and estimated the probability of its occurrence given a core meltdown accident to be equal to or substantially less that the WASH-1400 estimate (e.g., SERG estimated the upper bound to be 10^{-2} with a mean of 10^{-3}-10^{-4}).

Given such an estimated, one must consider other containment loadings that might be a threat from an FCI; e.g., ex-vessel interactions causing dynamic or static pressure loads or hydrogen production. The reasoning is that molten fuel could contact water in a number of specific locations in the containment ex-vessel. Therefore, one must be cognizant of the particular reactor-containment geometry and any particular vulnerabilities. In this presentation we will also focus on these topics with a few particular illustrations of FCI related containment loading:

(1) Fuel-coolant mixing and possible limits under various circumstances

(2) Hydrogen generation from FCI's regardless of the explosivity

(3) Fuel quenching in the presence of the water coolant.

These examples will include examination of some specific reactor geometries to illustrate these loadings.

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Status of Cooperative Efforts on the VICTORIA Code for Fission Product Release and Transport

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Modeling of the release of radionuclides from reactor fuel and the subsequent behavior of these species is an essential aspect of severe reactor accident analysis. Previous models for in-vessel fission product release and transport, such as CORSOR and TRAPMELT, were relatively simple and do not account for many effects now known from recent experiments. For example, releases predicted by CORSOR depend only on temperature and do not account for oxidizing or reducing environments that can affect some species by an order of magnitude. TRAPMELT likewise overlooks many chemical effects and is thus unable to handle revaporization, which depends on surface reactions. Further, CORSOR and TRAPMELT are not integrated, and mass transport effects that can affect releases by factors of 2 or more are lost. In addition, releases from some stages of core damage, where fuel has lost its normal geometry, been reduced to rubble, or melted, are not modeled at all

To take advantage of recent experimental information and to eliminate many weaknesses in current models, development of the VICTORIA code was initiated at Sandia National Laboratories. Table 1 shows the elements of release, transport,

Table 1. Model elements considered for incorporation in the VICTORIA code.

General Areas and Specific Models
Radionuclide Release from Fuel
Intact Rods
Rubble
Melt
Quenching
Oxidation
Transport and Deposition in the RCS
Non-Radioactive Sources
Aerosol Physics
Vapor-Surface Interactions
Vapor-Aerosol Interactions
Revaporization
Evolution of Chemical Forms
Re-Entrainment

and revaporization that are being considered in the VICTORIA code.

Each element is being evaluated with regard to complexity and importance to determine if and how it will be modeled in the code. The status of modeling in each of these areas will be presented.

Within the last 12 months, heightened interest in the VICTORIA code has been expressed, particularly abroad. Consequently, the NRC has made arrangements for the accelerated development of VICTORIA and is modifying its contents to address current interests. In doing this we have brought together several modeling groups with outstanding experience in this field to work cooperatively on the development and validation of VICTORIA. Table 2 identifies the

Table 2. Modeling groups working cooperatively on the development and validation of VICTORIA.

Principal Staff	Organization	Area of Modeling
R. O. Meyer T. J. Walker	NRC	Overall coordination
A. J. Grimley D. A. Powers	Sandia (NRC)	Code Custodians, Architecture, Chemical Kinetics
D. A. Williams N. Johns N. Chown	Winfrith (UKAEA)	Aerosols, Chemistry
F. C. Iglesias	Chalk River (AECL)	Oxidation Releases
J. Rest	Argonne (NRC)	Release from degraded fuel geometries
A. L. Wright	Oak Ridge (NRC)	Aerosols, Chemistry

principal members of these groups and the area where they will provide modeling input for VICTORIA. An intensive modeling effort is now underway to produce a first releasable version of VICTORIA by the end of 1989. The code is being validated by comparison to a large number of experimental results including radionuclide release tests (FLHT, ST, HI, VI) and transport tests (LACE and MARVIKEN).

As of fall 1989, the VICTORIA code is operating at all of the laboratories indicated in Table 2, but no general release of the code has been made. Documentation is being prepared, and the first general release of the VICTORIA code will be made in early 1990. Foreign distribution will, however, be limited at that time to those organizations that have shared in the development costs through their cooperative agreements with the NRC.

SAND89-1880A

The MELCOR Code System*

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and

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MELCOR is a computer code system that is being developed to calculate fission product source terms and phenomenological behavior associated with severe nuclear reactor accidents. The primary role of MELCOR is to support risk assessment studies. As such, it is designed to treat all relevant phenomenological regimes in an integrated framework in order to provide coupling between competing physical processes. Both in-vessel and ex-vessel phenomena are simulated as are the responses of the various engineered safety systems. Consequently, accident sequences can be simulated from accident initiation through containment failure and release to the environment.

The modeling approach in MELCOR can be characterized as engineering-level. This means that models are selected to represent the key phenomena at a level that is justified based on the available data. Hence, simplified (or parametric) modeling is used extensively where large phenomenological uncertainty exists. This approach allows complete, integrated calculations to be performed at reasonable cost and also allows for extensive sensitivity/uncertainty analysis to be performed.

MELCOR is the next generation NRC source term analysis code. It has been built upon the work of the Source Term Code Package, STCP, which it supersedes. Certain models in the STCP (e.g., CORCON) have been incorporated into MELCOR. In addition, models from CONTAIN that are appropriate for MELCOR have also been included. As work progresses in other areas, additional models developed in other NRC programs will be incorporated.

MELCOR has been completed to the point that detailed documentation, assessment, and application of the code are warranted. These activities provide a cost-effective means of improving the code. Currently, assessment work is being performed at Sandia, Oak Ridge, and Brookhaven. The activities

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at Oak Ridge and Brookhaven have been directed toward core melt progression modeling, while those at Sandia have mainly been directed toward containment phenomena. The notable exception at Sandia has been the analysis of the TMI-2 accident.

The Brookhaven work has shown that MELCOR is capable of simulating the PBF SFD 1-1 test. While there were many sources of uncertainty introduced during the performance of the test, the calculated results showed good overall agreement with test data and with the NRC's severe accident mechanistic code SCDAP.

In addition to the assessment activity, MELCOR is being applied in a number of risk assessment studies. Oak Ridge is using MELCOR as part of their analysis of the Mark III containment improvement work. Sandia is using MELCOR in the LaSalle level 3 PRA work. Other applications of MELCOR have been initiated related to risk assessment studies of DOE reactors.

MELCOR is emerging as a useful and versatile tool to support risk assessment studies. The primary areas of emphasis in the project are on assessing and documenting the code and incorporating the most important modeling into the code.

STUDIES OF CORE DEBRIS INTERACTIONS WITH CONCRETE

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The interactions of core debris with structural concrete are classic issues of severe reactor accident analysis. The U.S. Nuclear Regulatory Commission (USNRC) has sponsored experimental and analytical investigations of these interactions for several years. The findings from these studies have been augmented by the complementary BETA test program at the Kernforschungszentrum Karlsruhe in West Germany and more recently by an on-going test program conducted by the ACE consortium conducted at Argonne National Laboratory.

The USNRC-sponsored studies have led to the development of the CORCON model of core debris interactions with concrete and the VANESA model of aerosol generation and radionuclide release associated with these interactions. Developments of a more prototypic data base, validation of these computer codes and resolution of the remaining issues have been the foci of the USNRC program over the last two years. To achieve these objectives, the USNRC has sponsored a program of integral core debris/concrete interactions tests at Sandia National Laboratories, programs of separate effects and simulant tests at both Sandia National Laboratories and at Brookhaven National Laboratory, and a program of fundamental chemistry research at Battelle Columbus Laboratory. The USNRC is also a member of the ACE consortium cited above.

The data base available for development of the CORCON and VANESA computer models in the past has, consisted predominantly of steel melts interacting with various types of concrete. More recent experiments to validate these models, which are used for containment integrity analyses and risk evaluations, have been directed toward using more nearly prototypic simulants for core debris. Some of the recent developments in the model validation effort are:

- o The SURC-3 and SURC-4 tests have confirmed the predicted effects of unoxidized zirconium in core debris on debris interactions with concrete and demonstrated the need to consider condensed phase chemical reactions of zirconium in models of the interaction. Modifications to CORCON have been made to reflect these results.

- o The sustained, large-scale interactions of UO_2-ZrO_2-Zr melts with concrete in the SURC-1 test demonstrated that heat transfer models developed from data for metal melt/concrete interactions are applicable to oxide melt/concrete interactions.

- o Data from the WITCH-5 tests were used to validate model of gas holdup in core debris that is used in the calculation of both heat transfer and aerosol generation.

- Data from separate effects tests have been used to formulate a model of interphase mass and heat transport in core debris interacting with concrete.

- Mass spectroscopic studies have shown UO_2-ZrO_2-Zr mixtures are very non-ideal.

- CORCON and VANESA have been integrated into a single code.

Currently, it is believed that most of the issues of core debris interactions with concrete have been or are being resolved. A critical issue that remains outstanding is the effect of water will have on these interactions. Initial examinations of the phenomena of simultaneous core debris/concrete/coolant interactions have been conducted with high temperature materials at Sandia National Laboratories (SWISS and FRAG tests) and simulant materials at Brookhaven National Laboratory and the Kernforschungszentrum Karlsruhe. In these tests, water pools did attenuate substantially aerosol generation associated with core debris/concrete interactions. Non-condensible gas generation also enhanced the boiling heat flux from core debris to the coolant. Further, larger scale, tests of core debris/concrete coolant interactions are planned as part of the SURC test program at Sandia National Laboratories the MACE program at ANL and the BETA test program at the Kernforschungszentrum Karlsruhe to identify the limits of core debris coolability with water.

Interests are now turning toward application of the technology developed for predicting core debris/concrete interactions to issues of specific reactor systems. The most immediate issues are the spreading of core debris in the drywell of the Mark I boiling water reactor and attack by the debris on the containment shell of this reactor. Tests with simulant materials have been used to develop correlations of the ability of core debris to spread. Separate effects tests of the heat transfer from core debris attacking concrete to steel structures have been completed. Further large-scale tests of the spreading and liner attack by prototypic melts are now being considered.

Melt Attack and Coolability Experiments (MACE) Program Elements and the Scoping Test

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SUMMARY

The most important issue in the ex-vessel progression of the severe accidents is whether sufficient energy (heat) can be removed from the discharged melt that the threat to containment integrity is avoided. This issue is pertinent for both the initial and the longer term (hours) interactions of the discharged melt with the BWR and PWR containments. For example, if the superheat of the melt discharged in the Mark-I drywell is removed during the initial interaction, the spreading of the melt could be stopped and the melt attack on the liner precluded(1). Similarly, for high pressure melt ejection (HPME) in a PWR containment cavity, effective heat removal in the initial melt-structure(2) and/or melt-coolant interactions(3) greatly reduces the direct containment heating loads.

Perhaps the most efficient and commonly available agent for removing heat from the discharged melt in the PWR and BWR containments is water. It has a relatively large latent heat for the phase change and has the ability to serve as a heat transport medium through boiling at one location in the containment and condensing at another location (e.g., on structures, in fan coolers, in suppression pools). Also containments, in general, have engineered safeguard systems for delivery of water through sprays into the various containment regions.

The initial/short-term coolability of the melt discharged from the vessel depends upon the containment configuration and the placement of water. Melt-concrete interactions do not impact short-term coolability. Dynamic melt-water interaction may occur and large heat removal rates may be achieved(4). Since the initial/short-term coolability of the melt in the PWR and BWR containments is so very much initial-condition dependent and plant specific, it is not the subject of the Melt Attack and Coolability Experiments (MACE) program.

The MACE program addresses the generic question of the coolability of the discharged melt spread out to a part of the basemat floor under the reactor vessel. This situation results after the initial/short-term interactions of the melt with water and/or structures found in the BWR and PWR containments. The key issue then is whether the melt, if flooded with water, will quench and remain coolable, so that molten corium concrete interaction (MCCI) is avoided and containment integrity maintained.

Previously, Theofanous and Saito(5) and Greene(6) performed simulant material experiments investigating heat transfer from melt surface to a coolant. They found large increases in heat transfer when gas injections representing a MCCI

were maintained. Sandia National Laboratories (SNL) performed FRAG[7] and SWISS[8] tests using heated steel balls and steel melt, respectively, reacting with concrete slabs and cooled by a water overlayer. Crusts were formed at the top surface in these tests which precluded com-plete quenching. These were, however, small-scale (22 cm. diameter) tests.

The MACE program is designed to extend and unify the other existing research programs by investigating, at a reasonably large scale, the coolability of molten corium, in interaction with concrete, by a water overlayer. Specifically, the MACE tests will:

a) accommodate the possible initial dynamic interaction (i.e., steam explosions) and measure the heat transferred during that process,
b) employ prototypic corium materials (including UO_2) in integral tests,
c) pursue the questions of stability and strength of prototypic material crusts, in reasonable scale, and with gas input from the melt concrete interactions,
d) directly determine coolability and quenchability in both large and slow heat removal phases,
e) investigate the time taken for quenching the melt with water.

The other technical issues investigated will include (1) effect of water layer or aerosol source, (2) hydrogen generation, and (3) scaling.

Observations from a scoping experiment, employing approximately 150 Kg of molten corium containing approximately 7 Kg of Zirconium reacting with limestone-sand concrete and cooled by a water overlayer, performed at Argonne National Laboratory (ANL) in August 1989 will be presented. The MACE program elements and test matrix will also be described.

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ASSESSEMENT OF EX-VESSEL STEAM PRESSURE SPIKES
IN BWR MARK II CONTAINMENTS

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This paper addresses the pressure response of a Mark II containment from rapid steam formation during postulated severe accidents. For example, the flow of hot corium from a breached reactor vessel into water can lead to rapid steam generation. The resulting maximum pressure depends on the rate and duration of steam formation, the containment volume, and the relief capacity of venting flow paths.

The rate of corium discharge largely determines its progression in the containment and the rate of steam formation when contact with water occurs. If the corium enters a dry region, there is no immediate steam pressure spike. If corium spreads on a floor, arrival at vertical downcomers can provide a flow path to pool water below.

Although there are numerous postulated scenarios which could yield a steam pressure spike, phenomena which determine the corium progression and pressure-time response include:

1. Corium discharge rate, composition, and properties;
2. Corium progression through containment regions;
3. Corium arrival, attack, and flow through downcomers;
4. The configuration geometry of corium entering water;
5. The heat transfer rate from corium;
6. The rate of steam formation from hot corium;
7. Steam introduction, pressurization, and venting in various containment regions;
8. Water inertial response to steam formation.

The objective of this paper is to provide first order analyses which help display how various parameters affect both the corium progression and magnitude of steam pressure spikes from postulated severe accidents. State-of-the-art corium discharge computations are used to estimate the pressure response of a typical Mark II containment. Results of the analysis indicate that steam pressure spikes are not large enough to cause containment failure.

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11. ABSTRACT (200 words or less)

This report contains summaries of papers on reactor safety research to be presented at the 17th Water Reactor Safety Information Meeting at the Holiday Inn Crowne Plaza in Rockville, Maryland, October 23-25, 1989. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, USNRC. Summaries of invited papers concerning nuclear safety issues from the electric utilities, the Electric Power Research Institute (EPRI), the nuclear industry, and from the governments and industry in Europe and Japan 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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Mon AM	Accident Management **Session 1**	Primary System Integrity **Session 2**	
Mon PM	Plant Performance, Testing & Analysis **Session 3**	Piping & NDE **Session 4**	Equipment Qualification of Valves **Session 5**
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Tues PM	Plant Aging **Session 9**	Organization & Reliability Research **Session 10**	
Wed AM	Severe Accident Research **Session 11**	Earth Sciences **Session 12**	**ROCKVILLE**
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