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Transactions of the Eighteenth Water Reactor Safety Information Meeting

To Be Held at
Holiday Inn Crowne Plaza
Rockville, Maryland
October 22–24, 1990

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PREFACE

This report contains summaries of papers on reactor safety research to be presented at the 18th Water Reactor Safety Information Meeting at the Holiday Inn Crowne Plaza in Rockville, Maryland, October 22-24, 1990. 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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D. A. Petti
D. J. Osetek
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SUMMARY

Results from integral effects core melt progression experiments and from the examination of the damaged core of the Three Mile Island Unit 2 (TMI-2) reactor are reviewed to gain insight on key severe accident phenomena. The experiments and the TMI-2 accident represent a wide variety of conditions and physical scales, yet, several important phenomena appear to be common to core melt progression. Eutectic interactions between core materials cause the formation of liquids and loss of original core geometry at low temperatures (~1500 K) in a severe accident. The first liquids to form are metallic in nature and relocate to lower elevations in the core where they may freeze into a crust that forms a partial flow blockage. At temperatures above ~2200 K, fuel liquefaction causes the accumulation of fuel-bearing debris in the core above the metallic lower crust. The liquefied material oxidizes in steam as it relocates and the accumulated melt can incorporate unmelted fuel rod debris. The result is the formation of a molten ceramic pool above the metallic crust. This molten pool can be uncoolable, as was the case in TMI-2 accident, but failure of the peripheral crust can cause a coherent relocation of core melt to the lower plenum of the reactor, and fragmentation of the melt in water to form a coolable debris (as occurred in the TMI-2 accident).

Fission product release early in a severe accident is controlled by diffusion through solid fuel and is strongly influenced by microsructural features such as cracks and grain boundary porosity interlinkage. Cracking due to rapid cooling (e.g. during reflooding) can enhance fission product release as can liquefaction. Fission product release from the molten pool is controlled by bubble dynamics and the oxygen potential within the pool. Some inventory of volatile, among other, fission products remain in the melt, even after relocation to the lower plenum.

RESULTS of the ACRR DF-4 BWR CONTROL BLADE-CHANNEL BOX TEST

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ABSTRACT

The DF-4 in-pile fuel damage experiment addressed the behavior of boiling water reactor (BWR) fuel canisters and control blades in the high temperature environment of an unrecovered loss of coolant accident. This experiment, which was carried out in the Annular Core Research Reactor (ACRR) at Sandia National Laboratories, was performed under the USNRC's internationally sponsored severe fuel damage (SFD) program. The DF-4 test is described in some detail herein and results from the experiment are presented. Important findings from the DF-4 test include the low temperature melting of the stainless steel control blade caused by reaction with the B_4C, and the subsequent low temperature attack of the Zr-4 channel box by the relocating molten blade components. Hydrogen generation was found to continue throughout the experiment, diminishing slightly following the relocation of molten oxidizing zircaloy to the lower extreme of the test bundle. A large blockage formed from this material continued to oxidize while steam was being fed into the test bundle. The results of this test have provided information on the initial stages of core melt progression in BWR geometry involving the heatup and cladding oxidation stages of a severe accident and terminating at the point of melting and relocation of the metallic core components. The information is useful in modelling melt progression in BWR core geometry, and provides engineering insight into the key phenomena controlling these processes .

EXPERIMENT-SPECIFIC ANALYSES IN SUPPORT OF CODE DEVELOPMENT

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Experiment-specific codes have been developed since 1986 by Oak Ridge National Laboratory Boiling Water Reactor (BWR) severe accident analysis programs for the purpose of BWR experimental planning and optimum interpretation of experimental results. These experiment-specific models (or codes) have been applied to large integral tests (ergo, experiments) which start from an initial undamaged core state. The tests performed to date in BWR geometry have had significantly different-from-prototypic boundary and experimental conditions because of either facility limitations and/or experimental difficulties. These experiments (ACRR: DF-4, NRU: FLHT-6, and CORA) were designed to obtain specific phenomenological information such as the degradation and interaction of prototypic components and the effects on melt progression of control-blade materials and channel boxes.

Design and interpretation of these experiments and the subsequent development of models for the phenomena of interest require accurate detailed quantitative representation of the conditions in the experiment. That is, the structural components, boundary and experimental conditions, and thermal hydraulics must be represented accurately. The detailed modeling required for the experimental analysis may not be available (or may only be approximated) in the severe accident analysis codes that represent the entire BWR core.

The experiment-specific codes supplement and support the "whole-core" accident analysis codes. They allow the analyst to accurately quantify the observed experimental phenomena and to reduce the effect of known uncertainties. They provide a basis for the efficient development of new models for phenomena that are currently not modeled (such as material interactions). They can provide validated phenomenological models (from the results of the experiments) that may be incorporated in the "whole-core" codes.

Experiment-specific codes are important in the design and interpretation of individual experiments, incorporating the facility limitations, experiment configurations, and experimental boundary conditions necessary for detailed experiment analysis. Application of the codes specific to the ACRR DF-4 and KfK CORA-16 experiments will be discussed and significant findings from the experimental analyses will be presented.

Heavy Section Steel Technology Program Overview*

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SUMMARY

Technology required for the accurate assessment of fracture prevention margins in commercial nuclear reactor pressure vessels has been under continuous development for the past 25 years. The Nuclear Regulatory Commission (NRC) funded Heavy-Section Steel Technology (HSST) Program was created to act as a focus for this essential safety-related development activity. The HSST program, which is managed by Oak Ridge National Laboratory (ORNL), integrates input from a broad spectrum of national and international fracture mechanics research organizations to achieve the program objectives.

Early emphasis in the HSST program was on development of the fracture mechanics methodology and the associated materials fracture toughness database. These developments were required to support the generation of regulatory instruments and national consensus Codes and Standards for the analysis and control of fracture prevention margins in commercial reactor pressure vessels. This was followed by a phase of fracture technology validation tests with an emphasis on large-scale fracture mechanics tests using test articles fabricated from prototypical heavy-section reactor pressure vessel (RPV) steels. Results from these tests, and similar tests conducted by a number of overseas fracture mechanics research organizations, identified a number of issues with the existing RPV fracture prevention technology that need further study.

In a parallel development, RPV surveillance program data from commercial nuclear power plants were analyzed to identify areas in which a fully validated fracture technology was most urgently required to support continued operation for the duration of their current licensing periods. Plant-license extension studies provided further identification of areas in which additional fracture mechanics technology development was required. Data from materials irradiated in research reactors provided additional identification of fracture issues with near-term licensing significance.

Emphasis in the current phase of the HSST program is on the resolution of these high-priority fracture technology issues. This paper presents an overview of some of the fracture mechanics research activities currently in progress within the HSST program at ORNL and at associated universities and research laboratories.

Transferability of fracture toughness data generated using laboratory-scale test specimens for application to large-scale reactor vessel structures is the subject of a major segment of the ongoing research program. Concern for the transferability of fracture toughness data from small-scale specimens to large-scale applications was brought into sharp focus by the inconsistent crack-initiation toughness results obtained in the large-scale

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fracture toughness tests performed on the National Institute of Standards and Technology 27-MN test machine. Two parallel research activities within the ongoing program address this problem. One seeks to develop additional fracture toughness correlation parameters which can be used to improve the transferability of data. The other uses a micro-mechanical model to predict the onset of fracture based on the stress-state-dependent material ductility limits. A near-term application planned for the products of this research is the development of technology for the assessment of fracture prevention margins in circumferential welds where tensile strains exist parallel to the crack front.

Low-upper-shelf (LUS) material exists in a number of reactor vessels in service within the U.S. The need to perform both deterministic and probabilistic safety assessments of these vessels has prompted a critical review of the technology used to predict ductile tearing behavior in LUS material. An international workshop on this subject confirmed the previously suspected deficiencies with the technology in its present form. Research addressing this problem is ongoing in the U.S., Europe, and Japan. Within the HSST program the primary research efforts relating to this topic are (a) development of procedures to improve extrapolations of the J-R curve, (b) analysis of large-scale fracture experiments to better understand the fracture technology performance problems, (c) support of the ASME Section XI Working Group on Flaw Evaluation in their effort to develop criteria for the evaluation of fracture prevention margins in reactor vessels where the upper-shelf Charpy energy has fallen below 68 J (50 ft./lb), and (d) assessment of the impact of ductile tearing on the stability of cracks during a pressurized-thermal-shock event. Included in this latter assessment is an evaluation of the effects of an extended crack-arrest toughness curve. Demonstration of extended crack arrest behavior in pressure vessel steels is a prior product of the HSST program.

Shallow cracks have been shown to have enhanced fracture toughness properties due to the relief of crack-tip constraint resulting from the proximity of a free surface to the crack tip. This effect is of importance in the analysis of pressurized-thermal-shock events since a majority of the predicted crack initiations are associated with shallow cracks. Preliminary analyses conducted within the HSST program have shown that enhanced shallow-flaw fracture toughness can have a significant impact on the outcome of a probabilistic reactor vessel PTS analysis. Preparation for fracture toughness testing of shallow flaws in prototypical reactor vessel material is proceeding with a target of initiating testing in FY 1991.

NRC has under evaluation the possible use of the K_{Ic} fracture toughness curve in place of the K_{IR} curve in fracture margin assessments required by Appendix G of 10 CFR 50. In support of this NRC evaluation, a high priority activity has been initiated within the HSST program to assess the adequacy of the K_{Ic} curve as a lower-bound, crack-initiation curve. Fracture phenomena included in this ongoing evaluation include pop-ins, low-toughness sites, and crack initiation from an arrested cleavage crack.

The final element of the HSST program to be included in this review concerns the effect of stainless steel cladding on the fracture behavior of surface and near-surface flaws in a nuclear pressure vessel. Data from materials testing programs have shown cladding to have a low-tearing toughness in both the irradiated and unirradiated conditions. An ongoing element of the program seeks to evaluate the effect of low clad-tearing toughness on the initiation and propagation of cracks in clad vessels. Previously completed elements of the program provided test data on the effect of cladding on the propagation of a fast-running crack.

POTENTIAL IMPACT OF ENHANCED FRACTURE TOUGHNESS DATA ON PRESSURIZED-THERMAL-SHOCK ANALYSES*

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SUMMARY

The HSST Program is involved with the generation of "enhanced" fracture-toughness data of prototypical nuclear reactor vessel steels. This data is termed enhanced because it has distinguishing characteristics that could potentially impact PWR pressure vessel integrity assessments for the pressurized-thermal shock (PTS) loading condition which is a major plant-life extension issue to be confronted in the 1990's. Governing criteria associated with PTS are included in "The PTS Rule" (10 CFR 50.61) and Regulatory Guide 1.154.

Over the past several years, the Heavy Section Steel Technology (HSST) Program at Oak Ridge National Laboratory (ORNL) has performed a series of large-scale fracture mechanics experiments. These experiments have produced K_{Ia} data with the distinguishing characteristic that the values are considerably above 220 MPa • $\sqrt{m}$, the implicit limit of the ASME Code and the limit used in the Integrated Pressurized Thermal Shock (IPTS) studies. The IPTS studies also assumed the onset of unstable ductile tearing to occur at 220 MPa • $\sqrt{m}$. The results of the IPTS study are particularly important because they contributed to the establishment of the PTS governing criteria.

Deterministic fracture mechanics analyses were performed using OCA-P, a program developed at ORNL, to examine the influence of the enhanced K_{Ia} data on the cleavage fracture response of a nuclear reactor pressure vessel subjected to PTS loading. The results of the analyses indicated that the effect of the enhanced K_{Ia} data is to increase the probability of crack arrest; however, unstable ductile tearing must also be considered as a possible failure mode before concluding that the enhanced crack arrest potential decreases the probability of failure. Approximations for the onset of unstable ductile tearing, i.e., the crack depths corresponding to stress levels beyond which ductile tearing becomes unstable, were determined to be 240 MPa • $\sqrt{m}$ for vessels containing low upper-shelf weld material (LUSW) and 370 MPa • $\sqrt{m}$ for A533B steel, a prototypical reactor pressure vessel steel. It appears the potential benefit from crack-arrest events corresponding to toughness values above 240 MPa • $\sqrt{m}$ for low upper-shelf material (LUSW) and above 370 MPa • $\sqrt{m}$ for A533B will usually be negated by unstable ductile tearing. The potential benefit of the enhanced K_{Ia} data appears more likely to exist for vessels subjected to lower pressure transients and vessels that do not contain LUSW material. The application of the K_{Ia} data above 220 MPa • $\sqrt{m}$ should, in principal, result in a reduced calculated probability of failure relative to the original IPTS study.

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Recent investigations at the University of Kansas have shown that elastic-plastic fracture toughness is dependent on the crack depth of the fracture specimen. Experiments performed on A36 steel (low yield strength, high strain hardening) and A517 steel (high-yield strength, low-strain hardening) have produced data with the distinguishing characteristic that the fracture toughness for shallow-cracked specimens is elevated when compared to the fracture toughness determined using conventional, deep-notched (a/w = 0.5) specimens. It is anticipated that A533B will also show an increase in fracture toughness for shallow flaws since the stress-strain curve of A553B is bracketed by those of A36 and A517. The HSST program is currently investigating shallow flaw behavior to verify and quantify this phenomena for A533 material.

Probabilistic fracture mechanics analyses performed in the IPTS studies indicated that a substantial percentage of predicted crack initiations, and subsequently predicted failures, originate with shallow flaws. Deterministic fracture mechanics analyses were performed, using OCA-P, to evaluate the influence of elevated fracture toughness data on the cleavage fracture response of a nuclear reactor pressure vessel subjected to PTS loading. Specifically, the anticipated elevated shallow-flaw data for A533B was approximated by interpolating the University of Kansas data for A36 and A517 steels based on the yield strength of the materials. OCA-P was modified to interpolate the approximated A533B enhanced shallow flaw fracture-toughness data with respect to flaw depth and to interpolate/extrapolate the data with respect to temperature. The results of the analyses indicated that the effect of enhanced shallow flaw fracture-toughness data significantly reduces the number of crack initiations, and therefore failures, relative to the results of the IPTS.

Application of enhanced fracture toughness data has been shown to have potential for reducing the calculated probability of failure in an IPTS-type probabilistic fracture mechanics analyses. This could prove to be significant since Regulatory 1.154 references the IPTS study as an acceptable methodology for performing the probabilistic fracture mechanics portion of the plant-specific safety analysis. The plant-specific safety analysis is required for any plant to operate beyond the screening criteria defined in the PTS rule.

Multivariable Modeling of J-R Data

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It is necessary to estimate J-R data for reactor vessels, piping, and components to analyze plant safety issues related to radiation damage and remaining life. In many cases, field-aged specimens from reactor surveillance programs are either unavailable for the specific weld or heat of material in service, or they are available at radiation conditions that do not match the case to be analyzed, or they are available only in the form of Charpy specimens. In each of these cases, it is necessary to estimate J-R curves from the available data. For this purpose, a multivariable model for predicting J-R curves based on variations in material chemistry, radiation exposure, tensile properties, and Charpy data is needed.

The project reported here builds on the earlier modeling and data collection efforts by Materials Engineering Associates and Babcock & Wilcox. The public databases collected by these organizations were combined to form a reasonably comprehensive sample of completely characterized, public test data, with and without irradiation. Ferritic materials used in both nuclear pressure vessels and piping were included. An improved three-parameter model for the dependence of J on crack extension was developed, and advanced transformation analysis tools were used to identify the key variables and the optimal form for modeling the parameters. A deliberate attempt was made to represent the key effects with the simplest possible model.

Several models are developed, depending on the type of data available for prediction. The models divide into two classes, depending on whether Charpy data are available for the conditions of interest. They also can be divided into models for weldments and for all product forms. These models are presented and discussed in the full paper.

Heavy-Section Steel Irradiation Program Overview*

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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 releases. The RPV is one of only two major safety-related components 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 primary goal of the Heavy-Section Steel Irradiation (HSSI) Program is to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior, and in particular the fracture toughness properties, of typical pressure vessel steels as they relate to light-water 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 postirradiation 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 impact energy, dropweight nil-ductility temperature (NDT), and tensile properties. Models based on observations of radiation-induced microstructural changes using the field ion microprobe and the high resolution transmission electron microscope provide improved bases 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 postirradiation properties are most frequently limiting in the continued safe operation of commercial RPVs. In addition, a limited effort focuses on stainless steel weld overlay cladding, typical of that used on the inner surface of RPVs, since its postirradiation fracture properties have the potential for strongly affecting the extension of small surface flaws during overcooling transients.

Of particular interest are the efforts in the past year concerning the shifts in fracture toughness and crack arrest toughness in high-copper welds, the unirradiated examination of a low upper-shelf (LUS) weld from the Midland reactor, and the continued investigation into the causes of accelerated low-temperature embrittlement recently observed in RPV support steels. In the Fifth and Sixth Irradiation Series, designed to examine the shifts and possible changes in shape in the ASME K_{Ic} and K_{Ia} curves for two irradiated high-copper welds, it was seen that both the lower bound and mean fracture toughness shifts were greater than those of the associated Charpy-impact energies, whereas the shifts in crack arrest toughness were comparable. Even though the shifts in fracture toughness exceeded those of the Charpy tests, the irradiated data were fully encompassed by the appropriately indexed ASME K_{Ic} curve when it was shifted according to Revision 2 of Regulatory Guide 1.99 including its margins. The beltline weld which was removed from the Midland reactor, fabricated by B&W using Linde 80 flux, is being examined in the Tenth Irradiation Series to establish the effects of irradiation on a commercial LUS weld. A wide variation in the unirradiated fracture properties of the Midland weld were measured with values of RT_{NDT} ranging from -22 to 54°F through its thickness. In addition, a wide range of copper content from 0.21 to 0.32 wt % was found, which is much lower than the 0.42 wt % previously reported. The remainder of the unirradiated fracture testing and preparations for the irradiations of this material are currently in progress. A theoretical examination of the detailed irradiation mechanics which exist in low-temperature irradiations such as those which produced the accelerated embrittlement of the HFIR pressure vessel has led to the tentative conclusion that the cause of the acceleration is the high fraction of very low-energy thermal neutrons which existed rather than the low rate at which the fluence was accumulated. This conclusion is being investigated in two experiments. Specimens are being irradiated at low temperatures and high-fluence rates in spectra with and without large components of thermal neutrons. Other specimens are being exposed in the cavity of a pressurized water reactor at low temperatures and low fluence rates. As a result of the two experiments it should be possible to establish the mechanism primarily responsible for the accelerated low-temperature embrittlement.

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 LUS 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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Consequence Evaluation of Radiation Embrittlement of Trojan Reactor Pressure Vessel Supports

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Summary

The consequence evaluation of radiation embrittlement of reactor pressure vessel (RPV) supports of nuclear power plants offers a more direct and less controversial approach to the safety concerns addressed by Generic Safety Issue 15(GSI-15) identified by the U.S. Nuclear Regulatory Commission (NRC) because this approach depends on more conventional methodologies widely accepted by the engineering community. The success of this evaluation may permit a satisfactory resolution to GSI-15 by demonstrating that even under the most unfavorable circumstances, i.e., complete failure of all RPV supports, there is no undue risk to public safety.

This evaluation is divided into two phases. Phase 1 is a pilot study on a selected nuclear power plant. Phase 2 is a parametric study undertaken in an attempt to generalize the conclusion of the pilot study to other nuclear power plants. The Trojan nuclear power plant was selected for the pilot study because its RPV supports are located in the high radiation zone and are subject to high tensile stresses. The pilot study comprises a structural evaluation and an effect evaluation and assumes that all four RPV supports have completely lost their load carrying capability. The current paper addresses Phase 1 results and conclusions.

The structural evaluation considers two load combinations: the combination of dead weight, operating pressure, and the safe shutdown earthquake and the combination of dead weight, pressure, and a loss-of-coolant-accident (LOCA). Both load combinations are classified as Level D Service Limits in accordance with ASME Boiler and Pressure Vessel Code. Rules containment in Subsection NB in conjunction with Appendix F, Division 1, Section III of the ASME Code, which permit linear elastic analyses, are following by the structural evaluation.

A preliminary structural evaluation based on an existing computer analysis model of the nuclear steam supply system (NSSS) of the Zion nuclear power plant, which is similar to the Trojan plant, indicates that the ASME Code Appendix F requirements are satisfied by each of the load combinations considered in the analysis, leading to the preliminary conclusion that the Trojan RCL piping is capable of transferring RPV loads to steam generator (SG) and reactor coolant pump (RCP) supports. A subsequent final structural evaluation based on a computer model developed for the Trojan NSSS confirms the preliminary conclusion and, additionally, concludes that the SG and RCP supports have sufficient design margins to accommodate additional loads transferred through the RCL piping.

The effect evaluation, employing a systems analysis approach, investigates initiating events and the reliability of the engineered safeguard systems, which are designed to mitigate some of the initiating events, as the RPV is subject to movements caused by the support failure. As a result, the evaluation identifies the following areas of safety concern:

(1) The multiple rupture of instrumentation thimble tubes or the guide tubes that penetrate the bottom head of the RPV could result in a LOCA that may lead to core uncovery.

(2) The tilting of the flywheel and the deformation of the RCP casing may respectively effect the coastdown ability of the RCP and cause impellers to bind, resulting in loss of natural circulation.

(3) The control rods could bind in the event of tilting of the RPV and the ability to insert control rods during a reactor trip may be affected.

(4) The rupture of two or more of the 10-in. safety injection lines could impair the ECCS function.

Further investigation, however, concludes that the failure of the Trojan RPV supports will not result in consequences of safety concern because:

(1) A structural analysis of thimble guide tubes indicates that RPV movements will not cause tube rupture.

(2) An assessment of the RCP indicates that the pump should be able to sustain the motion without loss of its function during either the coastdown phase or the natural circulation stage.

(3) Based on information provided by the NSSS vendor, the control rods will not bind as the RPV is subject to the tilting caused by the RPV support failure.

(4) An anlysis of the 10-in. safety injection lines demonstrates that the RPV movements will not cause rupture of these lines.

TMI-2 VESSEL INVESTIGATION PROJECT METALLURGICAL PROGRAM*

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The TMI-2 Vessel Investigation Project Metallurgical Program at Argonne National Laboratory is a part of the international TMI-2 Vessel Investigation Project being conducted jointly by the U.S. Nuclear Regulatory Commission and the Organisation for Economic Co-operation and Development (OECD). The overall project consists of three phases, namely (1) recovery of material samples from the lower head of the TMI-2 reactor, (2) examination and analysis of the lower head samples and the preparation and testing of archive material subjected to a similar thermal history, and (3) procurement, examination, and analysis of companion core material located adjacent to or near the lower head material.

The specific objectives of the ANL Metallurgical Program, which accounts for a major portion of Phase 2, are to prepare metallographic and mechanical test specimen blanks from the TMI-2 lower head material, prepare similar test specimen blanks from suitable archive material subjected to the appropriate thermal processing, determine the mechanical properties of the lower vessel head and archive materials under the conditions of the core-melt accident, and assess the lower head integrity and margin-to-failure during the accident. The ANL work consists of three tasks: (1) archive materials program, (2) fabrication of metallurgical and mechanical test specimens from the TMI-2 pressure vessel samples, and (3) mechanical property characterization of TMI-2 lower pressure vessel head and archive material.

It was anticipated that the amount of material actually obtained from the lower head of the TMI-2 reactor would not be sufficient to carry out all of the mechanical tests and microstructural studies necessary to thoroughly assess the integrity of the lower head during the accident. The archive materials activity was therefore created to provide supplemental material for these studies. Unfortunately, no actual archive material from the TMI-2 lower head was available, and it was necessary to obtain material as similar as possible from another source. The source selected for this alternative "archive" material was the lower head of the Midland nuclear reactor (which was never operated due to cancellation of the plant before completion) in Midland, MI. The Midland reactor was a sister plant to TMI, and the lower heads of both vessels came from the same supplier and had virtually identical fabrication histories.

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

The Midland material was obtained in the form of four plates, each approximately 0.3 x 1.2 x 0.14 m thick (12 x 48 x 5-5/8 in. thick). Chemical analyses and checks on hardness and microstructure were carried out to verify that the Midland material was suitable for use in the program.

A heat treatment program was conducted on the archive material to produce a set of standard microstructures for comparison with those observed in the actual TMI-2 lower head material. Three types of thermal cycles were studied, each designed to simulate the thermal history at some point in the TMI-2 lower head during the accident. The results of the heat treatment program, along with those of supplementary hardness studies, demonstrated that those regions in the TMI-2 lower head where the maximum temperature exceeded the 727°C ferrite-to-austenite transformation temperature during the accident should be identifiable on the basis of microstructural observations.

A series of round-robin mechanical tests and microstructural studies on the as-received archive material was conducted to better characterize this material and to determine the level of variability in mechanical test data obtained by the OECD laboratories participating in the program. This test series consisted of tensile tests at room temperature and 600°C, as well as short-term stress-rupture tests at 600°C. The results of these tests and examinations indicate good agreement among the various laboratories.

Most recently, 15 actual samples removed from the TMI-2 lower head including 4 samples with the remains of instrument penetration nozzles, have been received. One sample, from a region of relatively severe damage, contained a crack in the cladding ≈5 mm (0.2 in.) wide and extending for ≈75 mm (3 in.) on the sample surface. This crack was associated with and encircled an instrument penetration nozzle adjacent to the location from which the sample was removed. Microstructural and SEM examinations of metallographic sections through the crack determined that it penetrated through the stainless steel cladding on the vessel surface but extended for only a short distance into the underlying A533B base metal. The total depth of the crack was ≈8 mm (0.3 in.). Extensive oxidation of the base metal was observed at the bottom of the crack, and control-assembly material from the core was present on the crack surfaces. However, only minor incidental fuel fragments were detected, thus indicating no intrusion of the molten fuel into this crack. A comparison of the microstructures and hardnesses of samples from the archive heat treatment program with those in the base metal of this sample indicate that the temperature of the TMI-2 lower head at this location substantially exceeded 727°C.

U. S. Nuclear Regulatory Commission's Policy Statement on Below Regulatory Concern

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The United States Nuclear Regulatory Commission has developed a policy statement on Below Regulatory Concern (BRC) which will serve to establish a framework for future rulemaking and licensing decisions exempting certain activities involving radioactive material from regulatory control on the basis that the risks are so small that further efforts to reduce them are not warranted. The policy is also intended to allow the NRC, Agreement States, and licensees to focus their resources toward addressing more significant risks from radioactive materials. The policy statement does not, in itself, exempt any materials from regulatory control, but will serve as a framework for potential applications such as release of formerly utilized sites for unrestricted public use, exempt distribution of consumer products containing small quantities of radioactive material, the disposal of very low-level radioactive waste, and the recycle or reuse of slightly contaminated equipment and materials. The policy will contain numerical criteria for both individual and collective dose which will define when further efforts to reduce dose from a practice are not necessary in keeping with the ALARA concept. The policy will also describe the NRC plans for implementation of the policy, and outline some of the information that would be necessary to support a request for exemption of a practice.

OVERVIEW OF THE NUCLEAR REGULATORY COMMISSION'S REVISED 10 CFR PART 20

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The principal Nuclear Regulatory Commission (NRC) regulations covering radiation protection are in Part 20 of Title 10 of the Code of Federal Regulations (10 CFR Part 20). These regulations apply to almost all NRC licensed activities including nuclear power and research reactors, nuclear medicine, industrial radiography and the use of most radioactive isotopes in research. The regulations were initially issued in 1957 and, although over 90 amendments have been made in the intervening 30 years, a complete revision has only recently been completed. This revision was initially started in 1978. An Advance Notice of Proposed Rulemaking asking for public comments and suggestions for this revision was issued by the NRC in 1980 and a proposed rule was issued for public comment in 1986. Over 800 sets of comments were received and considered in the development of the final rule.

The revised 10 CFR Part 20 is based on the 1977 recommendations of the International Commission on Radiological Protection (ICRP) in ICRP Publication No. 26, implements the 1987 Federal Radiation Guidance on Occupational Exposure and is generally consistent with the 1987 recommendations of the U.S. National Council on Radiation Protection and Measurements (NCRP) in NCRP Report No. 91. This means that the revised Part 20 incorporates some departures from previous methods of assessing, controlling and recording doses. Adoption of the ICRP approach entails use of an effective dose (a risk-weighted sum or organ doses) instead of separate dose limits for each organ. The rule also requires that the total effective dose equivalent (the sum of the external deep-dose equivalent and the internal comitted effective dose equivalent) be controlled rather than controlling external and internal doses separately. Because of this change, control of external doses and control of internal doses are given more equal emphasis than in prior regulations where more emphasis was placed upon limiting radionuclide intakes and internal doses.

There are a number of areas where the Revised Part 20 employs new and more up-to-date scientific information and concepts. One major change is in the use of revised lung and GI tract models and more recent metabolic retention data to calculate Annual Limits on Intake (ALI's) and Derived Air Concentration limits (DAC's). The Appendix B in the Revised Part 20 contains data on occupational ALI's and DCA's and radionuclide concentration limits for releases to the general environment (air, water, and sanitary sewer systems) for over 750 radionuclides (about 500 more than the 260 radionuclides listed in the existing Part 20).

The final rule is expected to be published early in September with an implementation date of January 1, 1992. The NRC staff is developing regulatory guides on the new concepts and procedures. These guides are expected to be issued for public comment approximately one year before the effective date of the rule.

THE DEBRIS MODULE: AN EFFECTIVE TOOL for the ANALYSIS of MELTDOWN PROGRESSION in LWR's

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ABSTRACT

The DEBRIS module was developed in response to the need for an effective and efficient tool for the analysis of core meltdown processes in Light Water Reactors (LWR's). The work was performed at Sandia National Laboratories under the auspices of the USNRC as part of the internationally sponsored severe fuel damage program. In particular, a model was needed to describe the "late phase" of the core meltdown process, that is, the period following the loss of intact rod geometry. This phase of a postulated core meltdown accident is characterized by the melting and relocation of ceramic rich materials through a rubblized medium composed mostly of fuel pellets and oxidized cladding fragments. Of particular interest are the dynamics of the melting process in a multi-component debris bed, the relocation of the various components, the formation of crusts due to freezing of relocating materials, the retention of molten materials by the crust, and the remelting of crusts. The set of models that constitute the DEBRIS module solve the two-dimensional (r,z) momentum equation to account for melt relocation due to both gravitational and capillary forces, the continuity equation to assure mass conservation, and the energy equation. The equations for fluid motion balance the viscous drag which is modeled by a modified Darcy law formulation against gravity and pressure differential. The motion of solid debris under the influence of gravity forces is also included. The energy equation is solved in a decoupled manner utilizing an explicit solution to account for enthalpy transport due to material motion and an implicit two-dimensional solution to account for conduction and radiation heat transfer. The model currently accommodates 4 material species (UO_2, ZrO_2, Zr, and Fe). A phase diagram for the low melting temperature Zr-Fe solution is incorporated as well as a model for the dissolution of UO_2/ZrO_2 by molten Zr. A phase diagram for the melting of the UO_2/ZrO_2 system has been added to estimate component stratification during the melting process. The DEBRIS model has been applied to a number of accident configurations with varying debris bed constituents. Some of the results of these studies will be discussed. A specially modified version of the DEBRIS module was developed to simulate the geometry and configuration of the Melt Progression (MP) experiment series being conducted in the Annular Core Research Reactor (ACRR) at Sandia Laboratories. This version of the DEBRIS module has been used to analyze the first experiment in the series (MP-1). Thus the MP-1 experiment provides a convenient vehicle to begin DEBRIS model verification. The preliminary results of the experiment appear to confirm both qualitatively and quantitatively the DEBRIS code predictions for the important processes including material relocation and heat transfer. The application of the DEBRIS models to the MP-1 experiment will be presented in comparison to measure data from the experiment. The Debris module appears to have significant potential for analysis of the "late phase" meltdown processes and can be effectively used both in a stand-alone mode and in conjunction with the severe accident sequence analysis and consequence analysis codes (MELCOR,SCDAP). In addition, by using modified flow resistance parameters (Darcy formulation) and enabling oxidation models the code may prove very effective in treating early phase processes as well.

LOWER HEAD FAILURE ANALYSIS

by

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The methodology and preliminary results from the NRC-sponsored lower vessel head research program are described within this paper. Objectives of the analysis are to investigate plausible modes of reactor vessel failure to determine (a) which modes have the greatest likelihood of occurrence during a severe accident and (b) the range of core debris and accident conditions that lead to these failures. A wide range of reactor designs and thermodynamic conditions leading to various types of vessel failure were investigated using analytic closed-form approximations to assess the important governing parameters in non-dimensional form. Preliminary results, which are described in detail in the final paper, are summarized below.

For melt plugging or bulk freezing, results indicate that higher conductivity metallic melts, low velocity melts associated with reduced system pressures, and melts moving into smaller diameter penetrations are more likely to freeze prior to exiting the boundaries of the lower head. In cases where the melt moves beyond the lower head boundaries, results indicate that vessel penetrations, such as empty control rod guide tubes or instrumentation tubes, will fail. Analyses of melts controlled by conduction limited freezing (the melt freezes as a film on the tube surface rather than freezing uniformly across the tube as in the case of bulk freezing) indicate that axial penetration distances of melt moving into the tube are longer and heatup of the tube walls is reduced compared to that for bulk freezing.

A simple combination numerical/analytical model was developed to evaluate the critical parameters for tube ejection and penetration tube rupture. Application of this model to a range of conditions representative of PWR and BWR designs indicate that the "as built" gap between the tube and vessel wall was a critical parameter for these two mechanisms. For temperatures significantly below 1400 K, where the tube strength approaches zero, neither tube ejection nor tube rupture are predicted in cases where a large temperature difference between the tube and vessel walls exists since the gap is predicted to close.

Simple thermal calculations were also performed to evaluate the relative timing of lower head failure for conditions and designs representative of PWR and BWR designs. These calculations, which evaluated (a) adiabatic heatup of different core designs, (b) dry out of debris in the lower plenum, and (c) the heatup rates of the lower head, indicate that BWRs may experience lower head failure later in time than comparable PWRs under similar accident conditions.

ACE PROGRAM PHASE A: CONTAINMENT FILTRATION EXPERIMENTS

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Filtered venting of reactor containments has received considerable attention recently as a method for avoiding containment failure due to overpressure during severe accidents. Several countries, such as Sweden, Finland, France, the Federal Republic of Germany, the Netherlands and Switzerland, have already committed themselves to using containment venting systems, while the potential benefits, drawbacks, and costs of such systems are still being assessed in other countries, such as the USA.

In light of these considerations a decision was taken to test several of the proposed filtration devices in the internationally sponsored Advanced Containment Experiments (ACE) Program, such that a self consistent comparison of the aerosol removal characteristics of these systems could be obtained. The tests were carried out at the Hanford Engineering Development Laboratory.

Considering the different design, requirements and operating conditions of these devices, a direct comparison is not possible, nor appropriate. Nevertheless, large scale models, utilizing full scale elements of the various devices whenever feasible, have been tested with consistent mixtures of aerosols and carrier gases.

The aerosols chosen for the test program were hygroscopic CsI and CsOH, because they represent the risk dominant fission products expected in a severe reactor accident, and nonhygroscopic MnO representing structural materials. Cesium and manganese vapor are injected into a mixing vessel along with gaseous hydrogen iodide, nitrogen and steam. A plasma torch is used to vaporize elemental manganese in an argon carrier gas, whereas the cesium is vaporized in an oven. Reactions with the steam then produce and age the required aerosols in the aerosol mixing vessel. Because the aerosol sizes produced by this method are fairly large, in the range of 2 - 3 μm, it was decided to also perform standard dioctyl phthalate (DOP) tests with air as the carrier gas to supplement the main experiments, and test the filter efficiency in the submicron size range.

The test matrix for the experiments was developed by the project's Technical Advisory Committee, which also served to review the test results and provide a quality assurance function for the data.

Two experiments were generally performed on each device; one with a mixture of nitrogen and thirty weight percent steam, and the other with almost pure nitrogen. This strategy provides the opportunity to assess condensation effects on filtration performance, and to allow hygroscopic particles to grow in size. The model filter systems, which were designed for these experiments, were sized to accommodate the carrier gas flow rates available in the test facility. All the experiments were performed at essentially atmospheric pressure.

For each filter the material, concentration, and size distribution of the input and effluent aerosols are measured, and decontamination factors subsequently determined based on this information. Overall mass balances and decontamination factors are also obtained by weighing the quantities of material injected, and the amounts collected in, and downstream of, the test filter.

To gain familiarity with the experimental equipment and develop confidence in the sampling and measurement techniques, the first set of experiments were used to determine the decontamination factors in water pools of varying depth and temperature. These results were subsequently found to be consistent with, the extensive available data base on pool scrubbing. The tests with simple water pools are also appropriate since all BWRs utilize suppression pools, and because several of the commercially available filtration systems involve devices that are submerged in a pool of water.

Systems that have been tested as part of the ACE program, which incorporate water pools in their designs, include the submerged gravel scrubber from the US DoE, the submerged cyclone separator from the USSR, and the submerged venturi scrubbers from ABB-Atom in Sweden, and Siemens in the FRG.

The remaining systems in the ACE program, which do not involve water pools, include a gravel bed, a sand bed and the KfK fiber metal filter from the FRG. All of these devices are normally dry and at ambient temperatures. Subsequently, upon use, they become wetted by condensing steam and, if flows of superheated steam or high temperature gases persist, they can in fact dry out once again.

This complex thermal behavior meant that the tests on these systems could not be carried out under steady state conditions, but rather that the filtration efficiency measurements had to be performed as transient tests.

All the tests in this research program were recently completed, though analysis is still continuing. The results will be reported to the project participants for their use, within their own countries, to meet their design, safety, and licensing requirements.

ACE Program Phase B: Iodine Behavior In Containment
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The objects of this phase of the ACE project are to perform experimental work to evaluate potentially important sources and sinks for volatile iodine species in containment during the course of severe accidents and to provide data for the testing/validation of iodine behavior models. The work includes a combination of laboratory studies, intermediate-scale experiments, and a large-scale test.

The laboratory studies are producing basic data in three areas. Measurements are being made of the gas phase sorption rates and sorption capacity for elemental iodine, methyl iodide, and hydrogen iodide with a variety of potential aerosol materials at temperatures up to 100 C. Other experiments have been evaluating the effects of pH, temperature, gamma radiation, inital speciation, and epoxy paint on iodine partitioning between aqueous and gaseous phases and on iodine-paint interactions. Lastly, a hydrogen burner and associated equipment are being used in an attempt to understand the mechanism of cesium iodide oxidation in hydrogen flames.

The intermediate-scale experiments are composed of a series of radiolysis effects tests in the Radioiodine Test Facility (RTF) and also a set of hydrogen combustion effects tests in the Containment Test Facility (CTF). The RTF consists of a 0.35 cubic meter vessel equipped with gas and liquid circulation/sampling loops, a Co-60 radiation source capable of dose rates up to 10 kGy/hr, and associated equipment and instrumentation enabling controlled operation from room temperature up to about 80 C. The parameters varied in these multi-effects tests include radiation dose, initial aqueous phase iodine speciation, aqueous phase pH, and a painted or unpainted vessel interior. Each test is run for several days during which gas and aqueous phase iodine concentrations, speciation, and system conditions (pH, temperature, pressure, etc.) are followed continuously. At present three test series have been completed and a fourth is scheduled for early next year.

The CTF is a 6.3 cubic meter spherical vessel that has been used to conduct numerous hydrogen combustion experiments are various conditions. For the ACE tests the facility has been equipped with a pneumatic device for dispersing several grams of fine cesium iodide powder into a pre-mixed hydrogen containing atmosphere in the vessel. The gas mixture is then ignited and following the resulting deflagration the atmosphere in the vessel is sampled, and the samples are analyzed chemically for the presence of oxidized forms of iodine. A total of five tests have been conducted at different combinations of hydrogen concentration, steam content, and cesium iodide dispersal conditions. Some tests resulted in cesium iodide oxidation but others did not.

The large-scale iodine test was conducted in November 1989 at the Containment Systems Test Facility (CSTF). The CSTF is an 850 cubic meter cylindrical vessel with associated aerosol generation equipment and all the auxiliary facilities needed to conduct containment behavior experiments. The test was carried out over a five day period with the vessel containing a saturated steam-air atmosphere at about 100 C. Before the test the interior surfaces of the vessel were painted with a DBA-qualified epoxy coating material. Phenomena studied during the test included uptake of vapor iodine species (both hydrogen iodide and elemental iodine) by high concentration aerosol suspensions (mixtures of cesium hydroxide and manganese oxide), organic iodide formation and buildup, surface deposition and retention of iodine, accumulation of iodine and aerosol components in the water sump at the bottom of the vessel, and the effect of changing the pH of the sump water late in the test period. Highly sensitive chemical analysis procedures were developed to permit measurement of iodine in the many gas, liquid, and deposition samples that were collected during the test. At present all sample analyses have been completed and a final report of the test details and results is being prepared for later distribution to ACE participants.

ABSTRACT

An Integrated Structure and Scaling Methodology for Resolving Technical Issues Relevant to Severe Accidents

by

The Technical Program Group*

To provide the basis for an efficient resolution of technical issues, the Technical Program Group (TPG) has developed a physically based methodology that integrates experiments, analyses and uncertainties. The Integrated Structure for Technical Issue Resolution (ISTIR) consists of the five components in Figure 1. The necessary integration is achieved by identifying and ranking the requirements to resolve a technical issue (Component I), and expressing them in terms of specifications for both experiments (Component II) and analysis (Components IV and V). Technical issue resolution is achieved by means of special models (Component III) and their uncertainties, or code calculations and their uncertainty quantification (Component V). The ISTIR provides a proper balance (sufficiency) between experiments and analysis to insure a cost-effective (efficiency) and timely resolution of a technical issue.

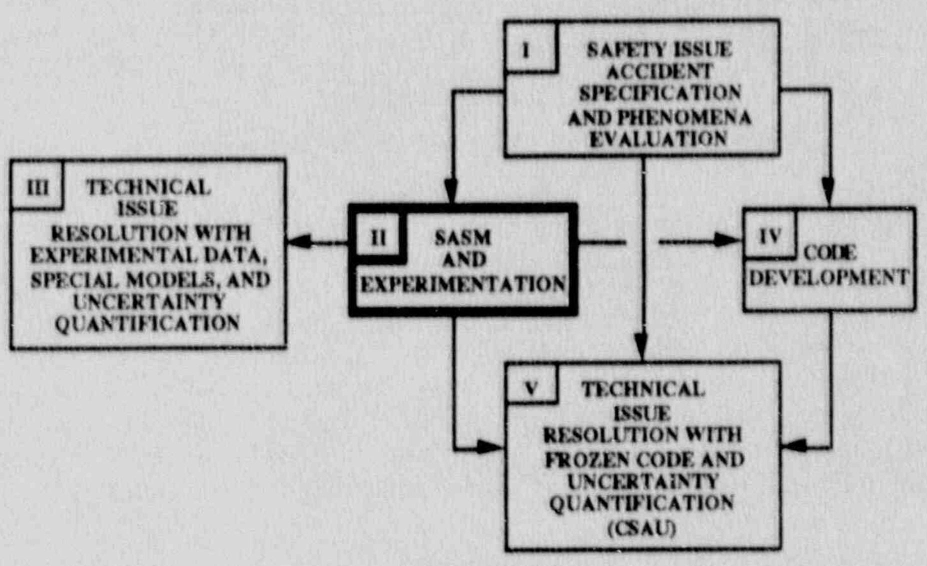

Figure 1. Integrated Structure for Technical Issue Resolution (ISTIR).

Scaling has been identified as a particularly important element in the resolution of severe accident technical issues. Accordingly, and as part of the ISTIR, the Severe Accident Scaling Methodology (SASM) has been developed. Based on a hierarchical

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approach, SASM provides:

1. A scaling rationale for a particular scenario,
2. Similarity criteria that combine the system (top-down) and process (bottom-up) view points,
3. A procedure for conducting comprehensive reviews of facility designs, and test specifications and results, and
4. A procedure to assess and quantify the effects of scale distortions.

The SASM consists of eleven steps which are grouped in three key elements (Figure 2):

1. *Experimental Requirements*, in which the experimental objectives are specified in terms of the technical issues defined in Component I of the ISTIR,
2. *Evaluation and Specification for Experiments and Testing*, in which the experimental objectives are reflected in terms of scaling rationales that are necessary to insure both separate and integral effects data are applicable to full scale reactors, and include the phenomena important to the specified accident scenario (or class of scenarios), and
3. *Data Base and Documentation*, in which the data base(s) appropriate to issue resolution is established and documented for its subsequent use in Components III, IV and V.

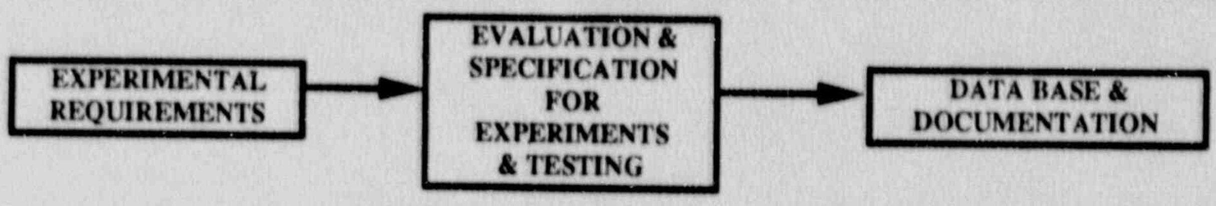

Figure 2. SASM (Component II) flow diagram.

The SASM, and its foundation (Component I of the ISTIR), have been tested, and are demonstrated, by application to a postulated direct containment heating scenario, in which the containment loads (over-pressure, temperature, etc.) were the primary areas of focus. The results demonstrate that the methodology is comprehensive, systematic yet practical, auditable and traceable, as required by a regulatory agency.

In this paper, Part A provides a general overview of the SASM in its generic context, including its integration with the ISTIR. Part B then provides more specific details, together with scaling criteria, which are associated with an example application to a postulated direct containment heating scenario.

SCALING LAWS FOR SEVERE ACCIDENT PHENOMENA IN BWRs

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The predicted thermal-hydraulic response of reactor and containment systems to postulated severe accidents is needed for determining mitigative features and emergency procedures. Severe accident behavior typically is predicted from experimental and analytical investigations. It is essential that experimental programs preserve the important full size phenomena in order to be useful both in predicting actual system response, and in providing representative data for the verification of analytical models. Subscale or laboratory experiments will preserve full size behavior if they can be designed to accommodate appropriate scaling laws. Sometimes an experiment cannot fully accommodate the scaling laws, and the response is distorted from that of full size. When this occurs, it is useful to have an analytical procedure for estimating the effect of distortions. Such a procedure also would be useful in predicting accident response in a given system from the available response of a reference system which may not be fully similar.

Scale modeling laws are developed in this study to assist the design and interpretation of representative small scale tests, useful in predicting severe accident phenomena expected in full size boiling water reactor and containment systems. Example phenomena considered include core melt progression inside the vessel, molten core debris discharge, core debris spreading on a floor, immobilization by freezing due to an overlying water layer, and containment response. A procedure also is described for estimating the effect of scale model distortions on the prediction of full size behavior. The same procedure is extended for predicting the response of a nuclear plant under postulated severe accident conditions from the given response of a reference plant under similar conditions.

DEBRIS DISPERSAL FROM REACTOR CAVITY DURING LOW TEMPERATURE SIMULANT TESTS OF DIRECT CONTAINMENT HEATING (DCH)

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ABSTRACT

The Ishii-Katoaka entrainment correlation for fully developed steady-state annular two-phase flow was modified for application to the undeveloped and transient conditions prevailing during blowdown from a reactor pressure vessel into a reactor cavity during Direct Containment Heating (DCH). Good correlation for the debris dispersion or entrainment fraction was obtained for the Winfrith simulant data obtained with a constant gas flow rate in a 1:25 scale of the Sizewell reactor cavity. The same correlation was found to apply to an equivalent circular reactor cavity over scales ranging from 1:132 to 1:21. The five fluids utilized in the Winfrith constant pressure reservoir tests were brought together by the gas Euler number and the property grouping $2P_c\sqrt{2P_c}\ t/\sigma\sqrt{\rho_f}$ where P_c is the reactor cavity pressure, σ and ρ_f the fluid surface tension and density, and t is the blowdown time.

Blowdown tests with simulant fluids in the Sizewell, Zion, Surry and Watts Bar scaled reactor cavities were correlated by the same equation and they exhibited the same dependence upon the initial gas Euler number, the same property grouping and an equivalent blowdown time. However, in contrast to the constant gas flow rate tests, the gas density was found to influence debris dispersal during blowdown tests. Also, a significant decrease in entrainment was obtained with the use of woods metal as a simulant. The decrease appears to be associated with increased Euler numbers and it may be related to a change in flow pattern entrainment or a slowdown of the gas velocity by the accelerating debris particles.

All tests showed a dependence upon the size of the orifice simulating the reactor vessel hole. This could be due to the fact that the orifice size may influence the thickness or the velocity of the gaseous boundary layer formed above the fluid being entrained.

Blowdown conditions required to initiate entrainment can be predicted from the correlations developed herein. The values so determined for constant gas flow rate approach the Kutateladze prediction when they are adjusted for the actual blowdown gas velocity in the boundary layer near the floor of the cavity. The Euler numbers for entrainment inception decrease with orifice (vessel hole) size. For blowdown tests, they increase with reduced gas and increased debris density.

For illustration purposes, the methodology was applied to predict debris dispersion in the Zion and Surry plants. The predicted dispersions are high when the reactor vessel hole is large or the vessel blowdown pressure high.

A System Level Scaling Analysis For Direct Containment Heating

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A systematic and practical scaling analysis has been completed for the DCH problem. The analysis provides the similarity groups necessary to ensure that experiments and model development can be performed in a suitable parameter range. In addition, importance ranking of DCH processes is possible, thus providing a means to focus and prioritize research. However, the scaling analysis cannot replace system level computer codes for making quantitative reactor predictions.

Conclusions drawn from the scaling analysis are contingent upon an assurance that all processes potentially important to DCH are reflected in the analysis. The specific choice of six source processes and five sink processes for this study was strongly influenced by a process screening study prepared by the SASM TPG.

The scaling analysis follows a top-down approach that incorporates the boundary conditions into the conservation equations written for a control volume. Dependent variables and their derivatives are normalized to order unity, which automatically incorporates the initial conditions. DCH is characterized by large uncertainties in its initial and boundary conditions.

Constitutive relations, which quantify heat and mass transfer, play a crucial and unavoidable role in the ranking process. They are equally important in forming similarity groups because they link parameters (e.g., debris particle size) back to the initial and boundary conditions imposed on the analysis. Constitutive relations represent a source of possible phenomenological uncertainty.

Evaluation of the similarity groups has been performed for the Surry plant with a prescribed set of initial and boundary conditions that included only a small amount of water. The evaluation indicates that there are four dominant source processes: debris entrainment, debris oxidation, debris/gas heat transfer, and hydrogen combustion. Only one mitigative process, debris trapping, was shown to be important.

Existing SURTSEY experiments have been reevaluated using the new scaling analysis. The scaling analysis also provides a justification for the design, specification, and operation of future SURTSEY experiments.

MELCOR Analysis of the TMI-2 Accident*

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Summary:

The MELCOR computer code is being developed by Sandia National Laboratories for the Nuclear Regulatory Commission for the purpose of analyzing severe accidents in nuclear power plants. The primary role of MELCOR is to provide realistic predictions of severe accident phenomena and the radiological source term. The results of these calculations are to be used as an integral part of probabilistic risk assessment studies. In particular, MELCOR calculations serve as a means of guiding the "back-end" analyses and can be used to answer important questions that arise in the formulation of accident progression event trees.

The first four phases of the TMI-2 standard problem have been analyzed with MELCOR. The purposes of these analyses were twofold. First, while MELCOR has been used extensively to analyze BWR plants, it has not been used to analyze PWR plants. Therefore, one goal of the analysis effort was to identify PWR specific features that needed to be added to MELCOR.

Second, the analysis of the standard problem allowed for the models in MELCOR to be compared to plant data and to the results of more mechanistic analyses. This exercise is therefore valuable for verifying and assessing the models in the code. As will be shown, the major trends in the TMI-2 accident are reasonably well predicted with MELCOR, even with its simplified modeling.

This paper describes the analyses of the TMI-2 standard problem that have been performed with MELCOR. A comparison of the calculated and measured results is presented. Based on this comparison, conclusions can be drawn concerning the applicability of MELCOR to severe accident analysis.

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

PROGRESS IN IMPROVING NDE RELIABILITY[a]

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SUMMARY

This paper reports on progress achieved under the NRC program entitled "Evaluation and Improvement in Nondestructive Examination Reliability for Inservice Inspection of Light Water Reactors (LWR)" (NDE Reliability Program). This program consists of five major tasks as follows: 1) Code Activities, 2) Pressure Vessel Inspection, 3) New Inspection Criteria, 4) Field Problem Consulting, and 5) Piping Inspection. Activities conducted under these tasks are summarized below.

The NDE Reliability Program objectives are to quantify the effectiveness of inservice inspection (ISI) techniques for LWR primary system components through independent research and to establish means for improving the overall reliability of ISI systems and processes. Significant progress was achieved during the past year in several of these areas.

Participation in ASME Section XI Code activities continued toward achieving industry acceptance of NRC-funded PNL research and development work. In culmination of a multi-year effort, a new Appendix VIII on "Performance Demonstration for Ultrasonic Examination Systems" was published in the 1989 Addenda to Section XI, which was issued March 31, 1990. This activity also contributed to the successful acceptance of a new Code Case that allows eddy current examination of coated (containment) surfaces in lieu of visual examination. Proposed new Code rules/requirements are being developed to address computerized UT imaging systems in ASME Section V. NUREG/CR-4882 entitled "Qualification Process for Ultrasonic Testing in Nuclear Inservice Inspection Applications" was completed and published.

The equipment interaction matrix study has developed a technical bases for revising the equipment operational parameters currently specified in ASME Section XI. This work culminated in recommendations provided to the Section XI Subgroup on Nondestructive Examination (SGNDE). This study showed that the equipment performance tolerances currently specified in Appendix VIII should be revised in one area, while also showing that the other specifications for equipment tolerance parameters are both appropriate and adequate. The recommended changes were to narrow the center frequency tolerance for narrow

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

band systems, whereas the current tolerances for wide band systems were shown to be adequate and appropriate.

A re-analysis of PISC-II round robin test data has been completed, and the results of this analysis have been compiled in a formal report for distribution to the international technical community. This work is being performed to quantify the capability of ultrasonic inservice inspection to detect and size defects in nuclear reactor coolant boundary components.

Continued participation in the PISC-III program with production of thermal fatigue cracks into specimens for the austenitic steel tests capability studies. Characterization of the wrought stainless steel reliability specimens was completed and compiled into integrated documentation for submittal to the Joint Research Centre in Ispra, Italy. The final call was made on several of these studies, a testing schedule was established, and testing was begun.

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. Probabilistic risk analysis (PRA) methods have been utilized to assess the effectiveness of current inspection requirements and practices by reviewing failure data and histories of ISI experience. An operating nuclear power plant site was visited to obtain plant-specific information for use in studies of risk-based inspection, and a plant walkdown was conducted to acquire data for this study. An experts workshop was conducted in May; and this very productive meeting provided valuable insight based on the participants review of the extensive data base that PNL had accumulated, collated, and integrated over a period of several months.

Work continued in addressing the inspectability of coarse-grained materials, and a topical report has been prepared for distribution to the technical community. A cooperative program in conjunction with the Center for NDE at Iowa State University, under EPRI funding, is being conducted to develop engineering requirements for surface specifications of ASME Code components. Currently, the ASME Code requirements for surface conditions are contained in a Nonmandatory Appendix that is largely based on engineering judgement and field experience. This project will provide a quantified, technical basis for specifying surface conditions relative to the actual need for ultrasonic inservice inspection requirements.

EVALUATION OF COMPUTER-BASED NDE TECHNIQUES AND REGIONAL SUPPORT OF INSPECTION ACTIVITIES[a]

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SUMMARY

This paper provides the program objectives and a summary of the technical progress during fiscal year (FY) 1990 for the program entitled "Evaluation of Computer-Based NDE Techniques and Regional Support of Inspection Activities." The objectives of this program are:

- Evaluate the reliability of computer-based ultrasonic inservice inspection.
- Develop guidelines for NRC regional staff to monitor and evaluate the effectiveness of inservice inspection performed using computer-based ultrasonic examination.
- Develop a steam generator tube mockup to evaluate the effectiveness of eddy-current inspection systems.

The program is comprised of four tasks: 1) Review Computer-Based Ultrasonic Systems, 2) Review Procedures for Computer-Based Ultrasonic Systems, 3) Technology Transfer and Training, and 4) Eddy Current Examination Reliability.

Review Computer-Based Ultrasonic Systems

During the past year, a seminar was developed for regional staff. The seminar provided instruction on basic knowledge required to review and evaluate computer-based ultrasonic systems. The seminar was two weeks in duration and included such topics as:

- Why Computerized UT/ISI?
- What is an Image?
- How Do You Generate an Image?
- How Do You Display an Image?
- Basic Computerized UT System
- Variables that Effect Computerized Images
- Flaw Sizing from Images

Review Procedures for Computer-Based Systems

(a) Work supported by the U.S. Nuclear Regulatory Commission under Contract DE-AC06-76RLO 1830; FIN L1100; Dr. J. Muscara, NRC Program Manager.

During the past year, guidelines were developed for reviewing procedures that are used with computer-based systems. The guidelines outline methods for conducting the procedure review from a human factors viewpoint and a technical viewpoint. The most significant finding from a review of several field procedures was that the procedures were generic and should be supplemented with a technique sheet that provides specific guidance for each component being examined.

Technology Transfer and Training

During the past year, a series of three test block sets was designed. The proposed series of test blocks included:

- 4 x 14 x 0.6 in. plate samples with EDM notches ranging from 5% to 95% through-wall.
- 10 in., Schedule 80 pipe rounds with EDM notches, thermal fatigue cracks, and stress corrosion cracking.
- 6 in. segments of cast stainless primary coolant pipe with thermal fatigue cracks.

The test blocks will be used to train NRC staff and could be used to test computerized systems during an audit. The plate samples were completed and shipped to NRC Region I.

Eddy-Current Inspection System Reliability

Results from a recently completed NRC research program on the reliability of eddy-current (ET) inspection techniques to detect and size degradation in steam generator tubes indicated a need for improvements in the ET inspection process.[a] In addition to updating Regulatory Guides 1.83 and 1.121 governing steam generator tube inspection and plugging criteria and working with the ASME Code, Section XI to improve ET inspection requirements, the NRC is funding research at PNL to develop performance demonstration qualification requirements for ET. In mid-FY 1990 the NRC initiated a task under this program to develop a steam generator tube mockup. The purpose of the mockup is to monitor and evaluate the effectiveness of ET inspection systems at the reactor site. Activities discussed in this paper include 1) statistical test design calculations to determine the feasibility and significance of various sizes of tube mockups, 2) design features of a portable tube mockup, and 3) initial efforts toward fabrication and characterization of flawed tube samples for incorporation in the mockup.

(a) R. J. Kurtz, et al. 1990. Steam Generator Tube Integrity Program/Steam Generator Group Project - Final Summary Report, NUREG/CR-5117. Pacific Northwest Laboratory, Richland, Washington.

ADVANCED NDE TECHNOLOGIES AND CHARACTERIZATION OF RPV FLAW DISTRIBUTION[a]

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SUMMARY

The NRC in its early years developed several technologies that have been undergoing validation and acceptance by the nuclear industry. The emphasis has been on refining field procedures, conducting field validation testing, providing training for NRC headquarters and regional staff, and working with ASME Code for the use of these advanced technologies. This work has focused on acoustic emission (AE) for continuous monitoring and synthetic aperture focusing technique for ultrasonic testing (SAFT-UT).

Significant accomplishments for the past year on the AE portion of the program involved on-reactor validation testing, acceptance of a Code Case on the use of AE technology, and completion of the final report. AE equipment was installed at Philadelphia Electric Company's (PECO) Limerick Unit 1 reactor to monitor a flaw indication in an inlet nozzle safe-end weld in March 1989. AE monitoring has continued since that time with the intent of correlating AE indications with conventional ISI inspection results at the time of the next refueling outage, which will start about September 1990. NRC-owned instrumentation is being used in this work which is being funded by PECO.

An ASME Code Case N-471 was developed as a vehicle to define the technology and procedure for applying AE for on-line monitoring of nuclear reactor components. This Code Case has finally been formally approved. It will permit the use of AE technology in a fashion designed for successful application to complex nuclear components. It is anticipated that this Code Case will permit an in-depth data base to be assembled for pursuing Code approval of a nonmandatory appendix to Section XI.

The AE technology development effort was completed and a final report documenting all of the significant work performed by PNL to evolve the technology sufficiently to meet the needs of nuclear applications was prepared.

Significant accomplishments on the SAFT-UT portion of the program focused on upgrading the SAFT-UT field system, continuing work to put the SAFT

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[a] Work supported by the U.S. Nuclear Regulatory Commission under Contract DE-AC06-76RLO 1830; FIN B2913 and L1099; Dr. J. Muscara, NRC Program Manager.

technology into ASME Code Section V, and testing the technology at the EPRI NDE Center on the performance demonstration test for IGSCC detection.

The thrust of the work to upgrade the SAFT-UT system was to replace the Vax 11/730 with a MicroVax III computer. The MicroVax III is a 3 MIP machine versus the Vax 11/730 which is a 0.3 MIP machine. At the same time the interconnect between the data acquisition 11/23 and the MicroVax was upgraded from a serial link to a parallel link which will make the data transfer rate much faster. The archival data storage media was upgraded from a magnetic tape to an optical disk. As a result of the new equipment, the system is being reduced from 3 racks to 2 racks of equipment. This has resulted in a much faster system and one more structured to handling the large data bases created when inspecting thick section nuclear components such as reactor pressure vessels.

The work with the ASME Code has been directed to Section V. A special task group was set up to generate a new appendix for computer processed imaging. This activity has been driven by PNL with an example constructed around the SAFT technology. This drafted version was approved as a model and sent to all current manufacturers of computer-based systems. They were requested to review this and to provide a comparable document on their system. This information was to be provided by the end of summer. At that time, the acceptable input was compiled and the appendix was reviewed by the task group and started through the formal review process.

The SAFT-UT system was to be taken to the EPRI NDE Center in September 1990 to pass the performance demonstration test for IGSCC detection. In a previous attempt where only an abbreviated SAFT inspection was performed, this test was not passed. With the upgraded system and the ability to use the required full SAFT inspection procedure, the test results should be greatly improved.

The program for the characterization of fabrication defects in U. S. reactor pressure vessels is to develop a technical data base for fabrication flaws that exist in nuclear reactor pressure vessels including the density, location, type, and size distribution. During this year, the material that is available for this kind of study was compiled and it was found that there is very little material representing the RPVs from the earliest to the latest ones constructed. Consequently, the inspection work will have to focus on more in-depth study on available material. A design for the systematic inspection of the available material was developed and a plan for validating the inspection results was also prepared.

The best source of available material is the information already collected on a series of Midland RPV blocks and the PVRUF RPV. The focus of the next year's work will be on thorough inspection of the PVRUF vessel.

The SAFT-UT system was also used in support of training NRC staff in understanding the fundamentals of how imaging systems work and their physical principles and limitations.

Improved Eddy-Current Inspection for Steam Generator Tubing*

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ABSTRACT

Oak Ridge National Laboratory has been engaged in the research and development of eddy-current tests for a wide range of different problems. We have been developing tests to monitor the radiation damage in steel, we are developing neural networks for the analysis of eddy current data, and we are developing new and improved probe types for the inspection of steam generators. Rotating reflection probes have been developed that allow a more sensitive test of steam generator tubing.

Recent advances have been made on our multiple property techniques. This technique generates a set of coefficients that correlate the readings from an eddy-current instrument to the properties of the test that produce the readings. While this technique will work with reflection probes, pancake probes, or bobbin probes, we have concentrated on the latter since this type of test is the most widely used in the commercial inspection of steam generators. The test properties varied include tube supports, tube sheets, copper deposits, magnetite deposits, denting, wastage, pitting, cracking and IGA.

While our multiple property technique has given good results for several years, recent advances in personal computers have considerably improved the results. Fits have been run for the differential bobbin probe that have included over 95,000 different sets of property values and their corresponding readings. Multiple property fits of these readings have given defect size fits with root-mean-square errors under 5% of the wall thickness for ASME Section XI standards. Although the actual measurement of the defect depths is not that good (without corrections), the signal to noise is very good, even at copper and magnetite interfaces. Different types of function fits have been tested for the various types of probes and defects, and optimum functions have been determined for each. More complex standards have been tested with larger sets of property values, up to 150,000. These use data taken with the Zetec MIZ-18, and the coefficients derived will be applicable to the data generated in field tests using this instrument.

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ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS

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Environmentally assisted cracking of piping and pressure vessels in light water reactors (LWRs) is an important concern as extended reactor lifetimes are envisaged. Intergranular stress corrosion cracking (SCC) of austenitic stainless steel (SS) piping in boiling water reactors (BWRs) has required research, inspection, and mitigation programs that have cost several billion dollars. More recently, cracks that initiated in sensitized stainless steels have propagated into low-alloy ferritic steels in BWR pressure vessels, and extensive cracking has occurred in upper shell-to-transition-cone girth welds in PWR steam generator vessels.

Stress Corrosion Cracking of Ferritic Steels

Fracture-mechanics crack-growth-rate (CGR) tests have been performed on a high-sulfur heat (0.018% sulfur) of A533-Gr B pressure vessel steel. In addition to conventional specimens, other specimens plated with either nickel or gold and a composite specimen of A533-Gr B/Inconel-182/Inconel-600 were tested. The composite specimen was also plated with nickel to better simulate a clad ferritic steel vessel, where the only low-alloy steel exposed to the environment is the crack surface. A comparison of the conventional and plated specimens provides insight into whether electron transfer through the oxide film on the bulk surface of the ferritic steel is an important rate-limiting process. The results will also indicate whether data obtained on specimens without cladding can be used to analyze the behavior of a clad ferritic vessel.

The conventional and plated specimens were tested in simulated BWR environments under cyclic loads at 0.08 Hz with load ratios R of 0.25–0.95 and maximum stress intensities, K_{max}, of 20–75 $MPa{\cdot}m^{1/2}$. The CGRs in the gold- and nickel-plated specimens were, in general, significantly higher than those in the conventional specimen, although the actual differences in CGRs depended on the specific loading conditions. Under high R loads (0.8–0.95), no steady-state crack growth occurred in either the conventional or the plated specimens at maximum stress intensity values below 39 $MPa{\cdot}m^{1/2}$. Additional tests have begun on nickel-chromium-plated specimens, which better simulate the oxide films associated with austenitic SS cladding.

In the composite specimen, crack growth occurred in the In-182 weld metal at a rate of 2.5×10^{-10} $m{\cdot}s^{-1}$ under high R loading (0.95) with K_{max} ≈30 $MPa{\cdot}m^{1/2}$, i.e., virtually the same rate as in sensitized Type 304 SS under similar loading conditions. No steady-state crack growth was observed in the ferritic steel at this stress intensity level. K_{max} was then increased in steps of ≈5 $MPa{\cdot}m^{1/2}$ to determine a threshold level for steady-state cracking in the ferritic steel. Even at a K_{max} of 80 $MPa{\cdot}m^{1/2}$, the apparent steady-state crack growth rate was less than 2 x 10^{-10} m·s^{-1}. However, according to the compliance measurements made during the test, each step increase in K_{max} produced a transient spurt of crack growth, which varied in length from 0.4 to 1.9 mm. At the end of the test, the specimen was examined metallographically. The final crack length was consistent with that obtained from the compliance measurements. The mode of cracking was transgranular. Additional tests are in progress to better determine the dependence of the transient crack growth on the loading history.

Fatigue of Type 316NG SS

Fatigue tests on Type 316NG SS in a simulated BWR environment showed that at strain ranges of >0.3%, lives in the environment were shorter than those in air at 288°C. The decrease in life N depends on the strain rate $\dot{\epsilon}$, (or frequency). At a strain rate of $\approx 10^{-4}$ s^{-1}, lives in the environment are less than one-third of those in air. Over the range of strain rates examined, this dependence can be described in terms of a power law, $N \sim \dot{\epsilon}^{0.18}$. In most cases, lives were above the ASME Section III design curve even at the lowest strain rates tested. However, the life for a strain range of 0.3% at $\dot{\epsilon} = 10^{-4}$ s^{-1} falls below the design curve. In contrast to behavior at higher strain ranges, lives in the environment at a strain range of 0.25% are more than four times as long as the corresponding life in air. This change in behavior roughly corresponds to the transition between the limitation of fatigue life by crack growth (relatively high strain range, short fatigue life) and limitation by crack initiation (low strain range, long fatigue life).

Stress Corrosion Cracking of Alternative Materials

Previous CGR tests on a heat of Type 347 SS showed that this material was very resistant to SCC in simulated BWR environments. In additional tests on another heat of Type 347 SS, the CGRs were comparable to those observed in Type 316NG SS. However, the CGRs in Type 347 SS appear to depend strongly on heat treatment, e.g., the rates in a forged material subjected to slow cooling in air were comparable to those in Type 316NG SS, but the rates in the same material when solution-annealed and water-quenched were lower by a factor of five. Neither heat treatment produced any measurable sensitization. The effect of the heat treatment on SCC may be attributed to possible effects on crack-tip deformation rates.

Previous tests have demonstrated the very low susceptibility of high-ferrite weld metal to SCC. In subsequent tests, the effect of ferrite content on the SCC of cast SS was evaluated. As expected, susceptibility to SCC is strongly dependent on ferrite content. In CGR tests in simulated BWR water, a cast CF-3M specimen with 5% ferrite cracked at an average rate of 9.3 x 10^{-11} m·s^{-1} at a K_{max} of 22 MPa·m$^{1/2}$, whereas a CF-3 specimen with 15.6% ferrite cracked at a rate of 1.0 x 10^{-11} m·s^{-1} at a K_{max} of 29 MPa·m$^{1/2}$. Thus, the CGR is almost an order of magnitude slower in the high-ferrite material even though the stress intensity is significantly higher. The CGR of the high-ferrite CF-3 SS is also about an order of magnitude slower than that of the slow-cooled Type 347 SS specimen and is about one-half of that of the Type 347 SS in the solution-annealed and water-quenched condition, which exhibited greater resistance to SCC.

ESTIMATION OF FRACTURE TOUGHNESS OF CAST STAINLESS STEELS IN LWR SYSTEMS

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A program is being conducted to investigate the low-temperature embrittlement of cast duplex stainless steels under light water reactor (LWR) operating conditions and to evaluate possible remedies for the embrittlement problem in existing and future plants. The scope of the investigation includes the following goals: (1) develop a methodology and correlations for predicting the toughness loss suffered by cast stainless steel components during normal and extended life of LWRs, (2) validate the simulation of in-reactor degradation by accelerated aging, and (3) establish the effects of key compositional and metallurgical variables on the kinetics and extent of embrittlement.

Microstructural and mechanical property data are being obtained on 25 experimental heats (static-cast keel blocks and slabs) and 6 commercial heats (centrifugally cast pipes and a static-cast pump impeller and pump casing ring), as well as on reactor-aged material of CF-3, CF-8, and CF-8M grades of cast stainless steel. The ferrite content of the cast materials ranges from 3 to 30%.

Charpy-impact, tensile, and J-R curve tests have been conducted on several experimental and commercial heats of cast stainless steel that were aged up to 30,000 h at temperatures of 290 to 400°C (≈555 to 840°F). The results indicate that thermal aging at these temperatures increases the tensile strength and decreases the impact energy and fracture toughness of the steels. In general, the low-carbon CF-3 steels are the most resistant to embrittlement, and the molybdenum-containing high-carbon CF-8M steels are the least resistant. Ferrite morphology has a strong effect on the degree or extent of embrittlement, and the kinetics of embrittlement can vary significantly with small changes in the constituent elements of the cast material.

Mechanical property data, including results from studies at Framatome, Electricité de France, Central Electricity Generation Board (U.K.), and George Fischer Co. (Switzerland) have been analyzed to develop a procedure and correlations for predicting fracture toughness and tensile properties of aged cast stainless steels from known material parameters. The "saturation" fracture toughness of a specific cast stainless steel, i.e., the minimum fracture toughness that would ever be achieved for the material after long-term service, is estimated from the degree of embrittlement at saturation. Degree of embrittlement is characterized in terms of room-temperature Charpy-impact energy. The variation of the impact energy at saturation for different materials is described in terms of a material parameter Φ, which is determined from the chemical composition and ferrite morphology. The fracture toughness J-R curve for the material is then obtained from correlations between room-temperature Charpy-impact energy and fracture toughness. Fracture toughness as a function of time and temperature of reactor service is estimated from the kinetics of embrittlement, which is determined from the chemical composition. Correlations for estimating the tensile properties of aged cast stainless steels are also presented.

IRRADIATION-INDUCED SENSITIZATION IN AUSTENITIC STAINLESS STEEL OF IN-CORE COMPONENTS

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A program is being conducted to develop an understanding of the metallurgical phenomena that occur in BWR and PWR structural components as a consequence of extended service within and outside the irradiation environment, and to assess the impact of these phenomena on structural integrity. Microstructural changes such as radiation-induced segregation (RIS) and depletion of alloying and impurity elements may have significant effects on the integrity of structural materials. These changes can influence mechanical and corrosion properties and can increase susceptibility to stress corrosion cracking. In recent years, failures of reactor-core internal components after accumulation of relatively high fluence have increased in both BWRs and PWRs. It appears that as plants age and the neutron fluence increases, a wide variety of apparently nonsensitized austenitic materials become susceptible to intergranular failure. Although most failed components can be replaced, some components, such as BWR top guide, shroud, and core plate, would be very difficult to replace.

RIS, which leads to an irradiation-induced sensitization of grain boundaries in austenitic stainless steels (SS), is known to be strongly dependent on irradiation temperature, fast-neutron flux (dose rate), and total fluence, and hence is difficult to simulate in accelerated test-reactor experiments. The failures of austenitic stainless steels and high-nickel alloys after accumulation of high fluence have been attributed to irradiation-assisted stress corrosion cracking (IASCC), in which irradiation-induced sensitization of grain boundaries by segregation and depletion of elements such as Si, P, S, Cr, and Ni has been implicated. However, the identity of the elements that segregate or become depleted and the extent to which irradiation-induced sensitization contributes to enhanced susceptibility to IASCC are not clear. These must be better understood in order to assess potential problems as reactors approach a significant fraction of their design life. In the present study, Type 304 SS components were obtained from a number of operating BWRs and examined to determine the nature and extent of irradiation-induced sensitization and to correlate it with susceptibility of the components to intergranular failure.

High- and commercial-purity heats of Type 304 SS, irradiated in different BWRs, were examined by Auger electron spectroscopy to characterize grain boundary segregation and depletion of alloying and impurity elements. The materials were sectioned from BWR control blade sheaths and absorber rod tubes after irradiation to fluence levels of 2×10^{20} to 2×10^{21} n/cm^2. The specimens were fractured in-situ in the ultrahigh-vacuum environment of the Auger electron microscope after hydrogen charging for 48 to 50 h. The fracture mode in the high-purity tubes was predominantly ductile regardless of fluence, and regions of intergranular failure were negligible. In contrast, the fracture surfaces in the commercial-purity tubes at comparable fluences showed significantly more intergranular failure.

Ductile-intergranular fracture behavior was correlated with results from Auger characterization of the grain boundary segregation and depletion. In the high- and commercial-purity irradiated absorber tubes, as well as in an unirradiated Type 304 SS, ductile fracture surfaces showed relatively high levels of C, O, and S contamination from the vacuum environment of the Auger microscope. The intergranular fracture surfaces in the irradiated commercial-purity material were characterized by lower contamination levels and with relatively high levels of Si, P, and Ni segregation. The intergranular fractures were also associated with an Auger energy peak at 59 eV, which indicates either segregation of an unidentified element or formation of an unidentified compound on the grain boundary. The intensity of the 59-eV peak increased with increasing Ni and Si segregation and may be associated with the formation of a thin film of a Ni-Si compound on grain boundaries. No correlation was observed between fracture behavior and Cr depletion or S segregation. In contrast to that in the commercial-purity material, segregation of the impurity elements in the high-purity material was negligible even at a fluence of 1.4×10^{21} n/cm^2. These observations on the roles of Si, S, and the unidentified compound in irradiation-induced sensitization in Type 304 SS are different than those reported previously for test-reactor-irradiated materials. The present results are also discussed in relation to initial data from slow-strain-rate-tensile tests in simulated BWR water.

SHORT CRACKS IN PIPING AND PIPING WELDS

by

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This program started on March 23, 1990, and has a duration of 4 years. The objective of the program is to develop and verify analyses by using existing and new experimental data for circumferentially cracked pipes, so modifications and improvements can be made to LBB and in-service flaw evaluation criteria. There are 7 technical tasks with the following specific objectives. In general, they deal with circumferentially cracked straight pipe under quasi-static loading.

Task 1, Short TWC Pipe Evaluations, has the objective to modify and verify analyses for short through-wall-cracked (TWC) pipe using existing and new data on large diameter pipe. The regulatory impact of these results can help to refine analyses that have been used for LBB crack sizes in large diameter pipe. There are four subtasks in this task: (1.1) material characterization of pipes to be tested, (1.2) facility modifications for large diameter pipe experiments, (1.3) large diameter pipe experiments, and (1.4) analysis modification and verifications. FY-90 efforts are: assist in defining weld procedures to be used in the pipe fracture experiments, and start tensile and J-R curve toughness testing on the welds in Subtask (1.1), initiate modifications to test facility in Subtask (1.2), conduct three large pipe experiments in Subtask (1.3), and start improvements on short TWC analyses and comparisons to existing data in Subtask (1.4).

Task 2, Short SC Pipe Evaluations, has the objective to modify and verify analyses for short surface-cracked (SC) pipe using existing and new data on large diameter pipe. These results have the regulatory impact to verify and possibly refine analyses that have been used for in-service flaw evaluations such as those in ASME Section XI. There are four subtasks: (2.1) material characterization of pipes to be tested, (2.2) small diameter pipe experiments, (2.3) large diameter pipe experiments, and (2.4) analysis modification and verifications. FY-90 efforts involve: defining weld procedures to be used in the pipe fracture experiments, start tensile and J-R curve toughness testing on the welds in Subtask (2.1), start tests on 6-inch-diameter pipe in Subtask (2.2), and start improvements on SC.TNP and SC.TKP internal surface crack analyses and comparisons to existing data in Subtask (2.4).

Task 3, Bi-metallic Cracked Pipe Evaluations, has the objective to develop experimental data and analyses for cracks in bi-metallic welded pipe. These results will help to establish criteria for LBB and in-service flaw evaluation criteria for cracks in bi-metallic welds. There are three subtasks in this task: (3.1) material characterization of a bi-metallic weld, (3.2) large diameter pipe experiments, and (3.3) analysis developments and verifications. No efforts are planned for FY-90.

Task 4, Dynamic Strain Aging and Crack Jump Evaluations, has the objective to develop a screening criterion to assess which ferritic steels are susceptible to dynamic strain aging and unstable crack growth at LWR temperatures. This methodology is applicable to both LBB and in-service flaw evaluations for ferritic pipe. There are three subtasks: (4.1) material characterization, (4.2) laboratory specimen tests analyses, and (4.3) pipe fracture analyses. FY-90 efforts will involve starting to conduct tensile, hardness, and C(T) tests on five different ferritic steels that exhibit different degrees of dynamic strain aging in Subtask (4.1). The hardness tests will be conducted at various temperatures to see if the ratio of hardness at LWR to room temperature can be used as a simple screening criterion.

Task 5, Anisotropic Fracture Evaluations, has the objective to assess if anisotropic fracture toughness properties can cause failure stresses to be less than those typically calculated by current LBB analyses. This has a direct impact on LBB analysis procedures. There are two subtasks: (5.1) Parametric analyses and (5.2) screening criteria development. FY-90 efforts which are in Subtask (5.1) involve: conducting an FEM analysis to determine the crack driving force relations for a crack that starts growing in a helical angle from a circumferential crack, conducting tensile and C(T) tests with different angled crack orientations (this will be done for one pipe where angled crack growth occurred in a prior pipe experiment), and an FEM analysis will be conducted for modelling an angled growing cracked pipe experiment from this program.

Task 6, Crack-Opening-Area Evaluations, has the objective to improve the crack-opening-area predictions used in LBB leak-rate analyses. These results will help to refine analyses that have been used for leak-rate analyses in LBB evaluations. There are four subtasks: (6.1) combined load prediction improvements, (6.2) improvements for short TWC, (6.3) improvements for cracks in welds, and (6.4) SQUIRT code modifications. FY-90 efforts are in Subtask (6.1) and involve starting improvements for combined loading. This will involve implementing a crack-opening-displacement prediction capability in the LBB.ENG TWC fracture analysis for combined loading.

Task 7, NRCPIPE Improvements, has the objective to improve the NRCPIPE PC code for circumferential TWC pipe fracture analysis, and to create a surface crack version. These efforts will incorporate the analysis developments from this program into a user-friendly PC code. There are four subtasks: (7.1) efficiency improvements in current version, (7.2) improvements for short TWC and cracks in welds, (7.3) surface crack version of PC code, and (7.4) manuals for codes. FY-90 efforts involve making efficiency improvements in the current NRCPIPE code. A separate code for Ramberg-Osgood fitting of stress-strain data will be made with options for fitting the data over various strain ranges as part of Subtask (7.1).

There is also a separate task to develop international cooperation, interact with Section XI of the ASME code, and perform program management functions. Cooperative efforts are underway with several international organizations (France, Italy, Japan, and W. Germany) in exchanging analysis results and experimental data.

INTERFACING SYSTEMS LOCA (ISLOCA) PRESSURE CAPACITY METHODOLOGY AND DAVIS-BESSE RESULTS

by
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A loss of coolant accident resulting from the potential overpressurization by reactor coolant fluid of a system designed for low-pressure, low-temperature service has been shown to be a significant contributor to risk in several probabilistic risk assessments. In this paper, the methodology developed to assess the probability of failure as a function of internal pressure is presented, and results developed for the controlling failure modes and locations of four fluid systems at the Davis-Besse Plant are shown. Included in this evaluation are the tanks, heat exchangers, filters, pumps, valves, and flanged connections for each system. The variability in the probability of failure is included, and the estimated leak rates or leak areas are given for the controlling modes of failure. For this evaluation, all failures are based on quasistatic pressures since the probability of dynamic effects resulting from such causes as water hammer have been initially judged to be negligible for the Davis-Besse plant ISLOCA.

The pressure capacities of the pipes and vessels are evaluated using limit state analyses for the various failure modes considered. The capacities are dependent on several factors, including the material properties, modeling assumptions, and the postulated failure criteria. A major source of uncertainty in the failure criteria is the expected strain resulting in failure. All welds are full penetration and the probability of failure at membrane strains below yield is considered to be quite low. On the other hand, biaxial strains and gage length effects, as well as strain concentrations and bending, significantly reduces the expected hoop strain at failure when compared to elongation data developed from standard specimen ultimate tests. Since test data from vessel tests are extremely limited, considerable variability is introduced not only in the failure criteria, but in analytical modeling and other assumptions. In particular, the limited data that do exist are related to finite length cylinders with internal pressure loading only, and no test results are available for such effects as thermal or bending strains in pipe, strain concentrations at branch connections, or nozzle loads on tanks. Since many of the base parameters are random and the methods used to evaluate the capacities are subject to some uncertainty, the pressure capacity for any failure mode is also considered to be a random variable.

Design stresses in piping systems and pressure vessels include provision for stresses resulting from deadweight, thermal expansion, nozzle loads, earthquake and other loads as well as internal pressure. Stresses from other than internal pressure may constitute a major or even the controlling portion of the design allowable stress. At overpressure conditions, however, the percentage of available strength required to resist the nonpressure loads may be expected to decrease (i.e., the deadweight stress in a piping system does not increase with an increase in pressure, and while thermal stresses may increase above the design case, they are not expected to be the controlling load). Thus, the failure criteria developed for pipe and pressure vessel burst is concentrated on the internal pressure effects, as reflected in the hoop stress in a cylinder, while still retaining some consideration for other loads such as bending or branch connections in pipe, or nozzle loads in tanks. The goal was to develop criteria which could reasonably include these additional effects without requiring a detailed evaluation in order to obtain the actual magnitudes of the bending stresses, etc., at every location and temperature.

A different approach must be used to evaluate the pressure capacities for gasketed flange connections, valves, and pumps. Unlike the failure modes for piping, vessels, and heat exchangers, which lend themselves to evaluation by conventional structural mechanics techniques, the failure modes for gasketed flange connections, valves, and pumps are very complex and evaluation must rely primarily on the results from ongoing gasket research test programs and available vendor information and test data. The pressure capacity and the associated leak rate of gasketed flanges depend on a number of parameters including bolt material characteristics and bolt preload, flange flexibility, initial gasket stress and relaxation, and gasket stiffness characteristics.

It is assumed that the pressure capacities have a lognormal distribution. This assumption is made because a lognormal distribution has been shown to be a valid description of the variability in material strengths. In addition, for a random variable that can be expressed as the product and quotient of several random variables, the distribution of the dependent variable tends to be lognormal regardless of the distributions of the independent base variables.

Uncertainties will exist in the estimated pressure capacities due to differences between the analytical idealization of the structure and the real conditions. There are numerous possible sources of modeling uncertainties. Examples of the sources of modeling uncertainties include: assumptions used to develop the internal force distributions, failure criteria, and the use of empirical formulae. Moreover, since the uncertainties are dependent on the particular failure mode under consideration, they must be evaluated on a case-by-case basis.

EVALUATION OF THE LEAKAGE BEHAVIOR OF PRESSURE-UNSEATING EQUIPMENT HATCHES AND DRYWELL HEADS[1]

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Summary

The primary purpose of the paper is to present the results of a recent research program to investigate the leakage behavior of pressure unseating equipment hatches. Because of the similarity in the sealing mechanism between unseating equipment hatches and drywell heads, the prediction equations that have been developed and validated during this program should also be valid for the prediction of the leakage behavior of drywell heads. The research activities described herein are a part of the Containment Integrity Programs, which are managed by Sandia National Laboratories for the U.S. Nuclear Regulatory Commission.

The overall goal of the Containment Integrity Programs is to develop a set of test validated methods to predict the ultimate pressure capacity, at elevated temperature, of the containment pressure boundary. These methods will be used to assess the performance of the containment in the unlikely event of a severe accident. As a part of the Containment Integrity Programs, a series of scale model containment tests have been conducted at Sandia including a 1:8-scale steel model and a 1:6-scale reinforced concrete containment model. Also, a 1:10-scale prestressed containment model was recently tested in the United Kingdom.

Because of the reduced scale and limited number of tests, the model tests could not include an adequate representation of the many different designs and types of penetrations. Thus, separate test programs have been conducted to better investigate the ultimate behavior of typical penetration designs. Completed penetration research programs include tests of a personnel airlock, electrical penetration assemblies (EPAs), compression seals and gaskets, and inflatable seals. Also, a series of tests are under way to investigate the severe accident behavior of the bellows that are used at process piping penetrations in steel containments. The results of the completed penetration programs, as well as the model tests, are documented in reports to the Nuclear Regulatory Commission.

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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.

Recently, a series of tests were conducted on the pressure unseating equipment hatch of the 1:6-scale reinforced concrete containment model. This hatch was virtually undamaged by the overpressurization test of the model; thus, it served as an ideal specimen for further investigation of pressure unseating hatch covers. The tongue and groove sealing configuration used in the hatch is typical of many unseating equipment hatch and drywell head designs in the United States. The results of the equipment hatch tests and the developed analytical method for predicting leakage of unseating hatch covers are the primary emphasis of the paper.

The tests were designed to provide engineering data that could be used to validate analytical methods to predict leakage from pressure unseating equipment hatches and drywell heads. Parameters that were varied during the tests include bolt preload, bolt stiffness, gasket material, gasket aging (thermal only), and temperature (up to 700°F). The effect of each of these parameters on the containment pressure at which leakage initiates and the leak rates that arise was observed.

An analytical method has been developed to estimate the pressure and temperature at which significant leakage first occurs and the rate of leakage for pressure and temperature above this level. The test results were used to improve and validate the analytical method, which can be broken into the following three steps:

(1) The structural response, in particular, the separation displacement (relative motion of the sealing surfaces) is determined based on a strength of materials approach.

(2) The maximum separation displacement for which the gasket can prevent leakage is evaluated using an empirical parameter for gasket performance, S_p, which is a measure of the available springback. S_p depends on the sealing configuration and geometry, aging history of the gasket, gasket material, and accident temperature.

(3) Risk significant leak rates are calculated from fluid mechanics equations for choked flow through a duct of known area, where the area is determined from steps 1 and 2.

Comparison of the test results to the predicted behavior shows that, although the actual hatch behavior is not uniform, the average structural response of the hatch (around the circumference) can be reasonably predicted with the proposed method. The containment pressure at which the bolt preload is overcome (loss of metal-to-metal contact) and the containment pressure at the onset of significant leakage also compares reasonably well with predictions. However, the agreement was not as good between the predicted and measured leak rates that occurred after leakage onset. The error in the predicted leak rate appears to be caused by the difference between the assumed and actual leak areas. The leak area depends on the degree of gasket degradation, the relative position of the gasket within the groove at the time of leakage, and the amount of 'out-of-flatness' or warping of the hatch cover. These parameters are difficult to quantify and tend to cause varying leakage openings around the circumference, which violates the assumption in the leakage calculations that the leak area is uniform around the circumference.

"LARGE - SCALE SEISMIC TEST PROGRAM at HUALIEN, TAIWAN"

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SUMMARY

The Large-Scale Seismic Test (LSST) Program at Hualien, Taiwan is a follow-on to the soil-structure interaction (SSI) experiments at Lotung, Taiwan. The planned SSI studies will be performed at a stiff soil site in Hualien, Taiwan that historically has had slightly more destructive earthquakes in the past than Lotung. The objectives of the LSST project is as follows:

- To obtain earthquake-induced SSI data at a stiff soil site having similar prototypical nuclear power plant soil conditions.
- To confirm the findings and methodologies validated against the Lotung soft soil SSI data for prototypical plant condition applications.
- To further validate the technical basis of realistic SSI analysis approaches.
- To further support the resolution of USI A-40 "Seismic Design Criteria" issue.

These objectives will be accomplished through an integrated and carefully planned experimental program consisting of; soil characterization, test model design and field construction, instrumentation layout and deployment, in-situ geophysical information collection, forced vibration test, and synthesis of results and findings. The LSST is a joint effort among many interested parties. EPRI and Taipower are the organizers of the program and have the lead in planning and managing the program.

Other organizations participating in the LSST program are U.S. Nuclear Regulatory Commission (NRC), the Central Reasearch Institute of Electric Power Industry (CRIEPI), the Tokyo Electric Power Company (TEPCO), the Commissariat A L'Energie (CEA), Electricite de France (EdF)and Framotome. The LSST was initiated in January, 1990 and is envisioned to be five years in duration.

MANAGEMENT OF THE AGING OF CRITICAL SAFETY-RELATED CONCRETE STRUCTURES IN LIGHT-WATER REACTOR PLANTS*

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The Structural Aging Program has the overall objective of preparing a report which provides USNRC license reviewers and licensees with: (1) identification and evaluation of the degradation processes that affect the performance of structural components; (2) issues to be addressed under nuclear power plant continued service reviews, as well as criteria, and their bases, for resolution of these issues; (3) identification and evaluation of relevant inservice inspection (or structural assessment) and repair programs in use, or needed; and (4) quantitative methodologies for assessing current, or predicting future, structural safety margins. The results of this program will provide an improved basis for the USNRC staff to permit continued operation near, at, or beyond the nominal 40-year design life of a nuclear power plant. The program has three technical tasks which address concrete material systems: materials property database, structural component assessment/repair technology, and quantitative methodology for continued service determinations.

The objective of the materials property database task is to develop a hard copy handbook and an electronically accessible version (Structural Materials Information Center) containing information on the time variation of concrete and other structural material properties. The Structural Materials Information Center has use in the prediction of deterioration of critical structural components in nuclear power plants. Two complementary database formats have been developed: *Structural Materials Handbook* (hardcopy version), and *Structural Materials Electronic Database*. Presently, the Structural Materials Information Center contains data on the performance of concrete, conventional steel reinforcement, prestressing and structural steel materials.

The structural component assessment/repair task has objectives of: (1) developing a systematic methodology which can be used to make quantitative assessments of the presence, magnitude, and significance of any environmental stressors or aging factors which can adversely impact the durability of safety-related concrete structures in NPPs; and (2) providing recommended inservice inspection or sampling procedures which can be utilized to develop the data required both for evaluating the current condition of concrete structures as well as trending the performance of these components for continuing service assessments. A structural component significance assessment methodology for concrete structures in NPPs has been developed. The methodology utilizes numerical ranking and relative weighting procedures to evaluate and categorize concrete structures in NPPs by their functional importance, safety significance, environmental exposure, and function of their subelemental parts. The methodology has been applied to three commercial NPPs.

The quantitative methodology for continued service determinations task has the objective of developing a procedure which can be used for performing current condition assessments and making reliability-based life predictions of critical concrete safety-related components in NPPs. Probabalistic models have been developed to evaluate the reliability over time of concrete containment structures. Structural loads arising from service, extreme environmental and accidental conditions are modeled as stochastic processes as well as environmental conditions under which the plant structures operate. Initial testing of the reliability-based life-prediction models developed included two mechanisms: (1) corrosion of steel reinforcement and (2) detensioning of the tendons in prestressed concrete containments.

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Probabilistic Methods for Condition Assessment and Life Prediction of Concrete Structures in Nuclear Power Plants

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Introduction

Nuclear power plants are granted operating licenses for a finite period of time, often 40 years. During the next 15 years, the operating licenses for a number of nuclear plants will expire. Faced with the prospect of having to replace the lost generating capacity from other sources and substantial shutdown and decommissioning costs, many utilities are expected to seek extensions to their nuclear power plant operating licenses. A major concern in evaluating such applications is in ensuring that the capacity of the safety-related systems to mitigate extreme events has not deteriorated due to structural aging during the previous service history. Studies have shown that although major mechanical and electrical equipment in a plant could be replaced, if necessary, replacement of or major repairs to certain concrete structures in the plant would be economically unfeasible. Thus, an application must be supported by evidence that safety-related concrete structures in their current (service) condition are able to withstand future extreme events within the license extension period with a level of reliability sufficient for public health and safety. The goal of the current research is to develop a methodology to facilitate quantitative assessments of current and future structural reliability and performance of concrete structures in nuclear power plants. This methodology takes into account the stochastic nature of past and future loads due to operating conditions and the environment, randomness in strength and in degradation processes, and uncertainty in nondestructive evaluation techniques. This research is in support of the NRC Structural Aging Program.

Technical Approach

The research has four components: (1) Identification of existing condition assessment methods and damage or life prediction models; (2) Assembly of pertinent data for use in these models; (3) Development of reliability-based condition assessment methods for analysis of current and future reliability; and (4) Validation of methodology using laboratory or prototypical structure data.

Research to date has focused on the development of probabilistic models to assess time-dependent reliability and deterioration of concrete structures subjected to stochastic loads (component 3). To illustrate the basic concepts, consider a structural component subjected to a sequence of discrete stochastic load events. Most design-basis events occur infrequently in time and have a relatively short duration. The magnitudes, S_k, of the load events in the sequence are assumed to be identically distributed and statistically independent random variables described by a cumulative probability distribution function, $F_S(x)$. Assume, for the present, that the initial strength of the structural component, R, is deterministic and equal to r, and that the strength of the component deteriorates with time (due, e.g., to corrosion of reinforcement) as,

$$R = r\,g(t) \tag{1}$$

in which g(t) = fraction of the strength remaining at time t. If n events occur within time interval (0,t) at deterministic times t_k, k = 1,...n, the reliability function, L(t), defined as the probability of survival through time t, is

$$L(t) = P[S_1 < g_1 r \cap S_2 < g_2 r \cap \cdots \cap S_n < g_n r] \tag{2}$$

$$L(t) = \prod_{k=1}^{n} F_S(g_k r) \tag{3}$$

in which $g_k = g(t_k)$, the fraction of strength remaining at time of load occurrence t_k. In general, the loads occur randomly at times $T_1, T_2, \ldots, T_n$. A Poisson occurrence model can be used to model the occurrence of rare events such as extreme environmental loads, accidents, etc. Accordingly, the deterioration function, g_k, is random as well. As a final step, the conditioning on initial strength, R = r, is removed to take the randomness in strength into account; accordingly,

$$L(t) = \int_0^\infty L(t|r)\, f_R(r)\, dr \tag{4}$$

in which $f_R(r)$ is the density function of R. In the absence of periodic inspection or maintenance, L(t) is a decreasing function of time. The probability of unacceptable performance is 1 - L(t). Work is in progress to extend the methodology to systems of components.

Summary

Appropriate degradation functions and load process statistics must be identified to utilize the above methodology. It is expected that additional information will become available later in the Structural Aging Program from this and other tasks. Reliability functions can be used to determine the extent to which structural reliability deteriorates over an increment of time. This information could be used to select appropriate licensing extensions or to determine required intervals of inspection and maintenance necessary to maintain reliability at an acceptable level.

DEVELOPMENT OF THE STRUCTURAL MATERIALS INFORMATION CENTER*

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ABSTRACT

The U.S. Nuclear Regulatory Commission has initiated a Structural Aging Program at the Oak Ridge National Laboratory to identify potential structural safety issues related to continued service of nuclear power plants and to establish criteria for evaluating and resolving these issues. One of the tasks in this program focuses on the establishment of a Structural Materials Information Center where data and information on the time variation of concrete and other structural material properties under the influence of pertinent environmental stressors and aging factors are being collected and assembled into a database. This database will be used to assist in the prediction of potential long-term deterioration of critical structural components in nuclear power plants and to establish limits on hostile environmental exposure for these structures and materials.

Two complementary database formats have been developed. The *Structural Materials Handbook* is an expandable, hard copy handbook that contains complete sets of data and information for selected portland cement concrete, metallic reinforcement, prestressing tendon, and structural steel materials. Baseline data, reference properties and environmental information are presented in the handbook as tables, notes and graphs. The handbook, which will be published in four volumes, serves as the reference information source for the electronic database. The *Structural Materials Electronic Database* is accessible by an IBM-compatible personal computer and provides an efficient means for searching the various database files to locate materials with similar properties. Properties will be reported in the International System of Units (SI) and in customary units whenever possible.

The database formats have been developed to accommodate data and information on the time-variation of concrete and other structural material properties. Open literature reports and results from testing of prototypical samples are being used as reference sources. To date, the database includes information on concrete, reinforcement, prestressing and structural steel materials.

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US NRC STRUCTURAL DAMPING RESEARCH PROGRAM

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The US NRC Structure Damping Research Program has the following objectives:

Assess the structural damping values used for the design of nuclear power plants, Regulatory Guide (RG) 1.61, or evaluation of existing plants.

Investigate the treatment of proportional and non-proportional damping in dynamic analysis methods, assess the adequacy of these treatments, and propose guidelines on their use.

Investigate the treatment of energy dissipation in nonlinear response analyses. Such analyses have applicability to the evaluation of structures excited by motions beyond the design basis, e.g., seismic PRA and margin reviews. Quantify the effect on structure response of different treatments.

Perform a case history study of the Watsonville telephone building which experienced the Morgan Hill Earthquake of 1984 and the Loma Prieta Earthquake of 1989 including modeling of the building by design analysis, equivalent elastic, and systems identification methods. Damping and frequency characteristics will be evaluated for the two earthquake levels.

These objectives correspond to separate tasks, whose status is summarized. Significant progress has been achieved for Tasks A and B as described herein. Tasks C and D will be described in detail in later publications.

Task A focuses on the first objective, to review the adequacy of current licensing provisions on damping specified in RG 1.61, and recommend revisions or additions as necessary. Specific activities for Task A include: collection of available damping data for actual structures and scale-model specimens; compilation and evaluation of the data; and review, assessment, and proposed revision of RG 1.61.

A literature survey was performed to identify potential sources of measured damping data from actual full-scale structures or laboratory specimens subjected to earthquakes, vibration tests, and other types of input. Bibliographies, engineering literature data bases, conference proceedings, etc. were reviewed to identify individual reports and papers which may include structural damping values. Slightly over 1,000 references were identified. Approximately 75% of these references were surveyed to verify that they contain relevant data. Reported damping values, and candidate-important parameters, were extracted from the available references and compiled in a data base. Over 30 parameters accounting for structure, foundation, input, response, and other characteristics that may influence the structural damping data were identified. Structure parameters include construction material, dynamic load-resisting system, geometry and configuration, mass, etc. Damping values obtained from actual structures may include soil-structure interaction effects. Foundation parameters which may influence SSI include foundation type, geometry, soil shear wave velocity, etc. Important input and response parameters include input type, data reduction method, input and response accelerations, stress level, and modal frequency. Not all of the data collected will be appropriate for inclusion in the final data base. The data collected are being screened and sorted according to quality and level of documentation available. The final data base will be data considered to be of acceptable quality for which sufficient documentation is available to identify the dominant parameters.

Once the final damping data base is assembled, regression analysis will be performed to determine what parameters significantly influence structure damping, and to quantify damping in terms of these parameters. Limited, preliminary reviews indicate that stress/response level and SSI effects may be important to concrete structures. Stress/response level and the presence of nonstructural elements may be significant to steel structures. The results of the statistical analysis will be used to assess the adequacy of current RG 1.61 criteria. Revisions will be recommended, as appropriate.

Task B consists of the identification, review and evaluation of analysis techniques that are currently available for use in the treatment of non-proportionally damped structural systems. Non-proportional damping is a form of linear viscous damping which produces a damping matrix not uncoupled by the undamped modal coordinates. This class of systems is divided into two categories: assemblies of interacting subsystems having different damping characteristics; and soil-structure systems for which radiation damping as well as material damping is present.

The study of the treatment of interacting subsystems consisted of first conducting a literature search to identify currently used methods, both rigorous and approximate. From the methods identified, three approximate methods were selected for evaluation--the commonly used mass-proportional and stiffness-proportional composite modal damping methods, and a method in which the off-diagonal terms of the generalized damping matrix obtained from the undamped modal coordinates are neglected. A rigorous method provided a basis for the evaluation of the approximate methods. Two benchmark problems were evaluated. A simplified model of a steel-frame super-structure supported by a reinforced concrete sub-structure was sufficiently small. The model for this problem was sufficiently small (4 dynamic DOFs) that an extensive parameter study was performed--varying ratio of subsystem masses, subsystem frequencies, and damping ratios of each subsystem. A model of a reactor building internal structure and a main steam line supported at a number of points comprised the second benchmark. This problem was run for a limited number of parameter variations. Both problems were studied using input motions consistent with US NRC RG 1.60. Responses in the structures calculated by the approximate methods were compared with those from the rigorous method to evaluate the approximate techniques.

A PERSONAL COMPUTER CODE FOR SEISMIC EVALUATIONS OF NUCLEAR POWER PLANT FACILITIES

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SUMMARY

The program CARES (Computer Analysis for Rapid Evaluation of Structures) is an integrated computational system being developed by Brookhaven National Laboratory(BNL) for the U.S. Nuclear Regulatory Commission. It is specifically designed to be a personal computer(PC) operated package which may be used to determine the validity and accuracy of analysis methodologies used for structural safety evaluations of nuclear power plants. CARES is structured in a modular format. Each module performs a specific type of analysis i.e., static or dynamic, linear or nonlinear, etc. This paper describes the various features which have been implemented into the Seismic Module of CARES.

The development of the Seismic Module is based on an approach which incorporate all major aspects of seismic analysis currently employed by the industry into an integrated system that allows for carrying out interactively computations of structural response to seismic motions. The process of seismic analysis of nuclear power plant structures and components generally involves the following steps:

- Definition of the design criteria at a given site.
- Evaluation of the free-field motion.
- Evaluation of the structural response and floor response spectra including soil-structure interaction effects.

The capabilities required to accomplish the above steps have been implemented into the Seismic Module with special emphasis on the areas of regulatory requirements pertaining to structural safety of nuclear power plants. The Seismic Module of CARES is organized in eight options which can be summarized below:

Option 1: General manager
-- deals with main computational aspects associated with input/output interfacing, time/frequency domain transformations and development of response spectra.

Option 2: Convolution analysis
-- performs convolution analysis for horizontally layered soil profiles and generates strain-compatible soil properties.

Option 3: SSI structural data preparation
-- prepares the stick model for the superstructure and the SSI model representing dynamic stiffnesses of the foundation.

Option 4: SSI input motion preparation
-- prepares Fourier components of free field motion for SSI analysis.

Option 5: SSI analysis
-- performs SSI analysis using inputs generated by Option 3 and 4 and producing Fourier components of the in-structure response time histories for selected locations. It also computes transfer functions which can be subsequently used to develop PSD's at various floor locations.

Option 6: Earthquake simulations
-- generates artificial time histories compatible to given design target response spectra, performing computation of power spectral density functions for given time histories and directly generating response spectrum from a target PSD function.

Option 7: PSD-related acceleration/spectra analysis
-- performes simulation of time histories consistent with given spectra and the minimum PSD requirements described in the recent SRP, Sec. 3.7.1 Rev.2(1989).

Option 8: Plot utility
-- Generates graphical presentations of time histories, response spectra, amplification functions and PSD spectra, as well as general 2-D x-y curve plots.

It should be pointed out that, although different phases of the seismic analysis can also be performed separately by other existing codes, the uniqueness of CARES. however, lies on its ability to perform all required steps of the seismic analysis in an integrated manner. It also has input-output interfacing compatibility which often poses difficulties in conversions between input and output data when different codes are used to perform different phases of seismic analysis. To this end, CARES becomes more reliable in terms of avoiding errors within the process of a complete seismic analysis. Finally, CARES is a completely interactive system with minimum number of input data. user friendly features as well as quick turn-around and provides comprehensive post-processing capability to display results graphically or in tabular form so that direct comparisons can be easily made.

LIFE ASSESSMENT PROCEDURE FOR LWR METAL CONTAINMENTS[a]

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Summary

The Aging Assessment and Mitigation Project is a part of the U.S. Nuclear Regulatory Commission (NRC) 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) components and structures and to develop life assessment procedures so that the impact of aging on the safe operation of nuclear power plants can be evaluated and managed. This paper reports the current accomplishments and future plans of the project, and presents a generic life assessment procedure for the LWR metal containments.

The major effort of the project consists of integrating, evaluating, and updating the technical information relevant to aging from current or completed NRC and industry research programs and from plant operating experience. The project is divided into five steps: a) identify and prioritize the major LWR components, b) identify degradation sites, mechanisms, stressors, and potential failure modes of each component and structure and then evaluate current inservice inspection methods, c) assess advanced inspection and monitoring methods and evaluate mitigating methods to reduce aging damage, d) develop life assessment models and procedures, and e) support the development of technical criteria for license renewal. The first two steps have been completed, which include qualitative aging assessment of twenty-two major components. A thorough assessment of emerging fatigue and acoustic monitoring methods and advanced material properties evaluation methods is being conducted as part of Step 3. Life assessment procedures have been developed for LWR cast stainless steel components and PWR steam generator tubes. The procedures for PWR reactor pressure vessels, LWR metal containments, and LWR concrete containments are currently being developed. The project results are being used to develop technical guidelines for making license renewal decisions.

The life assessment procedure is a systematic procedure to identify the actions required for an extended safe operation of a given major component during the current or renewed license period. The procedure is general in nature, allowing plant-specific evaluations to account for differences among individual power plants with a similar component. The procedure may be divided into four parts: a) assess the damage state of the component at the beginning of the next operating period, b) estimate the additional damage expected during the next operating period, c) evaluate component integrity at the end of the next operating period to ensure that acceptable safety margins exist, and d) recommend inservice inspection methods to detect, size, and trend the aging damage, and

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recommend appropriate mitigation measures and maintenance activities to reduce this damage.

The life assessment procedure for metal containments accounts for the major degradation effects of several corrosion and fatigue mechanisms on the containment vessels and bellows. Corrosion causes wall thinning of the metal shell that may significantly affect its structural integrity. Wall thinning would reduce the structural strength of the metal shell and has a potential to develop a leakage path for the interior environment of the containment. Extensive corrosion of the vessel wall would significantly reduce the fatigue life of the metal shell. Some metal containments have experienced extensive corrosion. Large misalignment and surface damage, if present, would significantly reduce the fatigue life of the bellows.

The first part of the life assessment procedure requires a review of several documents and records for estimating the current state of damage in a given containment at the beginning of the next operation period. The documents include design reports; construction and quality control documents; preoperational test records; inservice inspection, test, and maintenance records; and operational records. Corrosion and fatigue damage data from other plants with similar containments are also reviewed.

In the second part of the procedure, the review results from the first part are used to estimate corrosion and fatigue damage at the susceptible sites in the containment at the end of next operating period. Some examples of susceptible sites are containment vessel walls near concrete embedments, drywell walls in contact with sand pockets in Mark I containments, and pressure suppression pool walls near water line in BWR containments.

The third part of the procedure evaluates containment integrity at the end of next operating period. The evaluation determines whether the estimated minimum wall thickness of the metal shell is larger than the required minimum wall thickness by an acceptable margin. The evaluation also determines whether the fatigue usage factor for bellows is less than one by an acceptable margin.

The last part of the procedure identifies actions, if any, to be taken to ensure safe operation of the containment during the next operating period. The actions include use of enhanced inservice inspection methods, implementation of mitigation measures, and specific maintenance activities. Some examples of possible actions are ultrasonic inspection with focused transducer to measure thickness of extensively corroded wall, magnetic particle testing to detect flaws in coated surface, cathodic protection for surfaces that are inaccessible to coating, and repair of sealant at the metal-concrete interface.

DEVELOPMENTS IN RISK EVALUATION OF AGING

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Age related degradation of nuclear power plant components and systems are of considerable concern because of the impact of such aging on plant safety. This paper presents the developments made in prioritizing and determining risk significance of aging and in the analyses of the light water reactor (LWR) operating data to quantify such effects.

The approach developed to evaluate risk effects due to component and structural aging uses a probabilistic risk analysis (PRA) and component aging models. Linear and non-linear aging behavior have been incorporated. Aging occurring after some minimum time, known as threshold, can also be handled. As an important part of the evaluations, maintenance and surveillance strategies can be explicitly evaluated to determine their effectiveness in controlling aging impacts on system unavailability, core melt frequency and public risk. Both point evaluations and uncertainty evaluations can be carried out, and detailed contributors to the aging effects can be identified and prioritized. For the applications, two NUREG 1150 PRAs, one pressurized-water reactor (PWR) and one boiling-water reactor (BWR), were used to calculate the average increase in core melt frequency due to aging of active components when a specific maintenance and surveillance program was employed. A NUREG has been issued (NUREG/CR-5510) detailing the methodology and the applications performed.

With regard to the analyses of LWR operating data, methods have been developed to quantify aging effects in component failure and maintenance data. The methodology which has been developed allows any component failure or maintenance data containing times of failure or time of repair to be analyzed. The methodology determines the degree to which aging is exhibited in the data and determines the age-dependent failure rate. Both linear and non-linear failure trends can be handled. The uncertainty bounds associated with the failure rate are also determined. For the applications, the methodology has been used to determine aging effects in motor-operated valve, diesel generator, circuit breaker, controller and starter data. Computer software, suitable for personal computer (PC) applications, have been developed incorporating the methodology and the associated statistical and graphical techniques enhancing the efficiency and capabilities of the evaluation process.

The approaches which have been developed for prioritizing and quantifying risk significance of aging can be very useful in developing risk-effective test and maintenance guidelines directed towards aging and degradation control. Data collection guidelines can also be developed for effective monitoring of aging effects. Regulatory analyses have been carried out using these approaches for license renewal rulemaking purposes. Recent developments will be discussed.

DEGRADATION MODELING WITH APPLICATIONS TO AGING AND MAINTENANCE EFFECTIVENESS EVALUATION

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This paper presents a degradation modeling approach for analyzing component degradation and failure data to understand the processes in and reliability implications of the aging of components. The particular modeling employed, focuses on the analysis of the times of component degradations to depict how degradation rate changes with the age of the component. The models of the changes in degradation rate and failure rate with age are important since they quantify the aging reliability behavior of the component.

Analyses of component degradations provide significant information on the component aging process. Compared to failure data, degradation data are more abundant since degradations occur at a higher rate than failures. Analyses of degradation occurrences can identify the effect of aging on components before these effects become severe enough to result in significant reliability and risk implications. Markov modeling is used to construct a maintenance effectiveness parameter which relates component degradations to failures. This parameter identifies the effect of maintenance in correcting degradations and in preventing occurrences of failures.

Specific applications are performed to obtain component degradation rates and component failure rates from available plant-specific data. As part of the data analysis, statistical techniques are developed which identify aging trends in failure and degradation data. RHR pump data are analyzed using the degradation modeling approach. A definite "bathtub" curve in the RHR pump degradation rate is identified. The corresponding RHR pump failure rate has not yet shown the aging trend that is indicated by the degradation rate.

Initial estimates of maintenance effectiveness in limiting degradations from becoming failures also were developed. These results are important first steps in degradation modeling and show that degradations can be modeled to identify aging effects. The theoretical methodology which is developed is another advancement, in that degradation characteristics are explicitly related to failure rate effects and, hence, ultimately to risk effects. Further application of the methodology and statistical techniques is to develop and validate practical procedures for using degradation information in defining aging maintenance requirements, and in estimating aging failure rates from degradation data.

Status- Risk Evaluation from Aging of Passive Components

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The risk of core damage at all nuclear power plants is being determine using Probabilistic Risk Analysis (PRA) techniques. These assessments do not consider aging and consider the failure of passive components to only a limited degree. As a result of the low failure probability of the relatively new passive components, the failure of passive components is often ignored or not given adequate consideration in PRAs. Because of the large number of passive elements (e.g., many feet of pipe, numerous valve bodies, and pump casings), the large consequence when these components fail, and the increasing failure probability due to aging, passive components should be considered in PRA calculation of core damage risk.

The purpose of this project is to develop techniques to incorporate the effects of passive element failure into PRAs. The increased risk of core damage will be calculated as a result of the aging of passive components. This effort will contribute to the U.S. Nuclear Regulatory Commission Nuclear Plant Aging Research Program.

There are many observed aging mechanisms causing increased probability of failure in components of light water reactor systems. Examples of this are erosion/corrosion of feedwater and steam piping; fatigue of feedwater and safety injections piping; and stress corrosion cracking of pressurizer heater sleeves, steam generator tubes, and boiling water reactor (BWR) recirculation piping. This first year of the project, one component was selected to develop the technique to determine the failure probability because of one mechanism. The probability of failure of the aged passive component was calculated and a PRA was modified to reflect the increased probability. The component selected was a relatively small pipe in the auxiliary feedwater system. The failure mechanism was a type of thermal stratification that causes a cyclic stress in the pipe. This type of cyclic thermal stratification has caused cracking in safety injection lines at Farley and Tihange, and a residual heat removal line at Genkai. Passive elements usually have not failed, and increased failure probability should be theoretically calculated. Even though this mechanism has caused some failures they are to few to evaluate an increasing probability of failure, and theoretical calculations are used. A computer code that uses the structural probabilistic calculational method is generally used to perform this analysis.

The PRAISE computer code was selected to calculate the failure probability as a result of the aging mechanism in the auxiliary feed water pipe. The PRAISE code is a probabilistic fracture mechanics code that has been widely used to calculate the failure probability of piping components because of fatigue in both BWR and pressurized water reactor systems. In this case, the code was used to calculate the probability of failure because of the fatigue caused by the thermal cycling. The code was modified to determine the increasing failure probability of components as a result of changing material properties. To incorporate the effects of changing material properties modification of the code involves the identification of the random variables (those parameters that introduce the probabilistic nature into the calculation), and the modification of the statistical distributions of the random variables. To accurately model the failure mechanism, data that defines the distribution parameters (e.g., distribution type, mean, and standard deviation) should be found or generated.

An existing PRA was modified to account for the failure of this aged passive component. The failure of this pipe was added to the PRA fault trees and the effect of this pipe failure was modeled. The potential for the pipe to cause flow loss from multiple auxiliary feed water pumps was investigated. The event tree was examined to determine the effect of this component failure on the event tree. Finally, the increasing failure rate was used to calculate the increased initiation frequency and the increased core damage frequency as a result of this aged component.

If warranted by the 1990 work, the research will continue to investigate probabilistic structural analysis techniques to evaluate the failure probability on other passive components and aging mechanisms. Aging of multiple components will be considered. The calculations of component failure probabilities due to aging will be optimized by incorporating additional material aging data into the probability distributions. For high failure rate components where many failures have been observed, probabilistic failure predictions can be compared with known failure rates, to ensure accurate predictions by the code. To determine the risk significance the PRA will be modified to account for the increased failure probability of other aged passive components.

AGE-DEPENDENT RISK-BASED METHODOLOGY AND ITS APPLICATION TO PRIORITIZATION OF NUCLEAR POWER PLANT COMPONENTS AND TO MAINTENANCE FOR MANAGING AGING USING PRAs

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SUMMARY

This paper is based on a study to demonstrate several important ways that the age-dependent risk-based methodology developed by the Nuclear Plant Aging Research (NPAR) Program may be applied to resolving important issues related to the aging of nuclear power plant systems, structures, and components (SSCs). The study was sponsored by the NPAR Program of the Division of Engineering, Office Of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission (NRC). The study was performed by Pacific Northwest Laboratory (PNL) and Science Applications International Corporation (SAIC).

Initiated on the basis of a Users Need Request (memorandum from the Director, Office of Nuclear Reactor Regulation to the Director, Office of Nuclear Regulatory Research, April 9, 1987), the age-dependent risk-based methodology has been under development by the NPAR Program for several years. In this methodology, the time-dependent change in a component's risk contribution is the product of two factors: 1) the risk importance of the component (e.g., the change in its risk contribution when it is assumed to be totally unavailable to perform its intended safety function) and 2) the change in its unavailability with time. This change in the component's unavailability with time is a function of the component's aging rate and plant inspection and maintenance practices. The methodology permits evaluations of the age-dependent risk contributions from both single- and multiple-components.

Principal results and conclusions generated by the methodology demonstrations are the following.

1) The priority of a component's contributions to plant core damage frequency can be substantially altered by aging effects. When compared to priorities generated by standard, non-age-dependent PRAs, the priorities of many aged components were significantly increased. Selections made before aging effects are adequately considered might thus omit significant risk contributors from further consideration.

2) Because the age-dependent risk-based methodology models component risk contributions as a direct function of plant maintenance practices, its

application results in prioritizing maintenances on individual components and systems.

3) The impact of aging on plant risk increases can be controlled by focusing effective maintenance on the high priority risk contributors, component-types, and/or systems. The use of maintenance practices typical of predictive/preventive and reliability-centered maintenance were evaluated and were demonstrated to have substantial risk-reduction benefits.

Other important results and conclusions generated by the methodology demonstrations are:

1) The risk impacts from the aging of interacting components were large compared to impacts from single components.

2) The risk impacts from the aging of passive components, which had been separately calculated, were large and were important contributors to the increase in plant core damage frequency from aging.

3) Because of the high priorities observed for multiple aged components and for passive components, more comprehensive PRAs, in terms of the inclusion of multiple system and component interactions and passive components, would allow more detailed identification and prioritization of important-to-safety SSCs.

STUDIES OF AGED CAST STAINLESS STEEL FROM THE SHIPPINGPORT REACTOR

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The cast stainless steels used for primary coolant piping in many pressurized water reactors and for valve bodies, fittings, and coolant pump casings in most light water reactors are subject to embrittlement after extended service at reactor operating temperatures. Most studies pertaining to embrittlement of cast stainless steels involve simulation of end-of-life reactor conditions by accelerated aging at ≥400°C since the time period for operation of a power plant (~40 y) is far longer than can generally be considered for laboratory studies. Thus, an assessment of the end-of-life mechanical properties is almost always based on an extrapolation of the accelerated test data. Because the the embrittlement mechanisms and kinetics are complex, microstuctural studies and mechanical testing of actual component materials that have completed long in-reactor service are needed to ensure that the mechanisms observed in accelerated aging experiments are the same as those occurring in reactor. Cast stainless steel materials from the decommissioned Shippingport reactor offered a unique opportunity to validate and benchmark the laboratory studies.

Cast stainless steel materials were obtained from four primary coolant system check valves, two manual hot-leg isolation valves, and two pump volutes. One of the volutes is a "spare" that had seen service only during the first core loading, while the other was in service for the entire life of the plant. The materials were characterized to determine the chemical composition, hardness, grain structure, ferrite content, and ferrite morphology. All materials are CF-8 steel with ferrite contents in the range of 2-16%.

Microstructural examination of the cast materials indicates that the primary mechanism of thermal embrittlement is the same as that of laboratory-aged materials, i.e., spinodal decomposition of the ferrite to form chromium-rich α' phase. Other phases, such as nickel- and silicon-rich G phase precipitated in the ferrite, and the presence of carbides at the austenite/ferrite phase boundary also contribute to embrittlement.

Charpy-impact, tensile, and J-R curve tests were conducted on several cast stainless steels from the Shippingport reactor. Baseline mechanical properties for unaged material were determined from tests on either recovery annealed material (i.e., annealed for 1 h at 550°C and water quenched) or material from the cooler regions of the component. The aged materials show relatively modest decreases in impact energy. Room temperature impact energies for the cold-leg check valves CA4 and CB7 and the hot-leg main valve MA1 are 145, 183, and 299 J/cm^2 respectively; the 125-J/cm^2 (74-ft·lb)

Charpy transition temperatures (CTT) for the materials are 2, −34, and −127°C, respectively.

The results show good agreement with estimations that are based on accelerated test data. Correlations for estimating aging degradation of cast stainless steels indicate that the degree of embrittlement of the Shippingport materials should be low. The estimated minimum room-temperature impact energies that would ever be achieved after long-term aging are >120 J/cm^2 (>73 ft·lb) for all materials. The estimated activation energies for the kinetics of embrittlement range between 95 and 190 kJ/mole. The Shippingport cast materials were aged further at 350 and 400°C for up to 10,000 h to check these predictions.

INDIVIDUAL PLANT EXAMINATION PROGRAM - SUMMARY AND STATUS

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In the Commission policy statement on severe accidents in nuclear power plants issued on August 8, 1985, the Commission concluded, based on available information, that existing plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on any regulatory requirements for these plants. However, the Commission has recognized, based on NRC and industry experience with plant-specific probabilistic risk assessments (PRAs), that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low cost improvements. As part of the implementation of the Severe Accident Policy, the Commission issued Generic Letter 88-20 on November 23, 1988, requesting that each licensee conduct an individual plant examination (IPE) for internally initiated events.

Generic Letter 88-20 set forth the requirements for reporting the results of the IPE for accidents initiated by internal events to the Nuclear Regulatory Commission (NRC). Following a workshop on the conduct of the IPE, held in Fort Worth, Texas, early in 1989, the NRC issued Supplement 1 to Generic Letter 88-20 (dated August 29, 1989), transmitting NUREG-1335 and starting the IPE "clock." NUREG-1335, "Individual Plant Examination: Submittal Guidance," contains the guidance to the industry for the conduct and reporting of results of the IPE for internal events. This document had been issued as a draft for public comment prior to the workshop and had been revised by the staff prior to its final issue, based on comments received.

Generic Letter 88-20 had required the utilities to advise the NRC of the IPE methods and schedules within 60 days after the beginning of the process. Those plans have been received and approved by the staff. All utilities but one have committed to performing at least a Level I probabilistic risk assessment (PRA), that is, the core melt frequency will be determined by probabilistic methods. The different methodology, classified by the utility as an "other" methodology, was a combination of deterministic defense-in-depth and probabilistic methods, requiring review by the staff for applicability. Various plans for performing the "back end" analysis (which covers methods of evaluating containment performance and radioactive material release and transport) have been submitted. Several utilities will use probabilistic methods, that is, they will perform a Level II PRA, while others will utilize the "template" method suggested at the Fort Worth workshop, supported by plant-specific calculations where necessary. More than half of the submittals are expected to be received at the end of, or after, the time period allotted. Review of the reasons for those expected to be late showed them to be acceptable to the staff. A few utilities intend to evaluate external events on the same time schedule as for internal events. However, the staff is preparing separate guidance on the treatment of external events and has established a separate schedule for completion of the external event evaluations.

The staff intends to review the IPE submittals via a two step process. There will be a short review for all plants and a more in-depth review of questionable submittals. The basic intent of the staff review is to ensure that the utility has taken the process seriously and that the examination was sufficiently complete so as to reasonably expect it capable of finding vulnerabilities. The staff will also be evaluating the approach used by the utility to identify and dispose of vulnerabilities.

INDIVIDUAL PLANT EXAMINATIONS - AN INDUSTRY PERSPECTIVE

BY

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The purpose of this paper is to provide an overview of industry's approach and programs for implementing the 1985 Commission Policy Statement on Severe Accidents in order to achieve timely closure of the issue for operating plants. The emphasis of the paper is on the Individual Plant Examination (IPE) for internal and external events.

NUMARC, since 1987, serves as the United States nuclear power industry's principal mechanism for conveying industry views, concerns, and policies regarding generic regulatory issues to the 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 has placed high priority on resolution of the severe accident issue and hence, has established the NUMARC Severe Accident Working Group (SAWG). The mission statement of the SAWG is:

> "The purpose of the NUMARC Severe Accident Working Group is to coordinate industry activities and serve as the focal point for industry/NRC interactions in attaining resolution and closure of the severe accident issue. Specific items to be addressed by this Working Group are:
>
> - o Industry response and implementation of the requirements of the NRC;s generic letter on individual plant evaluations.
> - o Definition, development and implementation of severe accident management programs.
> - o Consideration of the need for individual plant evaluations of external events; and development and implementation of appropriate methodologies, if necessary.
>
> The Working Group will also focus on industry/NRC dialogue and develop industry position as necessary for other significant severe accident issues. Examples of such issues are: Containment Performance, Safety Goal Implementation, and Source Term Research."

The first and third items of the SAWG mission address the IPE. NUMARC has urged utilities to expeditiously proceed with implementation of IPEs for internal events. With regard to external events, the SAWG has directed the NUMARC staff to interact with NRC Staff to discuss the need for examination of plants from a severe accident perspective. The industry position with regards to external events is based on technical studies developed from the Industry Degraded Core Rulemaking (IDCOR) Program. IDCOR concluded that the potentially important external events (e.g., seismic, fire, external flooding, high winds) have been either conservatively included in the original design basis of operating plants, or have been or will be examined extensively as generic issues subsequent to their initial operation. It appears that no further cost justifiable gain could be obtained from probabilistic risk assessment of the external events.

Consideration of External Events in
the Individual Plant Examination Program

John T. Chen, USNRC

In the Commission policy statement on severe accidents in nuclear power plants issued on August 8, 1985, the Commission concluded, based on available information, that existing plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on any regulatory requirements for these plants. However, the Commission has recognized, based on NRC and industry experience with plant-specific probabilistic risk assessments (PRAs), that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low cost improvements. As part of the implementation of the Severe Accident Policy, the Commission issued Generic Letter 88-20 on November 23, 1988, requesting that each licensee conduct an individual plant examination (IPE) for internally initiated events including internal flooding.

Many PRAs have indicated that the risk from external events could be a significant contributor to the core damage in some instances. However, the examination for externally initiated events is being carried out on a later schedule to allow the staff to: (1) identify which external hazards need a systematic examination, (2) identify acceptable examination methods and develop procedural and submittal guidance , and (3) coordinate the IPEEE with other ongoing external event programs. In December 1987, an External Events Steering Group (EESG) was established to make recommendations regarding the scope, methods, and coordination of the IPEEE.

The EESG has recently completed its work and has made recommendations for the treatment of external events. Based on the EESG recommendations, the staff has prepared Supplement 4 to Generic Letter 88-20 describing the objectives, scope, and schedule of the individual plant examination for external events (IPEEE) and the acceptable methods of examination. The staff also prepared a detailed guidance document, NUREG-1407, on the conduct of the IPEEE and on the structure and the content of the IPEEE submittal. This paper provides a brief discussion of Supplement 4 to Generic Letter 88-20 and NUREG-1407. More specifically, the discussion will be centered on the IPEEE objectives, the external events that need to be included in the IPEEE, the acceptable methods, and the coordination needed between the IPEEE and other programs. Detailed discussion of the seismic methodology will be presented in the seismic engineering session.

CONTAINMENT PERFORMANCE IMPROVEMENT PROGRAM

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The Containment Performance Improvement (CPI) program has been one of the main elements in the US Nuclear Regulatory Commission's (NRC's) integrated approach to closure of severe accident issues for US nuclear power plants. During the course of the program, results from various probabilistic risk assessment (PRA) studies and from severe accident research programs for the five US containment types have been examined to identify significant containment challenges and to evaluate potential improvements. The five containment types considered are: the boiling water reactor (BMR) Mark I containment, the BWR Mark II containment, the BWR Mark III containment, the pressurized water reactor (PWR) ice condenser containment, and the PWR dry containments (including both subatmospheric and large subtypes). The focus of the CPI program has been containment performance and accident mitigation, however, insights are also being obtained in the areas of accident prevention and accident management. Recommendations relative to BWR plants with Mark I containments were made in January 1989. One, hardening of the wetwell vent, is being implemented either voluntarily by the licensees or by invoking the backfit rule (10 CFR 50.109). Other recommended changes are being explicitly reviewed within the Individual Plant Examination (IPE) program to examine individual plants for vulnerabilities to severe accidents. These other changes include: (a) alternate water supply for drywell sprays and vessel injection, (b) enhanced reactor vessel depressurization system reliability, and (c) improved emergency procedures and training. Recommendations on the other containment types were presented to the Commission in March 1990. In general, the same containment challenges and potential improvements were examined for the BWR Mark II and Mark III plants as in the Mark I program, with the addition of improvements related to the hydrogen igniters for Mark III plants. For the PWR ice condenser and dry containments, containment by-pass and direct containment heating are issues. In addition, improvements related to hydrogen igniters have been considered for ice condenser plants. Primarily because the benefits of proposed changes are perceived to be less or because of large design differences among plants, the case for generic recommendations is not so clear cut as for the BWR Mark I plants. Therefore, the NRC staff has not identified any recommended generic improvements that would be applicable to all containments of a given type, but has identified improvements to be considered further on a plant-specific basis as part of the IPE program. Improvements to be included in the accident management program and areas requiring additional research have also been identified.

Summary and Status of Generic Issue Resolution

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The Nuclear Regulatory Commission staff reviews proposed generic issues and prioritizes them, primarily based on safety significance, as high, medium, low, and drop. Issues that are prioritized as high are reviewed to determine whether they should be designated as an Unresolved Safety Issue (USI). During the past several years the Nuclear Regulatory Commission has made considerable progress in the resolution of generic issues. All of the issues designated as a USIs have been resolved. Most other issues designated as having high or medium priority have also been resolved.

All of the remaining issues that were prioritized as high or medium are being actively worked on. This paper summarizes the status of the generic issue resolution program and presents the technical findings and current status of the following six issues:

Generic Issue B-56, Diesel Generator Reliability

Generic Issue 15, Radiation Effects of Reactor Vessel Supports

Generic Issue 23, Reactor Coolant Pump Seal Failures

Generic Issue 57, Effects of Fire Protection System Actuation on Safety-Related Equipment

Generic Issue 87, Failure of HPCI Steam Line Without Isolation

Generic Issue 30, Essential Service Water System Failures at Multiplant Sites

SOURCE TERM UPDATE--SUMMARY AND STATUS
by Leonard Soffer
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Postulated fission product releases (so-called "source terms") have played a major role in U. S. regulatory requirements both for reactor siting and in setting plant design requirements. The source term used at present is derived from report TID-14844, issued in 1962, which postulates the instantaneous release into containment of 100 percent of the noble gas inventory of the core, 50 percent of the Iodine fission products (half of which are assumed to deposit on interior surfaces very quickly), and 1 percent of the remaining fission products. Current regulatory guidance also assumes that Iodine is present primarily in the form of elemental Iodine. There is general agreement, based upon research accomplished since its issuance, that although use of the TID source term has provided a high level of plant mitigation capability, the present recipe is not compatible with a realistic understanding of severe accidents.

In connection with the review of advanced light water reactor (ALWR) designs, the NRC staff has begun a program to implement the insights of source term research in revised regulatory guidance. This will include a revision of the TID source term, associated Regulatory Guides and sections of the Standard Review Plan.

TID-14844 was intended to represent releases for a severe core damage accident. Analyses of such accidents, using the NRC's Source Term Code Package (STCP) indicate that they would result in the release into containment of a quantity of radioactivity generally comparable to that assumed in TID-14844. The species are calculated to vary, however, in that significant quantities of Cesium, some Tellurium, and small quantities of other elements would be expected to be released in addition to Iodine and the noble gases. A major change would be in the timing of such releases. The instantaneous release currently assumed has long been recognized as highly conservative and the staff is currently examining accident analyses to develop a time-dependent model. Iodine chemical form will also be modified and is expected to reflect current research results that Iodine would be present mostly in particulate form, with some quantities of elemental and organic Iodine also present. Research studies currently underway at BNL, ORNL, and INEL to support this effort as well as an estimated schedule are described in more detail in the paper.

REVIEW OF THE IMPACT OF ENVIRONMENTAL FACTORS ON HUMAN PERFORMANCE
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The purpose of this project is to determine the effects of various environmental factors such as vibration, noise, heat, cold, and illumination on task performance in U.S. nuclear power plants. Although the effects of another environmental factor, radiation, is of concern to licensees and the Nuclear Regulatory Commission (NRC), much less attention has been paid to the potential effects of these other environmental factors. Performance effects from these environmental factors have been observed in other industries; for example, vibration can impair vision and noise can cause short- or long-term hearing loss. A primary goal of this project is to provide the technical basis for determining the likelihood of these factors affecting task performance in nuclear power plants, and thus the safety of the public.

This project consists first of a review of studies conducted in other industries and in the military. The review of the literature on vibration and noise has been completed. We will also review computer-based human performance models such as HUMAN, which was developed for the Department of Defense to evaluate the impacts of noise, heat, cold, and illumination; and the Maintenance Personnel Performance simulation (MAPPS), which addresses illumination and temperature. Finally, we will visit select nuclear power plants to estimate the degree to which workers are exposed to these factors. The results of these reviews will be used to support development of environmental exposure x human performance curves.

Vibration has documented effects on worker comfort, manual tracking ability, and visual acuity. These effects can be presented in exposure-effect curves, which depict the effect on task performance of a vibration of a given frequency and acceleration. Duration of exposure is also considered in these curves.

Noise, which is defined as an unwanted sound, is likely to be problematic, as workers in nuclear power plants are frequently exposed to noise (e.g., in the turbine building). Noise can impair task performance through effects on communication, either through masking speech or affecting workers' hearing. The effects on workers' hearing can be temporary or permanent. Temporary threshold shift (TTS) occurs when the workers' ability to detect quiet sounds is temporarily impaired after exposure to noise. Noise-induced permanent threshold shift (NIPTS) occurs when louder noises prolonged exposures, or both lead to a permanent hearing loss. TTS and NIPTS can be aggravated by the normal loss of hearing that occurs as all workers age. The effects of TTS and NIPTS can be shown in exposure-effect curves.

Other effects of noise on task performance are more complex and difficult to generalize because they are related to arousal curves. If a worker is under-aroused (i.e., bored), noise is likely to improve task performance. If a worker is over-aroused (i.e., under a lot of stress), noise is likely to impair task performance. Because arousal curves vary from person to person and because the same person will be on different points of his or her arousal curve in different situations, it is impossible to produce an exposure-effect curve that accurately depicts the effects of noise on task performance. Instead, we are determining the levels at which decrements in performance are first observed and using these levels as minimum acceptable levels of noise.

When we complete the review, we will produce three documents. The first will be a handbook to be used by the NRC and licensee personnel to determine whether the various levels of the environmental factors they may encounter will be likely to impair human performance. The second document will consist of the critical literature reviews upon which the handbook was based. The third document will provide an extensive annotated bibliography of the literature reviewed for this project.

An International Comparison of Manpower and Staffing Regulation and Practice in Commercial Nuclear Power Plants: Changes Over the Past Decade

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This paper describes trends in staffing practices in selected countries, including Canada, France, the Federal Republic of Germany, Great Britain, Japan, and Sweden. The focus of the investigation is the decision making process regarding staffing in nuclear power plants. The discussion includes:

- the government regulatory framework
- current policies and practices regarding nuclear power plant staffing and composition, including the number of plant staff, staff qualifications, job positions and shift configurations
- processes used in making decisions about staffing, including who makes decisions and how those decisions are implemented
- criteria used in making decisions about staffing, including the role of plant safety, cost, efficiency, and power production in making modifications to staffing policies and practices

The analysis of staffing strategies and practices concentrates on four areas; management, engineering and technical services, operations shift staff, and maintenance. Of particular interest in the decision making process is how staff levels and qualifications have needed to change in order to adjust to new developments in the nuclear industry. Two specific examples of factors likely to affect staffing are considered; the introduction of advanced computer technology and plant aging.

This work is based on a study, "Nuclear Power Plant Staff Composition," currently being conducted for the U. S. Nuclear Regulatory Commission Office of Research, FIN B56969.

Comparative Study of Alertness and Performance on 8-hour and 12-hour Evening and Night Shifts for Nuclear Power Plant Operators.

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Twenty males aged 25-40 years, many with extensive shiftwork and/or power plant operations experience, volunteered for a laboratory study to compare 8-hour and 12-hour shift schedules. Experiments were conducted in workplace simulation laboratories of the Human Alertness Research Center (HARC) at the Institute for Circadian Physiology in Boston, MA. This unique facility includes a simulated control room with process control simulator, control panels, and self-contained residential apartments where subjects sleep and spend off-duty hours.

The design of the study is as follows: The 8-hour shift experiment begins with two nights of overnight sleep in the laboratory so that subjects become accustomed to the laboratory laboratory environment and to recording techniques. This is followed immediately by 6 consecutive 8-hour evening shifts (3 p.m.-11 p.m.), four days off, then 6 consecutive 8-hour night shifts (11 p.m. - 7 a.m.). The 12-hour shift experiment also begins with two consecutive overnight adaptation nights, then four 12-hour night shifts (7 p.m.- 7 a.m.), three and one half days off, then 4 more 12-hour night shifts (7 p.m.- 7 a.m.). The two protocols are therefore equalized for total number of hours worked (96), although the 96 hours of simulated "duty shift" are distributed over more total days in the 8-hour shift protocol (16 days) versus the 12-hour shift protocol (11.5 days).

Subjects learn the operation of the process control simulator in multiple practice sessions prior to the experiment. They also practice computer-based performance tests, learn work shift protocols and assignments, and procedures for keeping subjective rating scales for alertness, mood and performance. By the time the first experimental shift begins, all subjects are prepared to perform their simulated work shift duties. These duties include monitoring of four video display terminals for silent (visual cues only) and auditory (visual and auditory) alarms, acknowledging the alarms with computer controls, and keeping log records of alarm activity. The process control simulator simultaneously keeps records of subject performance of these tasks by recording storing all times of alarm activity and alarm acknoledgement by the subject.

Every hour on the half-hour throughout the experiment, subjects are also asked to complete a 10-15 minute computer-based performance battery (Walter Reed Army Research Institute Performance Assessment Battery : PAB), which is comprised of 8 subtests. By completion of either study, each subject performs a total of 96 such performance tests and approximately 40 hours of process control simulator monitoring. Subjects also complete hourly self-assessment scales on mood, alertness, fatigue and subjective performance level.

Every two hours, Multiple Sleep Latency Tests (MSLT) are admininstered. These tests provide a sensitive and validated measure of the physiological sleepiness. In this test, latency to sleep onset is measured as subjects lie in bed in a darkenened room with instructions to relax and try to fall asleep if they feel sleepy. If sleep onset does not occur within 20 minutes, the test is terminated. If sleep onset does occur, it is interrupted within two minutes so that subjects do not accumulate sleep over the four (8-hour protocol) or six (12-hour protocol) successive MSLT tests given on each work shift.

After completion of each assigned work shift, subjects are allowed up to one hour for meals and relaxation, then spend a fixed 8-hour sleep period in a darkened bedroom for sleep. Sleep patterns are recorded continuously throughout this period via electrodes for measuring central electroencephalogram, electrooculogram, and electromyogram. If subjects awaken from this sleep period, they are asked to remain in bed in a darkened room for the entire 8-hour period, with the exception of brief bathroom breaks. No stimulant or depressant substances or medications, including caffeine and alcohol, are allowed for the entire period of laboratory studies. Subjects are also asked to refrain from the aforementioned substances during scheduled days off, but no verification blood testing was performed.

In this presentation for the WRSM, 8-hour and 12-hour shifts will be discussed and compared in terms of operator performance and alertness on shift, based on objective evidence derived from performance of simulated work activities (process control simulator), standardized performance tests, and subjective evaluations by the subjects. Changes in physiological sleep tendency at different times of night and on different nights of shift will be discussed. Finally, the important variable of sleep efficiency during the fixed daytime sleep period will be described.

Evidence will be presented which indicates that alertness problems on shift are associated primarily with the first and second night shifts of either 8-hour or 12-hour shift schedules. Furthermore, the experimental results indicate advantages of the "slow" shift rotation strategies, which allow shiftworkers to adapt to blocks of night shifts scheduled in succession, even if there are intervening days off in the schedule. Finally, 8-hour and 12-hour shifts are compared in terms of the potential for operator fatigue and performance decrements.

INITIAL EXAMINATION OF THE EFFECTS OF OVERTIME AND SHIFT SCHEDULING ON NUCLEAR POWER PLANT SAFETY PERFORMANCE

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The research reported here uses currently available data to begin to document overtime and shift scheduling practices in the nuclear industry and assess their safety consequences. Overtime and the 12-hour shift schedule have been found to affect employee fatigue and human error in other industries but neither of these practices have been systematically researched within the nuclear industry.

Overtime has long been a standard practice in the nuclear industry and, recently, several licensees have adopted a 12-hour shift schedule for control room operations crews. Although 12-hour shifts have certain benefits for the employer and employee alike (e.g., can require fewer total employees and provide longer periods of time-off) there is some concern that the 12-hour shift may produce greater employee fatigue. Moreover, the 12-hour shift may exacerbate the negative effect of overtime. The potential effect of human error in the nuclear industry for the health and safety of both nuclear power plant workers and the general public requires that these practices be assessed.

Data on overtime practices was obtained from the International Brotherhood of Electrical Workers (IBEW) for plants covered by the bargaining unit--approximately 63% of the total number of operating plants. Information on operator shift schedule was collected from all operating plants in 1987 by the NRC. The plant safety performance measures used in this research are publicly available from the NRC and include the set of performance indicators currently being used by AEOD to trend plant safety performance, the total number of violations levied against each plant by the NRC, a combined score for key Systematic Assessment of Licensee Performance (SALP) ratings, cause-coded License Event Reports (LERs), and the collective radioactive exposure level. Although these data provide an opportunity to begin to investigate the safety consequences of overtime and shift scheduling practices, the research conducted can only be seen as a preliminary investigation due to the fact that not all factors that potentially affect the plant safety performance measures can be controlled.

In spite of the above caveat, the research produced fairly strong evidence that greater amounts of overtime worked by operators is associated with a greater number of operational problems. Average operator overtime is significantly associated with poorer SALP ratings, the number of NRC safety violations, and with two of the five measures used by AEOD to trend plant safety performance. It is, however, noteworthy that average operator overtime does not have a significant effect on the incidence of LERs involving operator error. The overall general consistency of the findings for many of these performance measures does, nevertheless, warrant concern with fatigue-induced operator error.

The results for overtime worked by the technical staff, also, indicate that overtime for this functional group is a potential safety problem, but there are fewer significant associations. The amount of overtime worked by the technical staff has positive associations with the level of collective radiation exposure and with one of the AEOD measures.

Findings involving the amount of overtime worked by the maintenance staff are more ambiguous. When investigated independently, maintenance overtime has significant positive associations with only two of the safety indicators, collective radiation exposure and poorer SALP ratings. However, when the amount of operator and technical overtime is controlled, higher average maintenance overtime is found to be positively associated with two of the AEOD measures, negatively associated with two others, and negatively associated with the level of collective radiation exposure and the number of NRC violations. Thus, most of the relationships are in the direction opposite of that expected. Because maintenance overtime is so highly correlated with operator and technical overtime, it is picking up the effects of operator and technical overtime when investigated independently. If these effects are controlled, maintenance overtime does not have clear safety implications.

Finally, this research uncovered no effect of operator shift schedule. Further, it was not possible to investigate the joint effect of overtime and shift schedule as planned since only one of the 20 plants employing a 12-hour shift was IBEW-affiliated, meaning that overtime data was available for only one plant using a 12-hour shift schedule. Also missing from the shift schedule analysis, but crucial to a consideration of the effects of shift schedule, is information on the patterns of shift rotation. Until the 8- versus 12-hour shift schedule can be evaluated in the context of both overtime practices and shift rotation, the null findings reported here should not be prematurely accepted as indicating no safety concern.

In conclusion, future research efforts are recommended. One research area that should be given greater attention is an in-depth investigation of the effects of overtime by all categories of workers during outages. The amount of overtime worked by maintenance workers, supervisors/managers, and technical staff during outages is of particular interest. The quality of maintenance work as well as supervisory and technical oversight during outages may greatly affect the quality and safety of subsequent operations. Non-systematic information acquired during discussions with industry personnel in the conduct of this research indicated that overtime during outages may be the single biggest overtime issue currently existing in this industry.

Human Performance Information Management System: Inventory, Evaluation and Assessment of NRC Data Sources

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ABSTRACT

Background
The goal of this work effort is to develop enhanced methods for managing and accessing information on licensee personnel performance at nuclear power plants. Inadequate personnel performance is implicated in about half of the significant events at NPP's each year. An understanding of the factors which shape human performance can more effectively focus NRC's attention on the root cause of these incidents and guide regulatory actions. One key to understanding the root cause of human performance incidents is to develop better methods for managing existing data currently captured by the NRC. The tasks required for this effort are:

1. Evaluate existing data sources and coding schemes addressing human performance at NPP's.

2. Define the data needs of NRC end users.

3. Design and fully document a data management system on human performance during incidents/events at NPP's.

4. Download and integrate retrospective human performance data from available NRC documents and data sources.

5. Validate the usability for the human performance data management system through user feed back testing.

The final product is to define and develop a prototype system to support the systematic integration and retrieval of human performance information so that NRC personnel can more easily and effectively identify, prioritize and resolve human performance issues related to "root cause" incidents at nuclear power plants.

The first task of this multi-task initiative was the identification, quantification, and evaluation of the existing sources of human performance information generated by NRC. The categories of information analyzed included: operating event assessments, team inspections (Augmented Inspections and Incident Investigations), regional inspections, independent assessments of special problems, NRC "Morning Reports," and licensee event reports (LER's). We attempted to determine the accessibility and retrievability of these databases to support analysis of human performance problems and to guide resolution of performance issues. Several factors were found which make access and use of this data difficult and time-consuming: (a) the distribution of the data across a number of heterogeneous systems, and (b) the lack of any indexing for a major portion of the data. The existing NRC systems and configuration are thus currently neither compatible nor widely used to obtain human performance data.

Identification of Human Performance Information

There exist several NRC computer-based information systems, each designed for a different purpose, yet to some degree overlapping the others in coverage. These are NUDOCS, the PDR system which includes a subset of the NUDOCS database, SINET, (an evolving management information system which is planned to eventually tie together all the information systems in NRC), and SCSS (the Sequence Coding and Search System). Each of these systems is located on a different brand of computer in different physical locations (Bethesda, Washington, DC, Oak Ridge, Tennessee). Each system uses a different operating system and widely varying types of retrieval software. This distribution and dispersion of performance data across multiple computer-based systems causes great difficulty for end users attempting to access the data. Compounding this difficulty is the fact that much of the data is not indexed so that the data is not easily retrieved.

Preliminary Conclusions

There is no single "best" system to turn to for obtaining the NRC human factors data. In some cases, there is a preferred database for a certain class of information, such as SCSS for the LER's. In other cases, there is no clear advantage of one system over another. Before any improvement in information dissemination and analysis to support investigation of human performance issues can be effected, the existing information must be integrated, indexed, and made available via a single system which is more user-oriented than the existing systems currently available at the NRC.

One feasible approach which would both capitalize on existing NRC systems and avoid developing another stand alone system would involve building on the work which has already been accomplished with the SCSS at ORNL. It might be possible to use their numeric code system as the basis for a verbal indexing scheme, augmented as necessary, to fully cover the area of human performance. Although the NRC Division of Information Support Services has had an on-going thesaurus project for some time, it has not yet addressed the human factors vocabulary. If a modified SCSS code scheme could be converted to a structured vocabulary with necessary extensions, it could become the basis for improved indexing of the human performance literature in NUDOCS. This vocabulary could be used both to index whatever portion of the retrospective coverage is to be indexed as well as applied to new material to insure improved retrieval capability. A system tailored to the needs of the human performance research community could then be easily created by retrieving and downloading from NUDOCS the various desired categories of information.

Because the SCSS system includes all LER's, their coding scheme is not limited to the human factors area and therefore, this approach could serve as a model to demonstrate the feasibility of developing a prototype micro thesaurus which could be generalized to other areas in NRC. Given the variety of information systems in place at this time, it would seem expeditious to consider alternatives which can take advantage of any work which has already been accomplished to provide a system which capitalizes on existing strengths and minimizes the effects of the problematic areas.

ADVANCED HUMAN-SYSTEM INTERFACE DESIGN REVIEW GUIDELINES

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Advanced, computer-based, human-system interface designs are emerging in nuclear power plant control rooms as a result of several factors. These include: (1) incorporation of new systems such as safety parameter display systems, (2) backfitting of current control rooms with new technologies when existing hardware is no longer supported by equipment vendors, and (3) development of advanced control room concepts. Control rooms of the future will be developed almost exclusively with advanced instrumentation and controls based upon digital technology. In addition, the control room operator will be interfacing with more "intelligent" systems which will be capable of providing information processing support to the operator. These developments may have significant implications for plant safety in that they will greatly affect the operator's role in the system as well as the ways in which he interacts with it. At present, however, the only guidance available to the Nuclear Regulatory Commission (NRC) for the review of control room-operator interfaces is NUREG-0700. It is a document which was written prior to these technological changes and is, therefore, tailored to the technologies used in "traditional" control rooms. Thus, the present guidance needs to be updated since it is inadequate to serve as the basis for NRC staff review of such advanced or hybrid control room designs. The objective of the project reported in this paper is to develop an Advanced Control Room Design Review (ACRDR) Guideline suitable for use in performing human factors reviews of advanced operator interfaces. This guideline will take the form of a portable, interactive, computer-based document that may be conveniently used by an inspector in the field, as well as a text-based document. A key factor is to ensure that the Guideline will be a "living document," one that may be readily updated as further advances are made in the state of the art of control room design.

The methodology being employed to develop the ACRDR Guideline is based upon the need to meet two high-level program objectives : (1) to employ technology transfer in the development of an initial list of guidelines, and (2) to ensure that the guidelines developed are scientifically defensible and valid based upon the "internal" validity (research basis) and "external" validity (peer review) of the prior guidelines on which they are based. The first objective would insure that the guideline makes maximum use of relevant work done previously in other government, industry, or professional society efforts, as well as prior work performed within the nuclear industry itself. In addition, it institutes technology transfer as on ongoing activity to ensure that the guidelines always reflect the most current human engineering knowledge. The achievement of these objectives will ensure a sound basis upon which future new guidance may be developed in areas for which existing guidelines are lacking.

Guideline development is following several steps:

1. Significant related efforts (including nuclear and other industries) to develop human factors guidelines for advanced systems in government, industry and scientific groups have been identified, prioritized, and entered into a database.

2. The scope of the ACRDR Guideline has been defined based upon:
   - A review of the literature on advanced control room technology
   - Surveys of human factors reviewers of control rooms
   - Contacts with experts in advanced human factors technology
   - A review of the documents identified as important in Step 1.

3. The ACRDR Guideline organizational structure has been defined to include all important areas for which guidelines may be applicable including computer-based information displays, user input and controls, control/display integration, operator aids (such as expert systems), communications, and workstation design.

4. Identification will be made of areas for which new guidelines need to be developed.

5. Electronic document requirements will be defined and hardware and software selected.

6. The Guidelines document will be mocked-up and the prototype will be tested.

The paper describes each of these development steps, provides an overview of the content of the guidelines, and describes how the electronic evaluation document is intended to be utilized.

Development of a Performance Indicator of the Effectiveness of Human-Machine Interfaces for Nuclear Power Plants.

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With the development of computer graphic displays being increasingly proposed for NPP control rooms it becomes very importnat to be able to assess the degree to which they support improved operator performance. Three questions can be asked: (1) will a new interface improve the existing level of performance so that even operators with relatively low levels of skill can have their performance dramatically improved; (2) which displ;ay does this most effectively; and (3) is it possible to assess the level of skill of an operator when using a new or existing interface design?

The work to be described makes use of a well established phenomenon in cognitive psychology. In many fields of knowledge experts are able to recall displayed information and use it for decisions with a very much greater efficiency than novices or those of lower skill levels. This is however only true when the way in which the information is displayed maps the intrinsic relations among the variables in a way which matches the cognitive functions of the operator and his mental model of the controlled process.

The method to be described in this paper evaluates a modified version of the de Groot memory test as a vehicle for three tasks.

1. Can it be used to compare several displays of different design to see which supports retention and use of information the most effectively?

2. Can it establish the absolute level of performance obtained with the different displays?

3. Can it measure the level of skill of the operators for a given display and the way in which the level of skill interacts with different designs of interfaces?

In this paper the conceptual background of the research will be described and preliminary experimental work outlined.

Contract monitor: L. Beltracchi

A COMPUTERIZED SAFETY ASSESSMENT AND POST-TRIP ANALYSIS SYSTEM FOR THE FORSMARK UNIT 2 CONTROL ROOM, INTEGRATING A REAL TIME EXPERT SYSTEM AND A MODERN GRAPHIC DISPLAY SYSTEM

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SUMMARY

Records show that Swedish nuclear power plants have been operated without any serious accidents. They also have a high availability, recent figures indicate a yearly average energy availability of close to 90 percent for the nine BWRs. In spite of these good records, there is a continuous objective to improve safety at the Swedish nuclear plants. One way to meet the objective is to introduce computerized support systems in the control rooms and one such program is the development of a post-trip analysis system for the plant Forsmark Unit 2.

The purpose of the current program, called SAS II, is to develop a function oriented advisory system to assist the shift supervisor in his observation and evaluation task after plant disturbances leading to scram. To monitor this emergency shut-down process the supervisor today applies a set of function oriented emergency procedures and when SAS II is installed it will continuously give information to support the work with the EOP as well as alarm *if and why* critical safety functions are challenged.

A prototype including the two major components of the planned system has been demonstrated in the OECD Halden Reactor Project's experimental facility, HAMMLAB, in Norway. One of these components is the expert system shell G2 which evaluates the plant state with respect to the defined critical safety functions. The system also evaluates the functioning of the plant safety and protection system after scram with respect to the correct performance of automatically initiated safety sequences.

The other major component of SAS II is based on the graphic display system PICASSO

developed by the OECD Halden Reactor Project. A set of display formats has been designed to support the operators in their evaluation task in post-trip situations. These formats can be reached from the basic information presentation system which also is based on PICASSO. Operators from the Forsmark control room have been heavily involved in the display format design. Modern graphic techniques such as windowing and direct manipulation have been used in the MMI design.

During 1990 the SAS II system will first be installed at the compact-simulator located at Forsmark. Then an extensive verification exercise will be conducted. After the correct operation of the system is verified, the system will be subject to a validation exercise. Shift engineers and operators at Forsmark will be confronted with post-trip problems at the simulator. The aim with the study is to investigate whether the system contributes to improved performance in handling the transients.

If the verification and validation is successful the plan is to install and make the system operational in the Forsmark Unit 2 control room by the end of 1990.

The SAS II project is quite unique in the history of the OECD Halden Reactor Project. It combines the knowledge and experience of staff from plant operation, a state power board, a vendor, a licensing authority and a research institute in a single project to develop and install an advanced operator support system in a nuclear plant control room. The project is organised and run according to the ways devised as good practise for successful control room installations : end-user involvement in design, prototyping, a strict quality assurance program, verification program at a simulator before finally a validation study is carried out as a means to decide whether the system should be installed in the control room or not.

The use of advanced technology in hardware and software has not created any problems so far, on the contrary the involved operators are enthusiastic about the new man-machine interface possibilities offered by the PICASSO display system.

The end-user involvement in the design is believed to be crucial. Without their deep interest and efforts to bring over to the design group their knowledge and experience it would probably have been difficult to introduce this system at Forsmark.

When SAS II finally is taken into operation it is believed to be an improvement to safety for several reasons: the shift supervisor will easily and clearly, on his computerscreen at his own workplace get the information he needs when applying the emergency operating procedure, secondly the computerized system will warn him *if and why* - any of the defined critical safety functions are challenged - both during normal operation and in particular after scram.

The paper will describe this particular control room application, the approach for implementing it, the lessons learned thus far in developing the system with emphasis on the experiences with the real time expert system shell G2 and the PICASSO graphic display system.

AN ANALYSIS OF REPORTS OF PROCEDURE VIOLATIONS IN U.S. NUCLEAR POWER PLANTS

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The study described in this paper was undertaken in response to a request from the Nuclear Regulatory Commission's (NRC) Chernobyl Task Force. The objectives of the study were (1) to attempt to distinguish intentional procedure violations from those that are inadvertent, and (2) to assess the extent, nature, causes, and consequences of procedure violations in U.S. nuclear power plants. The focus on procedure violations, broadly defined here as failures to follow procedures, is a response to the key role that intentional violations and unintentional operator errors played in the Chernobyl accident.

Three categories of violations were defined, as follows:

> Level A Violation - A procedure violation which, in the judgment of the NRC, has been determined by a preponderance of the evidence to have been "willful," as defined in 10 CFR Part 2, Appendix C.

> Level B Violation - A procedure violation which may or may not have been "willful," as defined in 10 CFR Part 2, Appendix C, but for which either: (a) insufficient information is available to the NRC, after review, to make a determination of willfulness based upon a preponderance of the evidence; or (b) due to NRC resource limitations, the procedure violation has not been subjected to the scrutiny of such a review.

> Level C Violation - An inadvertent procedure violation which clearly was not "willful." These violations may be due to error, misjudgment, ignorance, or confusion.

Over 1,200 incident reports from the period extending from January of 1983 to July of 1988 were reviewed. The reports in which the violations were found consisted primarily of Licensee Event Reports (LERs) and NRC Inspection Reports (IRs). For each violation, an attempt was made to code the plant and region involved, the power level of operations at the time of the violation, the level of the violation committed (i.e., Level A, B, or C), the type of procedure involved, the job role of the person committing the violation, the probable cause(s) of the violation, and the consequences associated with the violation. The information coded from the incident reports was then statistically analyzed to address the central questions of the study.

The findings of the study indicated that both apparently willful and inadvertent procedure violations occurred in U.S. nuclear power plants during the study period. However, only a very small percentage of the violations reported in the data set of LERs and IRs were coded as Level A violations (13 out of 853). A larger number of the violations were categorized as Level Bs (140 out of 853). The very large majority of the violations identified in the study were characterized as errors (700 out of 853).

The distributions of the number of violations occurring at different plants in the data set varied depending upon the type of violation. That is, for the large majority of plants represented in the data set, no or only one violation that may or may not have been intentional was reported, and multiple violations of this type occurred at only a few plants. Inadvertent procedure violations were more widely distributed across the plants in the data set.

Findings regarding the nature of the violations indicated that all types of plant personnel, including licensed and senior reactor operators, and all types of procedural requirements, including technical specifications, were involved in the violations studied. Although the number of violations of each type of plant procedure varied in the data set, the distributions of violations appeared to be correlated with how often procedures of each type are performed in plants.

Additional findings regarding the nature of the violations that addressed power level, reactor type, and plant age were uninformative. Differences among the average number of violations found at plants in the different NRC regions were also assessed. Although slightly higher than average numbers of violations were found in Regions 4 and 5, the reasons for these findings could not be determined with these data.

The cause of procedure violations most frequently cited in the reports reviewed was the failure to use a procedure in performing a task. The Level B violations were most often caused by behaviors that could be characterized as misuse of the procedures, such as the decision to omit a procedure step. Causes cited for the Level C violations included an inadequate level of detail in the procedures, ambiguity in how the instructions were presented, and inaccuracies in the procedures.

The consequence associated with the procedure violations in the data set that was most frequently cited was no immediate safety consequences. In descending order, the next most frequently cited consequences were those that were unrelated to operational safety, automatic scrams, ESF actuations, and personnel exposures to radiation that exceeded or were within regulatory limits.

The total number of procedure violations at plants in the data set was significantly correlated with poor performance on several plant safety performance indicators used by the NRC's Office for the Analysis and Evaluation of Operational Data. The Level B violations were most strongly related to the plant performance data, and correlated with four of the seven indicators at moderate levels (r_s = .23 to .48).

Because of the manner in which U.S. nuclear power plants are designed, operated, and regulated, it is clear that the probability of a single procedure violation resulting in a major event is extremely small. However, the potential exists for procedure violations to act as precursors to serious events or to compound the seriousness of events as they occur. Thus, the potential importance to safety of procedure violations in U.S. plants suggests that the topic of procedure violations may warrant further attention.

Mechanical Properties of Cables Exposed to Simultaneous Thermal and Radiation Aging*

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SUMMARY

Sandia National Laboratories is 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. Three groups of twelve different cable products were aged for long times at relatively mild exposure conditions to nominal equivalent lifetimes of 20, 40, and 60 years. The three groups received respectively 3, 6, and 9 months of exposure to 10 krad/hr gamma radiation in aging chambers maintained at 100°C, resulting in total doses of 20, 40, and 60 Mrads.

Among the many CM techniques employed throughout aging, three mechanical properties of each cable's jacket and insulation materials were monitored: tensile strength, ultimate elongation, and hardness. Many short samples of both single and multiconductor cables were included in the aging chambers and were used for determination of hardness using an indenter developed at Franklin Research Center under EPRI sponsorship. Additional samples, consisting of strips of jacket materials and segments of insulation materials, were also included in the aging chambers and were used to measure tensile strength and elongation. The tensile samples were made by disassembling cables prior to aging. At monthly intervals throughout aging, a few samples of each cable type were removed from the aging chambers to obtain specimens with varying amounts of thermal and radiation exposure. After aging, some of the aged samples, as well as some unaged samples, were exposed to accident gamma radiation at ambient temperature. The nominal accident dose for this test program was 110 Mrad, but because of their locations in the test chambers, the tensile samples were exposed to a total dose of 95 Mrad at a dose rate of 500 krad/hr, while the hardness samples were exposed to a total dose of 100 Mrad at a dose rate of 530 krad/hr. The results of the mechanical measurements on some of these specimens will be presented as a function of exposure to the artificial aging environment.

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Recent Improvements in 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. Failures of swing check valves have occurred in several nuclear plants and have been largely attributed to severe degradation of internal parts resulting from instability (flutter) of check valve internals under normal plant operating conditions. Thus, in recent years, check valves have received considerable attention by the Nuclear Regulatory Commission (NRC) and the nuclear power industry.

In support of the NRC Nuclear Plant Aging Research (NPAR) program, ORNL has carried out an evaluation of three monitoring methods: acoustic emission, ultrasonic inspection, and magnetic flux signature analysis. These assessments have focussed on determining the capabilities of each method to provide diagnostic information useful in determining check valve aging and service wear effects (degradation) and/or indication of undesirable operating modes (e.g., instability).

Commercial suppliers of check valve monitoring systems utilizing these methods participated in a recent series of tests designed to evaluate the capability of each system to detect valve seat leakage and the position, motion, and wear of check valve internals. These tests were directed by the Nuclear Industry Check Valve Group (NIC) and were carried out at the Utah Water Research Laboratory on the Utah State University campus. Eleven check valves were utilized in the tests which included the use of hinge pins and disc studs in new condition and with simulated degradation, upstream flow disturbances, and reduced flow rates.

Descriptions of these three methods are provided in this paper, including the benefits and limitations associated with each method. Selected NIC test results are included that illustrate the capabilities of each method to detect simulated check valve degradation.

The ORNL Advanced Diagnostic Engineering Research and Development Center (ADEC), has recently developed novel check valve monitoring techniques that offer several improvements over existing methods. Full descriptions of these techniques are included in this paper, including laboratory and field test data.

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USNRC GATE VALVE TEST RESULTS CHALLENGE FLEXWEDGE GATE VALVE, MOTOR OPERATOR SIZING EQUATION.

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SUMMARY

The Idaho National Engineering Laboratory (INEL) under the sponsorship of the United States Nuclear Regulatory Commission (USNRC)[a] is performing research to provide technical input for the resolution of specific generic issues and to provide information to develop and improve industry mechanical equipment qualifications and operating and maintenance standards. This overall research effort includes a program that tested the operability (opening and closing) of six full scale motor-operated gate valves typical of those containment isolation valves installed in Boiling Water Reactor (BWR) Reactor Water Cleanup (RWCU) process lines and the High Pressure Coolant Injection (HPCI) turbine steam supply line. The valves were qualified and parametrically tested at, above, and below the pressures, temperatures, and flow conditions of a worse-case downstream pipe break in the RWCU and HPCI turbine supply lines outside of containment. One of the RWCU valves was also tested with steam to provide insights for the reactor core isolation cooling (RCIC) turbine steam supply line isolation valves. The purpose of the test program was to provide technical input for the USNRC effort regarding Generic Issue-87 (GI-87), "Failure of HPCI Steam Line Without Isolation." GI-87 is also applicable to the RCIC and RWCU Isolation Valves. All three of the GI-87 BWR process lines communicate with the

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

primary system, pass through containment, and have normally open containment isolation valves. The concern with these containment isolation valves is whether they will close in the event of a pipe break outside of containment. A high energy steam or water release in the auxiliary building could result in common cause failure of other components necessary to mitigate the accident. The test program also provides information applicable to the implementation of Generic Letter 89-10 "Safety-Related Motor-Operated Valve Testing and Surveillance" that is applicable to all light water reactor (LWR) safety related motor-operated valves (MOVs) as well as selected position-changeable MOVs in safety related systems.

Review of the test results provide convincing evidence that all of the valves required more thrust to open and close than would have been predicted by industry prior to the test program. This would indicate that the model industry uses to determine wedge gate valve thrust requirements, and ultimately the operator sizing, is flawed.

Analysis of the test results provide insights into where the inappropriate modeling might be occurring. All of the terms in the linear thrust equation can be measured during the course of a test with the exception of the valve disc (gate) sliding coefficients of friction. Industry has lumped all of the unknown gate valve performance characteristics into this sliding friction coefficient. From the analysis, it appears that other phenomenon, including fluid properties and pressure distributions throughout the valves, which are not accounted for in the equation, do not scale linearly. It also appears that the friction coefficients are higher than previously thought but were obscured by the other phenomenon taking place.

These insights are from an ongoing analysis from this recently completed test program. Actual numerical numbers should be available by the 18th WRSIM.

Reactor Internals Degradation Mechanisms*

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Summary

This paper is a progress report of a study on the aging effects on selected components of reactor pressure vessel internals. A data base of aging-related failure history of these components is being established based on information obtained from LERs, NPDRS data base and other published industry reports. The nature of the structural failure, its location and the "root cause" of the probable or suspected degradation mechanism(s) are identified. Conditions that are known to promote degradation mechanisms are compared to the design and operating conditions for the components with the goal of establishing of a linkage between the two. The failure information and the established linkage between operating and failure conditions form the basis for an assessment of the aging effects in reactor internals. The implementation of inservice inspection methods for reactor internals and its role in plant life extension will be addressed. The effectiveness of life extension and life prediction methods, including monitoring and flaw detection techniques, will also be discussed.

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Role of Recent Research In Improving Check Valve Reliability at Nuclear Power Plants

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Check valve failures at nuclear power plants in recent years have led to serious safety concerns, and caused extensive damage to other plant components which had a significant impact on plant availability. In order to understand the failure mechanism and improve the reliability of check valves, a systematic research effort was proposed by Kalsi Engineering, Inc. to U.S. Nuclear Regulatory Commission (NRC) under their Small Business Innovation Research (SBIR) program. The overall goal of the research was to develop models for predicting the performance and degradation of swing check valves in nuclear power plant systems so that appropriate preventive maintenance or design modifications can be performed to improve the reliability of check valves.

Under Phase I of this research, a large matrix of tests was run with instrumented swing check valves to determine the stability of the disc under a variety of upstream flow disturbances (elbows, reducers, butterfly valves, and multiple hole orifice plates as high turbulence sources), covering a wide range of disc stop positions (50 to 75 degrees) and flow velocities (up to 20 ft/sec) in two different valve sizes (3- and 6-inch). Phase I research led to the development of upstream flow disturbance factors which should be taken into account to determine the minimum velocity required to achieve a stable, full open disc position. The matrix of tests also quantified the severity of the disc fluctuations that can be expected when these minimum velocity requirements are not met. The results of Phase I research were published in NUREG/CR-5159.

The goals of Phase II research were to develop predictive models which quantify the anticipated degradation of swing check valves that have flow disturbances closely upstream of the valve and are operating under flow velocities that do not result in full disc opening. Two major causes of swing check valve failure are premature degradation due to hinge pin wear and fatigue of the disc stud connection to the hinge arm. A matrix of accelerated wear tests were performed using aluminum hinge pins and bushings in the 3- and 6-inch valves to quantify wear experienced in the hinge pin area. A special disc instrumented with strain gages was used in the 6-inch valve to measure the impact forces and their rate of occurrence to quantify the fatigue damage caused by tapping of the disc against the stop. Based on this theoretical and experimental research, wear and fatigue prediction models have been developed which show good correlation against laboratory test results as well as against a limited number of check valve failures at the plants which had been sufficiently documented in the past.

This research allows the inspection/maintenance activities to be focussed on those check valves that are more likely to suffer premature degradation. The quantitative wear and fatigue prediction methodology can be used to develop a sound preventive maintenance program. The results of the research also show the improvements in check valve performance/reliability that can be achieved by certain modifications in the valve design.

Aging Assessment of Circuit Breakers and Relays

by

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The aging of circuit breakers and relays is important since these devices provide critical services for safety related systems in nuclear plants. Failures of these devices can cause loss of vital functions as well as creating fire hazards. The aging assessments discussed in this paper were sponsored by the U.S. Nuclear Regulatory Commission's Nuclear Plant Aging Research (NPAR) Program. The Program Manager is Dr. Satish Aggarwal. The research has concentrated on identifying Inspection, Surveillance and Condition Monitoring (ISCM) techniques which are useful for detecting age degradation in these devices.

Significant elements of the research consisted of the following, laboratory tests, in-situ efforts and degraded parts in a laboratory environment. This includes five types of relays and three types of circuit breakers. Nineteen different ISCM techniques were evaluated. New and aged devices were used to determine the effectiveness of each technique to detect the effects of aging.

Aging effects were noted to be significant if they caused failure of the equipment or resulted in some other significant event, such as a fire hazard. The following techniques detected significant aging: Visual inspection, infrared thermal measurements, vibration monitoring, pick up voltage, drop out voltage, pole and contact resistance and smoke detection.

After the laboratory tests, the techniques which were effective, as well as some which still showed promise, were further evaluated in Duke Power Company's Catawba Nuclear Plant. This effort, referred to at the in-situ effort, was extremely valuable to the program. Catawba's staff aided greatly this effort because of their interest, support and knowledge. The practicability of the techniques was determined. Even though the laboratory tests had been performed to simulate plant conditions, utilizing the ISCM techniques in a plant provided a unique and realistic evaluation. Many of the difficulties of testing equipment installed in a plant had been effectively simulated in the laboratory. The additional difficulties encountered by moving test equipment into and out of radiation areas in the plant were not significant, but did provide the research team with better appreciation for the task. The Catawba maintenance and engineering staff were already performing some of the ISCM techniques and had developed procedures to minimize intrusion, to speed up testing and to assure accurate results. They benefitted from the in-situ effort since they were shown state-of-the-art ISCM techniques in infrared thermal scanning and vibration monitoring and analysis.

In the degradation tests, failure modes are purposely induced in samples of the circuit breakers and relays in order to determine the limits of the ISCM parameters before and at failure of a device. The results of the laboratory tests, in-situ effort and degradation tests are being analyzed.

The future efforts consist of collating the results and recommending the particulars of a program designed to manage aging of circuit breakers and relays. This effort has added significant insight into the aging of circuit breakers and relays and demonstrated new cost effective techniques for the monitoring and trending of safety significant failure modes.

Effects of Aging on Calibration and Response Time of Nuclear Plant RTDs and Pressure Transmitters

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Summary

This paper presents the key results of two experimental research projects conducted for the NRC on aging of safety system RTDs and pressure transmitters in nuclear power plants. These projects have addressed the following:

- Aging mechanisms in RTDs and pressure transmitters.
- Effect of aging on calibration and response time of RTDs and pressure transmitters.
- Oil loss syndrome in Rosemount pressure transmitters.
- Shelf life drift of RTDs.
- Drift of naturally aged RTDs and pressure transmitters.
- Test methods for on-line measurement of performance of RTDs and pressure transmitters.
- Assessment of useful life of nuclear plant RTDs and pressure transmitters.
- Testing intervals and replacement schedules.

The above topics will be discussed in detail and sample test results from laboratory measurements on typical nuclear grade RTDs and pressure transmitters tested in simulated reactor conditions will be presented. The emphasis will be on normal aging as opposed to accelerated aging.

It has been concluded that the steady state and transient performance of RTDs and pressure transmitters are affected by aging but the problem is manageable by in-situ response time testing and cross-calibration once every fuel cycle.

SUMMARY

Researching Organizational Factors

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Organizational processes at nuclear power plants should be sufficient to prevent accidents and to protect public health and safety upon the occurrence of an accident. The role of regulatory research is to (1) confirm that agency assessments of organizational processes are on a firm technical basis and (2) provide for improvements in the NRC programs. A firm technical basis is achieved by reducing uncertainties associated with methods and measures used to assess organizational processes.

The general objective for regulatory research is to confirm that (1) the agency has a coherent understanding of the organizational processes that are individually necessary and are collectively sufficient for safe operations, (2) methods are available to reliably characterize organizational processes, and (3) measures exist to monitor changes in the key organizational processes. The first specific objective was to develop a method to translate organizational processes into PRAs. Our research objective was neither to design organization charts nor to evaluate individual managers.

The discussion provides feedback and insights from experience (both successful and unsuccessful) with the past and the ongoing organizational factors research. That experience suggests a set of ingredients that appear proper for performing regulatory research on organizational processes. By keeping focused upon these proper ingredients, the research will contribute to the regulatory assessments of utility management through the use of improved methods and measures in investigations, inspections, diagnostics, performance indicators, and PRA insights.

TEAM PERFORMANCE MEASURES FOR ABNORMAL PLANT OPERATIONS

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In order to work effectively, control room crews need to possess well-developed team skills. Extensive research supports the notion that improved quality and effectiveness are possible when a group works together, rather than as individuals. The Nuclear Regulatory Commission (NRC) has recognized the role of team performance in plant safety and has attempted to evaluate licensee performance as part of audits, inspections, and reviews. However, reliable and valid criteria for team performance have not yet been adequately developed. The purpose of the present research was to develop such reliable and valid measures of team skills.

Seven dimensions of team skill performance were developed on the basis of input from NRC operator licensing examiners and from the results of previous research and experience in the area. These dimensions included Two-Way Communications, Resource Management, Inquiry, Advocacy, Confict Resolution/ Decision-Making, Stress Management, and Team Spirit.

Several different types of rating formats were developed for use with these dimensions, including a modified Behaviorally Anchored Rating Scale (BARS) format and a Behavioral Frequency format. The BARS format incorporated 7-point rating scales with anchors defining high, medium, and poor performance. The Behavioral Frequency scales required estimates of frequency of occurrence of specific behaviors. Following pilot-testing and revision, observer and control room crew ratings of team performance were obtained using 14 control room crews responding to simulator scenarios at a BWR and a PWR reactor.

Statistical analyses of the rating data revealed a general tendency of all raters to provide favorable ratings, with operator crews providing higher ratings of their own behavior than did observers. Inter-rater reliability coefficients for the Behavioral Frequency scales were, on average, greater than .50 while the average for the BARS was nearly 0. Internal consistency reliability analyses indicated substantial internal consistency for both types of scales across dimensions. Moderate support was found for convergent and discriminant validity of the dimensions. Exploratory analyses of the relationship between team skills ratings and measures of technical performance on a critical simulator parameter resulted in non-significant statistical values.

It was concluded, overall, that the Behavioral Frequency ratings appeared quite promising as a measure of team skills but that additional statistical analyses and other follow-up research are needed to refine several of the team skills dimensions and to make the scales fully functional in an applied setting.

Non-Nuclear Performance Indicators for Nuclear Power Plant Application

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The CTA/Concord team has been working since June 1989 on a project which is a part of the program to develop leading indicators of nuclear power plant (NPP) safety for the NRC, concentrating on management and organizational factors. Our specific approach is to examine other industries analogous in important ways to the NPP industry, to identify candidate leading indicators of safety in those industries, and to determine whether such indicators can be successfully transferred to the NPP industry. This approach was based on the knowledge that many other industries with public safety concerns similar to those of the nuclear power industry have been operating on a large scale for many years, and thus have had the opportunity to collect much more operating experience and data than the NPP industry has in its few decades of operation.

In an initial feasibility study phase, the team interviewed industry association personnel and corporate safety managers from various industries. The chemical/petrochemical manufacturing industry was selected as our primary focus of study. The chemical/petrochemical industry fulfills our criteria for critical analogies to the NPP industry, and also has been actively involved in the development and implementation of safety programs in the past several years. We began collecting data on safety indicators and outcomes from a number of chemical manufacturing companies last spring. Our efforts have included data collection and analysis to identify candidate leading indicators of plant safety, and to determine their relationships to safety outcomes at the plant level. In addition to plant-level analyses, we are examining data across plants within companies, to determine whether any company-wide or division-wide indicators can be identified, and are analyzing data across companies as well, to look for industry-wide indicators.

When the most promising candidate indicators have been selected in the chemical industry, the next step will be to investigate the feasibility of transferring those indicators to the NPP environment. This will be done with the assistance of a panel of experts from the NPP operations, regulatory, and research communities. The final candidate indicators, expressed in NPP terms, will be tested in a retrospective evaluation against existing data which has been collected by the NRC from nuclear power plants.

Amalgamation of Performance Indicators to Support NRC Senior Management Reviews

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Summary

US Nuclear Regulatory Commission has adopted a program of using performance indicator data to support the Senior Management Reviews of plant performance. Starting in 1987, seven indicators have been trended on a quarterly basis, with one additional indicator added in late 1989. The initial seven indicators are:--the number of automatic scrams while critical, the number of safety system actuations, the number of significant events, the number of safety system failures, the forced outage rate, the number of equipment forced outages per 1000 critical hours, and the collective radiation exposure. The indicator added in 1989 is that of the cause-code trends, which reflects trends in identified LER causes, such as operator errors, equipment failures, maintenance problems, and so on. These indicators are trended for each plant from quarter to quarters, and in comparison with classes of plants. NRC has adopted a policy of not making direct plant-to-plant comparisons.

In addition to these eight direct indicators, others are under development, specifically aimed at programmatic and organization issues. These indicators are intended to provide a leading or diagnostic capability for safety trends.

The problem with these indicators is, when taken individually, trends can be observed, but with multiple indicators it is difficult to identify a "big picture". The purpose of this amalgamation project is to provide a perspective to view these indicators in combination. This purpose is being accomplished by the creation of several frameworks: (1) a

diamond tree that describes how the various functions making up safety (reactivity control, heat removal, etc.) are influenced by both hardware and plant administrative programs; (2) an "onion" structure that describes, at different levels in the organization, the various influences that can impinge on the individual workers that are represented in the diamond tree; and (3) a framework for describing human performance at several levels in a nuclear plant.

The various safety facets of the plant reflected in the various indicators will be mapped onto these frameworks to provide the integrated perspective of safety.

Modeling the Effects of Dependency Between Humans and Hardware on Overall System Performance

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LWR Safety Conference, Washington, D.C., October 22–24, 1990

Summary

Current probabilistic risk assessment (PRA) studies employ event trees to define accident scenarios in terms of sequences of top events. The event tree top events typically model failures of hardware systems/subsystems or plant operators to mitigate the scenario. Since the likelihood of each scenario is determined using the conditional probabilities (sometimes called conditional split fractions) associated with each top event, the accuracy of an analysis relies heavily upon the correct assessment of these conditional probabilities.

Event trees have proven to be extremely useful, being ideally suited for treating scenarios involving functional dependencies between top events (e.g., the failure of one system causes the failure of another system) and shared equipment dependencies. However, they do not, nor are they intended to, literally simulate dynamic plant response during an accident. As a result, they may not provide the full context needed to assess dependencies during complex scenarios.

This paper describes a dynamic event tree approach designed to treat the time–dependent evolution of plant hardware states, process variable values, and operator states over the course of a scenario. The approach, which is similar in some ways to the accident progression trees used in NUREG–1150 and the DYLAM approach, allows improved characterization of the context for operator actions, and can lead to improved estimates of risk.

A dynamic event tree is an event tree in which branchings (which, in the case of a nuclear plant front–end model, reflect stochastic variability in the process) are allowed to occur at different points in time. Three characteristics of interest are: a) all possible combinations of system states must be considered at each branching point, b) branchings are performed at arbitrary, but discrete, points in time, and c) the number of event sequences can quickly grow to an unmanageable size if various approximations designed to limit the problem are not applied. The last point means that the practical application of the approach is likely to be a simulation–oriented one. Event sequences are generated by user–supplied rules as the analysis progresses, rather than specified in their entirety as an initial step in the analysis.

An application of dynamic event trees is characterized by a number of factors, including: the set of variables that determine the space of possible branchings at any tree node, the the set of variables that influence the probability assignments for the various branchings, the rules used to determine if a branching occurs at a given point in time, and the set of rules used to limit sequence expansion. In simple applications, the possible branchings are determined largely by the possible combinations of system hardware states; in more detailed applications, operator cognitions, emotions, and actions, for example, will also lead to branchings.

Even the simple applications of the dynamic event tree approach can provide useful information. For example, if the operators are modeled as strict procedure followers, the methodology can identify the effectiveness of procedures under a variety of conditions. In the context of PRA, the approach can be used to better specify success criteria for conventional event tree top events. When a more realistic operator model is used in defining tree sequences, the approach can explicitly identify chains of reasoning which, in concert with hardware failures and wrong operator actions, that can lead to plant damage.

DEPENDENT FAILURE ANALYSIS OF NUCLEAR POWER PLANT DATA BASES

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The paper presents an approach for identification of potential dependent failures in nuclear power plants, and determination of applicable defenses in normal plant operating practices to reduce the vulnerability to dependent failures. An application of the approach using plant-specific maintenance records for motor-operated valves (MOVs) is discussed.

The data analysis approach for identification of likely dependent failures centers on a) developing a categorizing scheme for component failure causes, b) identifying likely dependent failures based on the proximity of failures, and c) determining the occurrences of dependent failures using plots of failures of multiple components across systems. Analysis of MOV failures across four systems - low pressure injection (LPI), high pressure injection (HPI), emergency feedwater (EFW), and service water (SW) - results in identification of a significant number of dependent failures. The failure data analysis included both catastrophic and degraded failures. The inclusion of degraded failures in identifying likely dependent failures significantly enhances the dependent failure data base, compared to existing dependent failure data base used in PRA studies, and is useful in devising defensive strategies to reduce plant vulnerabilities to such failures. Details of the evaluations including the criteria for selecting the data bases for evaluation, failure cause categories for MOVs, cause-specific categorization of dependent failure occurrences and the risk-significance of the observed occurrences will be discussed.

The results of the data analysis are used to evaluate the applicability of various defensive actions. Since, as evidenc[illegible]e results, the vulnerabilities to dependent failures are plant-specific, the appropriate defenses a[illegible] general, also plant-specific. Generic insights are drawn by relating specific cause categories to applicable defenses.

CONVERSION OF A MAINFRAME SIMULATION FOR MAINTENANCE PERFORMANCE TO A PC ENVIRONMENT

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The computer model **MAPPS**- the Maintenance Personnel Performance Simulation has been developed and validated by the US NRC in order to improve maintenance practices and procedures at Nuclear Power Plants. This model has now been implemented and improved, in a PC environment and renamed **MICROMAPPS**. The model is stochastically based and users are able to simulate the performance of 2- to 8- person crews for a variety of maintenance tasks under a variety of conditions. These conditions include aspects of crew actions as potentially influenced by the task, the environment, or the personnel involved. For example, the influence of the following factors is currently modeled within the MAPPS computer code: 1) **Personnel characteristics** include but are not limited to intellectual and perceptual motor ability levels, the effect of fatigue and conversely, of rest breaks on performance, stress, communication, supervisor acceptance, motivation, organizational climate, time since the task was last performed and the staffing level available; 2) **Task variables** include but are not limited to time allowed, occurrence of shift change, intellectual requirements, perceptual motor requirements, procedures quality, necessity for protective clothing and essentiality of a subtask; and 3) **Environment variables** include temperature of the workplace, radiation level, and noise levels. The output describing maintainer performance includes subtask and task identification, success proportion, work and wait durations, time spent repeating various subtasks and outcome in terms of errors detected by the crew, false alarms, undetected errors, duration, and the probability of success. The model is comprehensive and allows for the modeling of decision making, trouble-shooting and branching of tasks.

As part of FY1989 efforts, a number of user enhancements were identified. It was determined that they would be implemented as part of the conversion process. These include allowing a greater number of subtasks to be processed, increasing the model precision, adding an "on-line" HELP function for first time users, reformatting the output in a form useful to the PRA risk analyst for inclusion in such NRC sponsored models such as IRRAS, adding "on-line" diagnostics, adding tasks to the task library and allowing for a greater number of iterations of the simulation, and using color to highlight important data. The User's Guide and Programmer's guide for MAPPS are also being updated to address the increased capability of the PC version **MICROMAPPS.**

Work conducted at the INEL has resulted in the successful completion of a preliminary MICROMAPPS software code. The code is undergoing sensitivity testing to ensure that all user enhancements operate properly and that the PC version yields the same results as the mainframe version. A workshop for NRC staff and their contractors is being planned for later this year.

Human Factors Assessment in PRA using Task Analysis Linked Evaluation Technique (TALENT)

by:
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Summary

Human error is a primary contributor to risk in complex high-reliability systems. 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. Since then, the USNRC has initiated research to fully and properly integrate human errors into the Probabilistic Risk Assessment (PRA) process. The resulting implementation procedure is known as the Task Analysis Linked Evaluation Technique (TALENT).

As indicated, TALENT is a broad-based method for integrating human factors expertise into the PRA process. This process achieves 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 that transcend the individual plant.

TALENT focuses on bringing together source data, quantification methods, and broad-based human factors expertise. TALENT takes advantage, whenever possible, of products from U.S. government agencies, national laboratories, U.S. commercial research organizations, and international research organizations. TALENT elements includes source data, data/information gathering tools, quantification methods, and the use of the Performance Evaluation Data (PED) file. Combined, these four elements provide the resources needed to integrate the human factors aspects of any analysis into the PRA process.

The TALENT procedure is being field-tested to verify its effectiveness and utility. The objectives of the field-test are to examine (1) the operability of the process, (2) its acceptability to the users, and (3) its usefulness for achieving measurable improvements in the credibility of the analysis. The field-test will provide the information needed to enhance the TALENT process.

Title: **Human Factor Insights from International Event Reports**

Author: **E. Swaton**

Organization: **International Atomic Energy Agency, Vienna, Austria**

SUMMARY

In the safe operation of nuclear power plants reliable and attentive human behaviour plays a major role. Apart from assisting the staff in the optimal performance of their duty from the managerial level to prevent undesired actions, it is also imperative to investigate actual events to determine their causes and implement remedial mechanisms. While analysis of single events can lead to specific conclusions related to the particular circumstances, the systematic screening of a collection of events will lead to generic insights and the identification of overall problem areas. The Incident Reporting System supported by the International Atomic Energy Agency is collecting reports on safety significant operational events on an international basis. Close investigation of these reported events revealed that in a large number of cases human intervention contributed to the initiation and/or development of these events. A systematic analysis developed within the IAEA termed Assessment of Safety Significant Events Technique (ASSET) was applied which provides a structured methodology to identify not only the direct cause explaining why an individual failed, but more important, provides insights into the root causes to determine why this latent deficiency was not detected earlier through the plant surveillance programme. The generic lessons drawn from the collected insights on human behaviour will be presented and a brief description of one or more symptomatic events illustrating these lessons will be given.

NRC's HLW Research Program Plan

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High-level radioactive waste (HLW) is (1) irradiated reactor fuel, (2) liquid wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuel, and (3) solids into which such liquid wastes have been converted. Following the enactment of the Nuclear Waste Policy Amendments Act of 1987, the US Department of Energy (DOE) has been pursuing plans to dispose of HLW in a deep geologic repository at Yucca Mountain, NV. Before it can dispose of HLW, DOE must obtain a license for disposal from the US Nuclear Regulatory Commission (NRC). Although NRC regulates the disposal of HLW, the US Environmental Protection Agency (EPA) is responsible for setting the overall performance standard, 40 CFR 191, for HLW disposal. NRC has issued its own HLW regulation, 10 CFR 60, which is a set of mandatory guidelines that DOE must follow in order to meet the EPA standard and includes the EPA standard by reference (10 CFR 60.112). Within NRC, the Division of High-Level Waste Management (DHLWM) of the Office of Nuclear Material Safety and Safeguards (NMSS) has the responsibility for licensing HLW disposal. The Waste Management Branch (WMB) of NRC's Office of Nuclear Regulatory Research operates a research program to support NMSS's HLW policies and decisions. This paper discusses WMB's HLW research program plan.

In its application for HLW disposal, DOE will have to demonstrate compliance with both procedural and technical requirements set forth in 10 CFR 60 and NRC will have to be prepared to assess DOE's compliance with those requirements. The NRC HLW research program is providing technical information that will be useful to DHLWM in assessing compliance with those procedural requirements that have technical aspects (e.g. 10 CFR 60.21) and the technical requirements that concern radiological safety of HLW after it is placed in a geologic repository. Other aspects of the safety of HLW, such as transportation and above-ground storage, are regulated by NRC divisions outside DHLWM and WMB's HLW research program does not support them.

For the purposes of designing a research program to support DHLWM, the WMB staff has worked closely with the DHLWM staff to analyze the technical requirements of 10 CFR 60 and identify technical information that NRC does not have in hand now but will need in order to make credible HLW regulatory decisions. The analysis has suggested that the most contentious part of HLW licensing will concern assessment of compliance with the Performance Objectives in Subpart E (Technical Criteria) of 10 CFR 60 and with other sections of 10 CFR 60 that support the Performance Objectives, especially content of the Safety Analysis Report (10 CFR 60.21(c)), Siting Criteria (10

60.122) and Design Criteria (especially 10 CFR 60.133-135). Because there are no existing HLW repositories than can be used as standards for comparisons in making licensing decisions, the general apporach of the NRC HLW research program is to ensure that NRC has an adequate understanding of the processes that affect repository performance and, where possible, to implement that understanding in mathematical models that can be used to estimate the effects of the processes over the long performance period of the repository.

Much of the information from the NRC HLW research program will be needed by 1998, when DOE plans to begin designing its application for a license to dispose of HLW. By this time, the HLW research program should have provided DHLWM with an independent understanding of the processes that affect repository performance, as defined by 10 CFR 60's Performance Objectives. Information that can be used for assessing compliance with quantitative measures of performance should be embodied in mathematical models that the NRC staff can use in its licensing decisions. DOE will have to provide NRC with information and data specific to the proposed repository facility.

In order to provide the information that DHLWM will need, WMB has devised an HLW research plan (NUREG 1406) with seven areas of research described briefly below. The research concentrates on finding out how radionuclides would be releas[illegible] through [illegible]rom HLW and transported through unsaturated fractured geologic media because the proposed repository horizon at Yucca Mountain is in unsaturated fractured welded tuff.

1. Controlled Release: Degradation of spent fuel and glassified HLW in oxidizing environments (10 CFR 60.113(a)(1)(i),(ii)(B)).
2. Containment: Corrosion of HLW packages in oxidizing environments (10 CFR 60.113(a)(1)(i),(ii)(A)).
3. Engineered System: Mechanical loadings on HLW due to tunnel deformation caused by seismic effects (10 CFR 60.122); performance of borehole and shaft seals (10 CFR 60.134); rock mass characterization (10 CFR 60.21(c)(1)(ii)(A); 60.31(a)(1)); coupled processes that affect repository performance (10 CFR 60.21(c)(1)(i)(F)).
4. Hydrology: Groundwater flow and non-chemical aspects of radionuclide transport (especially dispersion) in unsaturated fractured geologic media (40 CFR 191; 10 CFR 60.21 (especially (c)(1)(ii)(C)); 60.112; 60.113(a)(2); 60.122(b)(7)).
5. Geochemistry: Chemical and mineralogical effects of corrosion products and host rocks on the transport of radionuclides released from HLW (10 CFR 60.21(c)(1)(i)(E),(F); 60.21(c)(1)(ii)(A); 60.113(b)(3); 60.122(b)(1),(3); 60.122(c)(7),(8))
6. Geology: Tectonic and volcanic considerations in site suitability (10 CFR 60.122(b)(1); 60.122(c)(3),(12),(13)).
7. Performance Assessment: Integration of mathematical models of processes affecting radionuclide transport into a predictive methodology for assessment of compliance with Performance Objectives (40 CFR 191; 10 CFR 60.112; 60.113); validation of models (10 CFR 60.21(c)(1)(ii)(F)).

High-Level Nuclear Waste Research at the Center for Nuclear Waste Regulatory Analyses: A Progress Report

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As the Federally Funded Research and Development Center (FFRDC) for the U.S. Nuclear Regulatory Commission (NRC), the Center for Nuclear Waste Regulatory Analyses (Center) has been charged with conducting a broad range of research in support of the NRC mission to ensure public health and safety related to the storage and disposal of high-level nuclear waste (HLW). This mission, which is established under the Nuclear Waste Policy Act, requires that research be conducted to develop a sound technical basis for development and implementation of regulations and other guidance; to provide expertise and information required for independent assessment of license applications and prelicensing submittals; and to anticipate and evaluate problems of potential safety significance.

This paper presents the results to date on six research projects which have been developed consistent with the overall mission of the NRC, the identified needs of the licensing office of the NRC, and the HLW Research Program Plan.

A two-part Unsaturated Mass Transport (Geochemistry) study has been initiated to evaluate the extent to which the presence of zones of zeolitic tuffs beneath the proposed repository site at Yucca Mountain, Nevada will enhance the performance of the proposed repository site. In the first part of this project, experimental studies of the thermodynamic and ion exchange properties of clinoptilolite (the predominant zeolite mineral at Yucca Mountain) are being conducted. Studies to date indicate the importance of (a) applying rigorous thermodynamic principles to the study of ion exchange phenomena, (b) correctly calculating the nonideal behavior of aqueous solutions for precise description of ion exchange phenomena, (c) the nature of the experimental materials, and (d) the method of pretreatment of those materials. In the second part of this study, reaction path modeling capabilities are being developed for use in interpretation of groundwater and mineral chemistry and for identification of controls on the evolution of the Yucca Mountain geochemical system. In addition to developing a kinetic reaction path model for the evolution of groundwater and mineral chemistry at Yucca Mountain, modifications have been made to the computer program to treat nonisothermal kinetics and to incorporate a Rayleigh distillation model.

Validation of geochemical models such as those described above is a crucial technical and regulatory issue. Although a number of such studies have been conducted in the past, the Geochemical Analog Project will be the first such study which has as its focus the transport of contaminants in hydrologically unsaturated rocks such as occur at the proposed site. The aim of this project is to obtain fundamental data on time and space scales that are generally inaccessible in laboratory studies and, thus, significantly aid the validation of predictive models for geochemical transport.

The Stochastic Analysis of Flow and Transport research project has as its aim the development and evaluation of models for flow and transport in fractured spacially heterogeneous unsaturated rocks. Models of this type are candidates for evaluating compliance with siting criteria and performance objectives related to groundwater travel time. To adequately model the complex, heterogeneous flow and transport processes, it appears that it will be necessary to incorporate the effects of relatively small scale space and time variability in modeling large-scale and long-time unsaturated flow and transport.

In addition to the study of fluid flow under ambient-temperature undisturbed conditions described above, technical and regulatory concerns exist regarding thermally driven fluid flow which will result from emplacement of heat-generating waste in hydrologically unsaturated media. Technical issues and uncertainties associated with liquid- and vapor-phase transport are being evaluated in the Thermohydrologics Project laboratory experiments. To date, experimental apparatus has been developed and preliminary studies have been conducted to examine variations in parameters affecting fluid flow.

The Seismic Rock Mechanics research project has the twin focus of understanding the key parameters affecting repository performance under repeated seismic loading and evaluating current capabilities for calculating such effects. To support the laboratory testing needs of this project, a large dynamic direct shear test apparatus has been designed and constructed. Welded tuff rock specimens have been collected and prepared, and preliminary results have been obtained. In addition, a mine which experiences 10 to 20 seismic events per year with magnitudes greater than Richter 2.0, has been selected as the location for acquisition of both rock mechanical and hydrogeological response data which can be used in model validation studies. The initial phase of evaluation of computer codes for modeling seismic effects on the underground facility and surrounding rock mass has begun. Based on the results of four benchmark analytical problems, it appears that the discrete element code UDEC performs valid simulations of jointed rock behavior. However, the HONDO II finite element code demonstrated a limited capability.

The Integrated Waste Package Experiments project is a broad-based research program which has been initiated to investigate pertinent phenomena related to the long-term containment of radioactive materials within the waste packages. This project is organized to address localized corrosion, stress corrosion cracking, material stability under repository thermal conditions, microbiologically induced corrosion, and other degradation phenomena, such as hydrogen embrittlement. Early studies examined potential limitations and problem areas of existing laboratory corrosion testing techniques. To date, electrochemical potentiodynamic tests have been conducted on seven materials. In addition, microstructural evaluations of some high-nickel austenitic materials have shown significant chromium depletion of the surface relative to the bulk metal. Preliminary results of low-temperature sensitization in austenitic stainless steels and the long-term oxidation behavior of copper-based alloys have also been obtained.

Development and Evaluation of a Performance Assessment Methodology for Analyzing the Safety of a Geologic Repository for High-Level Radioactive Waste

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Beginning in 1976 the NRC directed research efforts towards the development of a performance assessment methodology to assess the long-term risks from radioactive waste disposal in deep geologic media. A performance assessment will be conducted by NRC to evaluate the compliance of a High-Level Waste (HLW) repository with the requirements in 10CFR60.21 and the quantitative performance objectives in 10CFR60.112 and 60.113 and release limits set forth in the EPA HLW Standard, 40CFR191. The performance assessment is envisioned to include the selection and screening of scenarios (events, features, and processes either singly or in combination which could contribute to the release of radionuclides from a repository), estimation of the consequences of the scenarios (quantitative estimates are usually obtained via computer simulation of the dominant processes), and the presentation of results and uncertainties according to the pertinent regulatory requirements.

Although the steps of a performance assessment are relatively straightforward, the application of a performance assessment methodology to an HLW disposal site requires understanding of the dominant hydrogeologic and geochemical phenomena of the site and the extrapolation of these site conditions over the thousands of years of regulatory concern. The complications and uncertainties involved with understanding current site conditions and estimating future conditions and quantifying their impact on repository safety have been the focus of research programs in performance assessment. Current research involves: 1) laboratory and field experiments to improve understanding of site characterization methods and dominant ground-water flow and radionuclide transport phenomena in unsaturated fractured tuff, 2) development of models to analyse and interpret field and laboratory experiments and for estimating scenario consequences, 3) review and development of techniques for performing sensitivity and uncertainty analyses, and 4) examination of the applicability or validity of the developed models and methods.

Laboratory and field experiments and theoretical studies being conducted at the University of Arizona (UA) are investigating: 1) field measurements necessary to characterize unsaturated fractured tuff adequately to estimate ground-water flow and radionuclide transport, 2) phenomena and parameters (e.g., related to fracture coatings) which control fracture matrix interactions in the unsaturated zone, and 3) appropriate methods for extrapolating small scale (i.e., meters and months) test results to the large scale (kilometers and thousands of years) of performance assessments. Research at UA is being conducted on a variety of scales (drillcore, block, and field scale) and includes a variety of processes (fluid flow, heat transport, solute transport, two-phase flow, and fracture flow). The UA investigations are being carried out in a tiered fashion (i.e., from simple to more complex) where later experiments build upon what was previously learned. The Center for Nuclear Waste Regulatory Analyses (CNWRA) has recently begun experimental and theoretical investigations complementary to the UA program. The CNWRA program is focussing laboratory experiments on the geochemical and thermohydrologic aspects of disposal in tuff.

The computational requirements of a performance assessment have necessitated the development of a number of computer programs to simulate radionuclide movement away from a repository and techniques for analysing sensitivity and uncertainty. Current research, being conducted at Sandia National Labortories and the CNWRA, is concentrating on developing models for testing different theories for fracture matrix interactions. Previously developed techniques (i.e., Latin Hypercube Sampling, LHS) for sensitivity and uncertainty analysis have been used many times on linear problems such as occur in the saturated zone. The CNWRA is currently examining these techniques for their applicability to the non-linear unsaturated zone problem.

NRC staff and contractors are currently participating in the international project called INTRAVAL to gain confidence in or validate the various strategies used to estimate radionulide transport in the biosphere. INTRAVAL is using a variety of laboratory and field experiments over a variety of scales and geologic media to examine the validity of different approaches to simulating radionuclide transport. Additionally, the project has included natural analogs (e.g., radionuclide migration from a uranium ore body) to provide transport "experiments" that offer spatial and temporal scales comparable to the corresponding HLW disposal scales.

LOW-LEVEL RADIOACTIVE WASTE RESEARCH PROGRAM

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Commercially generated low-level radioactive waste (LLW) is regulated by NRC's Office of Nuclear Material Safety and Safeguards (NMSS) or by State regulatory agencies authorized under NRC's Agreement State Program. The Office of Nuclear Regulatory research functions to assist NMSS or State regulatory agencies by conducting research to increase confidence in regulatory decisions.

Low-level waste refers to those radioactive wastes containing source, special nuclear, or byproduct material that are acceptable for disposal in a land disposal facility. Acceptability criteria are specified in 10 CFR Part 61. Generally, LLW consists of waste not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or byproduct material as defined in section 11e.(2) of the Atomic Energy Act (uranium or thorium tailings and waste).

LLW encompasses a wide range of radioactive wastes with an equally wide range of physical and chemical characteristics. All industries; hospitals; medical, educational, or research institutions; private or government laboratories; or facilities forming part of the nuclear fuel cycle (e.g., nuclear power plants, fuel fabrication plants) using radioactive materials as a part of their normal operational activities generate LLW just as they generate other types of hazardous and nonhazardous wastes. LLW consists of the radioactive materials themselves and other materials that have been in contact with radioactive materials and are contaminated or suspected of being contaminated. It is generated in many waste types, forms, and amounts. It ranges from trash that may be only slightly contaminated with radioactivity to highly radioactive materials such as activated structural components from nuclear power reactors. The form of the waste may be solid, liquid, or gaseous and it may consist of a variety of chemical forms. LLW ranges in activity from thousands of curies per cubic meter to less than a few microcuries per cubic meter.

In its efforts to ensure safe disposal of low-level radioactive waste (LLW), the Office of Nuclear Regulatory Research has developed a strategy for conducting research on issues of concern to the U.S. Nuclear Regulatory Commission. The resulting program plan has been published in NUREG-1380. The plan provides an integrated framework for the LLW research program to ensure that the program and its products are responsive and timely for use in NRC's LLW regulatory program. The plan addresses technical and scientific issues and uncertainties associated with the disposal of LLW, presents program goals and ways for achieving them, establishes a long-term strategy for conducting the confirmatory and investigative research needed to meet these goals, and includes schedules and milestones for completing the research. Areas identified for investigation include waste form and other material concerns, failure mechanisms and radionuclide releases, engineered barrier performance, site characterization and monitoring, and performance assessment. The plan calls for projects that (1) analyze and test actual LLW and solidified LLW under laboratory and field conditions to determine leach rates and radionuclide releases; (2) examine the radiological and chemical characteristics of decommissioning waste from nuclear power stations (10 CFR 61.55); (3) examine the short- and long-term performance of concrete-enhanced LLW burial structures and high-integrity containers; (4) investigate the long term effectiveness of disposal unit covers in controlling water infiltration into disposal units (10 CFR 61.51); (5) examine the information needed at time of closure for predicting future facility performance (10 CFR 61.13); (6) attempt to develop a predictive model of the rate of radionuclide release at the boundary of waste disposal units (10 CFR 61.13); (7) examine radionuclide transport behavior, pathways, uptake, and transfer coefficients of radionuclide releases from LLW disposal facilities (10 CFR 61.13); and (8) attempt to predict water movement and contaminant transport through low-permeability saturated media and unsaturated porous media (10 CFR 61.13).

LOW-LEVEL RADIOACTIVE WASTE CLASSIFICATION, WASTE FORM STABILITY, FAILURE MECHANISMS, AND RADIONUCLIDE RELEASES

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Regulation of low-level radioactive waste(LLW) disposal facilities requires that the NRC staff have an understanding of the complex scientific and technical variables associated with classifying LLW for radionuclide and chemical content and ensuring the waste forms produced by the nuclear power stations and disposed in licensed burial sites maintain stability for at least 300 years.

Licensing decisions concerning the classification and stability of LLW are made within the framework of 10 CFR Part 61. Part 61 requires a waste classification system for radionuclides in the waste and also establishes waste form characteristics the radioactive waste forms must have to protect the public health and safety from releases of radioactive materials, protect the general population from inadvertent intrusion, and ensure the stability of the low-level radioactive waste form after closure.

Part 61 requires that LLW be classified into three general classes, Class A, B, or C, depending on the radioactivity content and half-lives of the specific radionuclides. In addition, Part 61 also requires that Class B and C wastes be stabilized. Classification of LLW is described in Section 61.55 and Tables 1 and 2 of Section 61.55 list the concentration limits for Class A, B and C waste.

In Section 61.56, the characteristics and stability requirements for LLW is specified. LLW is required to be stabilized to prevent deterioration of waste forms when buried. The stability requirements of 10 CFR Part 61 are intended to ensure the LLW does not degrade structurally and affect the overall stability of the site through slumping, collapse, or other failure of the burial facility cover which may lead to water infiltration. Waste form stability is also intended to reduce potential exposure to an inadvertent intruder. Paragraph 61.56(b)(1) requires that LLW be structurally stability so the waste form can maintain its physical shape under normal disposal conditions such as overburden weight, microbial activity and internal factors that include chemical and radiation changes. Structural stability of LLW can be provided by the waste itself (activated metal), by processing the waste to a stable form (e.g. cement solidification), by placing the radwaste in a container(e.g. high integrity container) or structure that provides stability after disposal.

To ensure that LLW generated by nuclear power stations is processed to a stabilized waste form, an NRC technical position has been developed which provides test methods and acceptance criteria to the NRC staff for implementing 10 CFR Part 61 waste form stability requirements.

This research paper discusses the low-level radioactive research plan for conducting research on issues of concern to the U.S. Nuclear Regulatory Commission associated with LLW classification, waste form stability, failure mechanisms of waste forms, radionuclide and chemical releases from cement solidified waste forms, and past experiences involving unsuccessful cement solidification of radwaste. The topics discussed include the regulatory need for performing the research studies, research strategies, current research activities, applications of the research results to NRC's licensing and regulatory environment, and new research studies expected to begin in FY91.

Low-level radioactive waste collected from operating nuclear power stations will be used in the research programs described. Emphasis will be placed on performing research using decontamination LLW and other radwaste containing ion-exchange resins that have been shown to be difficult to solidify in formulated cement. Research studies involving activated metals and the determination of chemical concentrations in LLW that could lead to unstabilized solidified cement will be described. Included in the research described will be programs to address concerns about measuring hard-to-measure radionuclides in LLW, evaluating scaling factors used to determine activity levels of radionuclides difficult to determine at low concentrations, leaching of radionuclides and chelating agents from solidified decontamination radwaste, using field lysimeters containing solidified LLW and activated metals to determine stability and radionuclide releases and transport in soils under actual environmental conditions, and performing the waste form technical position tests on actual cement solidified radwaste.

HYDROGEOLOGY RESEARCH

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NRC's research in the field of hydrogeology addresses the regulatory issues in both the high-level waste (HLW) and low-level waste (LLW) management programs. Contractor research studies and NRC staff efforts examine state-of-the-art methods and theories in analyzing ground-water flow and radionuclide transport. This work has been incorporated into international cooperative efforts for studying ground-water flow models, HYDROCOIN, and geosphere transport models, INTRAVAL, related to geologic disposal of radioactive waste.

Low-Level Waste Management

(1) Regulatory Criteria

The LLW hydrogeology studies are in support of LLW staff and Agreement States' efforts to implement 10 CFR Part 61 technical criteria. For example, Section 61.50 dealing with site suitability states that "The disposal site shall be capable of being characterized, modeled, analyzed and monitored," and "The disposal site must provide sufficient depth to the water table that ground-water intrusion, perennial or otherwise, into the waste will not occur...In no case will waste disposal be permitted in the zone of fluctuation of the water table," and "The hydrogeologic unit used for disposal shall not discharge ground water to the surface within the disposal site." Section 61.51 dealing with disposal site design states that "Covers must be designed to minimize to the extent practicable water infiltration, to direct percolating or surface water away from the disposed waste...", and "The disposal site must be designed to minimize to the extent practicable the contact of water with waste during storage, the contact of standing with waste during disposal, and the contact of percolating or standing water with wastes after disposal."

(2) Program Objectives

The LLW research studies address regulatory issues dealing with (1) site characterization methodology and techniques, (2) baseline and operational monitoring, (3) field testing for hydraulic and transport properties, (4) numerical analysis of ground-water flow and contaminant migration, and (5) performance assessment of the engineered systems and natural ground-water flow systems. The emphasis has been on conducting small (10 cm) to moderate scale (10 m) flow and transport experiments for partially saturated heterogeneous porous media. These laboratory and field tests have produced significant results in (1) identifying the processes and parameters of concern, (2) assessing site characterization techniques, and (3) simulating flow and solute transport for two-phase systems.

(3) Research Contractor Studies

This work is now being expanded through a cooperative (i.e., MIT, New Mexico State University, PNL, University of Arizona, and SNL) research field program at the Las Cruces Trench site. Joint laboratory and numerical simulation studies precede and support detailed experimental designs of the field migration experiments. This program is providing unique and detailed data bases previously not available for testing flow and transport theories and validating models. Numerous modeling teams (e.g., SNL, LBL, PNL, MIT, NMSU, Princeton, KEMAKTA/SKI) are simulating these experiments as part of an international cooperative effort for studying validation of geosphere transport models, INTRAVAL.

Ongoing studies are examining the issue of characterizing hydrogeologic heterogeneities at various scales and determining appropriate means of estimating effective flow and transport parameter values for modeling radionuclide transport to the accessible environment. For example, the MIT stochastic theory for estimating mean flow and transport properties is being used to simulate large-scale migration as part of the Las Cruces Trench studies. These studies illustrate the interaction between theoretical and laboratory and field studies that test contaminant transport models. PNL and University of Arizona (UAZ) scientists are also evaluating state-of-the-art field techniques and methods for site characterization, and monitoring.

MIT, Princeton University, PNL, NMSU, SNL and HydroGeologic Inc. are using existing models or developing new codes and modeling approaches for simulating and studying ground-water infiltration, flow and transport of radionuclides. For example, HydroGeologic Inc. has developed the VAM2D code to model ground-water flow and transport of radionuclides and decay-chain daughters through partially-saturated soils. PNL has recently issued an "Infiltration Evaluation Methodology" which examines ground-water infiltration issues and approaches for estimating the temporal-spatial ground-water fluxes into subsurface engineered LLW facilities. Princeton University researchers have developed new numerical approaches to simulate sharp wetting fronts associated with rapid infiltration events, and have shown their applicability to modeling arid ground-water flow systems. This work has greatly enhanced the numerical capabilities and technical confidence in the next generation of LLW performance assessment models.

(4) Regulatory Significance

The ultimate success of the hydrogeology research program is its ability to inform the licensing staff and Agreement States as to the critical flow and transport processes that need to be idenified and characterized at the specific sites, and how to conservatively model radionuclide release rates to the accessible environment. This work is being used to review site characterization and design studies for unsaturated zone sites. The aforementioned studies have produced detailed technical guidance, numerical codes, data sets, evaluations of field and laboratory techniques and methods, and an understanding of the nature and extent of the remaining unresolved technical issues.

MELCOR Simulation of Long-Term Station Blackout at Peach Bottom*

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MELCOR is a fully integrated computer code that models all phases of the progression of severe accidents in nuclear power plants[1]. It is being developed for the U. S. Nuclear Regulatory Commission by Sandia National Laboratories (SNL), and is designed to provide an improved severe accident/source term analysis capability relative to the older Source Term Code Package (STCP)[2]. BNL has a program with the NRC to verify and apply the MELCOR code to severe accident analysis for several plants.

This paper presents the results from a MELCOR calculation of a Long-Term Station Blackout Accident Sequence with failure to depressurize the reactor vessel. Peach Bottom, a boiling water reactor with Mark I containment, was used in the analysis. The paper also compares MELCOR predictions with STCP calculations for the same sequence[3]. This sequence assumes that batteries are available for six hours following loss of all power to the plant. Station blackout sequences have often been determined to be important contributors to the risk from severe accidents. Following battery failure, the reactor coolant system (RCS) inventory is boiled off through the relief valves by continued decay heat generation. This leads to core uncovery, heatup, clad oxidation, core degradation, relocation, and, eventually, vessel failure at high pressure. STCP has calculated the transient out to 13.5 hours after core uncovery. MELCOR calculations have been carried out to 16.7 hours after core uncovery. The results include the release of source terms to the environment. The MELCOR Peach Bottom model consists of 19 control volumes, 33 flow paths, and 66 heat structures. The reactor core is modeled with 33 cells (3 concentric radial rings and 11 axial levels). MELCOR either explicitly or parametrically models all key in-vessel and ex-vessel phenomena.

The predicted timing of key events with MELCOR and STCP show similar trends. However, STCP predicts lower plenum dryout, vessel failure, and containment failure to occur 30 minutes to 1 hour earlier than MELCOR. This is partly because core relocation occurs more gradually and over a longer time period in MELCOR. Both codes predict deflagrations to occur in the secondary containment, shortly after drywell failure. The predicted duration of deflagrations is longer for MELCOR, because the MELCOR plant

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model considers many compartments in the reactor building, with a corresponding delay in burn propagation, compared to a single volume representation for STCP.

At the end of the calculation, MELCOR predicts much lower environmental release fractions of Te, Sr, La, Ce, and Ba, and STCP predicts lower fractions of I, Cs, and Ru. MELCOR and STCP predict similar release and retention of I and Cs from the fuel during in-vessel core meltdown; however, the higher environmental release fractions of I and Cs from MELCOR can be attributed to late revaporization from the RCS after the core debris penetrates the reactor vessel. This phenomenon is not modeled in STCP. The lower Te, Sr, and Ba releases is because MELCOR calculates debris release into the cavity over a much longer period of time, based on successive penetration failures in the 3 rings. The MELCOR meltdown model, therefore, results in less vigorous core concrete interactions than STCP, and this in turn results in lower release of the fission products associated with this phase of the accident.

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RESULTS OF RECENT ORNL FISSION PRODUCT RELEASE TESTS

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The effects of time, high temperature, and atmosphere were explored in ORNL tests VI-2, VI-3, and VI-4. These tests were performed using vertically oriented segments of Zircaloy-clad UO_2 fuel that had been irradiated to ~42 MWd/kg U in the Belgian BR3 reactor. Tests in steam were conducted at temperature plateaus of 2000, 2300, and 2700 K; test VI-4 was conducted in a hydrogen-helium atmosphere at 2450 K.

Results of test VI-2, which were run for 60 min at 2300 K, showed that 63% of the fission product cesium had been released. The release rate for cesium, expressed as a fraction of the remaining inventory released per minute, decreased tenfold during the test.

The fuel in test VI-3 was heated at 2000 and 2700 K for 20 min at each temperature. Essentially 100% of the cesium, krypton, and antimony were released. No measurable release of either cerium or europium was observed. In both VI-2 and VI-3, the steam oxidation of the Zircaloy cladding followed the Urbanick-Heidrick rate data.

The conditions used in test VI-4 (21 min at 2450 K) in hydrogen atmosphere allowed the Zircaloy cladding to melt and react with the UO_2 pellets. The behavior of the free-standing fuel specimen, which collapsed at ~2150 K, was somewhat similar to the fuel swelling observed in essentially identical ST-1 fuel at Sandia National Laboratories. The releases of cesium and krypton were close to those noted in steam-atmosphere tests. The release of fission product europium was much higher than that from tests in steam. However, the release of antimony was much lower, behavior which parallels that of tellurium.

Total iodine releases are not yet available for these tests. The amount of volatile iodine (I_2, HI, or methyl iodide) formed in these tests ~1 s after release from the fuel was less than 1% of the released amount. This also occurred when an oxidized stainless steel liner was used in the first thermal gradient tube of test VI-3.

The total mass of aerosol material produced in the hydrogen atmosphere of test VI-4 was less than half of that produced in a comparable steam-atmosphere test. The fraction of released cesium collected in vapor form, compared with that in aerosol form, was much higher in test VI-4 than in steam-atmosphere tests VI-2 and VI-3.

Our modeling work shows that Booth diffusion coefficients (random diffusion from fuel grains) correlate all of our test results very well. Burnup and grain size must be considered. Agreement with the ANS-5.4 Standard

Fission Product Release Model is very good. Results of the diffusion modeling studies show that the $Zr-UO_2$ reaction ("liquefaction") in test VI-4 did not significantly affect the release rates for krypton and cesium. Diffusion coefficients for the VI-series tests agree well with those from the HI-series tests.

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STATUS OF VICTORIA DEVELOPMENT AND ASSESSMENT*

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VICTORIA is a mechanistic computer code designed to analyze fission product behavior within the reactor coolant system (RCS) during a severe accident. It provides detailed prediction of the release and transport of radionuclides and non-radioactive materials during core degradation. These predictions account for the chemical and aerosol processes that affect radionuclide behavior. Finally, it has the potential to determine the long-term revaporization of deposited radionuclides.

The current release model uses a Booth term for diffusion from grains, surface and molecular diffusion for transport through the open porosity, molecular diffusion for transport through the gap and through openings in the damaged cladding, and advection and diffusion within the bulk gas. Anticipated release models include a FASTGRASS type model as an option to the Booth model, a model for oxidized fuel, and a model for highly degraded fuel, such as a rubble bed.

The current list of 25 chemical elements treated by VICTORIA includes not only the volatiles that are of primary concern in the event of an accident, but also those which may interact with the volatile species and inhibit their release, those which are easily measurable experimentally and so are beneficial for validation, and those which are important because of their quantity within the reactor vessel and cooling system. Transport and interaction of a set of 167 chemical species are analyzed by VICTORIA. Both chemical equilibria (determined by minimization of Gibbs free energy) and phase behavior are treated. Future developments include a complete accounting for structure surface and condensed film interactions, a preprocessor to enable the element and species set to be easily modified, and additional refinement of the thermochemical data base.

The aerosol model in VICTORIA accounts for condensation/evaporation, deposition onto surfaces, and agglomeration. Deposition mechanisms modeled are gravitational settling, turbulent deposition, Brownian motion, thermophoresis, diffusiophoresis, and inertial deposition in curved channels (bends). Agglomeration mechanisms include Brownian motion, relative gravitational motion, interactions in a shear field, and inertia in a turbulent field. Additional mechanism that are being added include re-entrainment of deposited aerosols, revaporization due to decay heating, and boundary layer phenomena.

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Much of the validation done to date has used the HI and VI out-of-pile tests done at Oak Ridge National Laboratory and the ST-1,2 in-pile tests done at Sandia National Laboratories. Each of these tests concentrated on release from fuel. Additional test that are slated for validation studies include the Marviken and LACE tests to assess aerosol behavior, separate-effects tests done in Canada to assess release from oxidized fuel, and integral tests, such as the French Phoebus FP experiments.

The VICTORIA code is now adequate for reactor accident analyses of radionuclide release during the early stages of core degradation, aerosol processes in the RCS, and vapor deposition in the RCS. Further development is needed to enable prediction of radionuclide release during later phases of core degradation, especially after vessel failure, resuspension of deposited materials at the time of RCS depressurization, and radionuclide entrapment and revaporization in ruptured steam tube accidents and other by-pass scenarios.

An international partnership--including four laboratories within the U.S. and one each within the United Kingdom and Canada--has been formed to continue the development and assessment of the code. The presentation discusses the models used, the experiments and tests necessary for validation, and what further developments are being undertaken.

SPARTA PROJECT: FACILITY DESCRIPTION AND FIRST TEST RESULTS

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ABSTRACT

Pool scrubbing is a key element in LWRs source term estimation. In fact, a number of pathways currently identified for the most risk significant sequences involve pathway segments through water pools (e.g. a transient without scram for a BWR and a containment bypass for a PWR). Previous work on pool scrubbing has shown that observed decontamination factors (DFs) are very sensitive to the test conditions. The steam content of the carrier gas, pool temperature and many other parameters have a strong influence on fission product retention processes. SPARTA experimental program (Suppression Pool Aerosol Retention Test Apparatus) has been planned at the ENEA-CRE Casaccia Laboratories to evaluate the overall DFs in water pools at different test conditions.

Features of the SPARTA facility are the aerosol generation system for MnO, CsI and CsOH fission product aerosol simulants, the "small pool" (2.5 m diameter, 4.0 m height) where parametric tests and scaling correlation will be carried out, the "large pool" (8 m diameter, 10 m height), which is the main feature of the plant, with its full scale X-quencher (as the one in the GE-BWR Mark III design) and horizontal vent (0.7 m hole diameter).

Service and carrier gas cylinders (N2, Ar, He), steam production by the near VAPORE facility together with the connecting pipes complete the plant. On-line process instrumentation, sampling stations and off-line samples analysis devices make up the plant instrumentation and allow to draw the test observations and results.

The paper describes in some details the main experimental features of the SPARTA facility and summarizes the preliminary observations following the first performed test.

LARGE-SCALE HDR-HYDROGEN MIXING EXPERIMENTS
- TEST GROUP E 11 -

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During severe accidents in light-water reactors, substantial amounts 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 distributions and associated mixing phenomena in multi-compartment geometry in order to plan for proper mitigation measures. In the case, that no mitigating measures are provided it has to be demonstrated at least that the containment integrity is not threatened by the combustion of hydrogen including local detonations.

A review of the four known past experiments on hydrogen distribution together with their major experimental characteristics and a qualitative overview of some important results has been given at the 16th WRSM. It was concluded that the combination of issues important for the physical phenomena controlling the H_2-distribution mechanisms, namely:

- sufficiently large scale of the experimental facility
- high H_2-release rates
- multi-compartment geometry with sufficiently large dome volume representation
- multiple steam and H_2-injection phases
- different axial positions for H_2-release
- examination of the efficiencies of mitigating features

has not been investigated yet.

As argued and demonstrated at the 16th WRSIM, the HDR-facility satisfies all of the criteria cited above. Therefore, the preliminary H_2-distribution experiment T 31.5, performed in December of 1987 during Phase II of the HDR-Safety Program constitutes a milestone in terms of hydrogen distribution experiments. This especially because not only 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 was generated, but also an assessment performed of the present status of computer codes and modelling in this area by virtue of a blind post-test exercise with broad international participation. Results of the comparisons between data and predictions were presented and discussed at the 17th WRSIM.

On the basis of the experiences gained both with respect to experimental and computational aspects, a complete, major test group, E 11, was designed and performed in the context of HDR-Safety Program Phase III (1988-1992) in the summer of 1989. This test series consisted of eight different experiments covering all aspects of H_2-distribution and mitigation features such as: small and large LOCA controlled containment atmosphere conditions, axially different injection positions, multiple injections of gas mixtures (He/H_2:

85/15 % vol plus 60/40 % vol) multiple steam injections into different compartments internal and external spray (on top of dome) initiations, boiling sump simulation, "dry" energy addition and venting measures at three different axial positions. Special attention was given to the completeness, accuracy and reliability of instrumentation especially with respect to local H_2-concentrations and steam/air concentrations. 48 H_2-sensors were applied throughout the facility with the same number of steam/air concentration sensors, and thermocouples in their immediate vicinity. In addition, velocities and structural temperatures were measured inside the containment as well as throughout the annular gap between steel shell and secondary concrete.

Experiment E 11.2 (this is now OECD-Standard Problem No. 29) and E 11.4 were chosen for two PHDR-Benchmark Exercises in the context of blind post-test predictions with broad international participation.

The paper will outline all major facility features used during the individual experiments and will present the detailed experimental procedures as performed.

Experimental results of Test E 11.2 will be used to demonstrate the major findings and the most important effects by mitigating measures on the containment atmosphere, especially the H_2-concentration throughout the facility.

The results will be shown both in the format of time histories of the important quantities as well as by virtue of axial profiles with time as parameter.

The institutions and codes participating in the PHDR-Benchmark Exercise on E 11.2 will be identified.

Results of comparisons between data and predictions will be presented pending the decision of the international working group meeting prior to the 18th WRSM.

ACE Program Phase C: Molten Corium Concrete Interaction (MCCI) and Corium Melt Coolability Experiments

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A series of experiments are being performed at Argonne National Laboratory (ANL), investigating the interaction of molten core material with concrete and its coolability with water. The experiments are being performed under the auspices of the ACE (Advanced Containment Experiments) and the MACE (Melt Attack and Coolability Experiments) Projects, which are sponsored by a consortium of 10 foreign countries and four domestic organizations, namely: 1) U.S. Department of Energy; 2) U.S. Nuclear Regulatory Commission; 3) Electric Power Research Institute (EPRI); and 4) Westinghouse Electric Corporation. EPRI is serving as the project manager for the ACE, as well as the MACE experiments.

The general objectives of the ACE MCCI experiments are to measure: 1) the releases of refractory fission product species i.e., oxides of Lanthanum, Barium, Cerium and Strontium; 2) the physical and chemical character of the aerosols generated; and 3) the thermal-hydraulic aspects of the interaction, including the concrete ablation rate. The motivation for performing these relatively complex experiments arose because of the lack of data, and the very different estimates of the releases predicted by the extant codes e.g., VANESA and MAAP for similar conditions of the MCCI.

The approach employed for the ACE MCCI experiments is to use real materials i.e., UO_2, ZrO_2, Zr, SSt, inactive fission products, representative concrete i.e., basalt limestone-common sand and limestone in a facility with tungsten electrodes supplying sustained internal heat generation in the melt to reach temperatures of approximately 2500K. The physical size of the corium and concrete interaction zone is 50 cm x 50 cm and approximately 300 kg of molten corium material, is used.

The test matrix for the set of MCCI experiments was developed with the advice and consent of the ACE Project Technical Advisory Committee. Both the PWR and BWR corium composition and the various (German, Soviet, U.S.) concrete compositions are represented and the initial Zr oxidation is varied from 30 to 100%.

Four experiments have been performed successfully. The fission product releases obtained have uniformly been quite small. Analyses of the thermal-hydraulic and the chemical interaction occurring during these experiments is proceeding at several institutions with codes like CORCON, VANESA, SOLGASMIX and MAAP. Comparisons of blind predictions with industry and NRC sponsored codes will be performed for 2 tests.

The principal objectives of the MACE project are to: 1) obtain data on coolability with water of corium melt interacting with basemat concrete; and 2) develop models for insertion in the industry and NRC-sponsored codes. These experiments will address the outstanding issue of accident management and termination through long-term coolability of the melt discharged into containment, on vessel failure, during a postulated severe accident scenario.

The approach employed for the MACE tests is similar to that for the ACE MCCI tests in that heated representative corium melt material is reacted with a representative concrete basemat and water is added on top of the corium melt after the MCCI begins. The steam and non-condensible gases produced are routed to a heat exchanger to obtain the rate of heat transfer between the melt and the overlying water.

A fundamental question is whether a thick stable crust is formed, which will substantially reduce the heat transfer rate. The stability of the crust formed is scale-dependent and the heat transfer rate and the long-term coolability are functions of crust cracking and allowing melt-water contact. A supporting project at University of Wisconsin is performing crust stability scaling analysis to determine and extend the applicability of the data obtained in the MACE experiments to prototypic situations.

A scoping MACE test was performed about a year ago and was reported at the 1989 Water Reactor Safety Information Meeting. That test employed approximately 130 kg of corium melt interacting with a 30 cm x 30 cm block of concrete and cooled by water. A crust formed in that test, was supported by the electrodes, which had moved inwards from their original position. The heat transfer from melt to water was substantially reduced due to the stable crust. The MACE test design is being changed to preclude support by electrodes. The next MACE test will employ about 450 kg corium melt interacting with a 50 cm x 50 cm block of limestone-common sand concrete. Further tests are planned in order to cover the other parameters which affect melt coolability.

THE PROBABILITY OF MARK-I LINER FAILURE

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We are proposing a probabilistic methodology* as an overall systematic, disciplined approach for addressing the Mark-I liner attack issue. The probabilistic framework encompasses the key features of the phenomenology, yet it is flexible enough to allow independent quantification of individual components as it may arise from independent research efforts. As a first step in this direction, we assembled, discussed, and took into consideration in the quantification proposed all relevant prior work. Furthermore, as an essential aspect of the overall methodology, we have asked most of those whose work has been referenced and/or used in this report (i.e., the experts) to comment. The details of this work, the comments received, and our responses are included in NUREG/CR-5423. As an even more important characteristic of the methodology, we hope that other quantifications (or information relevant to such) of independent components will become available in the future, such that we can aim for convergence and closure.

In the course of the work we discovered that the eventual development of the full probabilistic model (i.e., framework *and* quantification) requires an iteration between structuring the framework and delving into the physics that affect its components. Also, we came to the conclusion that for a complex phenomenological issue, such as the Mark-I liner attack, the multilayer of uncertainties can quickly get out of hand; thus, this issue-focused *gradual* approach attempted here is a much preferred medium as compared to the usual global PRA efforts. This is consistent with what we learned from our first application of this methodology*, which was for the perhaps even more complex issue of α-failure (steam-explosion-induced containment failure). Judgment is required to achieve a balance between what uncertainties to consider, and what is preferable to bound, while at the same time retaining a proper level of flexibility in the approach. Also, judgment is required to achieve the proper focus in the iteration mentioned above, between structuring the framework and sorting out the physics. Fortunately, the method is simple enough to set up and compute so that other points of view in structure can be readily incorporated.

We have limited our interest to the so-called low-pressure scenario, and the Peach Bottom Atomic Power Station has been used for the specifics. We have attempted to envelop the behavior by means of two melt release scenarios and two variations within one of them. For a flooded drywell, the results indicate conditional failure probabilities of 3.5×10^{-6}, 1.2×10^{-4}, and 6×10^{-5}, which according to our probability scale used in the quantification indicate that liner meltthrough can be considered as "physically unreasonable." By contrast, in the absence of water, the same quantification approach (i.e., a similar level of conservatism in creating the inputs) leads to failure probability levels of 0.63 and 1; that is, a virtual certainty. We would not exclude the possibility, however, that in this latter case our approach may turn out to prove unduly conservative in not recognizing certain (unknown as yet) mitigating aspects of the complex phenomenology.

The above, bottom line, results are the consequence of an intricate combination of physical processes associated with melt release and spreading behavior, corium-concrete interactions, and liner heat transfer, such that no single mechanism could be identified as dominating in the context of the quantification employed here. The only possible exception to the above is for oxidic melts where contact with the liner in the absence of water produces failure almost under any conceivable conditions (of pool depth and superheat). Thus we can only speak of key contributing factors, a summary of which, as identified in this study, is as follows.

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* Risk Oriented Accident Analysis Methodology (ROAAM)

- The rapid release of oxidic melts can be readily bounded in total volume and thus in resulting liner submergence. By contrast, metallic melts can involve considerably larger volumes but very slow release rates that favor retention in the pedestal area and, again, reasonable bounds on liner submergence.
- Melt superheats are low initially and short-lived once the melt is subject to interactions with coolant and concrete. In particular, even large quantities of zirconium in the melt cannot produce sustained superheated conditions; the role of chemical energy is rather limited to influencing the time duration of the superheated regime.
- The transient aspects of liner heatup couple crucially with the transient of the melt superheat in determining failure conditions.

Several potentially important aspects of the phenomenology were not considered as either premature at this time, or for lack of adequate data. They can be summarized as follows.

- Liner failure by melt attack can only be highly localized and the relief path created *could* easily plug (narrow clearance between the shell and concrete wall; insulation filling).
- Lower head failure *could* occur by creep rupture, with a highly superheated metallic phase circulating through a porous matrix of oxides.
- Slow melt release scenarios *could* favor significantly higher in-pedestal melt retention than considered here, and a slow, creeping process of spreading characterized by melting-and-freezing cycles. Although this would provide additional margin-to-liner-failure, it may be important in endangering the pedestal wall itself.
- Shallow melt pools (say up to ~30 cm) *could* be easily quenchable when flooded by water in a wide geometry (no wall effects). The currently planned, international, MACE program at ANL is likely to resolve this issue in the near future. Clearly, this would permit a less conservative treatment and the demonstration of even greater margins to failure.
- Local, intense fuel-coolant interactions, especially during the melt release and spreading phase *could* further enhance margins to failure by promoting melt quenching, retention within the pedestal, and widespread distribution of any melt that made it through the pedestal door.
- Crusts formed on the liner *could* come loose under the dynamics of the agitated melt. Although this possibility is considered rather unlikely, it may be worthwhile to devise an experimental test for it.

Further to the possible consideration of some of the above areas, we believe that ongoing work on various aspects of the pool-to-liner heat transfer and corium-concrete interactions will provide additional insights, and depth, to the quantification proposed herein. In particular, we suggest that better depiction of the regimes of interest can be obtained by utilizing crucibles with preheated side walls, low initial melt superheats, and prototypic power densities and melt compositions. Furthermore, we hope that the probabilistic inputs (*pdf*'s and *CR*'s) proposed in this study will receive a scrutiny, enrichment, and refinement in future efforts that aim for convergence and eventual resolution of this issue. This latter aspect is an integral part of the methodology proposed herein.

To conclude, we recommend that for an adequate understanding of the probabilistic estimates quoted in this summary one should read the whole report in detail. Indeed, the status of the issue, including the intangible uncertainty, can only be surmised by consulting carefully the comments received by the expert reviewers and our responses, both contained in Appendix F of NUREG/CR-5423.

Progress Report for the U.S. National Seismograph Network

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Good progress has been made in all aspects of the U.S. National Seismograph Network (USNSN) over the past year. Station sites have been selected for many of the USNSN stations (Figure 1) and cooperative agreements are being written with the local institutions and agencies. Funding is available for all of the stations shown in Figure 1.

As of mid-December 1989, signed contracts were in place for all major components of the USNSN. The equipment specified in these contracts meet or exceed all of the requirements which were originally determined to be important for the hardware at the beginning of this project. More amazing is the fact that the cost of the equipment under contract agrees almost exactly with the estimates made at the beginning of the project three years ago. There has been essentially no cost increase in the project. The USNSN equipment under contract include the following systems: 1) seismometers, 2) station processor for field deployment, 3) satellite telemetry for both field deployment and the master station, and 4) network processor for the USGS National Earthquake Information Center (NEIC).

The USNSN seismometers will be supplied by Guralp Systems Ltd., and will include both high-gain and low-gain sensors. The high-gain sensors will be a version of the CMG-3 seismometer with custom modifications made to meet specific requirements of the USNSN and will be available in both surface mounted and borehole configurations. The low-gain sensors will be essentially a stock version of the CMG-5 accelerometer, and will be surface mounted only.

Quanterra Inc., will provide the station processor for field deployment. The processor is a CMOS (low power) 68,000 microprocessor-based subsystem with an OS-9 real-time operating system. The processor will also contain the following: 1) 24-bit ADC's (one per seismometer input), 2) a precision clock, 3) seismometer control and calibration circuitry, 4) real-time state-of-health monitoring, 5) 1.4 Mbytes of memory, and 6) asynchronous RS-232C ports.

Scientific-Atlanta Inc. will provide the complete telemetry system. All data will be transmitted over a general purpose, bi-directional, host-to-host, computer network in a star configuration implemented by means of a time division multiple access (TDMA), Ku-band (14-16 GHz), very small aperture telecommunications (VSAT) satellite network. A VSAT unit, including a 1.8m (diameter) antenna, feedhorn electronics, and decoding and control electronics, will be located at each field site.

The network processor function is implemented on a local area network (Ethernet) of up to eight (depending on load) μ VAX 3800 processors supplied by the Digital Equipment Corp. Analyst access to the on-line data base is by means of VAX work stations.

UNITED STATES NATIONAL SEISMOGRAPH NETWORK

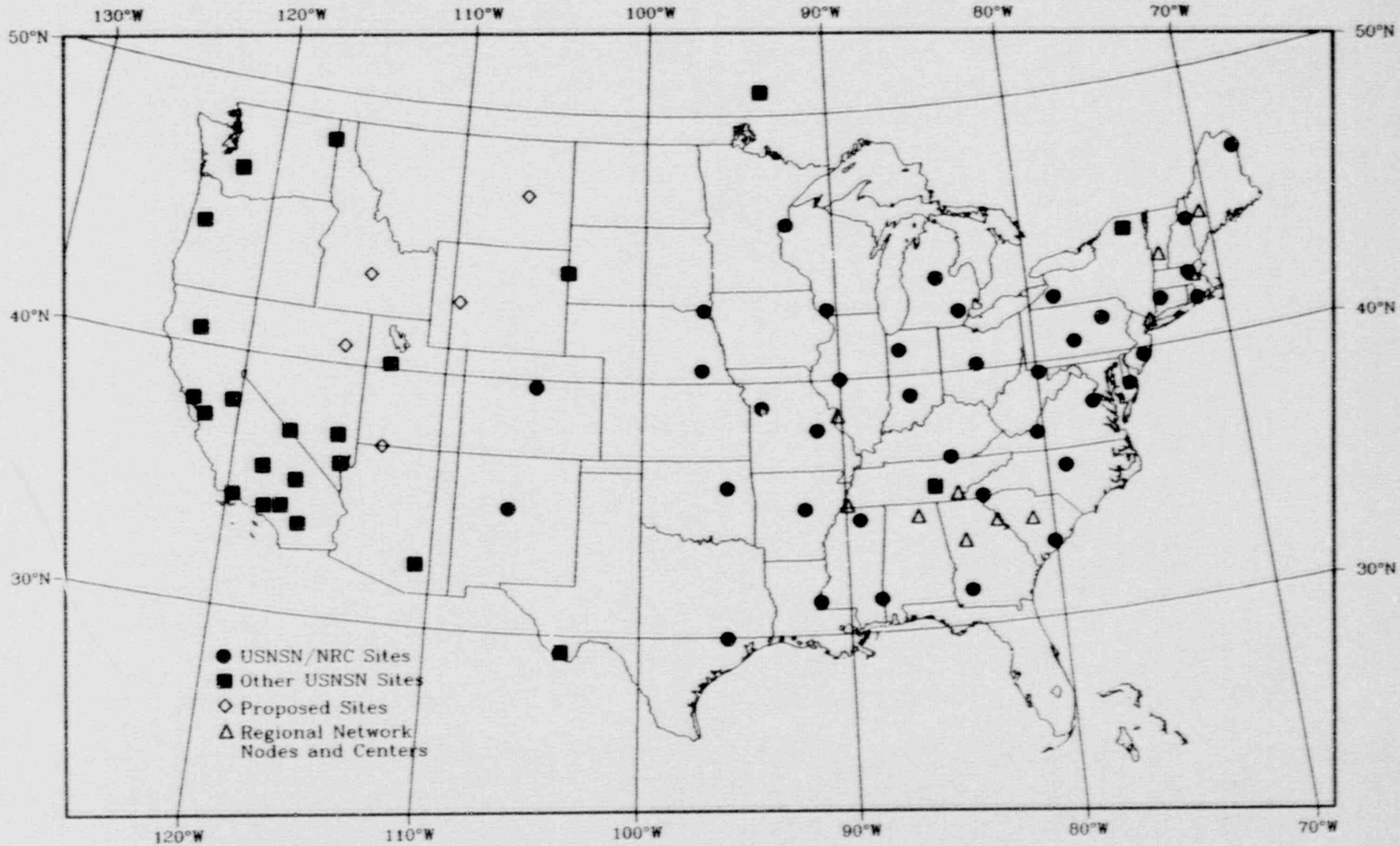

Figure 1

COOPERATIVE NEW MADRID SEISMIC NETWORK

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The development and installation of components of a U. S. National Seismic Network (USNSN) in the eastern United States provides the basis for long term monitoring of eastern earthquakes. While the broad geographical extent of this network provides a uniform monitoring threshold for the purpose of identifying and locating earthquakes and while it will provide excellent data for defining some seismic source parameters for larger earthquakes through the use of waveform modeling techniques, such as depth and focal mechanism, by itself it will not be able to define the scaling of high frequency ground motions since it will not focus on any of the major seismic zones in the eastern U. S.

Realizing this need and making use of a one time availability of funds for studying New Madrid earthquakes, Saint Louis University and Memphis State University successfully competed for funding in a special USGS RFP for New Madrid studies. The purpose of the proposal is to upgrade the present seismic networks run by these institutions in order to focus on defining the seismotectonics and ground motion scaling in the New Madrid Seismic Zone. The current networks focus on earthquakes in southeastern Illinois and the question of the northern extent of the New Madrid Seismic Zone, the New Madrid Seismic Zone proper, and the southern extent of the New Madrid Seismic Zone in Arkansas and Tennessee. While the present network of vertical component sensors has defined long, spatially linear seismicity patterns, depth control and the on scale recording of large earthquakes are problematic. Depth control, which is necessary for the proper three dimensional definition of seismicity, can only be obtained by having a very dense seismic network. Proper recording of ground motion requires both the recording of three components of ground motion and also recording with a much greater dynamic range than is possible with the present 12 bit systems. The justification for these improvements is that the New Madrid Seismic Zone is the most continually active region in the eastern U. S., with rates of activity high enough to permit a rapid improvement in knowledge of the earthquake process and ground motion scaling.

The proposed network is designed both to complement the U. S. National Seismic Network and to make use of the capabilities of the communication links of that network. At the lowest level, new three component sensors (seismometers, accelerometers and very broad band sensors) will be placed in field sites. Seismic data will be digitally transmitted over UHF radio links to a collection node. At the collection node, the data from the field sites will be time stamped and passed through an event detection algorithm. Waveforms of detected events will be buffered for transmission using the USNSN satellite telemetry system to regional network data centers at St. Louis and Memphis. These centers will also receive selected stations of the USNSN and will use the combined data sets to catalog the earthquake location, routinely determine earthquake source parameters, and archive the data for use of other researchers. The system is designed to have the capacity for additional data.

An important type of additional data, not yet funded, would be that from a borehole array of sensors in order to relate the "hardrock" motion at depth to the surface motion recorded at sites within the Mississippi River flood plain. The New Madrid region one of the few sites of earthquakes in the country which has very thick deposits of alluvium cover (600 - 1000 meters). There have been no studies on the ground motion amplification due to such a thick sedimentary column. Other data collection sites would include Corps of Engineer dams, DOE processing facilities, and metropolitan Memphis.

The relevance of this network to NRC needs lies primarily in seismic hazard analysis with input in two related areas. The refinement of the spatial distribution of seismicity and the delineation of coherent seismicity patterns will affect the definition of source zones. Recent analyses have focused on sites at some distance from New Madrid, in which case the specification of detailed source zone geometry is not so important. There are some critical facilities within 200 km of New Madrid, for which the source zone areal extent becomes critical.

The other area lies in reducing the variance in the ground motion estimation model used in the hazard analysis. Some recent earthquakes have demonstrated remarkable coherence in the radiation pattern of high frequency ground motion, even at distances of 100 km. The observed radiation pattern is related to the direction of motion on the fault. The incorporation of the expected mode of faulting into the ground motion estimation model may reduce the variance. With the current level of seismicity and the intent of the network design to learn from smaller earthquakes, the first two years of operation of the new network will provide significant constraints on the ground motion estimation models in terms of frequency dependent anelastic attenuation, site effects related to site geology, and source parameters related to seismic wave excitation.

Geodetic Measurement of Crustal Strain in the Eastern United States

by
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A network of stations whose positions are known at the $1:10^6$ to $1:10^8$ level is being established in the United States over an area reaching from the eastern edge of the Rocky Mountains to the Atlantic Ocean. Monitoring of this network will allow determination of crustal strain and deformation in the eastern United States. This network will have three primary components. The most accurate component will be a set of Very Long Baseline Interferometry (VLBI) stations and stations with continuous observations using Global Positioning System (GPS) technology. In most cases, those VLBI and GPS systems will be co-located. The differential positions of these fundamental stations, which are expected to be 10 to 15 in number, will be monitored with an accuracy of $1:10^8$. This fundamental network will also provide a firm tie to the worldwide coordinate system provided by the International Earth Rotation Service (IERS) through inclusion of several VLBI stations which contribute to the IERS and, through inclusion of GPS stations contributing to GPS satellite orbit determination, to the coordinate system in which GPS satellite positions are expressed. The second component will be a regional component of 75 to 100 stations established using transportable GPS receivers with a relative accuracy in the $1:10^7$ to $1:10^8$ range.

The final component, a densification component, will consist of 1000 to 2000 stations whose differential positions will have a relative accuracy in the $1:10^6$ to $1:10^7$ range.

As of June, 1990, this high accuracy geodetic network is well along toward establishment. There are twelve stations at which VLBI measurements have been made, six stations with large, fixed VLBI systems and six where mobile VLBI equipment has made measurements. The nucleus of the regional component of the network is also in place. In 1987, in support of the Nuclear Regulatory Commission (NRC) a 45 station strain network was established in the eastern United States using GPS. Reduction of this data demonstrated, through repeatability and comparison with VLBI that precisions/accuracies of 1 to $5:10^8$ were being achieved. Observations over this network were repeated in early 1990, with seven additional stations added, and are currently being reduced. Densification has also already begun, both in the form of GPS surveys performed specifically for crustal strain and deformation monitoring, and GPS surveys performed to meet other needs, which are accurate enough to additionally serve to support strain and deformation monitoring requirements. An example of the first type of survey is a densification network established in support of the NRC in southeastern Maine to monitor possible deformation in that area. This network was also established in 1987 and repeated in early 1990. Another example is a GPS survey in the New Madrid area in which NGS is working cooperatively with the New Mexico State University under U.S. Geological Survey funding. Examples of the second type of survey are statewide networks established in Tennessee (60 stations), in New Mexico (40 stations) and in Florida (145 stations) to support surveying needs. Comparisons at common stations between the Florida state network and 45 station strain network results showed agreement at the $1:10^7$ level.

Additional state networks and regional stations are scheduled to be established shortly in Wisconsin and Maryland. The Wisconsin work is scheduled to begin in late June, 1990. These surveys will be tied to the existing 45 station regional strain network. Additional surveys are under active consideration.

This paper will describe the requirements for the high accuracy crustal strain and deformation network in the eastern United States, the current status of network establishment, accuracies being achieved, and future plans.

Seismic Study of Soil Dynamics at Garner Valley, California

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The Garner Valley downhole array (GVDA) of force-balanced accelerometers was designed to determine the effect that near-surface soil layers have on surface ground motion by measuring *in situ* seismic waves at various depths. Although there are many laboratory, theoretical and numerical studies that are used to predict the effects that local site geology might have on seismic waves, there are very few direct measurements that can be used to confirm the predictions made by these methods. The effects of local site geology on seismic ground motions are critical for estimating the base motion of structures. The variations in site amplifications at particular periods can range over a factor of 20 or more in comparing amplitude spectra from rock and soil sites, e.g., Mexico City (1985) or San Francisco (1989). The basic phenomenon of nonlinear soil response, and by inference severe attenuation of seismic waves, has rarely been measured although it is commonly observed in laboratory experiments. The basic question is whether or not the local site geology amplifies or attenuates the seismic ground motion. Because the answer depends on the interaction between the local site geology and the amplitude as well as the frequency content of the incoming seismic waves, the *in situ* measurements must sample the depth variations of the local structure as well as record seismic waves over as wide a range as possible in amplitude and frequency.

The Garner Valley downhole array is about seven kilometers east of the San Jacinto fault and 35 km from the San Andreas. Geologically the site is on a prehistoric lake bed within the southern California granitic batholith. Drilling and seismic velocity logs show that the lithology is 19 m of soil, overlying 25 m of weathered granite with a granitic basement below. The shear wave velocity profile (from measurements made by the USGS) show 220 m/s for depths 0 - 19 m, 580 m/s for depths 19 - 63 m, 1310 m/s for depths 63 - 100 m. Based on velocity logs at the USGS Keenwild 300-m hole, located about six kilometers from GVDA, we estimate the shear wave velocity at 2650 m/s from 100 - 300 m.

With regard to seismicity GVDA is centrally located in an active part southern California. It is at the northern end of the Anza seismic gap, possible M > 6.5 expected, and at the southern end of the 1899 and 1918 (M > 6.5) ruptures. Earthquake hypocenters are routinely located by the USGS/Caltech southern California network. In addition to these high-gain vertical seismometers the USGS and the University of California, San Diego operate a 10-station, three-component, network of seismometers in the immediate vicinity of the Anza seismic gap.

At present GVDA consists of five, three-component, dual-gain accelerometers placed at depths of 0 m, 6 m, 15 m, 22 m and 220 m. An additional downhole accelerometer (46 m) and four additional surface accelerometers are to be installed within the next year. Each component has a high-gain channel, 0.1 g maximum, and a low-gain channel, 2.0 g maximum. Each channel is digitized at 16 bits (90 dB) and recorded at 500 Hz. The dual-gain output allows GVDA to record ground accelerations from 3×10^{-6} g to 2.0 g. The low-gain accelerometer channel has a flat amplitude response from 0 to 100 Hz; the high-gain channel has a flat acceleration response from 0.025 to 100 Hz.

Since operations began in July 1989 through April 1990, GVDA has recorded about 125 earthquakes ranging in magnitude from 4.7 to 1.2. Two-thirds of the earthquakes have magnitudes less than 2.5; however 43 earthquakes have magnitudes greater than or equal to 2.5, including five earthquakes with magnitudes greater than 4.0. The largest peak acceleration, 88 cm/s^2, was recorded at the surface on the vertical component for a M 4.2 earthquake (December 2, 1989) at an epicentral distance of 8.2 km. The peak acceleration due to a P-wave is about twice what the S waves generated on the horizontal components. The two largest magnitude earthquakes M 4.6 and M 4.7 were aftershocks of the February 28, 1989 Upland M 5.5 earthquake. Although these two earthquakes were 107 km and 110 km from GVDA, they generated very clean accelerograms with peak horizontal accelerations of 2.1 and 1.3 cm/s^2, respectively.

Although the ratio of peak acceleration at the surface to the peak acceleration at 220 m is highly variable from one earthquake to the next, the ratio is generally around four to five. A very stable and consistent estimator of the amplification (ignoring resonances) is the ratio of the seismic moment computed at the surface compared to that computed at 220 m. Seismic moment is a direct measure of the size of the earthquake. The seismic moment is directly related to the low frequency range of the Fourier displacement amplitude spectrum (the acceleration spectrum divided by frequency squared). Over almost five orders of magnitude in seismic moment (3×10^{18} - 8×10^{22} dyne-cm) the amplification factor from bedrock to the surface is 6.2. The amplification factor 6.2 is of even more concern given that the quality factor Q_s for shear waves ($Q_s = 1/2\beta$ where β is the damping factor in percent) in the upper 22 m is on the order of 20 while Q_s is about 500 or greater in the basement rock. Thus the observed amplification exists even after the seismic waves have passed through the highly attenuative near-surface weathered granite and soil. This attenuation is readily apparent when comparing acceleration amplitude spectra at various levels. The amplitude spectra at 220 m exceeds the amplitude spectra at 0 m for frequencies greater than about 40 Hz. Thus the attenuation is the dominant factor for frequencies above 40 Hz while amplification prevails for smaller frequencies.

In examining spectral ratios, surface spectra divided by the 220 m spectra, the amplification occurs over broad frequency bands rather than being localized as one would expect for resonances. Given the current velocity/depth structure we are in the process of generating synthetic accelerograms and the amplitude spectra to predict the resonant frequencies.

Although GVDA has been fully operational for only nine months, it has performed according to its design. The high-gain and low-gain accelerograms from the December 2 M 4.2 earthquake, which nearly saturated the high-gain channels, are indistinguishable. Amplitude spectra are usable from 0.017 Hz to 100 Hz. Both small and large earthquakes are recorded with a large dynamic range and high fidelity. The data from GVDA will provide the basis for calibrating other methods used to predict ground motion in a vertically heterogeneous medium as well as improve our understanding of the complex interaction between amplification and attenuation.

New Evidence for Great Earthquakes in the Pacific Northwest

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In the past 2000 years large parts of the coast along the Cascadia subduction zone have probably moved during earthquakes that were confined to four episodes each lasting no more than several centuries. The most recent episode, about 300 years ago, was marked by regional subsidence in southern Washington and northern Oregon, by localized subsidence in southern Oregon, and by localized uplift, subsidence, and thrust faulting in northern California. The penultimate episode, about 1100 years ago, brought subsidence to northern Washington, parts of southern Washington, and northern Oregon. This episode probably included two events of subsidence and at least one event of uplift and thrust faulting in northern California. Two earlier episodes occurred within a few centuries of one another about 1700 years ago. The coast of southern Washington and northern Oregon subsided during each of these two episodes, and subsidence and thrust faulting occurred in northern California during at least one of them.

For southern Washington and northern Oregon the coastwise extent of movement is better explained by interplate earthquakes on the Cascadia subduction zone than by intraplate earthquakes from small, unconnected sources. Such interplate seismicity probably continued southward, the ruptures splaying into a belt of upper-plate folds and faults that comes ashore in southern Oregon and northern California. Less probably, movement on the folds and faults occurred independently of interplate earthquakes, in which case great interplate earthquakes may have been restricted to southern Washington and northern Oregon. Interplate ruptures would have been located largely offshore in the case of southern Washington and northern Oregon, but largely onshore farther south. Although the lengths of most inferred ruptures are unknown, some individual ruptures probably ran only part of the length of the coast between northern Washington and northern California.

Great earthquakes on the Cascadia subduction zone probably recur at very uneven intervals. Such behavior complicates seismic forecasting for the Pacific Northwest.

PALEOLIQUEFACTION INVESTIGATIONS ALONG THE ATLANTIC SEABOARD WITH EMPHASIS ON THE PREHISTORIC EARTHQUAKE CHRONOLOGY OF COASTAL SOUTH CAROLINA

by

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The largest historical earthquake along the Atlantic seaboard occurred near Charleston, South Carolina in August of 1886. The cause of this enigmatic event has yet to be resolved and there remains considerable debate regarding the frequency of large earthquakes in the Charleston area and the potential for similar events to occur elsewhere in the region. Strong ground shaking associated with this event resulted in the formation of numerous seismically-induced liquefaction features in the near surface sediments. Previous studies have shown that older paleoliquefaction features, interpreted to be the result of prehistoric earthquakes, are also represented in the geologic record. The radiocarbon ages of these paleoliquefaction features suggest that during the 5000 to 6000 years prior to the 1886 earthquake, at least five other liquefaction episodes occurred near Charleston (600$\pm$100 YBP, 1200$\pm$100 YBP, 3200$\pm$200 YBP, 5150$\pm$500 YBP, and >5150 YBP). Based on the number and type of features and organic samples evaluated, liquefaction episodes 1200$\pm$100 YBP, and 3200$\pm$200 YBP are most strongly supported by the available data. The 600$\pm$100 YBP, 5150$\pm$500 YBP, and >5150 YBP episodes are based on reliable, but more limited information.

With the exception of the 1886 event, no moderate to large historical earthquakes have occurred along the Atlantic seaboard. However in many coastal areas extending from New Jersey to Florida, local geologic and hydrologic conditions are suitable for the generation and preservation of paleoliquefaction features. To obtain a prehistoric seismic record for this region, we have conducted a systematic search for evidence of prehistoric earthquake activity outside the Charleston area. Investigations have been completed at over 1000 potential liquefaction sites, extending from Charleston southward to the Georgia/Florida state line, and northward to Cape May, New Jersey. Although suitable sites have been investigated throughout the region, paleoliquefaction features have been found almost exclusively in South Carolina. The lone exception discovered during this investigation is located just north of the South Carolina/North Carolina state line.

Detailed radiometric age dating studies have been completed at ten paleoliquefaction sites located within South Carolina. Each site studied is located far south or north of Charleston. The ages of the southern paleoliquefaction sites all correlate to Charleston paleoliquefaction episodes and each is thought to be the result of an earthquake originating in the established Charleston source area. In addition to 1886 liquefaction features, the southern sites provide independent confirmation of the 600$\pm$100 YBP and 1200$\pm$100 YBP Charleston paleoliquefaction episodes. The ages of the northern sites correlate with and provide independent confirmation of these same two Charleston episodes.

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While all the southern and most of the northern paleoliquefaction sites and features probably result from large earthquakes occurring in the established Charleston source area, an 1800±200 YBP northern paleoliquefaction episode has no clear parallel Charleston area episode and may have been caused by a local earthquake. Alternately, the earthquake responsible for this episode may have originated in the established Charleston source area but associated paleoliquefaction features have not yet been identified there. Additional studies would be needed to confirm the existence of this postulated "new" earthquake source.

Our search for evidence of past large earthquakes targeted areas where present geologic and hydrologic conditions are conducive for the generation of seismically-induced liquefaction features. However, data suggest that due to variability in climatic conditions and fluctuations in sea levels, ground-water levels over much of our study area may have been much lower in the prehistoric past. Since saturated conditions are required for seismically-induced liquefaction to occur, such changes in ground-water levels would be expected to play a significant role in determining the spacial and temporal distribution of paleoliquefaction features. Based on a review of climatic data and sea level curves for the southeastern U.S., ground-water levels are thought to have been at or near present levels for only the past 2000 years. Consequently, the paleoliquefaction record is probably most complete for this period. During the period 2000 to about 5000 years ago, ground-water levels fluctuated in a range of about one to four meters below present level. The paleoliquefaction record for this interval probably includes only those earthquakes which occurred during periodic transgressive seas and/or wetter climatic periods. Prior to about 5000 years ago the climate in the region was drier and sea level was even lower. Such conditions would severely reduce or eliminate the potential for seismically-induced liquefaction in the deposits that we studied and may explain the absence of very old paleoliquefaction features even in the Charleston area prior that time.

The mean return period between liquefaction episodes identified in the geologic record (including both those originating in the Charleston area and the single event to the north) is about 1000 years. However, the time between liquefaction episodes has varied from about 2000 years to about 600 years in more recent times. The decrease in apparent return periods is probably related to "gaps" in the record due to the absence of liquefaction episodes associated with earthquakes that occurred during times of decreased liquefaction potential. The return periods between the past four large liquefaction associated earthquakes has exhibited a time-predictable pattern of about 600 years. Since only about 100 years have elapsed since the 1886 event, the probability of a similar earthquake occurring within the Charleston area over the next several decades is inferred to be low. While the potential for an earthquake large enough to produce significant features is low, the hazard presented by smaller earthquakes should not be overlooked.

With respect to seismic hazard elsewhere along the Atlantic seaboard, no paleoliquefaction evidence of large prehistoric earthquakes originating outside of coastal South Carolina was found. This is consistent with the concept of a unique seismotectonic setting there. However, the absence of paleoliquefaction features elsewhere must be viewed in the context of prehistoric climatic conditions and fluctuations in sea level. Given the impact of these factors on ground-water tables, the paleoliquefaction record is probably complete only for the last 2000 years, intermittent for the period 2000 to 5000 YBP, and is limited or non-existent for earlier times.

U.S. Accident Management Program

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The NRC is working with industry to achieve the goal of assuring that each operating utility has in place a framework and a plan for responding to a potential severe accident. The approach emphasizes planning to make maximum use of existing equipment and resources during the management of a potential accident. That is, no major design changes or new regulations are anticipated in order to achieve this goal. Further, the utility accident management plan must be structured to accept new information on severe accidents when it becomes available. It is the responsibility of industry to define and implement plant-specific accident management programs. The regulatory responsibility of NRC is to articulate overall accident management principles and evaluate industry guidelines and implementation. The role of NRC accident management research is to provide the base of knowledge needed to support the regulatory responsibility.

In the short-term, the research program is tasked to develop information to evaluate industry guidelines for accident management (A/M). The industry A/M program plan is centered around three products The first is a generic guidelines document for evaluating a utility's current A/M capability. This document will be issued by NUMARC in mid CY 1991. The second is a technical basis report on A/M guidance. This report will be issued by EPRI in mid CY 1991. The third is a set of vendor-specific A/M guidance to be issued by the individual owner groups in early CY 1992. The NRC generic letter on A/M will be issued in mid CY 1992 and will refer, as appropriate, to these industry products. The long-term research program is oriented to providing information that will focus on providing the technical basis for evaluating the five elements of an A/M framework: 1) strategies, 2) instrumentation. 3) guidance and analyses aids, 4) decisionmaking responsibility, and 5) training. Only a few strategies will be selected for detailed evaluation.

Although the concept of managing accidents is not new, the discipline of accident management itself is new and is still being defined. The primary regulatory goal is to have a functioning A/M program in place at each utility. The goals of the A/M research program are directly tied to achieving the primary regulatory goal. The three research goals are:

(1) Identify and assess candidate severe accident management <u>strategies</u>.

This research goal assures that new information on severe accidents is usefully defined and then practically applied to a better understanding of accident management. This research goal defines the main efforts expected in the long-term NRC accident management research program.

(2) Develop information to assess industry guidelines on the accident management framework which defines the necessary components of a functioning utility severe accident management plan.

This research goal supports the short-term needs of the regulatory goal to issue a generic letter on accident management in CY 1992.

(3) Assess options and develop methods and criteria for NRC review and audit of industry accident management framework and capabilities.

This research goal supports the long-term accident management audit requirements of NRR. After an accident management program has been defined, and is functioning at a utility, the NRC may need to periodically exercise an option to audit that capability. Such accident management audits are not anticipated for the next few years. During this longer-term period, research will be undertaken to investigate several options for performing this audit function.

SEVERE ACCIDENT NATURAL CIRCULATION AND DEPRESSURIZATION IN PWRS[a]

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BACKGROUND

During the latter stages of severe accidents in pressurized water reactors (PWRs), the coolant inventory in the core is depleted and fuel rod heatup ensues. Because of the radial power distributions in the core, the fuel temperatures in the center part of the core are higher than at the periphery. Subsequent heat transfer to the fluid (predominantly steam) results in less dense fluid near the core center. A buoyancy-driven flow pattern can thus be established with hotter fluid proceeding upward in the middle of the core, which is replaced with cooler fluid from the periphery. Computer code simulations [1, 2, 3] have predicted a natural circulation flow pattern within the reactor pressure vessel (RPV) that includes well-established flows within the core region and the upper plenum. Experiments conducted by Westinghouse [4] confirm these flows. In addition, the experiments indicate that hot leg countercurrent flow could be present. This flow involves hot fluid exiting the RPV through the upper portion of the hot leg and cooler fluid returning to the RPV through the lower portion of the hot leg. After mixing in the steam generator inlet plenum, some of the fluid enters steam generator tubes and exchanges heat with the tubes and secondary fluid.

RISK IMPLICATIONS

The net result of these natural circulation flows is to transport heat from the reactor core and deposit it in other structures, such as the upper plenum internals, hot leg pipe, and steam generator tubes. This has the effect of reducing core heatup rates, while causing the ex-core structures to heat up. Additionally, having hotter fluid in the loops when the power-operated relief valve (PORV) cycles exposes the pressurizer surge line and downstream piping to increased temperature loading. Ultimately, a failure of ex-vessel piping could occur due to creep rupture or even melting prior to the time of a lower head failure (if core melt progression is not terminated). The risk implication of such a failure (ex-vessel) is that ejection from the lower head would most likely be at low pressure, and direct containment heating (DCH) would not be of great concern. While that would reduce risk, having the RPV at low pressure could increase risk because of a higher probability of in-vessel steam explosions. If the ex-vessel failure occurred in the steam generator tube(s), the potential for containment bypass would exist. The failure may also be small enough (or late enough) so that limited depressurization of the RPV would take place, and DCH would still be a concern.

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DEPRESSURIZATION

The likelihood that natural circulation will result in a temperature-induced failure of the reactor coolant system (RCS) prior to lower head (or penetration) failure from corium attack can not be quantified with sufficient accuracy that this failure can be relied upon to prevent DCH. For this reason, intentional depressurization of the RCS is being studied as an accident management strategy. Results to date [5] have been obtained for the Surry nuclear power plant using simulated TMLB' conditions. These results indicate that intentional depressurization using the power operated relief valves (PORVs) and upper head vent can reduce the pressure to a sufficiently low value that DCH could be mitigated. To reliably accomplish this depressurization, the Surry Emergency Operating Procedures (EOPs) would need to be modified so that the functional restoration procedure that initiates depressurization would be entered during a TMLB' sequence. In addition, systems that support the long term operation of the PORVs during a station blackout, for example the batteries and air to operate the valves, would need to be modified to extend the time over which they are effective. The next step in the Surry evaluation is to examine the risk reduction capabilities of intentional depressurization including the potential effect of steam explosions resulting from relocation of the core material at low pressures.

A program is currently under way to evaluate the capability of all current PWRs to intentionally depressurize through application of the Surry depressurization results to other PWRs. The first step in this program is to identify plant parameters that are important for depressurization, for example, plant geometry and systems parameters and plant capacity and setpoint parameters. These parameters will be used to group PWRs and to relate the capability of these groups to the Surry results. For example, the capacity of the PORVs is expected to be important in determining how representative the Surry depressurization results are for other PWRs. If there are groups of plants where the Surry results are not applicable, additional depressurization analyses may be necessary.

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ASSESSMENT OF UNCERTAINTIES IN SEVERE ACCIDENT MANAGEMENT STRATEGIES

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Recent progress on the development of Probabilistic Risk Assessmet (PRA) as a tool for qualifying nuclear reactor safety and on research devoted to severe accident phenomena has made severe accident management an achievable goal. Severe accident management strategies may involve operational changes, modification and/or addition of hardware, and institutional changes. In order to achieve the goal of managing severe accidents, a method for assessment of strategies must be developed which integrates PRA methodology and our current knowledge concerning severe accident phenomena, including uncertainty. The research project presented in this paper is aimed at delineating uncertainties in severe accident progression and their impact on severe accident management strategies.

The paper describes two tasks currently under way. The first task as its focus, two workshops delinating uncertainties in severe accident management strategies. These workshops focused on:

- o dominant sequences with respect to core melt frequency
- o dominant sequences with respect to various risk measures
- o dominant threats which challenge safety functions
- o dominant threats with respect to failure of safety systems

Severe accident management strategies can be generically classified as:

- o use of alternative resources (i.e. air, water, power, etc.)
- o use of alternative equipment (i.e. pumps, generators, etc.)
- o use of alternative actions (i.e. manual depressurization, manual injection, etc.)

For each sequence/threat and each combination of strategy there may be several options available to the operator. Each strategy/option involves phenomenological and operational considerations regarding uncertainty. These include:

- o uncertainty in key phenomena
- o uncertainty in operator behavior
- o uncertainty in system availability and behavior
- o uncertainty in information availability (i.e. instrumentation)

The objectives of these workshops were to:

a) answer a set of questions regarding these uncertainties for a number of proposed strategies/options and

b) to determine whether or not the state-of-knowledge permits an early assessment of the viability of these strategies/options.

The second task will focus on the development of a methodology for assessing severe accident management strategies. The core of the methodology will be the ultimate use of validated (here validated in a long-term goal as none are now available) computer codes coupled with PRA techniques which will enable a determination of a) the state of the reactor, and b) the likely effects of intervention. This methodology will also include an assessment (snapshot) of the uncertainties, qualitatively and if possible quantitatively, of the state of knowledge with respect to accident management. A unique feature of the second task will be the inclusion of operational uncertainty (from operators actions to the observables available to him) as well as phenomenological (model) uncertainties. An event tree or other logic based formalism will be developed to both guide the workshops and to present the final results of the study.

The two tasks also deal with the robustness of existing tools and models for assessing accident management. This will involve selecting some key sequences and building tree structures that are phenomenologically based. Successful accident management requires sure actions at a branch to stop or slow the progression of the accident. Splinters off the branch based on phenomenologically possible paths (multi-dimensional branching) also will be considered.

Identification and Assessment of Containment and Release Management Strategies

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Brookhaven National Laboratory, under the auspices of the U. S. Nuclear Regulatory Commission, is investigating accident management strategies which could help preserve containment integrity or minimize releases during a severe accident. The objective is to make use of existing plant systems and equipment in innovative ways to reduce the likelihood of containment failure or to mitigate the release of fission products to the environment if failure cannot be prevented. Many of these strategies would be implemented during the later stages of a severe accident (i.e., after vessel breach) and sizeable uncertainties exist regarding some of the phenomena involved. A majority of the strategies identified go well beyond existing procedures and often depend on the specific containment type. Strategies for all of the five different containments used in the U.S. are being considered: BWR Mark I, Mark II, and Mark III, as well as PWR Ice Condenser and Large Dry Containments. Accident management strategies related to the in-vessel phase of a severe core melt accident are being dealt with under another NRC program.

For each containment type the most likely challenges are identified and existing emergency guidelines and procedures are reviewed as to how they address these challenges. Where needed and when possible, new strategies are devised. The feasibility and effectiveness of these new strategies are assessed, making due allowances for the complicated phenomena and associated uncertainties involved. Both beneficial and adverse effects of the suggested strategies are considered. The additional strategies are also evaluated for consistency with existing procedures and for practicality in terms of available resources of personnel and equipment. For both the identification and assessment of strategies, maximum use of is made of available existing information from such sources as NUREG-1150, other PRAs as well as industry and NRC reactor safety reports. The assessments place special emphasis on the indicators regarding plant status that the operators can realistically expect to have during a severe accident and which would enable them to implement a strategy and monitor its results. The influence of human performance on each strategy also receives special attention.

Challenges and relevant strategies are systematically identified and grouped via the development of safety objective trees for the containment under analysis. These trees start from a safety objective which is maintained by certain safety functions. The functions are threatened by particular challenges via a number of mechanisms. Each mechanisms can be addressed by one or more strategies. For example, one safety objective for all containments is preventing containment failure. This objective is met by the safety functions of containment integrity control, containment pressure control, and containment temperature and thermal attack control. Containment pressure control can be challenged by rapid pressurization or slow pressurization. In a Mark I containment, slow pressurization can result via the mechanisms of noncombustible gas buildup, mass and energy addition, gas burning, or loss of heat rejection. Strategies which address gas burning, for instance, are: maintaining an inert atmosphere, venting to control gas concentration, spraying, or diluting of atmosphere.

Once a comprehensive list of challenges and strategies has been established, the strategy assessment must take into account the effect of accident phase and previous history on strategy implementation. Strategies for similar challenges may vary depending on the accident stage during which the challenge occurs, or different considerations may become important for a strategy during different accident phases. At a minimum, strategy assessment must recognize whether a strategy is to be implemented before core melt, during core melt, in preparation for vessel breach, during the ex-vessel phase, or during the radiological release phase of the accident. For example, venting is an important containment and release strategy for BWRs. However, the implications of venting are vastly different for "clean" venting than for post core-melt venting. Under some circumstances, clean venting could be used to control containment conditions prior to fuel damage to prevent ECSS failure. For post core-melt venting, at least three different "strategies" can be identified with different concerns regarding diagnostics, downside risks, human factors, etc. These depend on whether (1) venting is done early to reduce containment pressure in anticipation of a large pressure increase resulting from vessel breach, (2) venting is required later in the accident due to slow pressurization from core-concrete interaction, or (3) venting via the wetwell is used to provide fission product scrubbing to reduce emissions from a containment leaking under pressure.

To account for the timing considerations alluded to above, accident management strategies are best assessed in the context of the sequences during which they may occur. In assessing the containment and release strategies for each containment type, a large enough number of sequences were selected to assure (1) all identified challenges to safety functions were covered, (2) potential failures of safety systems were considered, and (3) sequences with the potential for high consequences were included.

Examples of several strategy assessments will be presented.

Failure Of PWR-RHRS Under Cold Shutdown Conditions.

Experimental Results From The PKL Test Facility.

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The Residual Heat Removal System (RHRS) of a PWR is designed to transfer thermal energy from the core after plant shutdown and maintain the plant in cold shutdown or refuelling conditions for extended periods of time.

Initial reactor cooling after shutdown is achieved by dissipating heat through the steam generators (SGs) and discharging heat to the condenser by means of the turbine steam bypass system. When the reactor coolant temperature has dropped to about 160°C and pressure has been reduced to 30 bar the RHRS is placed into operation. It reduces the coolant temperature to 50°C within 20 hours after shutdown.
The time margin for establishing alternate methods of heat removal following a failure of the RHRS depends on the Reactor Coolant System (RCS) temperature, the decay heat rate and the amount of RCS inventory. During some shutdown operations the RCS may be partially drained (e.g. to perform SG inspections). Decreased primary system inventory can significantly reduce the time available to recover the RHRS's function prior to bulk boiling and core uncovery.

The PKL test facility, which simulates a 1300MW 4-loop PWR on a scale 1:145, was used to carry out a RHRS failure test based on the following scenario:

- reactor in cold shutdown condition, approx. 30 hours after shutdown, i.e. decay heat level at P=0.7% (26MW)
- core exit temperature 50°C (122°F), pressure p=1 bar
- RPV head still in place but RCS partially drained (loop ½-filled with water, the rest nitrogen)
- one steam generator with nominal secondary inventory (12.2m), temperature t=20°C, depressurized to p=1 bar, emergency feedwater available
- remaining three steam generators' secondaries empty i.e. filled with air and isolated.

Under these steady state conditions a failure of the complete RHRS was postulated.

The principal objectives of the test were to determine:

- energy transport from the core (steam/nitrogen mixing) and heat transfer mechanisms in the operational SG in the presence of large amounts of nitrogen
- the necessary pressure on the primary to reach steady state conditions under which the decay heat could be transferred to the secondary of the operating SG.

The test results show that after a failure of the RHRS saturation temperature is reached in the core within 20 minutes and the subsequent steam generation at near atmospheric pressure (low steam density, high void fraction) leads to formation of two-phase mixture which fills the hot legs and parts of the inlet plena of all SGs. The steam rises from the core through this two-phase mixture not only into the tubes of the operational SG1 (water-filled secondary)- the temperature of the N_2-filled primaries and air-filled secondaries of the isolated SG2,3 and 4 also follow quite closely the core exit temperature showing excellent mixing of steam and N_2 in the isolated SGs. In this transient phase, when the pressure rises at about 1.8 bar/h, all SGs act as heat sinks of different capacities, of course. While the secondary pressure of the operational SG1 is limited to p=2 bar by keeping the main steam line bypass valves fully open, the pressure on the primary reaches steady state at a pressure of p=6 bar. It follows that a temperature difference of ΔT=40K is sufficient to transfer all decay heat from the core to the secondary of one operating steam generator (pool boiling, EFW available).
It can be concluded that during the heat-up following a failure of the RHRS the steam formed in the core reaches all parts of the primary circuit previously filled only with nitrogen.
As soon as an alternative heat sink is available the steam quickly finds its way to it. The rise in pressure is determined by the amount of decay heat, the pressure level on the secondary and the number of heat sinks (SGs) available.

ACCIDENT SEQUENCE ANALYSIS FOR A BWR DURING LOW POWER AND SHUTDOWN OPERATIONS[a]

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SUMMARY

The purpose of this paper is to provide a brief overview of the work being conducted at Sandia National Laboratories (SNL) with regard to the identification and quantification of accident sequences initiated in modes of operation other than full power.

Traditionally, probabilistic risk assessments (PRAs) for nuclear power plants (including those of the recent NUREG-1150 analysis) have characterized the risk associated with accidents initiated while the plant is in full-power operation. This concentration of effort on full-power events was based on the judgment that the level of risk associated with accidents that could occur during full-power operation is greater than that associated with accidents that could occur during the other modes of operation, such as low power and shutdown. The primary justification for this claim is that lower decay heat levels are generally associated with these modes of operation, so more time is available to recover from adverse situations arising in these modes. However, there are additional factors that could influence the risk associated with accidents initiated during the shutdown modes. These factors include:

1. The greater need for operator actions to prevent core damage (due to disabling of automatic safety systems during some of the shutdown modes).

2. The increased unavailability of equipment due to planned maintenance. (This results from the demand for high equipment availability during power operation, which limits the amount and length of maintenance activities that can be performed while the plant is at power.)

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

3. Lack of containment integrity caused by the opening of penetrations and hatches. (These openings, which are allowed by Technical Specifications, in many cases are necessary before the activities planned for shutdown can occur.)

In addition to the above factors, the Chernobyl accident and other precursor events that have occurred during non-full-power operation have pointed to the need for a study of the risk associated with accidents initiated during modes of operation other than full power.

As a first step in assessing the risk associated with accidents initiated during these other modes of operation, SNL, at the direction of the U.S. Nuclear Regulatory Commission, has begun a low-power and shutdown accident sequence analysis for a U.S. Boiling Water Reactor (BWR). The objectives of this study are to (1) assess the frequencies of severe accidents initiated during plant operational modes other than full power for a U.S. BWR; (2) compare the estimated core damage frequencies, important accident sequences, and other qualitative and quantitative results of this study with those of accidents initiated during full-power operation; and (3) demonstrate methodologies for accident sequence analysis for plants in non-full-power modes of operation. The scope of this project includes accidents initiated during the following five modes of operation: (1) low power (up to 15% power), (2) startup, (3) hot shutdown, (4) cold shutdown, and (5) refueling.

For each of these modes of operation, initiating events (IEs) with the potential to lead to core damage have been identified. At present, five major groups of initiating events have been identified: (1) transient events, (2) loss-of-coolant accident (LOCA) events, (3) decay heat removal (DHR) challenge events, (4) special events (e.g., criticality and inadvertent overpressurization events), and (5) hazard events (i.e., internal fire and flood events).

For the IEs identified as applicable for each mode of operation, event trees have been constructed. As would be expected (given that five modes of operation are being examined), a large number of event trees are necessary to describe the possible accident sequences. For the most part these event trees are being analyzed using the same steps as for event trees in a full-power PRA: (1) construction of fault trees, (2) determination of event probabilities or frequencies, (3) quantification of sequences, and (4) documentation of results. Given the magnitude of such an effort and the number of sequences involved, a Coarse Accident Sequence Screening Analysis is under way using the detailed event trees that have been constructed. This coarse screening uses conservative numerical estimates for the events in the event trees based on simplified system models. Those sequences that could contribute significantly to core damage frequency and risk are then analyzed in detail.

The full paper will provide more details concerning the tasks of this project and will discuss the insights found thus far in the program.

STATUS OF THE SURRY LOW POWER AND SHUTDOWN PRA

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SUMMARY

Traditionally, probabilistic risk analyses[1] of severe accidents in nuclear power plants have limited themselves to consideration of the set of initiating events occurring during full power operation. However, some analyses of accident initiators during low power, shutdown, and other modes of plant operation other than full power have been performed. These studies as well as the Chernobyl accident[2,3] and recent operating experience at U.S. pressurized water reactors suggested that risks during low power and shutdown could be significant. As such, the analysis of the frequencies, consequences, and risks of these accidents was identified as one task in the Nuclear Regulatory Commission staff's study of the implications of the Chernobyl accident to U.S. commercial nuclear power plants.

This project is an ongoing high priority effort at BNL that is expected to continue for the next several years. This project is also closely coupled to a parallel project for the Grand Gulf plant (BWR) being conducted by SNL. As the project is ongoing and still in its early stages, this paper is necessarily limited to a report of project status and progress.

The major objectives of the project are:

- To assess the frequencies of severe accidents initiated during plant operational modes other than full power operation for a commercial pressurized water reactor, Surry; and
- To compare the estimated core damage frequencies, important accident sequences and other qualitative and quantitative results of this study with those associated with full power operation assessed in NUREG-1150.

A follow-on project (to be initiated at a later date by BNL) will combine this information with accident progression, source term, and offsite consequence analyses to yield estimates of severe accident risks from these plant operational modes which again will be available for direct comparison with the NUREG-1150 results.

The project will be performed in two phases. Phase 1 represents a coarse screening analysis to identity dominant accident scenarios as well as risk dominant plant configurations and operating modes. In Phase 2, a detailed PRA will be performed for the dominant accident scenarios/operating modes identified in Phase 1, including detailed human reliability analysis, refined fault tree and event tree analysis, uncertainty propagation, and sensitivity calculations.

The current status of each of the above tasks will be discussed in the full paper and the preliminary results of the first phase will be presented.

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DIABLO CANYON INTERNAL EVENTS PRA REVIEW: METHODOLOGY AND FINDINGS

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SUMMARY

The Diablo Canyon Probabilistic Risk Assessment (DCPRA) was undertaken, at least in part, as the result of a licensing requirement placed upon Unit 1 when it was granted an operating license. The licensing requirement called for Pacific Gas & Electric (PG&E) to perform a reevaluation of the seismic design of the plant in about ten years and to include the latest state-of-the-art techniques and knowledge. The requirement further stated that both probabilistic and deterministic tools should be applied. The DCPRA represents a full scope level 1 effort, that is, both internal and external events were included in the resultant calculation of core damage frequency (CDF). This paper concentrates on the internal event portions of the DCPRA.

As indicated in the title, there are two aspects of the DCPRA review that are highlighted in this paper. These aspects, methodology of the review and review findings, are interrelated in that the review approach and scope ultimately influenced the review findings. It is important to define the review approach taken with respect to the DCPRA. The DCPRA represents a new generation of PRA's that far exceed the older generation in terms of scope and level of detail. The full paper will provide a detailed description of the review methodology, its basis for selection and an evaluation of its effectiveness. For purposes of this summary it is noted that the review methodology called for a detailed examination of selected portions of the PRA within each of the major parts of the PRA (e.g., data base, fault trees, event trees, initiating events, etc.). The authors further note that the methodology was considered very effective in handling such a large PRA.

In terms of the findings of the review, it is believed that this is the first PRA or PRA review to provide a comprehensive importance analysis of all major aspects of a PRA. In this particular case, importance analyses were carried out for the initiators, the top events of the frontline systems, the top events of the support systems as well as the major human errors and recovery actions. In addition, pair importance calculations were carried out between frontline/support system top events and support system/support system top events as well as all of the individual split fractions that make up the dominant accident sequences (88% of the total non-seismic CDF). Selected portions of the importance analyses will be included in the full paper.

APPLICATION OF NUREG-1150 METHODS AND RESULTS TO ACCIDENT MANAGEMENT*

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The risk from five nuclear power plants was examined during the NUREG-1150 program. When the analyses of the plants were complete, an effort was undertaken to examine the implications of NUREG-1150 for accident management initiatives. The framework provided by the NUREG-1150 analysis presented a means within which current accident management strategies could be evaluated and future accident management strategies could be developed and assessed.

Five separate but interrelated phases of risk management were considered: (1) prevention of accident initiators, (2) prevention of core damage, (3) implementation of an effective emergency response, (4) prevention of vessel breach and mitigation of radionuclide releases from the reactor coolant system, and (5) retention of fission products in the containment and other surrounding buildings. A risk-based methodology was developed to identify and evaluate risk management options for each of these five phases. The methodology was demonstrated through quantitative examples for the first two phases of risk management listed above. In addition, the reduction in risk for several currently implemented risk management strategies at operating plants was quantified.

This initial risk management work provides the base for an expanded role of probabilistic risk assessment (PRA) in accident management and demonstrates its usefulness. Using the integrated framework of a PRA, operator actions that could mitigate or terminate a severe accident can be identified, and the impact of the action on risk can be quantified. In addition to identifying actions that could reduce risk, such studies can be used to assist the Nuclear Regulatory Commission (NRC) in prioritizing severe accident research such that detailed evaluations are performed for the operator actions that have the greatest potential for risk reduction.

The event trees developed during a PRA for evaluating the accident

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frequencies and accident progressions can be examined to identify areas where operator actions could mitigate or terminate severe accidents. The overall impact of implementing the actions that are identified can then be determined using PRA tools.

In order to quantify the significance of an operator action, the probability of successfully performing the action must be determined. This evaluation must consider human factors such as operator diagnostic skills and response time, as well as environmental factors such as equipment accessibility and equipment survival in the accident environment.

Next, any benefit of performing the operator action, considering all ramifications of the action, can be evaluated through the PRA integrated framework. By incorporating the action into the accident frequency and accident progression event trees, all possible outcomes, including unforseen detrimental effects, that could result from the operator action are identified. Many of these outcomes are not readily evident without these tools because of the complex interactions and dependencies among events that are inherent in severe accident phenomenology.

The impact of the action can then be evaluated by comparing the risk calculated without considering the action to the risk calculated when allowing for the action. PRA methodology allows multiple measures of risk to be used in this evaluation, such as accident frequency and health and economic risk. This allows judgments to be made based on the risk measure that is most significant for the particular application being investigated. The uncertainties in human actions and phenomonology, which must be addressed for meaningful evaluations of risk to be performed, are readily included through PRA methodology.

Sandia National Laboratories is beginning an accident management program for the NRC that will further demonstrate the feasibility of using risk methods in accident management studies. Initially, the program is focusing on quantifying the risk reduction for operator actions that are identified in other portions of the NRC's accident management research. The impact of intentionally depressurizing the reactor pressure vessel during station blackout sequences at Surry is being quantified first. The impact of additional operator actions on severe accidents at Surry and Grand Gulf will subsequently be quantified.

INTEGRATED RESULTS OF THE LEVEL 1 PRA ANALYSIS FOR THE LaSALLE PLANT*

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An integrated Level 1 probabilistic risk assessment (PRA) was performed on the LaSalle County Station nuclear power plant using state-of-the-art PRA analysis techniques. In this paper selected results from the accident sequence analysis are discussed, simplifications of the methods used in this PRA for use in the NUREG-1150 analysis are mentioned, and perspectives for Mark II plants relative to Mark I and Mark III plants are provided. The analysis was performed for the Nuclear Regulatory Commission (NRC) by Sandia National Laboratories (SNL) under the Risk Methods Integration and Evaluation Program (RMIEP).

LaSalle County Station is a two-unit nuclear power plant located 55 miles southwest of Chicago, Illinois. Each unit utilizes a Mark II containment to house a General Electric BWR-5 reactor. This PRA, which was performed on Unit 2, included internal as well as external events. External events that survived the screening process and were propagated through to the Level 3 risk analysis included earthquakes, internal fires, and internal floods. The internal event accident scenarios included transients, transient LOCAs, anticipated transients without scram (ATWS), and LOCAs. The transient group is dominated by station blackouts and accidents that involve failures of the containment heat removal systems. The accident scenario profile included accidents in which core damage resulted after the containment failed or was vented and accidents that progressed to core damage with an initially intact containment. The characteristics of these two types of accidents are markedly different.

The primary purpose of the LaSalle PRA was to develop methods for integrated uncertainty analysis and propagation. The LaSalle analysis was based upon the fact that the Interim Reliability Evaluation Program (IREP), performed by SNL, had previously developed modeling techniques for the analysis of plant systems to the subcomponent level and this was judged to be of sufficient detail for any PRA-type analysis of a nuclear power plant. Attention then turned to three other areas: (1) the representation, propagation, and quantification of uncertainties in the analysis; (2) the expansion of the methods and techniques to include external events on an equal basis with the internal events; and (3) the integration of all these analyses into one integrated evaluation. Some of the methods that were used in this analysis were simplified for use in the NUREG-1150 analyses performed by SNL for the NRC. Since NUREG-1150 was performed in parallel with the LaSalle analysis, the first application of some of the methods, in a simplified form that did

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not include integration of internal and external events, actually occurred in NUREG-1150. Some additional improvements have been incorporated into the methodology as a result of this experience.

In order to incorporate external events on an equal basis with the internal events, the same accident sequence event trees and fault trees used in the internal events analysis were used in the external events analyses. The system models were expanded to include active and passive failures of additional components important in the various external events analyses (e.g., multiple spurious actuations, piping, and cabling). Instead of putting all of the location information directly on the fault trees, subsidiary Boolean equations that mapped the pipe and cable location information into the trees were constructed. In order to complete the model, additional non-safety systems that could be used to mitigate accidents were included at an equal level of modeling detail (e.g., main feedwater, condensate, and their support systems).

Methods were developed for solving the event trees and fault trees with multiple random and location failures using the SETS code. Boolean equations were constructed for the external event accident sequences in the same fashion as for the internal events.

Uncertainty distributions were developed for all the data, both internal and external. The Top Event Matrix Analysis Code (TEMAC) was developed to perform the integrated uncertainty analysis.

The final integrated result for the core damage frequency for the LaSalle County Station Unit 2 is 5% = 5.34E-06, median = 2.92E-05, mean = 1.01E-04, 95% = 2.93E-04. This includes internal, seismic, fire, and flood initiated events. The dominant initiating events using the risk reduction measure are loss of offsite power; internal fires initiated in the control room, switchgear rooms, and auxiliary building; and a valve rupture leading to an internal flood. The dominant random failures are failure to manually extinguish the control room fire, failure to correctly operate the remote shutdown panel after a control room fire, failure to recover offsite power within 1 hr., common-mode failure of the diesel generator and ECCS cooling water pumps, non-recoverable isolation of RCIC after a loss of offsite power, and failure of the containment by leakage leading to severe environments in the reactor building. Failure of the operator to vent the containment actually has a negative risk reduction measure (i.e., risk increases) because successful venting leads to severe environments in the reactor building and possible core damage from subsequent equipment failure. For initiating events, the dominant contributors to the uncertainty are the control room fire frequency, the switchgear fire frequency, loss of main condenser, and loss of offsite power. For random failures, the dominant contributors to uncertainty are (1) control circuit failures for various AC power circuit breakers and LPCI, HPCS, DG cooling, and HVAC cooling systems MOVS, pumps, and fans and (2) relay failures in the LPCI, HPCS, DG, and DG cooling systems.

INTEGRATED RESULTS OF THE LEVEL 3 PRA ANALYSIS FOR THE LASALLE PLANT*

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An integrated Level 3 probabilistic risk assessment (PRA) was performed on the LaSalle County Station nuclear power plant using state-of-the-art PRA analysis techniques. In this paper selected results from the accident progression, source term, consequence, and integrated risk analyses are discussed; enhancements to the NUREG-1150 methods that were incorporated in this study are examined; and perspectives for Mark II plants relative to Mark I and Mark III plants are provided.

LaSalle County Station is a two-unit nuclear power plant located 55 miles southwest of Chicago, Illinois. Each unit utilizes a Mark II containment to house a General Electric BWR-5 reactor. This PRA, which was performed on Unit 2, included internal as well as external events. External events that were propagated through the risk analysis included earthquakes, fires, and floods. The internal event accident scenarios included transients, transient LOCAs, anticipated transients without scram (ATWS), and LOCAs. The transient group is dominated by station blackouts and accidents that involve failures of the containment heat removal systems. The accident scenario profile included accidents in which core damage resulted after the containment failed or was vented and accidents that progressed to core damage with an initially intact containment. The characteristics of these two types of accidents are markedly different.

The LaSalle PRA developed methods for integrated uncertainty analysis and propagation. Some of these methods were used in other PRAs performed by Sandia National Laboratories--the NUREG-1150 analyses done for the NRC and the N-Reactor PRA done for the U.S. Department of Energy. An integrated assessment for both external and internal events was done in the N-Reactor analyses but not in NUREG-1150. In light of the NUREG-1150 and N-Reactor PRA experience, some additional improvements have been incorporated into the methodology. The architecture of the Level 1/Level 2 interface was modified to allow a direct integrated evaluation of all accident types (i.e., all of the plant damage states (PDS) from both the internal and external events were evaluated in one integrated run). The result is a reduction in the number of files that must be passed through the analysis, an improvement in the efficiency and the quality control of the process, and an integrated assessment of all accident types.

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* This work was supported by the U.S. Nuclear Regulatory Commission and performed at Sandia National Laboratories which is operated for the U.S. Department of Energy under contract No. DE-AC04-76DP00789.

The number of potential release segments was increased from 2 to 3 in the source term analysis in order to improve the resolution of the fission product release characteristics. In the NUREG-1150 BWR analyses, the constraint of two release segments resulted in a much coarser representation of the release characteristics. The improved modeling is especially important for accidents that involve early containment failure since the in-vessel, vessel breach, and core-concrete interaction release rates and durations can be quite different. The source term analysis also allows for different containment failure locations during the various segments of the accident and, thus, allows the release pathway to change during the course of the accident. The NUREG-1150 BWR source term analysis did not have this capability. The partition process was modified to allow source terms to be grouped based on health effect potentials, characteristics of the accident progression, and source term characteristics. In addition, the consequence and partition architecture was modified to allow an integrated assessment of both internal and external events. Another enhancement over NUREG-1150 was that the NRC's new state-of-the-art accident analysis code, MELCOR, was used extensively to support the development and quantification of the accident progression and source term models. Three complete analyses were performed starting at accident initiation and ending when the radionuclide releases to the environment stabilized. The analyses were performed for a short-term station blackout (considered two subcases: RPV at high pressure, and RPV at low pressure), an extended station blackout with injection failure at 9 hours, and a TW sequence (loss of containment heat removal) with high pressure injection failure after late containment failure. These calculations and additional sensitivity calculations were used to assess the timing of key events in the accident progression, the applicability of NUREG-1150 expert elicitation results to the LaSalle PRA, and the uncertainty in accident phenomenon.

There are important differences between Mark I, Mark II, and Mark III containment designs that affect the accident progression and source term analysis. Both the Mark I and the Mark II containments are inerted with nitrogen for hydrogen combustion control, whereas the Mark III containment uses a distributed hydrogen ignition system. The result of this difference is that hydrogen combustion is not an important mechanism for containment failure in Mark I and Mark II containments; however for a Mark III, hydrogen can pose a severe threat to the containment integrity during a station blackout accident. The arrangement of the drywell relative to the wetwell provides another interesting difference between the three containment designs. This difference can affect both the containment failure probability and the source term release characteristics. The ultimate failure pressure of the various containment designs can drastically affect the timing and the consequences of the accident sequences. The Mark III design typically has a much lower failure pressure than either the Mark I or the Mark II (e.g., the assessed ultimate failure pressures for Peach Bottom (Mark I), LaSalle (Mark II), and Grand Gulf (Mark III) are 147 psig, 195 psig, and 55 psig, respectively). The impact that these design differences has on early containment failure probability, effectiveness of venting, source term release characteristics, and risk can be large.

OVERVIEW OF SEISMIC DESIGN MARGIN METHODOLOGY RESEARCH

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The Seismic Design Margins Program (SDMP) was initiated by NRC in 1984 to answer a need identified by the ACRS. The need was to develop an effective and efficient way to assess how much larger than the design basis earthquake can a plant withstand. The methodology was largely developed through efforts of an expert panel consisting of PRA, system, and fragility experts (Refs. 1 and 2). The methodology has used insights from seismic PRAs, and test and earthquake experience data to focus the review effort. Initiators, functions, and systems failures which were not major contributors to the core damage frequencies in past seismic PRAs are screened out from the review. Similarly, components known to have high seismic strength are also screened out from the analysis. A review level earthquake, higher than the design basis earthquake, is established at the beginning to implement the SDMP at a particular plant. Seismic hazard curves are not used; however, plant walkdown and integrated evaluation of plant response are retained. The outcome of a margin review is not the estimation the core damage frequencies, but is a statement regarding the capacity of the plant to withstand the given review level earthquake. The results of review are expressed as "High Confidence of Low Probability of Failure" (HCLPF) capacities for components, accident sequences, and the plant. A trial application of the NRC method was successfully carried out at the Maine Yankee Atomic Power Station in 1987 (Ref. 3).

Subsequent to the NRC effort, the Electric Power Research Institute (EPRI) also undertook the development of an industry sponsored SDMP approach. While utilizing insights from PRAs and earthquake experience data, the EPRI approach uses the "success-path" approach rather than the event/fault tree approach used in the NRC method (Ref. 4). In the EPRI approach, an operational sequence is identified to bring the reactor to a safe shutdown given a seismic induced initiator (e.g. loss of offsite power or small LOCA). Components in this sequence are evaluated with the basically same guidance as in the NRC method. A successful trial application of the EPRI method was carried out at the Catawba plant (Ref. 5). Recently, through the cooperative efforts of EPRI, Georgia Power, and NRC, a trial applications of both methods (application of the NRC method was limited to the system analysis aspects) has been carried out at the Hatch plant.

In addition to these trial applications, research has also continued into examining these methods to determine what type of risk insights can be developed from the application of these methods. These methods are also considered to be viable for performing the individual plant examination in response to the NRC's severe accident policy. However, certain enhancements have been identified in the NRC staff guidance for this application.

The purpose of this paper is to give an overview of activities since 1984 as the major activities associated with the development of SDMP methods are near completion. Future activities to incorporate lessons learned from the trial applications, and the other research described above are also presented. The regulatory uses and insights obtained are highlighted.

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METHODOLOGY FOR THE SEISMIC PORTION OF THE IPE
and
COORDINATION OF ONGOING SEISMIC ISSUES

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INTRODUCTION

The Commission has recognized, based on NRC and industry experience with plant specific probabilistic risk assessments that systematic examinations are beneficial in identifying plant specific vulnerabilities to severe accidents that can be fixed with low-cost improvements. In 1988 the staff requested that each licensee conduct an individual plant examination for internally initiated events. Examination for externally initiated events would proceed on a later schedule.

The NRC established the External Events Steering Group (EESG) to make recommendations to senior management on Severe Accident Policy implementation for external events. Described below are their recommendations on the seismic portion of the individual plant examination for external events (IPEEE).

EXAMINATION SCOPE

In implementing the Severe Accident Policy, the EESG recommended that an external events initiator need not be treated in the same manner as internally initiated events. The reasons are (1) uncertainties of core damage are different (possibly significantly larger) from those of internally initiated events, (2) the inappropriateness of comparing numerical estimates of core damage frequency between external and internal events, and (3) the inherent difficulties in relating the output of the various evaluation methodologies. Therefore, the IPEEE emphasizes a qualitative rather than a qualitative understanding of core damage.

EXAMINATION METHODOLOGIES

Two methods are recommended to identify potential vulnerabilities from seismic events at nuclear power plants. The first, applicable to all plants, is a seismic probabilistic risk assessment (PRA). The second, applicable to a majority of plants, is the seismic margins methodology. In addition, for a limited number of plants where the seismic hazard is low, a reduced scope seismic margins methodology is recommended.

Seismic PRA

The PRA should be at least a level 1 plus containment performance analysis. The basic elements include (1) hazard analysis, (2) plant system and structure response analysis, (3) evaluation of component fragilities and failure modes, (4) plant system analysis, and (5) containment and containment system analysis.

Seismic Margins Methodology

The margins methodology was developed using both seismic PRA information, and test and earthquake experience data. The methodology has retained two of the most important aspects of a seismic PRA. These are (1) plant walkdown, and (2) the ability to identify potential seismic vulnerabilities through an evaluation of a plant's response to the seismic event as an integrated system.

The end product of the seismic margins evaluation is a conditional high confidence statement that the probability of failure (core damage) is low (HCLPF). Mathematically, the HCLPF can be thought of as an estimate of the 5% failure probability with 95% confidence. HCLPF capacities are typically measured in terms of peak ground acceleration and an assigned spectral shape.

Two seismic margins methodologies are current available: one developed under the sponsorship of the Nuclear Regulatory Commission (NRC), the other developed under the sponsorship of the Electric Power Research Institute (EPRI). The two methods use different system analysis philosophies. The NRC method is based on an event/fault tree approach to delineate accident sequences. The EPRI method is based on a systems "success path" approach.

Methodology Enhancements

Enhancements to both the seismic PRA or seismic margins methodologies described in References 1 through 6 are needed prior to their use in the IPEEE.

For a seismic PRA the enhancements include (1) plant walkdown procedures, (2) the inclusion of relay chatter and, if applicable, soil liquefaction effects, and (3) the reporting of results in terms of high confidence, low probability of failure (HCLPF) levels.

For the margins methodology the enhancements include (e inclusion of relay chatter, enhanced plant walkdown guidance and, if applicable, the on of liquefaction effects for the NRC method; (2) guidance on alternative success and the use of the Generic Equipment Ruggedness Spectrum (GERS) for the EPRI method; and (3) the inclusion of nonseismic failures and human actions for both methods.

COORDINATION

Guidance is provided to coordinate the IPEEE process with ongoing programs. The first level of coordination is among the major elements of the severe accident policy implementation; that is, coordination between the IPEEE and the internal events IPE. The second coordination level is among the major elements of the IPEEE; for instance, earthquakes and fires. The third level of coordination is among the ongoing seismic programs; for instance, USI A-46 (equipment qualification).

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Risk Insights from Seismic Margin Reviews

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This paper will discuss the information that has been derived from the three seismic-margin reviews conducted so far, and the information that is potentially available from using the seismic-margin method more generally.

There are two different methodologies for conducting seismic margin reviews of nuclear power plants, one developed under NRC sponsorship and one developed under sponsorship of the Electric Power Research Institute. Both methodologies will be covered in this paper. The paper will begin with a summary of the steps necessary to complete a margin review, and will then outline the key technical difficulties that need to be addressed. After this introduction, the paper will cover the safety and operational insights derived from the three seismic-margin reviews already completed: the NRC-sponsored review at Maine Yankee; the EPRI-sponsored review at Catawba; and the joint EPRI/NRC/utility effort at Hatch.

The emphasis will be on engineering insights, with attention to the aspects of the reviews that are easiest to perform and that provide the most readily available insights.

Examples of insights to be covered are seismic-margin findings with respect to large structures, large items of equipment, electrical and fluid distribution equipment, control equipment, and so on. Also, insights about the role of operator errors, non-seismic failures, and the possibility of large releases from earthquake-initiated accidents will be discussed.

A method for examining relay chatter effects within a seismic margin review has been developed by EPRI and a trial application has been completed as part of the Hatch study. The insights from that effort will be discussed, and more general lessons will be presented about how to examine potential relay chatter issues.

Finally, the overall level-of-effort required for an effective seismic margin review will be discussed and commented on.

SEISMIC TESTING AND EVALUATION OF RELAYS
PAST AND FUTURE

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SUMMARY

INTRODUCTION

Since safe operation of relays in a seismic environment is critical for many electrical equipment classes, a special study on relays is being conducted at Brookhaven National Laboratory as part of the Component Fragility Program. The study consists of evaluating existing test data and performing additional tests. The first series of relay testing was conducted in 1988-89 at Wyle Laboratories and the partial results were presented at the 1989 Water Reactor Safety Information Meeting[1]. The evaluation of the test data has now been completed and the observations are discussed in this paper. In addition, the planning for a second test series is continuing at the present time. This paper will also present an outline of the background, objectives and the methods being adopted for the second test series which is expected to be completed by the end of 1990.

FIRST TEST SERIES

The primary objective of the first series of relay tests was a systematic study of the influence of parameters related to the relay design and the input motion on its seismic capability. To this end, single frequency sine dwell excitation was applied for better characterization of the specimen at a specific frequency. A total of thirty eight (38) relays were tested with single frequency inputs. All the operating, non-operating and transition electrical modes were simulated during shaking as recommended by the ANSI/IEEE std.[2]. Both normally closed and normally open contacts were electrically monitored for detecting a temporary or a sustained change of state. A duration of 2 milliseconds or greater was considered a failure. The single frequency data revealed that depending on the electrical condition, contact state and the direction of the input motion, a relay can be considered rugged or fragile at a particular frequency. Although the overall contact driving mechanism of a relay design exhibits one range of frequency sensitivity, there are other elements in the relay that also influence the relay capacities at various other frequencies. Most relays are relatively sensitive to excitation in the vertical direction, especially at high frequencies (e.g. greater than 30Hz). A substantial variation of the capacity levels was observed for multiple specimens of the same relay model. A shift in the sensitive frequency was also observed among these specimens.

SECOND TEST SERIES

The first test series raised some fundamental questions regarding the relay chatter acceptance criteria. In addition, there is a need to study the chatter tolerance behavior of a breaker circuit. Due to inductance of the circuit breaker trip coil, its energization for a short period (e.g. 2ms) may not allow the current to increase to a value where subsequent opening of the protective relay contact cannot interrupt the current flow; thus the circuit breaker will not trip. The second test series will address these issues by correlating the pulse time characteristics of breakers and relays.

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SEISMIC COMPONENT FRAGILITY DATA BASE FOR IPEEE

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SUMMARY

INTRODUCTION

As part of the Individual Plant Examination for External Events (IPEEE) Program most nuclear plants will perform either a seismic probabilistic risk assessment (PRA) or a seismic margin study. Both the PRA and the margin study will require a reliable data base of seismic fragility of various equipment classes especially those that are important for safety of the plant and, at the same time, have low seismic capacities. Based on a prioritization study conducted by Lawrence Livermore National Laboratory[1], Brookhaven National Laboratory has selected a group of equipment and generically evaluated the seismic fragility of each equipment class by use of existing test data. The progress in this effort and partial results were presented at earlier Water Reactor Safety Information Meetings[2,3]. This paper briefly discusses the evaluation methodology and provides fragility results on additional equipment pieces. It also includes a summary of earlier results for completeness of the data base. The following is a complete list of equipment: Motor Control Center, Switchgear, Panelboard, Switchboard, Power Supply, NSSS I&C Panels, Transmitters, Indicators, Switches, Transformers, BOP I&C Panels, Miscellaneous Instruments, Batteries, Battery Chargers, Inverters, Motors, Electrical Penetration Assemblies.

EVALUATION METHODOLOGY

Existing test data from various sources and for various models of a particular equipment class have been compiled and evaluated for each failure mode. The fragility has been defined as the threshold of occurrence of a failure and the corresponding vibration level has been measured in terms of the test response spectrum (TRS). The zero period acceleration (ZPA) and an average of the spectral accelerations (ASA) of the TRS in the frequency range of 4-16Hz have been used to represent each TRS. The ZPA's and ASA's for all models of the same equipment class have been statistically analyzed for determination of the median and variances due to uncertainties and randomness of the data. Finally, a high confidence value (HCLPF 95%-5%) has been computed for ready use in margin studies. If the available data points were considered inadequate for mathematical computation, the statistical parameters were obtained by use of judgment on the test results.

RESULTS

For each equipment, the median fragility data, uncertainty coefficients and the HCLPF value are presented in this paper. The results are obtained for each failure mode. For electrical equipment, electrical malfunction occurs at a vibration level lower than that required for an overall structural failure.

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SUMMARY OF RESULTS OBTAINED FROM THE TESTING OF REINFORCED CONCRETE SHEAR WALLS

by

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This paper summarizes the experimental results that have been obtained from the Nuclear Regulatory Commission's Seismic Category 1 Structures program over the past eight years. As part of the program, we have been investigating the seismic response of noncontainment Category 1 structures (diesel generator buildings, auxiliary buildings, etc.) subjected to beyond-design-basis seismic events. Results obtained at various stages of the program have been reported at previous Water Reactor Safety Information Meetings.

The low-aspect-ratio (height/length), reinforced concrete shear wall is the primary lateral-load-carrying structural element in noncontainment Category 1 structures. Numerous static, monotonic, and cyclic tests have been performed on these types of structural elements and reported in the open literature. However, these tests have primarily been concerned with the walls' ultimate strength rather than with their dynamic properties, such as stiffness and damping. This program has extended the knowledge of low-aspect-ratio shear walls' dynamic properties by performing simulated seismic shake-table tests on scale-model Category 1 structures and scale-model shear wall elements. Scale models were employed because the enormous size of a prototype structure and the need to test into the nonlinear response region precluded the testing of an actual building.

First, the paper briefly reviews the testing performed between FY 1982 and FY 1986. During this phase of the program, tests were performed almost exclusively on microconcrete scale-model structures. Included in this review will be a summary of the types of structural geometries that were investigated and the results obtained concerning stiffness, damping and similitude.

Results from tests that have been conducted on 12 shear wall elements between FY 1987 and FY 1990 are the focus of the paper. These structural elements had an "I" cross-section geometry with the web representing the shear wall. They were made out of microconcrete and conventional concrete, were reinforced with wire mesh and conventional deformed rebar, and were tested statically and dynamically. These tests helped us examine the stiffness and damping of these structural elements, verify similitude relationships for these structural elements' dynamic properties, and investigate the effects of subjecting these structural elements to seismic pulses of increasing amplitude.

Results from this last series of tests were significantly different from those obtained during FY 1982 through FY 1986. These differences are attributed to improved data reduction and test methods as well as to the use of improved instrumentation, some of which was not commercially available at the start of this program. At this point in the program the experimental results support the following conclusions:

1. Strength-of-materials analysis that accounts for shear deformation can accurately predict the static and dynamic stiffness of the ideal, monolithic shear walls up to a nominal-base shear stress (NBSS) in the range of 150 psi. The actual value of NBSS for which this statement holds true is primarily a function of the individual test specimen's concrete tensile strength. These results do not account for sources of stiffness reduction that could be found in prototype structures such as cracking resulting from differential settlement.

2. Before cracking is induced by the applied excitation, damping of the shear wall elements is in the range of 1% of critical. This value can be verified by several methods. The Reg. Guide 1.61 value of 4% at the OBE level should not be directly compared with this value because Reg. Guide 1.61 specifies system damping values (that includes the effects of construction joints, etc.) and the 1% value was obtained from a monolithic structural element.

3. Similitude in stiffness can be demonstrated between 1/3-scale models made with 3/8-in. aggregate and microconcrete structures of the same size. Results from both types of models can be scaled to the conventional concrete shear wall element (the prototype for the 1/3-scale models) that was also tested in this phase of the program.

THE EFFECTS OF REDUCED STRUCTURAL STIFFNESS ON PLANT RISK AND MARGIN

by

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SUMMARY

Since 1983, the U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research has been sponsoring the Seismic Category I Structures Program at Los Alamos National Laboratories. This program is investigating the static and dynamic response of Category I reinforced concrete shear wall structures (exclusive of containments) subjected to seismic loads beyond their design basis. From these tests, it was clear that there was a significant difference between the measured natural frequencies and stiffnesses, and those computed analytically. From early low level tests, measured stiffnesses were found to be as low as 25% of the computed values. Later tests, felt to be more reliable, found measured stiffnesses in the range of about 75% to 100% of the computed stiffness. However, in all the tests, when the structures were subjected to increasing shear stress levels, either static or dynamic loads simulating earthquakes, measured stiffnesses and frequencies further decreased.

The observed differences between calculated and measured stiffnesses and frequencies represents a potentially important issue in the design and safety of nuclear power plants. This is due to two major factors. First, most safety related equipment are located in Seismic Category I structures which typically have predicted fixed base natural frequencies in the 5 Hz to 20 Hz range. If the frequency reduction was as large as 50%, then these structures could have actual frequencies in the 2 Hz to 10 Hz range. Most broad band strong motion earthquake time histories have the majority of their energy in the 2 Hz to 8 Hz range. Therefore, the excitation of the structure could be much greater than was considered in the original design. Both the loads experienced by the structural members and in-structure floor acceleration spectra would be increased. The second important factor to consider is that safety related equipment could experience greater seismic loads due to a shift in the floor spectra near the equipment natural frequency.

In 1988, the U.S. Nuclear Regulatory Commission funded Sandia National Laboratories to assess the importance of this "frequency difference" issue. The program is assessing the impact of decreased natural frequencies on both the calculated seismic risk and the deterministic design calculations for several prototypical power plants. This paper describes the scope of the program and its current status.

In order to assess the importance of this "frequency difference" issue on power plant risk, three existing seismic probabilistic risk assessments

(PRA's) are being re-evaluated. The first PRA chosen for the initial re-evaluation using a preliminary model for stiffness reduction was Peach Bottom, for which the original seismic PRA was computed at Sandia as part of the NRC-sponsored NUREG 1150 program. In addition to this rock site BWR, a rock site PWR and a soil site plant will also be re-evaluated. The current choice for a rock site PWR is ANO-1. This plant was part of the TAP A-45 program and has current structural models and fragilities available. The soil site chosen for this program is Zion. This plant was chosen because the soil properties and depth are likely to enhance the effects of reducing the fixed-base natural frequencies of important structures. In addition, a full piping analysis of the NSSS at one plant will be performed, and Zion already has all piping models available as part of the SSMRP.

In addition to re-evaluating the three existing seismic PRAs above, the effect of reduction of structural frequencies on deterministic seismic design margin will also be examined for the three sites. To do this, a design-type calculation of structural loads and in-structure floor spectra is being performed for each site. Building models of the structures are being obtained from the original architect/engineering firm or developed from plant drawings where necessary. The original site design spectra, load combination rules, methods of inclusion of soil-structure interaction, design materials/soils properties, and structural damping values will all be re-used in the deterministic re-evaluation. Both "as designed" and "reduced" frequency cases will be considered in the deterministic "design type" calculations. This is necessary to make an evaluation of the decrease in the design margin of safety from reducing the natural frequency of the safety related structures at existing commercial nuclear power plants.

To date, the Peach Bottom Atomic Power Station has been re-evaluated. Re-evaluation of the original PRA was performed with a preliminary model incorporating both static and dynamic reduction in stiffness for the critical concrete structures at Peach Bottom. A suite of 10 recorded earthquake time histories (chosen to match the site conditions) were used as input for the analysis. The mean core damage frequency was found to increase from the original NUREG 1150 value of 7.66E-05 to 1.24E-04 per year, a 60% increase when using the LLNL developed hazard curves. When the EPRI hazard curves are used, the frequency increases from 3.09E-06 to 5.21E-06, a 70% increase. This increase was primarily due to two structures, the crib house and the cooling towers. These structures originally had natural frequencies in the 13 to 20 Hz range, which were reduced into the amplified acceleration region of the ground motion input. This increased acceleration resulted in higher net loads on safety related equipment in these two structures and on the structures themselves, causing the probability of failure to increase, resulting in a higher overall core damage frequency.

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 down significantly. In addition, calculated net wall loads and moments were increased by up to 30%. These results have significant implications for the design of nuclear power plants.

Completion of Thermal-Hydraulic Code Development: TRAC-PF1/MOD2, RELAP5/MOD3, and TRAC-BF1/MOD1

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During the last 15 years, the NRC Office of Nuclear Regulatory Research (RES) engaged in research to develop computer codes that would calculate realistically the response of light water reactors to transients and loss-of-coolant accidents (LOCA). The purpose of this paper is to apprise the technical community that the development program was completed with the release of the final versions of these codes (TRAC-PF1/MOD2, RELAP5/MOD3, and TRAC-BF1/MOD1) in 1990.

Thermal-hydraulic systems codes were developed to provide the NRC staff with an independent capability to analyze plant transients. Two versions of the Transient Reactor Analysis Code (TRAC) were developed, for PWRs (TRAC-PWR) and for BWRs (TRAC-BWR). The TRAC code treats the vessel in three dimensions and other fluid systems in one dimension. Its development dates from 1975. A third code, RELAP, treats the entire system in one dimension. Its development dates from 1966. The codes have now reached a state of sufficient maturity such that further work would not be expected to yield major gains in accuracy. Thus, no new versions are planned.

The specific functions for which the staff has utilized, and intends to utilize, the codes are to: confirm licensee analyses by performing audit calculations; understand operating reactor transient events and their broad implications (including "what if" studies); evaluate design and operational-related issues, including changes to technical specifications, operator guidelines and accident management strategies; investigate and resolve issues such as feed-and-bleed cooling, pressurized thermal shock, anticipated transient without scram, and BWR stability; provide information on risk-dominant accident sequences and the early phases of postulated severe accident scenarios; and evaluate advanced LWR designs.

The basic structure of each code consists of conservation equations for mass, energy, and momentum. The current codes are known as two-fluid codes, since they have six such equations, enabling the steam and liquid phases to be modeled separately. The advantage of the two-fluid code is that it allows a more accurate representation of off-normal conditions in fluid systems. The difficulty which must be overcome, however, is to obtain an accurate description of interphase relations, that is, the rates of transfer of mass, energy and momentum between steam and liquid. The models and correlations which describe these processes are termed closure relations, of which each code contains a large number. The task of demonstrating that a code containing such closure relations is applicable to the full-scale plant is considerable. Therefore, RES developed a Code Scalability, Applicability and Uncertainty (CSAU) Methodology that systematically addresses these issues.

Code development proceeded in three stages: (1) developing models within an appropriate numerical structure; (2) testing alternative models in the code and determining that models were implemented correctly (developmental assessment); and (3) evaluating code performance (independent assessment or validation).

It was recognized that improved practices were required with respect to documentation and software quality assurance. Thus, RES established requirements for documentation to cover the following topics: code structure, system models, and solution methods; user guidelines and code input requirements; detailed description of models and correlations; and documentation of developmental assessment. Moreover, formalized software quality assurance procedures were introduced, and an external audit of the procedures was performed.

While the codes were reaching completion, RES organized an international program to provide the codes to safety authorities of foreign countries in exchange for independent code assessment. The independence of the assessment process is important to the assurance of objectivity and to the establishment of credibility. This was effected during 1984-85, and the International Code Assessment Program (ICAP) was begun as a six-year effort to be carried out from 1986-91. The ICAP program provided a formal code users group structure that provided for software configuration control, software documentation, and documentation of independent assessment results. Such configuration control was vital in order to avoid proliferation of special versions, a potential source of great confusion when the codes are applied to plant safety analyses.

When TRAC-PF1/MOD1, RELAP5/MOD2, and TRAC-BD1/MOD1 were released to the international community, it was envisioned that usage of these codes would reveal modeling deficiencies that would require development of new versions. This indeed proved to be the case. A consensus was reached among the ICAP code users on the nature and cause of the deficiencies. In December 1987, agreement was obtained on a plan for development of the final versions of the codes, namely TRAC-PF1/MOD2 and RELAP5/MOD3. RES also determined that addition of fast numerics to the vessel component of TRAC-BF1 would be desirable, hence TRAC-BF1/MOD1 was initiated. The code development was carried out during 1988-90, with several ICAP participants making direct contributions. In its letter of June 15, 1989, the ACRS reviewed the closure plan for thermal-hydraulic code development and endorsed the termination of further major development.

Following the release of the codes in 1990, the ICAP participants are carrying out independent code assessment. By the end of 1991, the mission of ICAP will be completed, and a successor program will take effect to maintain a code-users group focused on: (1) code applications for plant safety analysis; and (2) code maintenance. A multilateral agreement is in preparation for a five-year program which will call for shared funding of code maintenance.

NUCLEAR PLANT ANALYZER DESKTOP WORKSTATION[1]

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SUMMARY

In 1983 the U. S. Nuclear Regulatory Commission (USNRC) commissioned the Idaho National Engineering Laboratory (INEL) to develop a Nuclear Plant Analyzer (NPA). The NPA was envisioned as a graphical aid to assist reactor safety analysts in comprehending the results of thermal-hydraulic code calculations. The development was to proceed in three distinct phases culminating in a desktop reactor safety workstation. The desktop NPA is now complete.

The desktop NPA is a microcomputer based reactor transient simulation, visualization and analysis tool developed at INEL to assist an analyst in evaluating the transient behavior of nuclear power plants by means of graphic displays. The NPA desktop workstation integrates advanced reactor simulation codes with online computer graphics allowing reactor plant transient simulation and graphical presentation of results. The graphics software, written exclusively in ANSI standard "C" and FORTRAN 77 and implemented over the UNIX/X-windows operating environment, is modular and is designed to interface to the NRC's suite of advanced thermal-hydraulic codes to the extent allowed by that code. Currently, full, interactive, desktop NPA capabilities are realized only with RELAP5.

The RELAP5 version of the desktop analyzer or RELAP5 Desktop Analyzer (RDA), is a complete, stand-alone, reactor safety analysis tool which allows interactive graphic display development, automated color-graphic display driver production, interactive RELAP5 execution, and simultaneous color-graphic depiction and online plotting of results via an intuitive graphical user interface. The RDA superimposes the results of a RELAP5 calculation on a graphical plant display schematic derived from the corresponding RELAP5 input model nodalization scheme. A display schematic is easily developed using the integrated graphics display generation module. The user simply chooses from available geometric shapes or freehand sketches by selecting available contours to assemble enclosed component representations on the color monitor screen. He then selects from a menu of pre-defined color graphic display techniques with which to drive the derived component representations color-graphically and is queried to identify the corresponding simulation code data channel names, conversion factors, thresholds and setpoints appropriate to that technique. The production of the color-graphic display driver is thereby automated transparently to the user. Pre-programmed color-graphic display techniques have already been developed to illustrate process parameter readouts, safety parameter display systems, annunciator alarm functions, void profile and mass distribution, fluid thermodynamic state (i.e., subcooled, saturated, superheated), core status and fuel rod stored energy, collapsed and two-phase mixture levels, plant component status (i.e.,

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

on-off, open-closed), and hydrodynamic flow regimes. During display development the user can also assign interactive control capabilities to corresponding RELAP5 input model interactive component representations by associating control functions with appropriate RELAP5 interactive input variable specification command strings. Upon display completion the interactive interface to RELAP5 is completely automated and the display itself becomes the graphical user interface. By using the mouse to select a graphical display component during an interactive simulation or playback, the user is able to ascertain the simulated component's status, the driver parameters and their current values, and available interactive component manipulation and control options by means of pop-up menus. These mouse-sensitive menus allow online editing and modification of display features, designation and automated online plotting of any RELAP5 data channel, and implementation of interactive component control commands to affect the course of the simulation online as it unfolds. The user is thus able to introduce malfunctions online, implement appropriate operator actions interactively, graphically interpret simulated plant response, and simultaneously accomplish the required analysis via online plotting of parameters as the calculation advances.

The NPA's ability to execute RELAP5 interactively and simultaneously depict the results graphically on a desktop workstation enables a reactor safety analyst to concurrently simulate, visually interpret, assimilate and analyze complex reactor transients on his desktop, at his own discretion and at no mainframe computational expense. No longer is analyst productivity hampered by: poor mainframe computer turnaround for problem setup, checkout and initialization; limited mainframe CPU allocation, accessibility and availability for transient advancement; or difficulty assimilating and interpreting printed numerical results. Dependence on mainframe computers has been eliminated for all but the most demanding realtime applications. However, what a computer can produce in seconds or minutes requires hours or days for an analyst to validate and comprehend. Not even the most highly skilled analyst is capable of credibly assimilating and analyzing the results of a reactor transient simulation in realtime. Nevertheless, RELAP5 is already capable of 50% realtime performance on a DECstation 5000 for many problems of practical interest, and faster than realtime desktop performance is assured by recent product announcements. Notwithstanding realtime performance considerations, the efficiencies inherent in being able to assimilate calculational results as they are produced not only reduces unproductive analyst time but also reduces analysis costs and enhances accuracy. The completion of the NPA desktop workstation heralds a new era of desktop reactor safety analysis and promises to revolutionize current methodology.

REACTOR SYSTEM ANALYSES OF ADVANCED, PASSIVE LWR DESIGNS[a]

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The Department of Energy (DOE) is supporting the development of two passive light water reactors (LWRs): the Westinghouse Advanced Passive 600 MW(e) (AP600) Pressurized Water Reactor (PWR) and the General Electric Simplified Boiling Water Reactor (SBWR). The final versions of these designs are expected to be submitted to the Nuclear Regulatory Commission (NRC) for design certification. In the certification process, NRC may have to use best estimate thermal-hydraulic codes to audit vendor analyses of the reactor and plant system responses to a range of postulated operational and accident conditions. This paper summarizes current work to support NRC's evaluation of the codes available for analysis of these conditions.

Both reactors were designed to meet goals set by DOE and the Electric Power Research Institute in response to lessons learned from past experience with LWRs. These goals are:

1. Assured safety, with features that minimize negative consequences of human error.
2. Significantly simpler designs with increased safety and performance margins in key operational parameters.
3. High reliability, lifetimes on the order of 60 years, and plant availability increased to over 85%.
4. Reduction in capital, operating, maintenance, and fuel costs to meet the economic competition of other energy production methods.
5. Designs that are standardized at a high quality level and predictably licensable.

The design of these reactors relies on experience for the basic reactor components and the use of new passive systems for the safety functions. The passive safety systems and other design changes lead to simpler support systems, further reducing the amount and complexity of equipment (components, piping, instrumentation and cabling). This finally results in expected lower cost, more reliable operation, and increased safety.

The current program at The Idaho National Engineering Laboratory consists of the following basic elements:

1. Design reviews to identify new or unique components and system behavior under postulated design basis accident conditions.
2. Code assessment to review the applicability of current models to the new components/phenomena and the experimental bases for those models.
3. Scoping calculations to explore the operating ranges of the new passive safety systems and to investigate code capabilities to model those systems.

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4. Development of an experimental program to complement existing experimental work as needed to support NRC's code assessment and demonstration/verification needs.

The AP600 is a two loop reactor system which uses proven PWR technology, but is significantly simplified compared to present generation PWRs. The safety systems for safety injection, reactor shutdown, residual heat removal, containment spray, and emergency ventilation are passive, relying on natural circulation, gravity, and gas pressurization as driving forces. These systems are supported only by station batteries, which are sized for three day duration, rather than by safety grade/redundant active systems such as diesel-generators, cooling water systems and HVAC systems. Systems supporting normal operation are non-safety grade. These include residual heat removal, chemical and volume control, component cooling water, spent fuel cooling, service water, and diesel-generators. As a result, the number of safety-grade and non-safety grade components is substantially reduced.

Review of postulated accidents and RELAP5 code models indicated that, despite the new passive safety systems, the AP600 can generally be modeled by the existing RELAP5 code. The most important difference in design between the AP600 and current PWRs, as well as in transient behavior and code modeling, is the coupling between the reactor system and the containment during accident response. This coupling will require extension of present models to provide best estimate analyses of the complex heat transfer across the containment shell, including the external convection and evaporative cooling.

The SBWR design is also based on current LWR technology. It is, in common with the AP600, designed for 600 MWe using a low power density core. It is, however, a natural circulation boiling water reactor. The natural circulation design eliminates many forced circulation components and also reduces the number of reactor vessel penetrations. Emergency core cooling is provided by the Gravity Driven Cooling System. Decay heat removal is accomplished by the Isolation Condenser which transfers the energy from the primary system or the drywell into a large pool and from there to the environment by evaporation.

Review of the SBWR design indicates that the present analytical capabilities (e.g. TRAC-B) are generally capable of simulating transient phenomena of interest. The Isolation Condenser is the component requiring most attention from the evaluation and modeling point of view. Its performance can be significantly affected by the presence of noncondensable gases drawn into the system from the containment.

In both designs, the performance of the passive safety systems is of concern because of the small driving forces and the possibly adverse effects of system and phenomena interactions. These interactions will be evaluated in a series of scoping and sensitivity calculations. The codes to be used for analysis of both designs are being evaluated for suitability for all key and controlling phenomena. This work will be supported next year with extensive code assessment using experimental data to quantify applicability ranges and simulation accuracy. New experimental programs will be recommended for areas not covered by the existing experimental data base or current vendor programs.

UPTF EXPERIMENT

FULL-SCALE TESTS ON COUNTERCURRENT FLOW BEHAVIOUR IN PWR DOWNCOMER

EFFECT OF INJECTION MODE AND NONCONDENSIBLES

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ABSTRACT

The Upper Plenum Test Facility (UPTF) Experimental Program, funded by the Ministry of Research and Technology (BMFT) is the German contribution to the trilateral 2D/3D project, and is performed within international cooperation among Japan (JAERI), USA (US NRC) and the Federal Repubiic of Germany (BMFT).

The UPTF constructed and operated by SIEMENS AG, UB KWU was designed to investigate the three-dimensional thermal hydraulic behaviour of fluid in a full-scale reactor pressure vessel and primary system during the end-of-blowdown, the refill and the reflood phases of a postulated loss-of-coolant accident (LOCA) of a pressurized water reactor (PWR).

The UPTF [1] simulates the primary cooling system of a 1300 MW PWR. A three-dimensional view of the primary system is shown in Fig. 1. The upper plenum including internals, the downcomer and the four connected loops are represented in 1:1 scale. The core is simulated with controlled injection of steam and water supplied from external sources.

The three intact loops are equipped with valves to simulate the reactor coolant pumps, and with steam/ water separators representing the steam generators. The hot and cold leg of the broken loop lead through steam/water separators and break valves to the containment simulator. Breaks of variable size can be simulated in the hot and in the cold legs respectively.

Countercurrent Flow Studies

Countercurrent flow of steam and water may occur in the downcomer of a pressurized water reactor during a postulated loss-of-coolant accident (LOCA) when emergency core cooling water (ECC-water) is being injected via the cold legs into the reactor vessel. For a cold leg break, depressurization takes place during the blowdown phase of a LOCA leading to a reverse core steam flow up the downcomer, with the potential to retard or even to prevent liquid penetration into the lower plenum. Flooding occurs as a result of instability at the interface of the two fluids flowing countercurrently.

The current understanding of downcomer countercurrent flow limitation and ECC bypass has been developed through numerous tests in scaled facilities as part of the USNRC ECC Bypass Program, which was concluded in 1981. In this program, steam/water tests were performed at 1/30, 1/15, 2/15 and 1/5 scale at Battelle Columbus Laboratories and at Creare[2],[3].

To investigate the thermal hydraulic phenomena in a full-scale downcomer of a PWR during end-of-blowdown and refill phases six separate effects tests (23 test runs) have been performed at UPTF (see Fig. 2). Special attention has been paid to the effects of the geometry and the injection mode on downcomer countercurrent flow and ECC bypass behaviour. A synopsis of the most significant events and a comparison of countercurrent flow limitation (CCFL) data from UPTF and 1/5 scale test facility of Creare are given. The CCFL results of UPTF are compared to data predicted by an empirical correlation developed at Creare, based on the modified dimensionless Wallis parameters J^*. Furthermore, based on UPTF test results, the following effects on the ECC bypass behaviour are quantified using the original Wallis-correlation:

- the ECC injection mode - cold leg or downcomer injection and
- the effects of noncondensibles.

It had been found, that

- the flooding correlation recommended by CREARE predicts the CCFL behaviour of the full-scale downcomer insufficiently.
- the presence of noncondensibles diminishes the condensation effects in the downcomer moving CCFL-conditions towards lower core mass flow rates.
- the injection mode - cold leg or downcomer injection - strongly effects the ECC delivery into the lower plenum; for direct downcomer injection an increased ECC bypass flow was measured.

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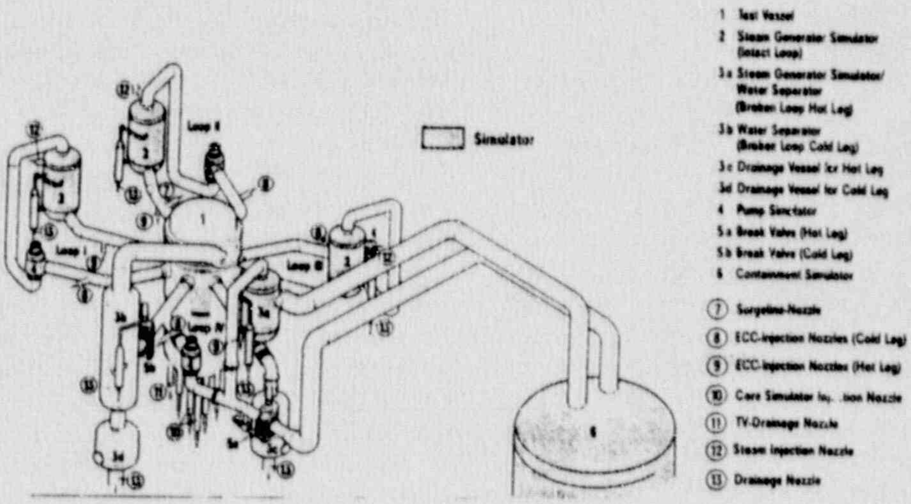

Fig. 1: Upper Plenum Test Facility - Primary System

Scaling - and Parameter Studies

Reactor Behaviour

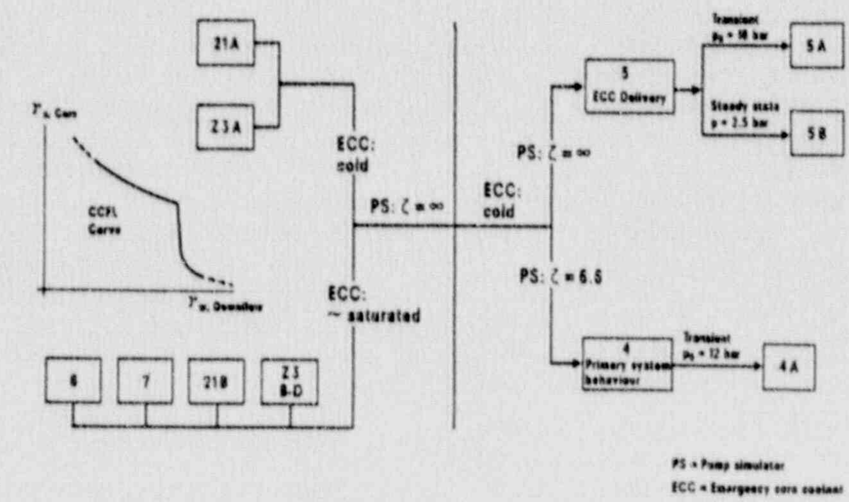

Fig. 2: Flow Phenomena in Downcomer: UPTF Separate Effects Tests (23 Runs)

UPTF RESULTS WITH RESPECT TO SELECTED REACTOR SAFETY ISSUES

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The Upper Plenum Test Facility (UPTF) located in Mannheim, Germany, was designed to study the end of blowdown, refill, and reflood phases of a large-break loss-of-coolant accident (LBLOCA). It is a full-scale simulation of a 1300 MWe German PWR with some modifications to include U.S./Japanese PWR design features. The upper plenum, downcomer, and the piping for four loops are realistically simulated while the core, steam generators, and primary coolant pumps are simulated only in terms of their functions during a LOCA.

A series of cold leg injection experiments in the UPTF showed that the emergency core coolant (ECC) started penetrating into the lower plenum well before the blowdown ended. Under simulated PWR conditions, the ECC started penetrating rapidly at a rate equivalent to about 90% of the injection rate starting at 8 to 11 bar, while the blowdown was still taking place, and refilled the lower plenum in about 16 seconds, or about 5 seconds after the blowdown ended. This refilling rate included liquid accumulation due to condensation of steam which was estimated to be about 80% efficient. The favorable delivery rate of ECC is achieved by the two-dimensional behavior of the downcomer; most of the ECC injected into the two cold-legs farthest away from the broken loop entered the lower plenum, while most of the ECC injected near the broken loop bypassed the core. TRAC-PF1/MOD2 calculations simulated these phenomena. The ECC penetration data from the UPTF showed much higher delivery rates of ECC than the rate predicted by the correlation based on previous small scale test results. The UPTF data also show that the liquid level in the downcomer is somewhat lower than the bottom of the cold leg during most of the reflood period because of entrainment of liquid by steam entering the downcomer from the intact loops, then going out the break. For a typical PWR condition, the liquid level in the downcomer during most of the reflood period is analytically estimated to be about 0.7m below the bottom of the cold legs. This reduction in liquid level corresponds to 11% of downcomer height measured from the bottom of the core to the bottom of the cold legs. This reduction is estimated to cause a peak clad temperature increase of about 13^{o} C.

For high ECC mass flow rates, water plugs formed in the cold legs and oscillated around the injection port. Water delivery to the downcomer occurred intermittently. During combined ECC injection, loop sealing occurred when loop steam flow was decreased because of condensation in the hot legs. The loop was cleared again as soon as steam was injected into the steam generator to simulate generation of steam. During low pressure injection of ECC into the cold legs, the predominant flow pattern was a stratified flow. When higher ECC flows were injected into the cold leg, flow instabilities occurred as a result of condensation. The mechanical loads measured were not in a range which could threaten the plant integrity. In fluid/fluid mixing tests in UPTF it was demonstrated that the cold ECC is significantly heated up before entering the downcomer.

Entrainment/deentrainment of the liquid in the core, hot legs, and the steam generators was also investigated. It was found that after the upper plenum, hot legs, and the steam generator inlet plenum had been "saturated," all of the entrained water reached the heat transfer area of the steam generator, causing steam binding.

For very high steam flows in the full scale hot leg of the UPTF, countercurrent flow flooding occurred. However, under the reflux condensation condition expected to occur during a small break LOCA, no flooding occurred; a stable countercurrent flow was maintained. ECC flows injected into the hot legs as part of the combined injection were also delivered to the upper plenum intermittently because of the intermittent plug formation in the hot legs near the injection locations.

USEFULNESS OF A REDUCED PRESSURE, REDUCED HEIGHT SCALED TEST FACILITY

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This paper summarizes the scaling concepts and the major findings resulting from six years of research on Small Break LOCA at the UMCP facility. A review of the most significant contributions to the scaling of SB-LOCA transients is presented and the goals of a scaled simulation are discussed. This discussion places the UMCP contributions in the framework of the research subsequent to the TMI accident. Bounding phenomena with respect to reactor safety (core uncovery and energy transport capability) are defined for the SB-LOCA transient and their scaling implications are outlined.

The theoretical and experimental portions of the paper are proceeded by the definition of operating states and boundary conditions for an integral system. The relation between local flow conditions and system operating conditions is explored and the concept of operating modes is introduced. The distinction between inherent system response and boundary conditions is clearly drawn. This distinction will have a fundamental bearing on the scaling of experimental test results. The important concept of open and closed system introduces the scaling rationale for reduced pressure and will be referred to in the analysis and experimental sections.

The analytical portion addresses the following three issues:

1. geometry scaling

2. inventory scaling

3. the pressure effect

The section on geometry scaling reviews the concepts that led to the design of the UMCP facility. The idea of reduced height and its implications on single and two-phase natural circulation are summarized. The major UMCP contribution is the concept of inventory scaling: the substitution of the temporal scale with the system liquid inventory scale is the principal technique which allows the translation of reduced

pressure, reduced height information into full pressure full height. Two aspects of this important issue are discussed; namely:

1. the inventory distribution and its impact on the transitions between operation modes

2. the inventory scaling for an idealized integral system

The influence of pressure on scaling procedures is analyzed by comparing two idealized facilities operating in two different pressure realms. The use of the term pressure effect as opposed to pressure scaling is a clear indication of the fact that pressure per-se is not a scaling parameter as it will be demonstrated by the experimental results. Therefore, it does not require to be scaled in order to relate information from high to reduced pressure.

The experimental portion is headed by the scaling methodology which outlines the criteria used to correlate data from MIST and UMCP facilities. A phenomenological description of the integral system inherent response to an SB-LOCA is also provided. Early experiments on the initial rapid depressurization transient which correspond to the initial drain of pressure are reported. This early work defined the concepts subsequently used in the interpolation of the UMCP test data. The first confirmation of the synthesis capability of inventory scaling is outlined in the findings related to the transition inventory.

Complete counterpart tests between MIST and UMCP are presented to demonstrate the scalability of SB-LOCA transients. A simple inventory depletion transient is very close to the system inherent response and is an excellent test case to prove that the two facilities operate along similar trajectories. By trajectories we intend sequences of events that are occurring in a chronological scale of global system liquid inventories. A second counterpart test which encompasses both inventory and energy depletion is presented to outline the generality of the scaling methodology and to define the limits inherent to the inventory scaling.

Concluding remarks summarizing the findings and outlining the future research needs are enclosed.

BWR Stability Research Program

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The NRC Office of Nuclear Regulatory Research (RES) began a program of research on BWR stability in 1988 based on a user-need letter from the Office of Nuclear Reactor Regulation (NRR). The request stemmed from the March 1988 power oscillation event at the LaSalle-2 reactor. NRR requested assistance in the form of RAMONA-3B and BNL Engineering Plant Analyzer (EPA) calculations. Two questions were posed: (1) What is the potential extent of fuel damage resulting from asymmetric regional neutron flux (power) oscillations if they are not detected and suppressed, and (2) what are the potential implications of instability with respect to ATWS events (where the oscillations might complicate the recovery).

A research program was developed to address the questions and to provide an independent review and audit capability for analysis of industry results. Four computer codes, each with unique capabilities, are being used, namely, LAPUR, EPA, RAMONA-3B, and TRAC-BF1. The research is coordinated amongst BNL, INEL, and ORNL. The results of the program will be used by NRR to support: 1) review of BWR Owners Group solutions for prevention and/or mitigation of power oscillations and 2) review of emergency procedure guidelines for ATWS.

Calculations of the LaSalle event performed by BNL with the EPA have produced oscillations without the need for artificially induced destabilizing model changes. These power oscillations are caused by a thermal-hydraulic instability in the core referred to as density-wave oscillations. The LaSalle oscillations resulted from a combination of 1) axial and radial power peaking, 2) flow reduction from trip of both recirculation pumps, and 3) feedwater temperature reduction from reduced feedwater heating.

Calculations performed by BNL with the RAMONA-3B code (using a BWR/4 model) produced both core-wide and asymmetric power oscillations. Density-wave oscillations that begin in a few high-powered fuel bundles can cause asymmetric power oscillations due to out-of-phase parallel-channel flow oscillations. Two types of asymmetric oscillations were calculated, azimuthal (side to side) and radial (center to periphery). The excitation threshold of each type (mode) of oscillation is a complex function of the core inlet subcooling, the local flow and power, and the subcriticality of the mode being excited. No cladding temperature escalations were calculated, hence no fuel damage, for asymmetric oscillations with amplitudes up to 300%. Oscillations of this magnitude would produce easily detectable average power range monitor readings.

It has been determined that the fully dynamic simulation of closed-loop feedback effects from the balance of plant is necessary to predict the nonlinear limit-cycle oscillations in the reactor core. The effect of the oscillations would be to increase the average core power. A description, developed at ORNL, of the connection between limit-cycle oscillation and increase in average power will be presented in the full paper. As the average power increases, the water-level controller increases feedwater flow. This increases the core inlet sub-cooling. The core inlet subcooling effects the period of oscillation and the period of oscillation effects the amplitude of the oscillation. This effect is explained in terms of the interplay between the void and doppler reactivities and the cooling time of the fuel between successive surges of the coolant flow, i.e., density wave.

Most of the effort in the program is on evaluating BWR response to an ATWS event, with emphasis on the role of oscillations during the recovery phase and the appropriateness of ATWS procedures. Calculations of the LaSalle event performed by BNL with the EPA indicated the possibility of very large limit-cycle oscillations if the reactor would fail to scram, i.e., ATWS. The amplitude of these oscillations remains bounded. The oscillations continue unless the feedwater flow is throttled or the feedwater temperature is increased. On the question of implications to ATWS events, the increase in average power causes the suppression pool heatup rate to increase over that for an ATWS without oscillations. The calculations have not shown density-wave induced oscillations during the lowered water level portion of an ATWS scenario.

The research program includes a set of ATWS calculations to estimate the uncertainty in the maximum temperature of the suppression pool. We are using an approach from the successful code scalability, applicability and uncertainty (CSAU) study undertaken for a large-break LOCA. That approach leads to a quantified estimate of the uncertainty. The full paper will describe the reference case scenario, list the parameters chosen for sensitivity calculations, and show some results of the calculations.

RESOLUTION OF US REGULATORY ISSUES INVOLVING BWR STABILITY

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SUMMARY

The U.S. Nuclear Regulatory Commission reviews of the March 1988 instability event at the LaSalle County Nuclear Station, Unit 2, identified several generic concerns regarding the stability of other boiling water reactors (BWRs). The NRC staff and its Office of Research began a reexamination of BWR instability characteristics and consequences. Discussions with the BWR Owners Group (BWROG) and General Electric Company (GE) resulted in the BWROG initiation of calculations by GE to explore the characteristics and consequences of (1) asymmetric oscillations and of (2) large symmetric oscillations in conjunction with anticipated transient without scram (ATWS) events. These were deemed to be the significant areas requiring further analytical exploration as a result of the LaSalle event. Through analysis, the BWROG found that the "high neutron flux" reactor trip from average power range monitor (APRM) signals does not provide protection against the local high neutron flux that can occur during out-of-phase modes of instability. Further, the calculations showed that the Critical Power Ratio (CPR) safety limits could be exceeded in violation of the General Design Criteria 10 and 12. Because of these results, interim operating recommendations to reduce the risk associated with instability were proposed by the BWROG and GE in a GE letter to BWR Utilities, of November 1988. The staff issued to all licensees Supplement 1 to Bulletin 88-07, which approved the proposed BWROG-GE interim operating recommendations with some additions. Regional inspections verified the implementation of the recommended actions. Supplement 1 also indicated that the staff would continue to work with BWROG to develop a generic approach to long-term corrective actions.

Based on analytical studies that GE performed for the BWROG during 1989-90, the BWROG has proposed alternate solutions that will be available for the licensees to choose among for implementation in their reactors. The solutions involve automatic protection features either to prevent operation in a predefined region of the power and flow map where instabilities are considered likely, or to detect and terminate instabilities prior to violation of specified acceptable fuel design limits. Generic resolution of the long-term solutions is expected to be complete by the end of 1990.

Both the NRC and the BWROG are evaluating the characteristics, sensitivities, and safety significance of large power oscillations that result from ATWS events. The studies will examine the need to modify the automatic and operator actions previously developed for response to an ATWS event because of oscillation effects not fully considered in previous studies. Operator actions under consideration include injecting boron earlier, reducing feedwater flow, and performing other measures to reduce core inlet subcooling. The effect of these actions on avoiding or damping oscillations and reducing heat load to the suppression pool are being specifically examined.

The current status of these studies and an assessment of actions needed for closure of the issue are presented.

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This report contains summaries of papers on reactor safety research to be presented at the 18th Water Reactor Safety Information Meeting at the Holiday Inn Crowne Plaza in Rockville, Maryland, October 22-24, 1990. 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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	PLAZA I	PLAZA II	PLAZA III
	Plenary Session	Plenary Session	Plenary Session
Mon AM	Severe Accident Research I **Session 1**	Pressure Vessel Integrity **Session 2**	Radiation Protection BRC and Part 20 **Session 3**
Mon PM	Severe Accident Research II **Session 4**	Piping & NDE **Session 5**	Containment Testing & Structural Engineering **Session 6**
Tues AM	Aging & Components I **Session 7**	Individual Plant Examination Program & Other Issues **Session 8**	Human Factors Research **Session 9**
Tues PM	Aging & Components II **Session 10**	Organizational Factors & Reliabiity Assessement **Session 11**	Waste Management Research **Session 12**
Wed AM	Severe Accident Research III **Session 13**	Earth Sciences **Session 14**	Accident Management **Session 15**
Wed PM	Probabilistic Risk Assessment Topics **Session 16**	Seismic Engineering **Session 17**	Thermal Hydraulics **Session 18**

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