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PREFACE

This report contains summaries of papers on reactor safety research to be presented at the 21st Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel in Bethesda, Maryland, October 25-27, 1993. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, U.S. NRC. Summaries of invited papers concerning nuclear safety issues from U.S. government laboratories, the electric utilities, the Electric Power Research Institute (EPRI), the nuclear industry, and from foreign governments and industry are also included. The summaries have been compiled in one report to provide a basis for meaningful discussion of information exchanged during the course of the meeting, and are given in the order of their presentation in each session.

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

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THE PROBABILITY OF CONTAINMENT FAILURE BY DIRECT CONTAINMENT HEATING IN ZION

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This report is the first step in the resolution of the Direct Containment Heating (DCH) issue for the Zion Nuclear Power Plant using the Risk Oriented Accident Analysis Methodology (ROAAM). This report includes the definition of a probabilistic framework that decomposes the DCH problem into three probability density functions that reflect the most uncertain initial conditions (UO_2 mass, zirconium oxidation fraction, and steel mass). Uncertainties in the initial conditions are significant, but our quantification approach is based on establishing reasonable bounds that are not unnecessarily conservative. To this end, we also make use of the ROAAM ideas of enveloping scenarios and "splintering". Two causal relations (CRs) are used in this framework: CR1 is a model that calculates the peak pressure in the containment as a function of the initial conditions, and CR2 is a model that returns the frequency of containment failure as a function of pressure within the containment. Uncertainty in CR1 is accounted for by the use of two independently developed phenomenological models, the Convection Limited Containment Heating (CLCH) model and the Two-Cell Equilibrium (TCE) model, and by probabilistically distributing the key parameter in both, which is the ratio of the melt entrainment time to the system blowdown time constant. The two phenomenological models have been compared with an extensive data base including recent integral simulations at two different physical scales ($1/10^{th}$ scale in the Surtsey facility at Sandia National Laboratories and $1/40^{th}$ scale in the COREXIT facility at Argonne National Laboratory). The loads predicted by these models were significantly lower than those from previous parametric calculations. The containment load distributions do not intersect the containment strength (fragility) curve in any significant way, resulting in containment failure probabilities less than 10^{-3} for all scenarios considered. Sensitivity analyses did not show any areas of large sensitivity. The feasibility of extrapolating containment loads distributions to most other Pressurized Water Reactors (PWRs) is explored.

INTEGRATED ANALYSIS OF DCH IN SURRY*

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The probability of containment failure during high pressure melt ejection (HPME) at the Surry plant was estimated. An approach was used that considered both the likelihood of a high pressure vessel breach occurring and probabilistic representations of the resultant containment loading (termed direct containment heating or DCH) and the containment strength. The evaluation was performed for the two dominant high pressure sequences identified in NUREG-1150: short-term station blackout (with loss of all emergency core coolant except accumulators and steam generator auxiliary feedwater at the initiation of the accident), and long-term station blackout (with initial operation of steam generator auxiliary feedwater, but subsequent failure which leads to core damage).

This work is part of a Nuclear Regulatory Commission effort on DCH resolution that involves three national laboratories and one university. The key elements in the DCH resolution effort are: (1) the probability that sequences will proceed to vessel breach while at high pressure, rather than being depressurized by failures such as stuck-open relief valves, pump seal leaks, or hot leg failure; (2) the conditions of the reactor coolant system (RCS), containment, and lower head debris at the time of vessel breach; (3) the containment pressurization accompanying HPME; and (4) the containment strength. Resolution was approached by performing in-depth evaluations for the Surry and Zion plants, and developing a methodology for extrapolating this information to other pressurized water reactor (PWR) plants. Other papers at this meeting focus on the analyses that were performed to estimate the probability of HPME occurring for various sequences at Surry and Zion and to estimate the containment failure probability for Zion for various HPME sequences. This paper focuses on an approach that addresses both of these aspects of the issue for Surry.

A probabilistic approach was used in which the various accident progression pathways were delineated in an event tree. Probabilities were estimated for the individual events that collectively define the various pathways, and distributions for variables such as melt mass and composition were established to reflect their uncertainties.

* This work is supported by the U.S. 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.

The probability of a sequence proceeding to vessel breach at high pressure was estimated by Idaho National Engineering Laboratory (INEL) staff, based primarily on SCDAP/RELAP5 analyses for short-term station blackout sequences. The probability of hot leg or surge line failure was estimated for four variations: stuck-open power operated relief valve (PORV), 250 gpm per pump leakage, 480 gpm per pump leakage, and no induced RCS failures. The RCS pressure at vessel breach was also estimated by INEL for these variations. We extended these probabilities to cover long-term station blackout sequences. By combining this information with the probabilities of a stuck-open PORV or pump seal leak, the probability of HPME was established.

The conditions of the RCS, containment, and lower plenum debris were estimated by scaling the distributions that had been established for Zion. In that evaluation, four scenarios were defined to envelop the credible melt progressions, and results for the four scenarios were presented separately. In this evaluation, the conditional probabilities of containment failure for the four scenarios were combined with the HPME probability to get an estimate of the containment failure probability for each of the four scenarios that incorporated the probability of depressurizing the reactor coolant system before vessel failure.

The two-cell equilibrium (TCE) model, which was used for the Zion loads evaluation, was also used in this Surry evaluation to provide estimates of containment pressurization. These loads were compared to the containment strength distribution from NUREG-1150 to give the probability of containment failure. The TCE calculations indicated higher conditional containment failure probabilities for Surry than had been calculated for Zion. However, the evaluations indicated that there is not a large threat from DCH at Surry when both the probability of HPME and the resulting DCH loads were considered.

CONTAIN calculations were performed to investigate potential conservatisms in the TCE calculations. Calculations were performed for selected melt conditions from the scenario with the most threatening loads (when using the TCE model). Cases were run with (1) melt conditions near the median of the distributions and (2) melt conditions at the high end of the distributions and at full system pressure. Several variations were run to examine the effect of DCH modeling assumptions for uncertain phenomena. The CONTAIN calculations indicated that the dominant modeling assumptions were neglect of heat transfer to surroundings and partial combustion of hydrogen on DCH time scales. Accounting for heat transfer significantly lowers DCH loads for these cases.

Experimental Results for DCH Tests at 1:6 and 1:10 Scale With Surry Geometry

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The Containment Technology Test Facility (CTTF) and the Surtsey Test Facility at Sandia National Laboratories are used to perform scaled experiments for the Nuclear Regulatory Commission (NRC) that simulate High Pressure Melt Ejection (HPME) accidents in a nuclear power plant (NPP). These experiments are designed to investigate the phenomena associated with direct containment heating (DCH). High-temperature, chemically reactive melt is ejected by high-pressure steam into a scale model of the Surry reactor cavity. Debris is entrained by the steam blowdown into a containment model where specific phenomena, such as the effect of subcompartment structures, prototypic atmospheres, and hydrogen generation and combustion, can be studied.

Three integral effect tests using a 1:6 scale model of the Surry NPP (CTTF) and one test at 1:10 scale in Surtsey were performed. Scale models of the Surry reactor pressure vessel (RPV), reactor support skirt, control rod drive missile shield, biological shield wall, cavity, instrument tunnel, residual heat removal (RHR) platform, RHR heat exchangers, seal table room (STR), seal table, operating deck, and crane wall were constructed inside the facilities. A 158-kg charge of iron oxide/aluminum/chromium thermite was used as a corium melt simulant in CTTF (30 kg in Surtsey). The facility's atmosphere were initially heated above saturation temperature, and then pressurized with a mixture of steam, air, and H_2. The amount of preexisting hydrogen in the facilities represented levels of hydrogen produced by partial clad oxidation during the core degradation process in a loss-of-coolant accident. The molten core debris was ejected using scaled amounts of superheated steam into the reactor cavity.

Pressures in basement, RHR platform, and seal table room closely tracked vessel pressure. Temperatures from the vessel operating deck-to-dome thermocouple arrays indicate peak bulk gas temperatures of 900-1500 K. The measured containment peak pressure increase ranged from 0.28 to 0.43 MPa.

These experiments were conducted for three primary purposes: (1) to measure the pressure load on scaled containments caused by energy transfer from the HPME; (2) to investigate the amount of hydrogen combustion by a HPME in a prototypic steam/airH_2 atmosphere; (3) to measure posttest debris distribution in a containment model; and (4) to validate DCH models in a prototypic, large-scale experiment.

Results of Separate Effects Simulation Experiment on Corium Dispersion in DCH

by

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The present research focuses on the corium dispersion in the direct containment heating in severe accidents. The degree of dispersion has not only the strongest parametric effect on the containment loading, but also the highest uncertainty in predicting it. In view of this, a separate effect experimental program has been initiated in March 1992 by the NRC at Purdue University to study the detailed mechanisms of corium entrainment in a reactor cavity and of trapping in a subcompartment. Four major objectives for this corium dispersion study are, 1) to perform detailed scaling study using newly proposed step-by-step integral scaling approach, 2) to perform carefully designed simulation experiments using water-air and woods metal-air in a 1/10 linear scale model, 3) to develop reliable mechanistic models and correlations for the corium jet disintegration, entrainment, droplet size, liquid film carryover, and subcompartment trapping, and 4) to perform stand alone calculations for prototypic conditions. The combination of water-air and woods metal-air as working fluid gives a unique data base over broad parametric ranges which can be used together with the integral test results to develop reliable models and correlations. The 1/10 linear scale model for the cavity and subcompartment was chosen based on two considerations. The first consideration is the relation of the prototype scale and the scale of the existing standard data base for liquid entrainment and droplet size. Since most of the annular flow data were taken in a system having a hydraulic diameter around 2 to 3 cm., the scale ratio to the prototype is about 1/100. The present 1/10 linear scale model increases the scale ration by a factor of 10. The property scale base can be significantly broadened by using water as well as woods metal which has very similar hydrodynamic properties as those of molten corium. When the existing models are improved by the use of these data, the scaling reliability will increase significantly. The second consideration is the scale relation to the integral test data base. The most extensive integral test data were obtained by SNL in their 1/10 scale facility. Thus, using the same linear scale, the separate effect results can be compared directly with the integral tests.

The first year simulation experiments were conducted using air-water in the Purdue facility. The focus of the experiments is the detailed measurements of following phenomena in the cavity and subcompartment.

i. Corium discharge and corium jet disintegration

ii. Liquid corium spread-out upon impact of the jet or droplets

iii. Liquid film motion and transport

iv. Liquid entrainment, droplet size distribution and droplet mass flux

v. Gas velocity in the cavity

vi. Subcompartment liquid trapping

vii. Liquid carryover into the dome section

The air-water simulation experiments have been carried out for the phenomena in the cavity and subcompartment with the Zion geometry and 0.35 m prototypic break size. The liquid film transport mechanism, entrainment process and subcompartment trapping mechanism have been measured by hot film probes, resistivity based film thickness probes, pitot tubes, isokinetic droplet sampling probes and several pressure transducers. It was observed that the entrainment process was rather rapid (~ 0.2 sec) and about half of the liquid which remained in the cavity at the time of the gas blow down was entrained into small droplets. The rest of the liquid was transported out as a liquid film. Prior to the gas discharge about 15% of the liquid came out by its own inertia. Thus approximately 46% of the liquid became droplets in the cavity. The volume median droplet size was in the range of 0.2 to 0.6 mm depending on the location and time. Approximately 3% of liquid was transported to the upper dome section through the seal table room holes and exhaust holes. The rest of the liquid was initially deflected into the horizontal directions by the bottom of the seal tables room. This impingement and flow diversion into a horizontal plane was very effective in trapping the liquid in the subcompartment. Smaller droplets after this impingement and flow diversion were lifted up by the continuous gas flow and carried over to the dome section. However, this process was limited to very small droplets. Most of the liquid subsequently fell on the floor of the subcompartment and formed a thick liquid film.

The present separate effect experiments not only gave clear flow visualization in the cavity, subcompartment and the bottom of the dome section, but also gave detailed quantification of various processes related to the corium dispersion. All the instrumentations have worked as they were designed and benchmark experimental data have been obtained under carefully controlled conditions. From these, a considerable improvement on understanding the corium dispersion and trapping mechanisms have been obtained. The facility is now modified to accommodate the experiments using the woods metal, which has a low melting temperature of ~ 75° C.

DEBRIS DISPERSAL EXPERIMENTS IN A REACTOR CAVITY WITHOUT AN INSTRUMENT TUNNEL

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To date the research into DCH phenomenology has been focused entirely on PWR designs with lower mounted instruments. These PWRs were typically housed in reactor cavities with large exit side tunnels. While experimental data on a subset of these cavities is quite complete, the ability to extrapolate the Debris Dispersal data obtained for these cavities to cavities of significantly different design is unclear. It was the purpose of this effort to expand the current debris dispersal data base by performing a limited investigation of debris dispersal from a reactor with top mounted instrumentation enclosed in a circular reactor cavity. The specific objectives of this effort were the following:

1. Characterize the hydrodynamics of the debris dispersal process.

2. Define the role of the missile shield and refueling pool in debris retention.

3. Provide data for use in evaluating the applicability of debris dispersal correlations.

The current test series consists of a 1:20 scale debris dispersal simulation experiment. The simulated RV hole size was between 2 and 5 square feet. To fully simulate the debris dispersal process from the reactor cavity to the upper compartment, the test facility modeled (1) a narrow annular gap separating the RV and the reactor cavity interior wall (simulating the reactor cavity region from the lower head to the upper head flange), (2) a wider annular gap in the upper annulus (simulating the area expansion in the reactor cavity above the upper flange elevation) (3) nozzle cutout areas connecting the cavity to the lower compartment (4) a missile shield and (5) the refueling pool. As a result of the variability among the many plant designs that fall into this design category a "composite" reactor cavity was constructed which typically included the most limiting attributes of the prototype cavity group.

Approximately 100 experiments were performed. The majority of the experiments used nitrogen to simulate the driving fluid and water as the corium simulant. Five tests were performed with nitrogen as the driving fluid and wood's metal corium simulant. To expand the data base in this geometrical configuration, several parametric studies were performed. These studies included (a) substituting helium as the driver fluid in several experiments (b) variations in driver pressure between 100 and 1000 psia, (c) variation in simulated corium charge (d) variation in simulated RV failure hole size, and (e) variation in missile shield location.

Preliminary review of data obtained from these experiments indicated the following:

1) Debris entrainment from the cavity is relatively complete even down to pressures as low as 100 psia.

2) Debris dispersal into the upper compartment is restricted by the interaction of the debris with the missile shield and flow through the nozzle cutouts.

3) The overall dispersal of corium into the upper compartment is dependent on (a) RCS pressure (b) lower plenum inventory and (c) RV failure area

4) The mechanism for particle fragmentation changes as the debris dispersal process changes from one of momentum driven film expulsion to a process governed by droplet "entrainment".

Materials Engineering Branch Activities in 1993

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The Materials Engineering Branch, Division of Engineering, Office of Nuclear Regulatory Research, has had an exceptionally busy year of work, encompassing the traditional research program, regulatory codes and standards development, and direct technical assistance to NRR. The research work is detailed in the following papers of this volume, and is highlighted in this summary; the regulatory codes and standards work and the direct technical assistance to NRR however, are not described elsewhere and so are included in more detail in this summary.

The research program continues to have a strong focus on fracture mechanics issues related to pressurized thermal shock (PTS). The Heavy Section Steel Technology (HSST) program at Oak Ridge has emphasized a number of technical issues within the area of crack-tip constraint, including biaxial loading conditions and shallow-surface crack effects. Accurate understanding and representation of these potentially competing effects is necessary to assess the degree of conservatism and margins in present fracture methodologies. The degradation of pressure vessel steel mechanical properties caused by neutron radiation embrittlement continues to be a critical issue. Irradiation is currently underway on a typical production submerged-arc low upper shelf weldment made with Linde 80 flux; and good progress has been made on developing models to predict transition temperature increase, drop in upper-shelf energy, and the response of both to post-irradiation annealing. Direct neutron flux spectrum measurements are being made on a VVER-440 mockup experiment at the Czech Nuclear Research Institute in a cooperative program; the Czech results are providing important verification of the neutron flux spectrum calculations done in the U.S. using transport codes. In addition to the work underway in the IPIRG (International Piping Integrity Research Group)-2 program on dynamic and seismic effects in piping components, a multi-disciplined effort has continued at Battelle Columbus Laboratories on "Short Cracks" to assess the applicability of current ASME Code rules for long cracks. Environmentally assisted cracking and corrosion continues at Argonne National Lab, wherein studies now show that previous theories on irradiation assisted stress corrosion cracking cannot be substantiated. Additional studies are underway to strengthen the data base on environmental effects on fatigue life particularly for carbon and low alloy steels. A wide range of activities is underway at Battelle-Pacific Northwest Laboratories on non-destructive examination. Progress has been made in the difficult area of ultrasonic (UT) inspection of cast stainless steel, and in evaluation of commercial UT systems so that NRC Regional staff can better interpret inservice inspection results from periodic plant inspections. Lastly, good progress is reported by Battelle PNL on improving the detailed plans for inservive inspection of power plant components, on the basis of risk posed by those components; this work is being done in cooperation with the ASME Research Task Force, and a Section XI working group.

Activities undertaken this year by the MEB staff in the preparation of a series of rule changes and Regulatory Guides, are as follows:

Proposed Revisions to Rules

- 10CFR50.61 "Pressurized Thermal Shock"
- 10CFR50 App. G "Fracture Toughness Rules"
- 10CFR50 App. H "Reactor Vessel Material Surveillance Requirements"

Proposed Rule

- 10CFR50.66 "Thermal Annealing Requirements"

Draft Regulatory Guides

- DG 1-027 "Format and Content of Application for Approval of Thermal Annealing of RPV"
- DG 1-025 "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"
- DG 1-023 "Evaluation of Reactor Pressure Vessels with Charpy Upper Shelf Energy Less than 50 ft-lb."

The rule changes correct a number of inconsistencies, clarify NRC's intent, and respond to Commission guidance on a number of issues. The proposed rule and Reg. Guide on thermal annealing for example, sets forth NRC's requirements for licensees that might be considering annealing in the future. The Reg. Guide on neutron dosimetry has been in preparation for several years, and is very complementary to the vessel integrity and annealing guidance; it integrates and "codifies" the best of current practice today for this technology. Finally, the Reg. Guide on low Charpy upper shelf energy was completed this year. The MEB staff used this methodology to perform generic bounding calculations to determine allowable upper shelf energies in support of the NRR staff's evaluation of Generic Letter 92-01 "Reactor Vessel Structural Integrity. This very critical guide shows a method acceptable to the staff to demonstrate that the margins of safety against fracture are equivalent to those required by Appendix G of the ASME Code, if the Charpy upper-shelf energy of reactor pressure vessel beltline materials fall below 50 ft-lbs during the plant operating lifetime. A final, intense effort this year involved preparation of a report "Evaluation of Steam Generator Tube Leak Rate Under MSLB Conditions,"[1] in support of NRR's reevaluation of interim plugging criteria for steam generator tubes. The evaluation provided a "best estimate" of the primary-to-secondary leak rate that would be expected due to a main steam line break (MSLB) and the resulting leakage from steam generator tubes at the Trojan nuclear power plant. MEB staff were also members of the task group which prepared the draft NUREG-1477 "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes," and will continue to be involved in steam generator activities in the future.

The Materials Engineering Branch will continue strong efforts in both the traditional management of research, and in regulatory code and standards development activities, providing input to both sides through its very close ties to NRR, the Regions, the U.S industrial community, and codes and standards groups, and through contacts and exchange with colleagues in foreign countries.

[1] E. S. Beckjord to T.E. Murley, January 15, 1993.

BIAXIAL LOADING AND SHALLOW-FLAW EFFECTS ON CRACK-TIP CONSTRAINT AND FRACTURE TOUGHNESS

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Postulated pressurized-thermal-shock (PTS) accident conditions remain an important safety assessment issue in the licensing of commercial nuclear reactor pressure vessels (RPVs), especially in the case of aging nuclear plants. The development of technology required for an accurate assessment of the margins against fracture in RPVs under PTS conditions is a focal point of the Nuclear Regulatory Commission-funded Heavy-Section Steel Technology (HSST) Program. Currently, the HSST Program is seeking to obtain an improved understanding of several issues that could significantly impact the fracture mechanic technologies employed in these safety-assessment procedures. One important area of research is that of crack-tip constraint, a topic that encompasses a number of factors relating to the material fracture resistance and to transfer of fracture-toughness data from small-scale specimens to large-scale structures. Factors affecting crack-tip constraint include structural and crack geometry, loading conditions, and material properties. Within these categories, far-field, tensile out-of-plane biaxial loading conditions, and shallow-surface crack effects have been identified as issues that could significantly impact RPV safety assessments. This paper provides an overview of ongoing HSST Program research aimed at evaluating the effects of biaxial loading conditions and shallow-crack geometries on constraint conditions and, consequently, on transfer of fracture-toughness data to RPVs. Development and evaluation of fracture methodologies for characterizing constraint conditions represent a major element of this research.

Shallow Crack Effects

PTS loading produces maximum stresses and minimum temperature adjacent to the inner surface of a reactor vessel, where irradiation embrittlement effects are also most severe. These effects combine to produce a preponderance of crack initiations in a probabilistic PTS analysis from shallow, inner-surface flaws.

The HSST Program is investigating the increase in effective fracture toughness of A533 B steel associated with shallow flaws and the implications of the shallow-flaw effect on life assessments. A series of thirty-eight tests were conducted on three-point-bend specimens having a beam depth (W) of ~100mm and normalized crack depth (a/W) varying from 0.1 to 0.5. The specimens were fully instrumented to determine both J-integral and crack-tip-opening displacement measurements of fracture toughness. Test data from these beams indicate a significant increase in the fracture toughness of shallow-crack specimens compared with deep-crack specimens in the transition region of the toughness curve for unirradiated A533 B steel. If the toughness increase present in the test specimens were also present in a reactor vessel, the impact on PTS analyses could be significant. To facilitate transferability of the specimen data to an RPV, posttest finite-element analyses have been performed on several test specimens and a reactor vessel for a single PTS transient. Interim results from these analyses indicate that potentially a greater margin of safety is derived from inclusion of shallow-crack effects in the model.

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Biaxial Loading Effects

Existing data suggest that shallow-flaw specimens under tensile out-of-plane biaxial loading (i.e., parallel to the crack front) will exhibit a toughness reduction compared to shallow-flaw uniaxially loaded specimens. Determination of the extent of this toughness reduction under conditions prototypic of an RPV is one of the main goals of the biaxial fracture toughness program. A cruciform specimen with a two-dimensional, shallow, through-thickness crack has been developed for biaxial fracture-toughness testing. The test section of the cruciform specimen was very similar to the shallow-crack beams, with a cross section of 91 × 102mm and a straight through-crack of uniform depth of 10mm. Five specimens were tested in the development phase, all of which yielded consistent results. The critical fracture load for each specimen was approximately the same, but the uniaxial specimen withstood substantially more deformation at failure than did the biaxial specimens. The plastic component of "work" at the crack tip was a factor of 3 greater for the uniaxial specimen than for the biaxial specimens. Posttest finite-element analyses were used to determine the critical J-integral value of each test. Results from these tests indicate that the shallow-crack toughness increase is partially, but not totally, removed by the application of far-field biaxial loading. In addition, the scatter of data from the admittedly small population of biaxial tests appeared to be significantly less than that obtained from uniaxial tests on the same material. However, additional data are required to solidify these conclusions. A proposed test matrix for additional uniaxial and biaxial testing is described in the paper.

Crack-Tip Constraint

Current pressure vessel fracture prevention technology relies on the use of fracture-correlation parameters (K) or (J) to characterize both the applied load and the resistance of material to crack initiation. Shortcomings of these conventional one-parameter, fracture-correlation methods, which impact issues associated with the transferability of small-specimen (i.e., surveillance-sized) toughness data to large-scale RPV applications, are being addressed through evaluations of various dual-parameter fracture methodologies.

The existing methodologies being investigated include stress-based fracture characterizations (i.e., J–Q methodology combined with Ritchie-Knott-Rice fracture criteria, and the Dodds–Anderson constraint correction technique) and stress-strain-based characterizations (i.e., plane-strain fracture ductility techniques due to Clausing, Barsom, Merkle, and other researchers). Determinations are being made concerning the bounds of applicability of the existing constraint effects correlation methodologies (i.e., how effective are they in matching existing data?). If the existing methodologies are found to be deficient, determinations will be made concerning whether or not they can be modified to make them work. If necessary, alternative constraint methodologies will be developed and validated.

Constraint effect correlation assessments of the stress-based and ductility-based characterizations applied to uniaxially and biaxially loaded, shallow-crack beam and cruciform data are currently under way. These data include those obtained from shallow-crack beams by ORNL and by the Naval Surface Warfare Center as well as full-thickness clad beams tested at NIST for ORNL.

In addition to measured structural response, test data employed herein include results from test specimen fractography focusing on locations of crack initiation sites. Correlations of crack initiation sites with local crack-tip fields may provide a basis for validation of the various fracture models.

Heavy-Section Steel Irradiation Program Progress—Recent Results from Midland Low Upper-Shelf Weld Studies*

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The Heavy-Section Steel Irradiation (HSSI) Program at the Oak Ridge National Laboratory includes several activities to examine various aspects of the fracture resistance of materials classified as low upper-shelf (LUS) welds. This class of welds was produced by the submerged-arc welding process using a specific type of welding flux (Linde 80). Linde 80 flux was deemed suitable for reactor vessel fabrication since it produced a very fine dispersion of inclusions within the weld and a resultant lower number of reportable defects observed by radiography than commonly obtained with other welding fluxes. An unfortunate by-product of this fine dispersion of inclusions was that they provided such a large number of microvoid initiation sites that the macroscopic resistance of these welds to ductile crack extension by the microvoid growth and coalescence process was significantly reduced. This problem was further aggravated by the fact that, at the time these welds were being fabricated, the deleterious effect of impurity copper on the radiation-induced degradation of toughness in pressure vessel materials was not yet generally recognized. Hence, a copper coating on the welding wire, a common practice to minimize corrosion products on ferritic welding wire and enhance electrical contact between the feed rollers and the workpiece, was also used. The combination of the use of Linde 80 welding flux and copper-coated welding wire has resulted in a significant number of pressure vessels in which major fabrication welds have both relatively low resistance to ductile fracture in the unirradiated condition and a high sensitivity to further degradation from neutron exposure. The principal, current activity within the HSSI Program to examine LUS welds is the Tenth Irradiation Series, in which the effects of radiation on the fracture toughness of commercially fabricated LUS submerged-arc welds from the reactor pressure vessel of the canceled Midland Unit 1 nuclear plant are being investigated. Additional LUS studies within the Eighth and Ninth Irradiation Series are in their early stages but, in combination, will eventually examine the detailed relationship between the Charpy V-notch (CVN) and fracture toughness degradation for the irradiated, annealed, and reirradiated conditions for a generic LUS weld.

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The welds from the Midland plant carry the Babcock and Wilcox Co. designation WF-70, a specific combination of weld wire and welding flux that exists in several commercial pressurized water reactors. The initial part of this study involves the determination of variations in chemical composition, reference temperature (RT_{NDT}), tensile properties, and fracture toughness throughout the welds. Four 1.17-m-long (46-in.) sections of beltline weld and two similar sections of nozzle course weld have been examined. For each section, drop-weight and CVN specimens were machined from locations through the weld thickness. The nil-ductility transition temperatures ranged from –40 to –60°C (–40 and –76°F). Because the Charpy impact energy did not achieve 68 J (50 ft-lb) at NDT + 33°C (60°F) at any of the locations, the RT_{NDT}s are all controlled by the Charpy behavior. The RT_{NDT}s vary from –20 to 37°C (–4 to 99°F) while the upper-shelf energies (USEs) varied from 77 to 108 J (57 to 80 ft-lb). Analysis of the combined data revealed a mean 41-J (30-ft-lb) temperature of –8°C (17°F) with a mean USE of 88 J (65 ft-lb). The tested Charpy specimens were used to determine chemical composition at each location. Even though all the welds carry the WF-70 designation, the bulk copper contents range widely, from 0.21 to 0.34 and 0.37 to 0.46 wt % in the beltline and nozzle course welds, respectively. Atom probe analyses of the matrix copper content yielded an average of about 0.10 wt %, indicating depletion of matrix copper from precipitation during postweld heat treatment.

Tensile and fracture toughness properties were determined on nozzle and beltline weld metals at six temperatures ranging from –100 to 288°C (–148 to 550°F). The yield strength of the nozzle weld metal was significantly higher than that of the beltline weld, on the order of 100 MPa (15 ksi). All the fracture toughness tests to characterize the unirradiated material, using compact specimens ranging in size from 1/2T to 4T, have been completed. Data to characterize ductile-to-brittle transition temperature were evaluated using a test standard currently under development by the American Society for Testing and Materials. For this evaluation, it was of interest to determine the position of a median fracture toughness transition curve (master curve), using only the data from six 1/2T compact specimens, to compare with data from large specimens. The proposed practice provides guidelines for selection of an optimum test temperature from a Charpy curve. Since the nozzle and beltline materials had been shown to exhibit the same average Charpy behavior, there was only one test temperature selection for both materials, –50°C (–58°F). The six 1/2T compact specimens of each material were tested at this temperature, and the "reference temperature" for the master curve was found to be –60°C (–72°F) for the beltline weld and –43°C (–45°F) for the nozzle weld. The irradiation of the Midland weld is in progress. The exposure of the first of the two large irradiation capsules, containing tensile, CVN, and fracture toughness specimens, to the primary target fluence of 1×10^{19} n/cm^2 (>1 MeV) has been completed and the second begun. Small fracture toughness and CVN specimens are also being exposed in low- and high-fluence scoping capsules to 5×10^{18} and 5×10^{19} neutrons/cm^2 to examine fluence effects over this range.

SUMMARY

Modeling of Irradiation Embrittlement and Annealing Data

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This project is developing improved irradiation embrittlement equations based on the increased amount of embrittlement data that has become available in recent years. Using data from the Power Reactor Embrittlement Database (PR-EDB) and the Test Reactor Embrittlement Database (TR-EDB) compiled at Oak Ridge National Laboratory, two approaches to modeling embrittlement are being investigated simultaneously: (1) development of models for transition temperature shift (TTS) and drop in upper shelf Charpy impact energy (ΔC_vUSE) due to irradiation and (2) development of a model for the entire Charpy curve as a function of irradiation exposure and other variables. In addition, the embrittlement project includes development of models for recovery in TTS and ΔC_vUSE due to annealing. The annealing data are primarily found in the TR-EDB.

Advanced statistical techniques are being used to identify potential modeling variables and their functional forms, including pattern recognition, transformation analysis, and regression analysis. Least squares fitting techniques are being used for fitting TTS and ΔC_vUSE embrittlement models and recovery models. Orthogonal distance regression analysis techniques are being used for fitting the Charpy curve model, where they can outperform standard least squares techniques because of the steep slopes in the transition region.

Results of the statistical analysis of embrittlement data will be presented to the extent they are available. Some potentially important independent variables may be discussed along with the results of substudies used to quantify variations in some important variables. Preliminary models for recovery in TTS and ΔC_vUSE due to annealing will be presented.

TRANSIENT ANALYSIS OF THE PIUS ADVANCED REACTOR DESIGN WITH THE TRAC-PF1/MOD2 CODE

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SUMMARY

The PIUS advanced reactor design was submitted for preliminary review to the Nuclear Regulatory Commission (NRC) by Asea Brown Boveri (ABB) Atom and its US Affiliate, Combustion Engineering (CE), in the form of a Preliminary Safety Information Document (PSID). Subsequently, ABB/CE submitted a Supplement to the PSID that documented changes to the PSID design.

Two related purposes are being served by our effort. First, we are gaining familiarity with the PIUS reactor's response to important transients and accidents. Second, we are identifying PIUS phenomena and processes and then assessing the capabilities of analytical tools to predict these phenomena and processes. With this information, the NRC can prepare plans to modify independent computer codes, as appropriate, for use during design certification. It has not been our purpose to determine the adequacy of the ABB/CE codes or the acceptability of reactor behavior, and no such conclusions will be reached.

In the PIUS design, reactivity is controlled by coolant boron concentration and temperature. There are no mechanical control rods. The reactor is submerged in a large pool of highly borated water, and the lightly borated water in the core is in continuous communication with the pool water through pipe openings called density locks that are always open between the primary system and the pool. The primary coolant pumps are operated so that there is a hydraulic balance in the density locks between the primary coolant loop and the pool, keeping the pool water and primary coolant separated during normal operation. A reactor scram is accomplished by either (a) activating a mechanical "scram-valve" system that permits the pressure-difference-driven flow of highly borated pool water through the scram-valve system piping into the primary while the primary pumps are operating, or (b) the introduction of highly borated pool water into the primary coolant via natural circulation anytime one or more of the four primary coolant pumps stop functioning, thereby eliminating the balance between the primary coolant loop and the pool.

The reactivity control of the PIUS reactor was raised as a policy issue in a formal Commission paper (SECY-93-092) dated April 8, 1993. This issue arises because General Design Criterion 26 in NRC's regulations requires that two independent reactivity control systems be provided. Further, one of the systems is required to use control rods, preferably using a positive means for insertion. The PIUS design does not have control rods.

Emphasis in our calculations has thus been placed on understanding the reactor trip and shutdown phenomena during transient and accident events. We have analyzed both the primary reactor trip with the "scram-valve" system and the backup passive reactor trip that activates the density locks and induces natural circulation between the pool and the primary system. Further, we have evaluated the performance of the PIUS reactor for both main-steam-line break and large-break loss-of-coolant events. These transients and accidents

* This work has been done under the auspices of the Reactor and Plant Systems Branch in the Office of Nuclear Regulatory Research of the US Nuclear Regulatory Commission.

embody a range of phenomena. Results of the calculations will be presented in the full paper and published in the transactions.

Our PIUS calculations have been performed with the TRAC-PF1/MOD2 code. Our primary TRAC model of the plant includes a three-dimensional model of the reactor internal structures, including the reactor core, lower and upper plena, riser and downcomer regions, the regions where the hot- and cold-leg introduce or remove coolant from the internal structures, and the pressurizer and upper portions of the pool. The remainder of the model, including the primary coolant loops, was developed using one-dimensional components. Although we have the capability to calculate multidimensional thermal and hydraulic phenomena, TRAC-PF1/MOD2 utilizes a point kinetics model of the core neutronic behavior under transient and accident conditions. While nonuniform boron distributions have been calculated with the three-dimensional model for some of the transients, the code does not currently have the capability of calculating coupled multidimensional core thermal-hydraulic and neutronic phenomena. TRAC may have to be modified for this code to adequately simulate this possibly important phenomenological feature of the PIUS design by incorporating a multidimensional neutronics computational model. Finally, we have also developed a fully one-dimensional PIUS model, and this model is used for scoping studies because it runs faster.

CANDU 3 TRANSIENT ANALYSIS USING AECL CODES[1]

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The CANDU 3 design was submitted for preliminary review to the NRC by Atomic Energy of Canada Lt. (AECL) through its U.S. affiliate, AECL Technologies (AECLT). As part of this preliminary review, AECL extended an offer to let the NRC and its contractor, INEL, use the Canadian nuclear design computer code suite for thermal-hydraulic and neutronic analysis. We accepted this unusual offer and undertook an analysis of several design-basis accidents and operational transients using DEC 5000 workstations, both at the NRC and INEL. We received training in the use of the code suite at AECL facilities in Canada, and ran a variety of cases using input decks supplied by AECL.

Two related purposes are being served by our calculations. One is to gain familiarity with the CANDU reactor's response to important transients and accidents. The other is to see what analytical capabilities are needed so plans can be made to modify the NRC's independent computer codes for use during the design certification. It has not been our purpose to determine the adequacy of the AECL codes or the acceptability of reactor behavior, and no such conclusions will be reached.

Emphasis in our calculations has been placed on the loss-of-coolant accident (LOCA) without scram. In the CANDU 3 design, this is an extremely unlikely accident sequence because of the two fast-acting independent scram systems that are present. However, CANDU 3 has a positive coolant void coefficient of reactivity that results from the nature of the design. Consequently, if coolant is lost and negative reactivity from shutdown scram devices is not inserted quickly, power will increase rapidly and fuel melting may occur.

The positive void reactivity coefficient of CANDU 3 was raised as a policy issue in a formal Commission paper (SECY-93-092) dated April 8, 1993. This issue arises because General Design Criterion 11 in NRC's regulations requires that a reactor be designed so that, in the power operating range, the net feedback characteristics tend to compensate for a rapid increase in reactivity. While the CANDU design may satisfy this criterion because of its slightly negative overall power coefficient during normal power operation,

1. This work was done under the auspices of the Reactor and Plant Systems Branch (RPSB) in the Office of Nuclear Regulatory Research (RES) of the U. S. Nuclear Regulatory Commission (NRC).

void reactivity increases dramatically during a LOCA. The calculations performed in this study will provide some of the background needed to deal with this issue.

Calculations by AECL indicated that horizontal two-phase flow, radiative heat transfer to the pressure tubes, and header models in TRAC and RELAP may have to be modified for these codes to adequately simulate the CANDU design during LOCA's. Decisions regarding NRC's code choice and schedules for its modification will be made later this year. Results of the calculations will be presented in the full paper and published in the Transactions.

DATABASE AND MODELING ASSESSMENTS OF THE CANDU 3, PIUS, ALMR, AND MHTGR DESIGNS

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SUMMARY

BACKGROUND

The NRC has been conducting preliminary reviews of the CANDU 3, PIUS, ALMR, and MHTGR designs. CANDU 3 (Canadian Deuterium Uranium Model 3) is an evolutionary heavy water reactor design submitted by Atomic Energy of Canada Ltd Technologies (AECLT); PIUS (Process Inherent Ultimate Safety) is an innovative pressurized water reactor design submitted by ASEA Brown Boveri and Combustion Engineering (ABB-CE); ALMR (Advanced Liquid Metal Reactor), also called PRISM (Power Reactor Innovative Small Module), is a metal-fueled, sodium-cooled fast reactor design submitted by the U.S. Department of Energy (DOE); and MHTGR (Modular High-Temperature Gas-Cooled Reactor) is a graphite-moderated, helium-cooled reactor design also submitted by DOE.

The preliminary reviews of the four designs are aimed at identifying key technical areas and policy issues that will have to be addressed for standard design certification. Among the research tasks associated with these preliminary reviews is the assessment of databases and modeling capabilities for use in design confirmation. This paper summarizes the database and modeling assessments performed for all four designs.

OBJECTIVE

The objective of the assessment work for each design is to provide an early identification and prioritization of areas where further development of databases and computational models may be desirable in preparing for NRC's confirmatory analyses. Recent experience with the review of Westinghouse's AP600 design suggests that an early evaluation of potential areas for additional database development is needed to enable timely planning of associated NRC research activities. Early planning can be especially important where establishment of a confirmatory database entails constructing or modifying a major test facility.

The essential question addressed by the database and modeling assessments is: "What additional work is needed to have databases and computational models for these advanced reactors comparable to those for current light water reactors?" The question regarding databases encompasses data available to the applicant as well as to the NRC. Thus, some of the holes that will be found in the databases may have to be filled by the applicant while others will be filled by the NRC. This work does not attempt to determine who should bear responsibility for generating any additional data. The question regarding computational models, however, addresses only the NRC's independent audit codes. Information from the modeling assessments will be used to plan NRC's code development efforts.

METHODOLOGY

Because database development potentially requires longer term planning than does the development of computational models, the major emphasis of this work is placed on providing an early assessment of databases. A rather formalized process is therefore used in assessing databases, whereas potential areas for modeling enhancements are assessed in a more ad hoc manner for this preliminary review stage.

The structure of the database assessment process follows the broad logic of the NRC's CSAU (Code Scaling, Applicability, and Uncertainty) methodology. Accordingly, the assessment process addresses *design*, *scenarios*, *phenomena*, and *data* in that order. The resulting database assessment process is not linked to particular codes or models. While most data are ultimately used for code validation, important exceptions exist, such as certain data for ECCS (Emergency Core Cooling System) criteria and SAFDLs (Specific Accepted Fuel Design Limits).

The database assessment process is designed to ensure that all important phenomena are covered. It is applied to each of the four designs as described in the following five steps:

1. Select and list, for each event category (EC-I, EC-II, etc.), a set of representative event sequences that exercises a broad range of important phenomena for sequences in that event category.

2. For each representative event sequence, tabulate the important phenomena, along with the types and ranges of data that, if available, would be used to validate related models or criteria. The types and ranges of such data are grouped in up to ten phenomenological areas, one table for each area. For water-cooled reactors, the ten areas would be: 1) Reactor Physics, 2) Thermal Hydraulics, 3) Fuel Behavior and Core Melt Progression, 4) Fuel-Coolant Interactions, 5) Reactor Vessel Failure, 6) High-Pressure Melt Ejection, 7) Core-Concrete Interactions, 8) Hydrogen Combustion, 9) Fission Product Release and Transport, and 10) Containment Failure. In general, no more than four of these areas (1, 2, 3, and 9) may come into play in sequences of EC-I or EC-II (Anticipated Operational Occurrences or Design Basis Events), whereas all ten may arise in EC-III or EC-IV (Severe Accident or Residual Risk) sequences.

3. To complement the preceding "bottom-up" tabulation of data types and ranges for individual key phenomena, generate a list of integral data types and ranges that would be useful in a "top-down" sense for confirming key phenomena interactions.

4. Generate a set of tables indicating whether, and to what extent, the types and ranges of data identified in the previous two steps are included in the existing and planned databases identified by the vendor.

5. In order of priority, list and comment on those data types not covered by existing or planned databases where additional data would be most helpful toward confirming important safety characteristics of the design.

RESULTS

Because the work is now underway, results are not currently available but will be in place by the time of the conference.

ACKNOWLEDGEMENTS

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MELCOR Development for Existing and Advanced Reactors†

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Recent activities in the MELCOR development project have been focused on addressing deficiencies identified by assessments of improvement needs conducted by Sandia in the past, by the MELCOR Peer review completed in 1991, and by the ongoing MELCOR Assessment Program at Sandia. Excellent progress has been made in addressing these deficiencies, resulting in a much-improved version of the code, MELCOR 1.8.2, distributed to users at the end of March 1993. This version of the code received substantial testing on a suite of plant and experiment calculations and has demonstrated improved robustness and reliability, with significantly reduced numerical sensitivities to time step, computer system, and minor user input changes not expected to change results.

Major new models implemented in MELCOR 1.8.2 include (1) a parametric model for direct containment heating (DCH) following high pressure melt ejection; (2) ice condenser modeling by adaptation of the Heat Structures package degassing model; (3) quenching of debris relocating from the core region to the lower plenum; (4) a new package of models for lower plenum debris behavior in BWRs developed at Oak Ridge National Laboratory (ORNL); (5) core materials interactions modeling for solid dissolution and eutectics formation; and (6) radial relocation of debris within the reactor vessel.

Significant improvements have been made in aerosol modeling, particularly with regard to condensation and evaporation of water aerosols, and in the condensation and evaporation of fission product vapors. An updated version of the CORSOR correlations developed at Battelle's Columbus Laboratories that incorporates improvements in the release formulation and coefficients, consideration of mass transport limitations, and use of the Booth diffusion model, has been implemented. Improvements have also been made in the modeling of interfacial momentum exchange in the hydrodynamics package.

Efforts are underway to address hydrodynamics difficulties associated with bubble separation from boiling pools, to include film resistance and high mass transfer effects in condensation on structures, to complete the modeling of fine-scale natural circulation within the reactor vessel, and to implement CORCON-Mod3 in MELCOR.

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Additional development needs are recognized, which include improvements in modeling core degradation behavior, especially failures of core structures, debris crusts and the reactor vessel. Modeling of coolable debris beds in the reactor cavity is needed, and several areas of fission product behavior need to be addressed, including resolution of pool scrubbing concerns, modeling of chemical reactions on surfaces and in water pools, and inclusion of turbulent flow and inertial impaction deposition mechanisms in the RCS. Further reduction of numerical sensitivities and an improved user interface, especially better user guidance and enhanced graphics capabilities are desired.

While MELCOR possesses flexibility in the modeling of existing light water reactor plant systems, some new features of the advanced light water reactor designs, e.g., AP600 and SBWR, require improvements to MELCOR's capability to handle calculations of transient phenomena expected in these cases. SBWR-specific modifications are being carried out by ORNL, while three major modification tasks have been initiated for the AP600 at Sandia.

The first of these involves the drainage of water films on heat structures; MELCOR currently allows films to drain to a control volume pool but does not permit them to drain to a lower heat structure. Treatment of the latter is necessary to model the behavior of the AP600 containment shell and cooling system with a vertical stack of heat structures.

The second AP600 task will provide the capability to treat heat transfer from the exterior surface of the lower head to the liquid pool surrounding the lower vessel, including the capability for the user to supply a parametric "downward facing" heat transfer coefficient.

Third, since the AP600 has no lower head penetrations, failure will likely occur by creep rupture due to the weight of core debris and wide temperature variations resulting from a central hot spot and external liquid cooling. The third AP600 task will add the capabiltiy to use Larson-Miller relations to predict such creep rupture failure modes.

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MELCOR Technical Assessment at SNL†

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MELCOR [1] is a fully integrated, engineering-level computer code that models the progression of severe accidents in light water reactor (LWR) nuclear power plants. MELCOR is being developed at Sandia National Laboratories for the U. S. Nuclear Regulatory Commission (USNRC). The entire spectrum of severe accident phenomena, including reactor coolant system and containment thermal/hydraulic response, core heatup, degradation and relocation, and fission product release and transport, is treated in MELCOR in a unified framework for both boiling water reactors (BWRs) and pressurized water reactors (PWRs). The MELCOR computer code has been developed to the point that it is now being successfully applied in severe accident analyses, particularly in probabilistic risk assessment (PRA) studies.

MELCOR was the first of the severe accident analysis codes to undergo a formal peer review process. One of the major conclusions of the recent MELCOR Peer Review [2] was the need for a more comprehensive and more systematic program of MELCOR assessment. Since then, a number of assessment analyses have been completed and documented by Sandia, including: the LACE LA4 containment-geometry aerosol deposition test (SAND91-1532), the FLECHT SEASET natural circulation tests (SAND91-2218), the ACRR ST-1/ST-2 in-pile source term experiments (SAND91-2833), the OECD LOFT integral severe accident experiment LP-FP-2 (SAND92-1373), the Marviken-V ATT-2b and ATT-4 aerosol transport and deposition tests in system geometries (SAND92-2243), and PNL ice condenser experiments 11-6 and 16-11 (SAND92-2165). Results for those analyses have been presented previously.

Recent assessment work at Sandia has concentrated on evaluating new code models added in version 1.8.2. Many of these models were developed and incorporated into the code in response to major deficiencies identified by the MELCOR peer review. MELCOR assessment analyses at Sandia (either recently completed or still in progress), whose results will be summarized in this paper, include the ACRR DF-4 fuel damage experiment (SAND93-1377), the SNL and ANL IET direct containment heating (DCH) experiments, the ACRR MP-1 late-phase melt-progression experiment, and PWR TMLB' calculations with and without direct containment heating.

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The in-pile DF-4 experiment provided data for early phase melt progression in BWR fuel assemblies, particularly for phenomena associated with eutectic interactions in the BWR control blade and zircaloy oxidation in the canister and cladding. MELCOR provided good agreement with experimental data in the key areas of eutectic material behavior, and canister and cladding oxidation. A number of sensitivity studies were performed varying core, heat structure and hydrodynamic parameters. These studies showed that the new eutectics model in MELCOR 1.8.2 played an important role in predicting control blade behavior. Numerics studies revealed slight time step dependence and no machine dependence in the MELCOR results. Comparisons made with the results from four best-estimate codes showed that MELCOR did as well as those other codes in matching DF-4 experimental data.

The MELCOR computer code also is being assessed against several of the IET direct containment heating experiments done at 1:10 linear scale in the Surtsey test facility at Sandia and at 1:40 linear scale at Argonne. MELCOR results are being compared to test data in order to evaluate the new high-pressure melt ejection (HPME) DCH model recently added in MELCOR version 1.8.2, with good agreement generally found, and are also being compared to CONTAIN calculations. The effect of various user-input parameters in the HPME model, which define both the initial debris source and the subsequent debris interaction, were investigated in sensitivity studies. Several other non-default input modelling changes involving other MELCOR code packages were required in our IET assessment analyses in order to reproduce the observed experiment behavior. Calculations also have been done to identify whether any numeric effects exist in our DCH IET assessment analyses, with no significant machine-dependency or time-step effects found.

The PWR TMLB' analysis was first done as a demonstration calculation for the MELCOR peer review. This analysis has been used as a test problem during the development of MELCOR version 1.8.2, and basecase results from MELCOR 1.8.2 have been compared to basecase results from the release version of MELCOR 1.8.1. The effects of individual code models changed or added (*e.g.*, CORSOR-Booth fission product release, interfacial momentum exchange and countercurrent flow limit (CCFL), core debris radial relocation, core material eutectics interaction, and direct containment heating) have also been evaluated through sensitivity studies. Machine-dependency and time-step studies provide a measure of the significant improvement and progress made by the code developers in addressing and eliminating numeric effects in MELCOR version 1.8.2, identified as a major concern for version 1.8.1 by the peer review.

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EVALUATION OF MELCOR IMPROVEMENTS: PEACH BOTTOM STATION BLACKOUT ANALYSES*

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Long-term station blackout analyses in Peach Bottom were first carried out in 1990, using MELCOR 1.8BC, as part of an overall program between the U.S. Nuclear Regulatory Commission (NRC) and Brookhaven National Laboratory (BNL), to provide independent assessment of MELCOR as a severe accident/ source term analysis tool. In addition to the reference MELCOR calculation, several sensitivity calculations were also performed to explore the impact of varying user-input modeling and timestep control parameters on the accident progression and radionuclide releases to the environment calculated by MELCOR. The sensitivity studies helped to assess MELCOR by evaluating the changes in results in response to changes in input parameters. In some cases, there was increased confidence in the code based on "physical reasonableness" arguments, where the changes in results could be justified based on the phenomena being modeled. In other cases, areas of concern emerged [1]. (These concerns have been addressed as noted below.)

One such area was the impact of the selection of maximum allowable timestep (Δt_{max}) on the calculational behavior of MELCOR. Complete sequence calculations were carried out selecting two variations of Δt_{max} (5 and 3 sec), in addition to the reference case using Δt_{max} = 10 sec. Both variations were seen to delay the occurrence of most key events compared to the reference calculation and to substantially increase the release of Cs and I to the environment. The 5 sec. case gave the largest deviation in timing. With the release of a newer version of the code, 1.8DN, the impact of Δt_{max} was re-examined using MELCOR 1.8DNX (1.8DN with corrections for two code errors). Complete sequence calculations were carried out this time selecting five variations of Δt_{max} (10, 5, 3, 2, and 1 sec.). Once again, the results showed significant differences in timing of key events, and a lack of convergence of the solution with reduction of Δt_{max}. This uncertainty in results indicated the need for further investigation of the solution algorithm. The maximum uncertainty in environmental release fractions was a factor of seven (for Ru) and within a factor of four for the rest of the radionuclide over the entire range of Δt_{max}.

These findings were reported to the NRC, SNL, and the MELCOR Peer Review Committee. This alerted the NRC, the code developers, and the MELCOR Peer Reviewers to the importance of correcting the numerical sensitivities. As a consequence, a significant effort was undertaken to eliminate or mitigate these sensitivities. The latest released version of MELCOR, Version 1.8.2, released in April 1993, contains several new or improved models, and has corrections to mitigate numerical sensitivities [2].

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

The purpose of this paper is to report on the results of rerunning the sensitivity calculations using MELCOR version 1.8.2. Results will be presented in terms of timing of key events, thermal-hydraulic response of the system, and environmental release of radionuclides. Comparisons will also be made to the earlier calculations using MELCOR 1.8DNX, to clarify the differences due to improvements in the code. The impact of some of the new models available in MELCOR 1.8.2 will also be evaluated, such as the falling debris quench model, boundary fluid temperature option, and Oak Ridge National Laboratory's BWR bottom head (BH) model [3].

The falling debris quench model allows the debris to lose heat to surrounding water in the lower plenum as it falls to the lower head, following failure of the core support plate in each radial ring. Different values for debris fall velocity will be used for evaluation of its impact. The boundary fluid temperature option, allows the user the option to use core cell fluid temperatures calculated by the COR package dT/dz model in the heat transfer calculations for heat structures facing the core, in this case the core shroud. Earlier versions of the code only allowed the user to use bulk fluid temperature of the CVH volume between the core and the surrounding heat structure. The new option is aimed at yielding more correct heat transfer calculations. The eutectic interactions model allows for some treatment of solid dissolution and eutectics formation in the core. Radial relocation of debris allows both conglomerate and particulate debris to relocate to adjacent rings, based on leveling of hydrostatic head. These models will be activated as they are improvements to the code. The BH model, available as an option, requires preparation of special input, and carries out more detailed calculations of the thermal response of the lower plenum debris, the heatup of the reactor vessel bottom head, and the release of core and structural materials from the reactor vessel to the drywell. BH calculations begin when the BWR lower plenum is dry, and when sufficient solid debris mass has accumulated to form the foundation of a debris bed. The impact of this model will be examined using a low pressure sequence, assuming the automatic depressurization system (ADS) operational.

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PCCS Modeling for the SBWR within the CONTAIN Code[1]

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ABSTRACT

The Passive Containment Cooling System (PCCS) for the GE Simplified Boiling Water Reactor (SBWR) consists of vertical tube condensers located in a pool outside the containment safety envelope. During design basis and severe accidents these condensers transfer heat from the containment to the environment through the evaporation/boiling of pool water. Recently, the University of California at Berkeley and the Massachusetts Institute of Technology completed experimental studies to determine the local condensation heat transfer coefficients in vertical tubes with downward flowing steam/gas mixtures. Experimental correlations were developed in these studies for the local condensation coefficients. This work summarizes the recent assessment of these correlations carried out within the CONTAIN code project and discusses the PCCS modeling approach that is currently being used in the CONTAIN code.

The modeling of steam condensation in tubes with noncondensables is complicated by the rapidly changing conditions within the tube: temperature and mixture concentration in the gas/vapor core, the liquid film thickness along the cool wall, and the gas boundary layer thickness at the liquid/gas interface varies with axial distance. In this paper, a model for the local condensation heat transfer rate is formulated and compared to the recent vertical tube condensation experiments. The model assumes laminar liquid film flow with interfacial shear and a heat and mass transfer analogy for steam diffusing through a gas boundary layer. By first solving the film and gas mixture equations with steam quality as the independent variable and then integrating with respect to quality, the tube distance that corresponds to a given quality is obtained. Agreement between this model and data obtained in the recent experimental studies is very good. In general, the model is an improvement over the experimental correlations. This model is used in the PCCS modeling in the CONTAIN code. Verification of the PCCS model is demonstrated by comparing code calculations to several tube condensation experiments.

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Validation of COMMIX with Westinghouse AP–600 PCCS Test Data*

by

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Test data for the Westinghouse AP–600 Passive Containment Cooling System (PCCS) have been used to validate the COMMIX computer code.[1–4] COMMIX solves a system of time-dependent and multidimensional equations for conservation of mass (of several species), momentum, energy, turbulent kinetic energy, and dissipation of turbulent kinetic energy as a boundary–value problem in space and as an initial–value problem in time domain. To evaluate the performance of the PCCS, a transient liquid–film–tracking model[5] has been developed and implemented in the COMMIX code. This model computes liquid–film thickness, velocity, and temperature on both sides of the containment steel wall. The inside film is formed by condensation of steam on the vessel wall, and the outside film is formed by water flooding from the top of the dome. The outside film evaporates into the air stream in the air annulus that is driven by buoyancy–induced convection. The effect of a wavy liquid film on heat and mass transfer is explicitly accounted for.

A set of heat transfer models and a mass transfer model based on heat and mass transfer analogy were used for the analysis of the AP–600 PCCS. It was found that the flow of the air stream in the annulus is a highly turbulent forced convection and that the flow of the air/steam mixture in the containment vessel is a mixed convection. Accordingly, a turbulent–forced–convection heat transfer model is used on the outside of the containment wall and a mixed–convection heat transfer model is used on the inside of the containment wall. The flow in the air annulus is calculated with a turbulence k–ε model. The results from the COMMIX calculations are compared with the Westinghouse experimental data for the average wall heat flux, vessel wall temperature distribution, average temperature rise of air flow in the annulus, evaporation rate, and containment vessel pressure; agreement is good. The COMMIX calculations also provide detailed

* Work supported by the U.S. Nuclear Regulatory Commission

distributions of velocity, temperature, and steam and air concentrations. It will be shown that the air/steam mixture is stratified inside the containment vessel.

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SCDAP/RELAP5/MOD3 Code Development[a]

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ABSTRACT

The SCDAP/RELAP5 computer code is designed to describe the overall reactor coolant system (RCS) thermal-hydraulic response, core damage progression, and fission product release and transport during severe accidents. The code is being developed at the Idaho National Engineering Laboratory (INEL) under the primary sponsorship of the Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission (NRC). The development of the current version of the code, SCDAP/RELAP5/MOD3[8X], was started in the spring of 1992. This version contains a number of significant improvements since the last version of the code, SCDAP/RELAP5/MOD3[7af], was released for beta testing. These improvements include the addition of several new models to describe the earlier phases of a severe accident, changes in the late phase models to provide more "physically intuitive" behavior for full plant calculations, and changes to improve the overall reliability and usability of the code. Over the past year, SCDAP/RELAP5/MOD3[8X] has undergone an intensive effort of verification testing to identify and resolve outstanding code errors. Phase I of this effort was completed in February of 1993 and was focused upon the incorporation and testing of code usability and reliability improvements that had been specifically requested by code users. Following the completion of this phase, Phase II verification testing was started and now has been completed. The objective of the Phase II verification testing was to identify and resolve as many residual code implementation errors as practical prior to the release of the code for assessment and plant applications.

The Phase II testing used both full plant and experimental analysis test problems using decks that have been developed to support NRC plant applications and code assessment activities. The full plant models included Surry, TMI-2, and Browns Ferry. The experimental analysis test problems included LOFT FP-2, ACRR DF-4, PBF SFD 1-4, and CORA-7. These experimental analysis test decks had been developed previously to support a systematic code-to-data assessment of previous versions of SCDAP/RELAP5/MOD3. The results of selected Phase II verification test problems are described in the paper.

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BWR Control Blade/Channel Box Interaction Models for SCDAP/RELAP5 *†

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SUMMARY

The core of a boiling water reactor (BWR) consists of an array of fuel assemblies with cross-shaped control blades located between these assemblies. There is one control blade for every four fuel assemblies. A fuel assembly consists of a fuel rod bundle surrounded by a Zircaloy channel box. A control blade consists of four stainless steel blade sheaths that surround four rows of small stainless steel absorber rods filled with B_4C powder. During a severe accident, material interactions between the B_4C, stainless steel, and Zircaloy have a significant impact on the melting and subsequent relocation of the control blade and channel box structures.

Degraded-core experiments that include BWR control blade and channel box structures have been performed in-pile in the Annular Core Research Reactor at the Sandia National Laboratory (the DF-4 test), and out-of-pile in the CORA test facility at the Kernforschungszentrum Karlsruhe (KfK), Federal Republic of Germany. Posttest analyses of the DF-4, CORA-16, and CORA-17 experiments have been conducted by L. J. Ott at the Oak Ridge National Laboratory (ORNL) using analytical tools that represent the specific geometry and boundary conditions of each experiment. These experiment-specific analyses have shown that accurate modeling of the control blade and channel box structures must account for the effects of B_4C/stainless steel interactions and stainless steel/Zircaloy interactions.

The objective of the project described in this paper is to incorporate BWR control blade/channel box interaction models within the SCDAP/RELAP5 code, which is being developed primarily at the Idaho National Engineering Laboratory (INEL). The modeling approach was to convert portions of the experiment-specific models, developed by L. J. Ott at ORNL, into a new BWR control blade/channel box component for SCDAP. These models are provided within SCDAP/RELAP5 as an additional "building block" that can be selected by the user when appropriate.

Based on the experimental evidence, the following processes are modeled by SCDAP's new BWR control blade/channel box component. As the control blade and channel box structures heat up during an accident, oxidation begins on the stainless steel and Zircaloy surfaces when

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temperatures exceed ~900 K, with the reactions becoming more vigorous at higher temperatures and when more steam is available. Melting of a control blade begins at the inner surfaces of the absorber rods, where the B_4C reacts with the stainless steel. The absorber rods melt at a temperature of ~1505 K, which is lower than the melting temperature of pure stainless steel. Stainless steel from the control blade then relocates downward and forms a blockage between the control blade and the channel box, where it reacts with the Zircaloy. The Zircaloy channel box forms a eutectic mixture with the stainless steel blockage and melts at a temperature of ~1523 K, which is much lower than the melting temperature of pure Zircaloy.

Within SCDAP, the control blade and the channel box are represented by an array of radial and axial temperature nodes. The actual control blade configuration of small absorber rods inside a stainless steel blade sheath is converted into an equivalent slab geometry. The solid structures interact with two RELAP5 hydrodynamic volumes: one for the fuel bundle region inside a fuel assembly and the other for the interstitial region between the fuel assemblies. All hydrodynamic parameters used by the BWR control blade/channel box component are obtained from the RELAP5 data base.

The input data for the BWR control blade/channel box component can be specified using either the original SCDAP input format without card numbers, or the new SCDAP input format with RELAP5-style card numbers, which is a new capability that has been added recently by the SCDAP development staff at INEL. The predicted results are printed in the output file as a part of each Major Edit at time intervals specified by the user. Also, information such as the nodal temperatures and the frozen crust thicknesses can be written to the restart-plot file for use with the Nuclear Plant Analyzer (NPA) plotting and animation program. The frozen crust thickness variables can be used with the NPA program to generate animated drawings depicting the melting and downward relocation of the control blade and channel box structures.

The new BWR control blade/channel box component has been implemented and tested within several developmental versions of SCDAP/RELAP5. The operation of this component has been tested using three input decks: (1) a simple test calculation designed for fast execution, (2) a full-plant calculation for the General Electric Company's Simplified Boiling Water Reactor, and (3) a preliminary calculation for the FLHT-6 experiment planned for the Canadian National Research Universal reactor. The BWR control blade/channel box component is scheduled for release to users in the next production version of SCDAP/RELAP5.

Integrated Fuel-Coolant Interaction Code : Assessment of Stand-Alone Version 6.0 *

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In the event of a severe reactor accident leading to core melt, it is possible that molten fuel materials will come into contact with water, producing a molten fuel-coolant interaction (FCI). A computational tool, the Integrated Fuel-Coolant Integration code, version 6.0 (IFCI 6.0), has been developed and demonstrated, in a stand-alone format, as capable of assessing explosive and nonexplosive Fuel-Coolant Interactions. The IFCI code is expected to fulfill the need to have one code that would handle mixing of hot particles with coolant, particle size variations, space or volumetric influences (size and scaling) and detonation triggers.

IFCI is a best-estimate code that attempts to model all phases (mixing, triggering, propagation, and expansion) of FCIs in as mechanistic a manner as possible.[1] IFCI is a two-dimensional, cylindrical finite difference solution of the Navier-Stokes equations, for vapor liquid and melt, including models for, momentum advection, surface tracking, Weber breakup, bulk boiling and subcooled surface boiling, and triggering. Modules and references associated with reactor cores, detailed structural information, and extraneous codes linkages have been removed in order to create a stand-alone tool to study all phases of FCI progression. Code size was reduced from 93138 to 35445 lines, and compile time reduced by 20%. The basic features of IFCI and the assessment using FITS, FARO, and IET experimental cases are presented.

A generic FITS-Type Pouring mode experiment was simulated to ensure continuity between this and previous versions of IFCI. Results of that numerical experiment have been previously issued as a report.[1] The results of this case are not discussed in the context of FCI phenomenology, but only as a comparison to the previously executed version of the code. Results of a 0.5 s IFCI 6.0 simulation, executed on a Cray-YMP, match exactly with those of Young. However, results from IFCI 6.0 when executed on the SUN Sparc2 workstation are noticeably different despite the use of double precision on the workstation. The cause has not been determined, and is the subject of further study. The SUN results demonstrate considerably larger (>50%) spatial variation (from cell to cell) in pressure and temperature. Melt volume fractions differ slightly, and pressures differ by as much as 10% between the two cases. The pressures do not diverge, however, as the simulation progresses. Any difference in compilation and executable may produce slightly different timesteps and results. Unfortunately, FCI progression is highly sensitive to perturbations in the fuel/coolant interaction zone.

Several simulations of the FARO Quench Test, performed at JRC-Ispra, were run, not only to demonstrate IFCI's capabilities for quenching of molten fuel, but also to provide insight into how user input affects the results of this experimental simulation. Results, especially the

* 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 DE-AC04-76DP00789.

progression of the melt through the coolant, were sensitive to axial and radial nodalization. However, global values, such as maximum liquid temperature, and steam pressure are consistent to within 10-20% during the course of the simulation. The maximum simulated pressures at the upper plenum location are within 10% of each other, but 20% higher than corresponding instrumented experiment location. Liquid temperatures, compared at 35 cm above the vessel floor, overpredict the measured liquid temperature rise of 17°C, with the maximum being 25°C, 33% more than experiment.

Two experiments (IET-8 A&B)[2] were performed in the Surtsey facility at Sandia National Laboratories, both resulting in energetic FCIs. IFCI 6.0 qualitatively captures all FCI phases, water ejection and has been used, posttest, to track pressure response to within an order of magnitude. The maximum pressure detected by a transducer near the cavity wall is about 3.5 MPa, and has been simulated to within a few percent in one IFCI 6.0 posttest analysis.

The stand-alone version of IFCI is shown to reproduce results of previous code versions, and to provide acceptable simulations for FCI-steam explosions of reactor materials using entirely mechanistic models, except for triggering. However, the operational assessment has demonstrated some convergence difficulties, sensitivity to nodalization, and machine dependency.

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VVER-440 DOSIMETRY AND NEUTRON SPECTRUM BENCHMARK*

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Light Water Reactor (LWR) benchmark experiments performed in the United States under the Surveillance Dosimetry Improvement Program (SDIP) measured reaction rates instead of neutron flux spectrum. The VVER-440 Neutron Spectrum Benchmark performed at the Nuclear Research Institute (NRI), Rez, Czech Republic, measured the neutron flux spectrum. A combination of a proton-recoil detector (for neutron energies between 10 keV and 1.5 MeV) and a stilbene scintillator detector (for neutron energies between 1.0 and 10 MeV) was used in the NRI experiments. These measurements provided a direct verification of the neutron flux spectrum calculations by transport methods and served as a better test of the current neutron transport methodology used in the United States.

Although the fuel assemblies of the VVERs were hexagonal instead of square as in the U.S. LWRs, the different geometries did not present any difficulties. The locations of the measurements were in the surveillance position, in front of the pressure vessel, and in the cavity behind the vessel. Two measurements were made in the vessel, and we hope to make comparisons with these locations after we receive the data. Two different groups made the measurements: one from NRI; the other, from Skoda Nuclear Engineering, Ltd.

The transport cross sections were taken from the ENDF/B-VI library for iron, oxygen, and hydrogen. A calculation was also performed using the ENDF/B-V cross-section files for comparison.

The comparison of the spectrum at the surveillance position shows an excellent agreement between the calculated and measured values. The average discrepancy stays consistently at about 5% below 5.0 MeV and 8% above 5.0 MeV. At the inner surface of the pressure vessel the calculations also agree remarkably well with the measured values. The average discrepancy stays below 1% throughout the energy interval 0.4 to 5.0 MeV, and it is about 5% if the energy range is extended to cover 0.1 to 10.0 MeV. At the outer surface of the pressure vessel (cavity), the calculations, using ENDF/B-VI cross sections, overestimate the measured values in all energy groups below 5.0 MeV. In the energy from 0.1 to 10.0 MeV, the average discrepancy approaches +30%, while between 1.0 to 5.0 MeV, it reduces to about 15%. An investigation is under way to identify the source(s) of this discrepancy in the cavity.

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SHORT CRACKS IN PIPING AND PIPING WELDS

by

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Summary

This program was initiated in March 1990 and has a duration of four years. The primary objective of this program is to develop and verify analyses by using existing and new experimental data for circumferentially cracked pipes. As a result of this work LBB and in service flaw evaluation criteria are being improved.

There are eight technical tasks that involve the behavior of short circumferential cracks in stainless steel and carbon steel pipes and piping welds under quasi-static loading. The objectives of each of these tasks and significant accomplishments to date are summarized below.

Task 1 Through-Wall-Cracked (TWC) Pipes
Objective: Modify and verify analyses for short TWC pipes using existing and new data on large diameter pipes.
Accomplishments: All full-scale pipe experiments and material characterization (tensile, hardness and C(T)) tests have been completed. Revised and corrected GE/EPRI functions for TWC pipes under bending, tension and combined bending and tension loads have been developed using 3D finite element analyses. The experimental results show that the ASME Section XI Z-factors are conservative for short cracks typically used in LBB analyses.

Task 2 Surface-Cracked Pipes
Objective: Modify and verify analyses for short-surface-cracked pipes using existing and new data on large and small diameter pipes.
Accomplishments: All material characterization tests and all but one full-scale pipe experiment have been completed in this task. The existing estimation schemes to analyze surface-cracked piping (SC.TNP and SC.TKP) are being revised and a new procedure LBB.ENG2 is being developed. The data have shown that for experiments where limit-load failures were expected, the ratio of the experimental maximum loads to the Net-Section-Collapse analysis predicted loads were dependant on the pipe radius to thickness ratio. Thinner pipe failed at lower loads than predicted by the Net-Section-Collapse analysis.

Task 3 Bi-metallic Cracked Pipes
Objective: Conduct experiments and develop necessary analytical procedures to assess the behavior of TWC and SC in bi-metallic welded pipe under combined bending and tension.
Accomplishments: This task was initiated during the last fiscal year (FY '93) and to date most of the material characterization efforts have been completed. Full-scale pipe experiments and analysis efforts will begin later this year.

Task 4 Dynamic Strain Aging
Objective: Evaluate and predict the effects of crack instabilities in ferritic piping believed to occur due to dynamic strain aging.
Accomplishments: All material characterization tests have been completed in this task. A screening criterion to predict if a material is sensitive to dynamic strain aging was developed. This criteria is based on high temperature hardness tests.

Task 5 Anisotropy Effects
Objective: Assess if anisotropy in fracture toughness can cause failure stresses to be less than those calculated by current LBB analyses.
Accomplishments: Skewed tensile and C(T) tests at different orientation to the pipe axes have been completed. A screening criterion to predict anisotropy effects is currently under development.

Task 6 Crack Opening Area Evaluations
Objective: Improve the crack-opening area predictions used in LBB leak-rate analyses.
Accomplishments: The primary accomplishment in this task has been the development of an new probabilistic analysis to predict leak rates and failure probabilities of cracked piping using advanced reliability methods. As part of this effort, statistical data bases were developed for material properties, N+SSE stresses of piping systems stresses, and crack morphology parameters for leak-rate analyses. An improved leak-rate analysis was also developed.

Task 7 NRCPIPE Computer Code
Objective: Improve the existing PC code NRCPIPE to analyze through-wall cracks in piping and create NRCPIPES to analyze surface-cracked pipes.
Accomplishments: A new and improved version (1.4f) of NRCPIPE has been released and a beta version of NRCPIPES has also been prepared.

Task 8 Additional Technical Tasks
Objective: Investigate additional items as a result of the findings during this program. Several tasks including J-R curve validity limits, fusion line toughness evaluation, pipe fracture data bases have been undertaken in this task.
Accomplishments: Several of these subtasks have been initiated and will be presented on when complete in topical and semiannual reports.

IRRADIATION-ASSISTED STRESS CORROSION CRACKING OF MATERIALS FROM COMMERCIAL BWR'S: ROLE OF GRAIN-BOUNDARY MICROCHEMISTRY*

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In recent years, failures of reactor-core internal components in both BWRs and PWRs have increased after accumulation of relatively high fluence (>5 x 10^{20} $n \cdot cm^{-2}$, E >1 MeV). The general pattern of the observed failures indicates that as nuclear plants age and neutron fluence accumulates, various nonsensitized austenitic stainless steels (SSs) become susceptible to intergranular failure. Some components (e.g., BWR control blade handle and sheath) are known to have cracked under minimal applied stress. Although most failed components can be replaced at a considerable economic penalty, some safety-significant structural components, such as the BWR top guide, shroud, and core plate, would be very difficult or impractical to replace. Therefore, the structural integrity of these components after accumulation of high fluence has become a subject of concern, and extensive research has been conducted to provide an understanding of this type of degradation, which is commonly known as irradiation-assisted stress corrosion cracking (IASCC).

Most of the safety-significant structural components in present BWRs and PWRs have been fabricated from solution-annealed austenitic SSs, primarily commercial-purity Type 304 SS and, to lesser extent, Type 316 SS. Component chemical composition, fabrication procedures, and reactor operational parameters, such as neutron flux, fluence, temperature, water chemistry, residual stress, and mechanical loads, have been reported to influence susceptibility to IASCC. However, results from research at several laboratories on materials irradiated under a wide variety of in-reactor and simulated conditions are often inconsistent and conflicting as to the influence of these parameters. Even for materials that have similar chemical compositions and similar irradiation histories, it is quite common to observe strong heat-to-heat variations in IASCC susceptibility.

Intergranular failures of austenitic SS after accumulation of high fluence have been attributed in the past without convincing evidence to radiation-induced segregation (RIS) or depletion at grain boundaries of elements such as Si, P, S, and Cr. However, the exact identities of the elements that segregate and the extent to which RIS contributes to the susceptibility of the core-internal components of LWRs to IASCC have been never resolved. This is particularly true for Type 304 SS, from which the majority of the safety-significant in-core components have been fabricated. Since the late 1960s, when failure of stainless steel fuel cladding was observed frequently, RIS of Si and P has been believed to be the primary metallurgical process for IASCC. Strong support for this belief appeared in the 1980s, when limited experience showed that a high-purity (HP) heat of Type 348 SS performed better than a commercial-purity (CP) heat (containing relatively high C, Si, P, S, and N). The better performance was thought to indicate that RIS of Si is the predominant process, because RIS of other impurities and Cr depletion in the HP and CP Type 348 SS were either similar or insignificant, according to analysis by field-emission-gun scanning transmission electron microscopy. It has also been reported that, after irradiation in a test reactor, Type 316L SS exhibited somewhat better resistance to stress corrosion cracking than Type 316NG SS, indicating a possible benefit of low N.

On the basis of experiences with Type 348 and 316 SS, HP Type 304 SS has been suggested as a better alternative to CP Type 304 SS. From subsequent demonstration, at least in a BWR, there has been an indication of better resistance of neutron absorber tubes

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(in the uncreviced environment) fabricated from an HP heat of Type 304 SS than those fabricated from CP heats. However, results obtained recently from other BWR swelling-tube tests and laboratory constant-extension-rate tensile (CERT) tests, conducted on different HP heats of Type 304 SS irradiated in test reactors or accelerators, indicate either an opposite or inconclusive trend. Therefore, it seems that the superior performance of HP Type 304 SS, and to a larger extent, the role of grain-boundary chemistry (e.g., concentration of Si, P, S, N, and Cr) on susceptibility to IASCC has by no means become well established.

In view of this background, and considering the strong influence of irradiation conditions and possible role of transmutation of elements by thermal neutrons, results obtained from specimens irradiated in test reactors and accelerators must be considered as tentative. Thus, a benchmark study on a larger number of well-characterized HP and CP materials from actual reactor components was considered necessary. For this purpose, Type 304 SS specimens were obtained from BWR components fabricated from three HP and two CP heats and irradiated up to 2.6×10^{21} $n \cdot cm^{-2}$ (E >1 MeV) in 4 BWRs, and CERT tests were conducted in simulated BWR water. Susceptibility to intergranular failure was also determined on the BWR specimens that were precharged with hydrogen and subsequently fractured in high vacuum. Grain-boundary microchemistry of the test specimens was analyzed by Auger electron spectroscopy (AES), and the results were correlated with the results of CERT and inert-gas fracture tests to determine the role of grain-boundary segregation and depletion of impurity and alloying elements on IASCC susceptibility.

No IG fracture could be produced in the irradiated materials in either vacuum or air without hydrogen-charging, indicating that any effect of helium-induced grain boundary weakening is negligible. Specimens tested in simulated BWR water exhibited significant heat-to-heat variation in their susceptibility to intergranular SCC (IGSCC) under otherwise similar irradiation and test conditions. For example, for a fluence of $\approx 2 \times 10^{21}$ $n \cdot cm^{-2}$, percent IGSCC observed on the fracture surface of a CP control blade sheath was as low as ≈3% and that of an HP neutron absorber was as high as ≈60%. Whereas the RIS of Si and P was pronounced in all CP heats at high fluence, RIS of the impurities was negligible for all fluence levels in all HP heats in which Si and P contents were low. No evidence of grain-boundary segregation of S was observed. Contrary to the previous belief, susceptibility to intergranular fracture in the specimens tested in simulated BWR water could not be correlated with grain-boundary concentrations of any of the impurities such as Si, P, Ni, C, or S. However, good correlation was obtained with grain-boundary Cr concentration that was measured by the AES depth-profiling technique. Cr depletion in the BWR components was significant in a zone as narrow as 2-3 nm near grain boundaries.

IG susceptibility of specimens charged with hydrogen and fractured in vacuum could be correlated well with grain-boundary concentration of an element that gives rise to a characteristic Auger peak at 58 eV. The characteristic peak, which is believed to be a combination of the secondary peaks of Ni and Li, was observed only in neutron-irradiated specimens or unirradiated specimens corroded in Li. This indicates weakening of grain boundaries by a complex synergism between hydrogen and transmutation-induced elements. That is, lithium atoms, produced from transmutation of boron and subsequently segregated on grain boundaries, interact with hydrogen that is produced on grain boundaries from transmutation of nitrogen segregated on grain boundaries. In support of this, evidence of grain-boundary segregation of nitrogen wac observed in neutron absorber tubes fabricated from HP heats that exhibited relatively higher susceptibility to IGSCC. In summary, results of this investigation show that performance of HP and CP heats of Type 304 SS, and most likely Type 316 and 348 SS as well, cannot be predicted on the basis of impurity levels such as Si, P, S, and C. Hence, the superior resistance to IASCC reported for some HP heats of Type 304, 316, and 348 SS appears to be due not to the low level of these impurities but to some other factor(s) not identified yet.

Fatigue of Carbon and Low-Alloy Steels in LWR Environments*

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Plain carbon and low-alloy steels are used extensively in pressurized water reactors (PWRs) and boiling water reactors (BWRs) as piping and pressure vessel materials. The steels of interest include A106-Gr B and A333-Gr 6 for seamless pipe and A302-Gr B, A508-2, and A533-Gr B plate for pressure vessels. ASME Code Section III (Division 1, Subsection NB) includes rules for construction of nuclear power plant Class 1 components. It recognizes fatigue as a possible mode of failure in piping materials and pressure vessel steels. Appendix I to Section III specifies fatigue design curves for applicable structural materials. The current Code fatigue design curves are based primarily on strain-controlled fatigue tests of small polished specimens in air at room temperature. To obtain the Code fatigue design curves, best-fit curves to the experimental data were decreased by a factor of 2 on stress or a factor of 20 on cycles, whichever was more conservative at each point. The factors were intended to account for uncertainties in translating the experimental data of laboratory test specimens to actual reactor components. The factor of 20 on cycles is the product of three subfactors: 2.0 for scatter of data (minimum to mean), 2.5 for size effects, and 4.0 for surface finish, atmosphere, etc. "Atmosphere" was intended to reflect the effects of an industrial environment rather than the controlled environment of a laboratory. The effects of the coolant environment are not explicitly addressed in the Code design curves.

Recent fatigue strain vs. life (S/N) data illustrate potentially significant effects of light water reactor (LWR) environments on fatigue resistance of carbon and low-alloy steels. In some cases, failures were observed below the ASME Code fatigue design curve. These results raise the issue of whether the fatigue design curves in Section III are appropriate for the purposes intended and whether they adequately account for environmental effects on fatigue behavior. The factors of 2 and 20 applied to the mean-data curve may not be as conservative as originally intended.

Available fatigue S/N data indicate that the magnitude of the decrease in fatigue life depends on alloy composition, temperature, strain rate, and concentration of dissolved oxygen (DO) in water. At the very low DO levels characteristic of PWRs and BWRs with hydrogen/water chemistry, environmental effects on fatigue life are modest at all temperatures and strain rates. Fatigue life decreases rapidly as DO increases over a rather narrow range of ≈0.1–0.3 ppm, but further increases up to 8 ppm cause only a modest

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decrease in life. In oxygenated water, fatigue life strongly depends on temperature and strain rate. At a given strain rate, fatigue life increases by a factor of 5 or more as the temperature is decreased from 288 to 200°C. For the same environment and strain range, fatigue lives can be decreased by a factor of ≈50 by reducing the strain rate from 0.1 to 0.0001%/s. Based on existing S/N data, Argonne National Laboratory (ANL) has developed interim fatigue design curves that account for the effects of temperature, DO level in the water, sulfur level in the steel, and strain rate.

The S/N data on carbon and low-alloy steels in water, however, are somewhat limited and do not cover the range of loading conditions found in actual reactor operation. For example, virtually all data in LWR water are at relatively high strain ranges. The data also do not extend over the range of strain rates normally encountered in service, e.g., some transients may have strain rates as low as 0.00001%/s. Extrapolation of available data to such low values would predict a reduction in fatigue life by a factor of >300. The relatively good service experience of carbon steel piping in BWRs suggests that the effect of strain rate on fatigue life must saturate at some level, although no such saturation has been observed experimentally. Limited data indicate that environmental effects on fatigue life are greater for carbon steel than for low-alloy steel. However, most low-alloy steels that have been investigated contain <0.007 wt.% sulfur. It is likely that differences between carbon and low-alloy steels are caused by the sulfur content of the steels.

The objectives of this program are to (1) conduct fatigue tests on carbon and low-alloy steels under conditions where information is lacking in the existing S/N data base, (2) establish the effects of material and loading variables on environmentally assisted reduction in fatigue life, and (3) validate and update the proposed interim fatigue design curves. Fatigue tests are being conducted on A106-Gr B carbon steel and A533-Gr B low-alloy steel in water and in air at 288°C.

For both carbon and low-alloy steels, environmental effects are modest in PWR water at all strain rates. Fatigue data in oxygenated water confirm the the strong dependence of fatigue life on DO and strain rate. The effect of strain rate on fatigue life saturates at some low value, e.g., at ≈0.0004%/s for A106-Gr B carbon steel in water containing ≈0.8 ppm DO. The data suggest that the saturation value of strain rate may vary with DO and alloy composition. Degradation of fatigue life of A106-Gr B carbon steel and A533-Gr B low-alloy steel with comparable sulfur levels is similar. Although the cyclic stress-strain and cyclic-hardening behavior of these steels is distinctly different, the reduction in fatigue life of the two steels is comparable or somewhat greater for the low-alloy steel. Carbon steel also exhibits pronounced dynamic strain aging, whereas strain aging effects are modest in the low-alloy steel. Environmental effects on fatigue crack initiation of carbon and low-alloy steel are also being investigated.

RELIABILITY OF NDE - CAST STAINLESS STEEL, SAFT-UT PERFORMANCE, PISC III PROGRAM STATUS AND EVALUATION OF COMPUTER-BASED UT/ISI SYSTEMS[1]

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This paper reports on the progress associated with three different NRC-funded programs. The topics cast stainless steel and PISC III program status are activities being conducted in the program entitled "Evaluation and Improvement in Nondestructive Examination Reliability for Inservice Inspection of Light Water Reactors (LWRs)" - FIN No. B2289, the work on SAFT-UT performance is part of the program entitled "Field Validation, Acceptance, and Training of Advanced NDE Technology" - FIN No. B2913, and the final activity is part of the program entitled "Evaluation of Computer-Based NDE Techniques and Regional Support of Inspection Activities" - FIN No. L1100.

This year, the cast stainless steel (CSS) work concentrated on determining the effectiveness and reliability of ultrasonic inspection techniques for LWR components containing CSS material. Low frequency ultrasonic techniques inherently insensitive to microstructural effects are being investigated. Development of specialized search units, employing advanced signal processing techniques, as well as utilization of a suitable combination of frequency, wave mode, and incident angles are all important for the optimization of techniques to inspect CSS materials. The low-frequency UT has been coupled with the synthetic aperture focusing technique (SAFT) to improve the resolution and signal to noise in the data. This combined SAFT and low-frequency technique was evaluated in a blind test as part of an evaluation being conducted by the EPRI NDE Center. This combination proved to be among the top performers of the techniques evaluated. Finally, fracture mechanics calculations have been performed to estimate "critical" crack dimensions in CSS. The results of these calculations have been used to support a basis for the design of performance demonstration CSS test sets for an ASME Code Section XI Appendix VIII Supplement.

As part of the program to develop the SAFT-UT technology, the technology is to be validated. A portion of the validation process includes blind testing of the technology by participation in the Programme for the Inspection of Steel Components, Phase 3 (PISC III) Action 2 concerning full scale vessel (FSV) testing. Several years ago the SAFT-UT system was taken to Stuttgart, Germany to perform inspections as part of the first phase of the FSV to assess the sizing accuracy of techniques and procedures. Following destructive examination of the areas to quantify the defects that had been placed there, data analysis was performed and a report prepared for review and comment. It is not possible to give the results in comparison with the other FSV participants

[1]Work supported by the U.S. Nuclear Regulatory Commission ; Dr. J. Muscara, NRC Program Manager. Pacific Northwest Laboratory is operated for the U.S. Department of Energy by Battelle Memorial Institute under Contract DE-AC06-76RLO 1830.

until the PISC III program officially releases them. However, it can be stated that the SAFT-UT technology performed extremely well and basically met our expectations. The SAFT-UT performance in comparison with theoretical estimates will be the focus of the presentation.

The PISC III program has eight different actions that involve various aspects of NDE: Action 1 concerns inspections on real contaminated structures (RCS); Action 2, the full scale vessel tests (FSV); Action 3, nozzles and dissimilar metal welds (NDW); Action 4, austenitic steel tests (AST); Action 5, steam generator tubing tests (SGT); Action 6, ultrasonic modelling; Action 7, human factors; and Action 8, codes and standards. The PISC III program began in 1986 and will be completed at the end of the calendar year 1993. All of the actions are scheduled to be completed at that time. This involves completing all round robin tests this year, performing destructive examinations, data analysis of the inspection results, and preparing final reports. This is very ambitious and will be a very busy time for the personnel performing this work. The PISC III program will have a final symposium during March 1994 which is planned to be held in Petten, Holland. This paper will present an overview of the various actions, state the objective of each action, work performed to date, and the status of the remaining work. It should be noted that this is a very large program that is and will have a significant impact on the NDE inspections being performed world wide.

One of the objectives of FIN L1100 is to evaluate the reliability and accuracy of interpretation of results from computer-based ultrasonic inservice inspection systems. In order to conduct this work, a background document was prepared that contains a description of what will be evaluated and the method to be used for the evaluation. This first volume is a NUREG/CR report entitled <u>Auditing Computer-Based Ultrasonic Inservice Inspection Systems</u> and it has gone through one full review cycle. A revised version is being reviewed by the NRC. This report describes the general functions of computer-based ultrasonic equipment, provides a review of selected systems, and presents the evaluation procedures to be used with specified systems. Systems that are to be evaluated will be reported as appendices to this report. The evaluation procedure has been performed on the P-Scan and Intraspect I-98 systems, a presentation of preliminary P-Scan results was made to the NRC Technical Advisory Group on NDE, and the Appendix for the P-Scan system has been prepared and submitted to the NRC for review.

The most interesting observations to date relate to the differences in the objectives of the two evaluated systems, and consequently in the gathering, analysis, and presentation of data they perform. P-Scan is a dedicated scanning and imaging system for pipes. The images produced are specialized to the evaluation of weld defects and corrosion. The Intraspect is a general-purpose system, provided in many customized variants depending on the application. It provides more sophisticated visualization of volumetric positions of reflectors. P-Scan normally acquires only imaging data, and by exception, a limited amount of RF data. Intraspect normally acquires full RF data, which is then analyzed for image presentation. Both systems performed all functions correctly. Analysis of additional systems may lead to a classification of system types according to scanning, acquisition, and analysis capabilities.

PROGRESS ON RISK-BASED INSPECTION GUIDELINES: APPLICATION OF SURRY-1 PILOT STUDY TO IMPROVED INSERVICE INSPECTION PLANS[1]

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Pressure vessels and piping at nuclear power plants are inspected for structural integrity on a routine basis following the rules of the ASME Section XI Code. The code prescribes all aspects of the inspection program including the method, timing and acceptance criteria for detected flaws. ASME Section XI rules for these inspections are based on engineering judgement, prior experience, and in large measure, on implicit considerations of risk. This paper describes efforts to develop improved inspection plans by explicitly quantifying the factors, including inservice inspections, that can increase or decrease the risk due to failures of pressure boundary components. Results of ongoing research programs are now being applied to make recommendations to ASME Section XI for improved inspection plans. These risk-based inspection programs have the objectives of: 1) establishing a level of inspection such as to maintain the contribution from structural integrity failures to be a small fraction of overall plant safety risks, 2) allocating limited inspection resources to those components with the greatest contributions to risk, and 3) ensuring that all prescribed inspections are capable of detecting degradation on a timely basis and provide a meaningful reduction in structural failure probabilities.

This paper describes risk-based studies performed at the Pacific Northwest Laboratory for NRC as part of the research program entitled "Evaluation and Improvement in Nondestructive Examination Reliability for Inservice Inspection of Light Water Reactors (LWRs). Plant-specific pilot calculations have focused on the Surry Unit-1 Nuclear Power Station in cooperation with the Virginia Electric Power Company. Prior efforts which ranked individual pipe segments on the basis of their contributions to core damage have been extended to address components of all the important systems of the Surry-1 plant. Attention this past year was directed to the high-pressure injection system, the residual heat removal system and the power conversion system. Planning is underway for a second pilot study, similar to that performed for Surry-1, but for a typical boiling water reactor plant. Cooperative arrangements are now being formalized with the utility operator of the plant.

Components within the individual pipe segments for the low pressure injection (LPI) system at Surry-1 have been ranked at the same high level of detail

[1]Work supported by the U.S. Nuclear Regulatory Commission ; Dr. J. Muscara, NRC Program Manager, FIN B2289. Pacific Northwest Laboratory is operated for the U.S. Department of Energy by Battelle Memorial Institute under Contract DE-AC06-76RLO 1830.

(welds, fittings, etc.) as currently addressed by ASME Section XI requirements. In identifying inspection priorities for the LPI system, it was found that less than 1% of the 644 total number of structural elements contributed over 50% of the estimated risk from the LPI system due to structural integrity failures. The dominant contribution was from potential thermal fatigue failures at the check valves separating hot and cold fluids at the interface between the LPI and primary coolant system.

Having selected high risk locations (based on both the consequences and probabilities of failure) as the focus for improved inspection programs, the effects of alternative inspection strategies have been quantified using structural reliability models. These models have used the PRAISE computer code to simulate ultrasonic inspections performed at different time intervals over the service life of the component. Each location was assumed to have a statistical population of initial fabrication flaws ranging from frequently occurring flaws of very small size to unlikely flaws of very large size. The calculations addressed ultrasonic detection reliabilities that covered the expected range for inspection teams that could pass the performance demonstration requirements prescribed by Appendix VIII of ASME Section XI. The more effective candidate inspection strategies were found to reduce the frequencies of structural failures by a factor of ten for the critical components of interest.

The studies at PNL have been carried on in close cooperation with efforts by the ASME Research Task Force on Risk-Based Inspection Guidelines, and with a newly formed Working Group of ASME Section XI on Implementation of Risk-Based Inspection. Complementary studies are being performed by Westinghouse Electric to quantify the economic aspects of the candidate inspection strategies. Strategies are first required to provide acceptable levels of safety. The methods of decision risk analyses have then addressed the costs associated with inspections and also the economic benefits of inspections derived from increased plant reliability. Estimated costs associated with each inspection strategy have included the costs of not only the inspections themselves but also of costs for repairs and for replacement power during unscheduled outages due to preventable failures.

Results and recommendations from risk-based studies by PNL and other contributing organizations will be published in a future document to be prepared by the ASME Research Task Force on Risk-Based Inspection Guidelines. While this document is being prepared, ASME Section XI has been briefed on progress and as a result, valuable feedback has been received to provide direction to the research activities. The Section XI Working Group has sought a code approach to maximize the benefits of risk-based inspection in a manner that has the minimum impact on the current format of the code. To this end, a special workshop meeting was held in June 1993 to develop trial revisions to some code tables that would bring the revised code into better agreement with findings of the risk-based studies. In addition, the Working Group has discussed a much longer term code approach to more fully gain the benefits of risk-based inspection. This approach would be more plant specific in nature and would involve detailed quantitative risk evaluations for each plant.

EXPERIMENTS IN A SCALED LOOP

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INTRODUCTION

Under loss of forced circulation, coupled with the loss or reduction in primary side coolant inventory, horizontal stratified flows can develop in the hot and cold legs of Pressurized Water Reactors (PWRs). Vapor produced in the reactor vessel is transported through the hot leg to the steam generator tubes where it condenses and flows back to the reactor vessel. Within the steam generator tubes, the flow regimes may range from counter-current annular flow to single phase convection. As a result, a number of heat transfer mechanisms are possible depending on the loop configuration, total heat transfer rate and the steam flow rate within the tubes. These include (but are not limited to), two-phase natural circulation where the condensate flows co-current to the vapor stream and is transported to the cold leg such that the entire reactor coolant loop is active, and reflux cooling where the condensate flows back down the interior of the coolant tubes counter-current to the vapor stream and is returned to the reactor vessel through the hot leg. While operating in the reflux cooling mode, the cold leg can effectively be inactive. Heat transfer can be further influenced by noncondensables in the vapor stream which accumulate within the upper regions of the steam generator tube bundle. In addition to reducing the steam generator's effective heat transfer area, under these conditions operation under natural circulation may not be possible and reflux cooling may be the only viable heat transfer mechanism. The SPWR Facility in the Nuclear Engineering Department at North Carolina State University is being used to study the effectiveness of two phase natural circulation and reflux cooling under conditions associated with loss of forced circulation, mid-loop coolant levels and noncondensables in the primary coolant system.

FACILITY DESCRIPTION

The NCSU Scaled PWR Facility (SPWRF) is a Freon based, 1/9 scale model of a two-loop Westinghouse Pressurized Water Reactor. The reactor core is simulated by electrically heated rods with heater power governed by either a point reactor kinetics model, or set manually at an operator designated power level. The reactor kinetics model is coupled through the system's instrumentation such that reactivity feedback effects (Doppler, moderator temperature, etc.) control the reactor's dynamic response. Both primary and secondary sides are represented including such normal balance of plant components as condensers, condensate and feed pumps, and feedwater heaters.

REFLUX COOLING EXPERIMENTS

A series of reflux cooling experiments have been run on the SPWRF to measure steady state heat transfer rates as a function of primary and secondary side pressure. In the first series of experiments, steam generator level was maintained such that the tube bundle region was completely flooded and the steam generator operated in its normal recirculation mode. The primary side of the SPWRF was drained to mid-loop coolant levels from nominal operating temperatures and pressures, and stabilized prior to initiation of significant secondary side steaming. Counter-current, horizontal stratified flow in the hot leg and stagnant conditions in the crossover leg were verified visually through the glass viewing windows located in these areas.

The reactor kinetics model receives reactivity inputs from three sources: (1) simulated control rod motion, (2) moderator temperature and (3) core power. The SPWRF will "load follow" in a manner similar to an actual power plant, based upon the average loop temperature as measured by the hot and cold leg RTDs. For steady state studies, the reactor power is controlled indirectly through the steaming rate and the corresponding steam generator pressure by manually opening and closing the main steam throttle valve. The kinetics model includes a preprogrammed rod worth curve which gives reactivity as a function of rod position. Primary side pressure and temperature are determined by control rod position.

Additional experiments have been run at relatively low steaming rates in the absence of feed flow. Steam generator level was allowed to decrease continuously while reactor power and primary and secondary side pressures went to steady-state. The steam generator was then refilled prior to changing the steaming rate. At no point did the steam generator level approach the minimum level necessary for recirculation to occur. Variability in the primary side pressure is much more evident in these results, illustrating the importance of feed flow rate and feed temperature on the heat transfer rate.

CORE TO SURGE-LINE ENERGY TRANSPORT IN A SEVERE ACCIDENT SCENARIO

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The thermal transient of a B&W PWR nuclear plant, undergoing core uncovery and clad oxidation with hydrogen production, is relevant to the assessment of potential failures in the upper structures and their relation to the reactor vessel lower head failure. The timing and nature of these failures are the decisive parameters determining the direct containment heating scenarios. For the plant considered, the hot legs and cold legs do not constitute complete flow loops since both the once-through steam generators are completely inactive (i.e. the feed water is unavailable) and their primary side is full of water up to the cold legs loop seals elevation.

The detailed scaling presented in the paper focusses on the hot legs. These components are crucial because it is anticipated that a possible failure point is at the weld between the pressurizer surge line and the hot leg nozzle which is at a significant distance from the reactor vessel. A scaling methodology for the transient energy transport in the primary system of a B&W PWR during a severe accident scenario is applied to experimental data recently obtained at the UMCP facility.

The UMCP experiments are conducted with SF_6 to simulate a high pressure, high temperature steam-hydrogen mixture. A nominal fluid temperature ramp of 1.0 °C/s is postulated from available TMI-2 event data. This rate is bracketed by temperature ramps of 0.5 °C/s and 2.0 °C/s to cover the range of potential prototypical transient scenarios. The ratio of prototypical-to-test-facility temperature ramps is used in transposing the measured heat-up rates to prototypical conditions at the surge-line nozzle location. Multiple tests performed at different operational conditions lead to consistent conclusions regarding the surge line failure time. This lends confidence to the scaling procedure.

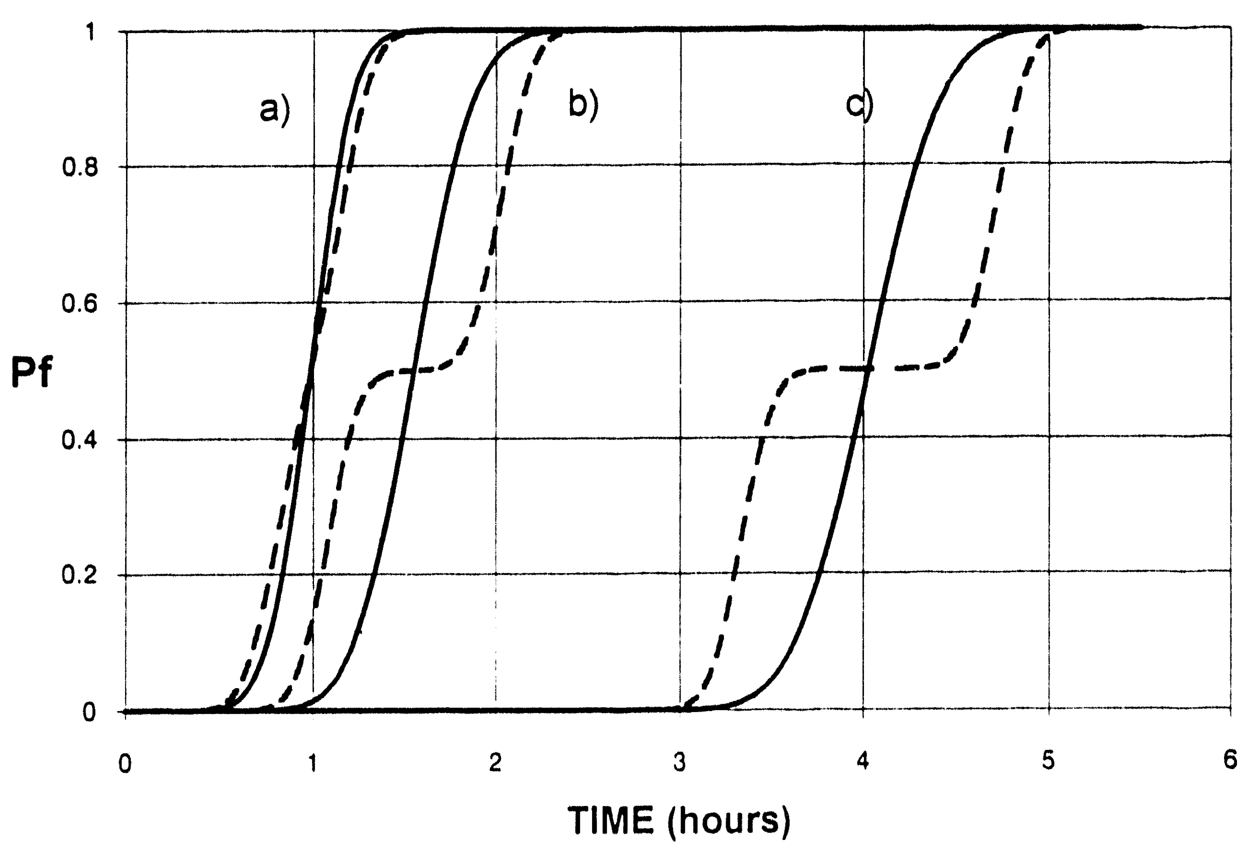
The experimental data obtained from the UMCP facility show a bi-furcation at about 500 seconds into the transient regardless of the power input. This bi-furcation is random in nature and results in one of the two hot legs to receive about twice the heat input of the other. These results are repeatable and are consistent over 14 tests performed at powers ranging from 17 to 57 kW.

The temperature in the metal at the hot leg surge-line connection exhibits this bi-modal behavior and the data extrapolated at the prototypical scale reflect this occurrence. Therefore, a probability failure at the hot leg surge-line nozzle is characterized for a range of possible prototypical fluid temperature ramps. In any case, the TMI-2 transient duration is less than the lowest possible failure time.

The bi-furcation observed in the UMCP facility is currently being evaluated in terms of its applicability to the prototype. The UMCP experiments are conducted at constant pressure and a venting system is necessary to achieve these conditions. The effect of the geometrical configuration of the venting lines is investigated as a possible cause of the bi-furcation. The venting lines configuration may originate two conditions: a) a pressure equalizing conditions between the two hot legs and b) a flow balancing condition between the two legs. The first option could be observed in the prototype while the second is a-typical. At the present time a definite conclusion has not been reach yet.

In spite of these uncertainties, a non-bifurcating behavior for a different set of boundary conditions has been observed and compared with similar bi-furcating behavior. The energy transport into both the hot legs confirms that the non-bi-furcating behavior is bounded by the two branches of the bi-furcation. Therefore, the results at the prototypical scale can be considered bounding if the mechanism originating the bi-furcation is found to be a-typical.

The figure illustrate the prototypical failure probability at the hot leg surge-line nozzle versus time for fluid temperature ramps of: a) 2.0 °C/s; b) 1.0 °C/s; and c) 0.5 °C/s. The dashed lines represent a bi-modal distribution of the failure probability while the solid line refers to the uni-modal distribution with similar associated uncertainties.



Assessment of the Potential for HPME during a Station Blackout in the Surry and Zion PWRs[a]

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The integrity of pressurized water reactor (PWR) containment structures could be challenged by direct heating associated with a high pressure melt ejection (HPME) of core materials following reactor vessel breach during certain severe accidents. Intentional reactor coolant system (RCS) depressurization, where plant operators latch pressurizer power-operated relief valves (PORVs) open, has been proposed as an accident management strategy to reduce the radiological risks associated with potential containment failures by mitigating or preventing the severity of the HPME. However, decay heat levels, valve capacities, and other plant-specific characteristics determine whether the required operator action will lead to effective RCS depressurization.

Without operator action, in-vessel, full loop, and hot leg countercurrent natural circulation flows could develop and redistribute core decay heat during severe reactor accidents. Those natural circulation flows could heat ex-vessel RCS pressure boundaries (surge line and hot leg piping, steam generator tubes, etc.) to the point of failure before vessel breach, providing an alternate mechanism for RCS depressurization and HPME mitigation. SCDAP/RELAP5/MOD3 calculations have been performed at the Idaho National Engineering Laboratory (INEL) to evaluate the potentials for RCS depressurization and HPME during a station blackout without operator action in the Surry and Zion PWRs.

The station blackout scenario was selected because it is expected to cover the possible range of RCS responses during potential HPME sequences. The specific station blackout scenario considered was a TMLB' sequence, which was initiated by the loss of all AC power and a simultaneous loss of auxiliary feedwater. The possibility of accident recovery was not considered. Code calcu-

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lations from accident initiation through the time of lower head failure were performed with and without hot leg countercurrent natural circulation, with and without reactor coolant pump (RCP) seal leakage, and with variations on some of the more important core damage progression parameters. Best estimate parameters were used as inputs where there is data or where the effects of the parameters are well characterized. For core damage progression parameters with a high degree of uncertainty, values were selected to minimize the time to lower head failure producing a conservative assessment of the potential for HPME. Throughout this assessment, it was assumed that HPME would not occur if the RCS pressure could be reduced to 1.38 MPa or less before failure of the lower head.

SCDAP/RELAP5/MOD3 results indicate that the potential for HPME during a station blackout without operator action in the Surry PWR is low. The RCS pressure is controlled by continuous PORV cycling in sequences that progress at full system pressures (without RCS leaks). Natural circulation of steam and steam flow through the PORVs leads to surge line and hot leg failures before failure of the lower head. RCS depressurization through the surge line and/or hot leg breach will depressurize the RCS and eliminate the possibility for HPME in that case. If the RCS is depressurized below the PORV set point by RCP seal leaks, PORV cycling terminates and a reduction in surge line heating follows. However, ex-vessel heating by hot leg countercurrent natural circulation continues. If seal leaks do not exceed 250 gpm per RCP (the most probable leak rate), hot leg failures that depressurize the RCS and prevent HPME will occur before lower head failure. If seal leaks are as high as 480 gpm per RCP (the maximum leak rate in Westinghouse RCPs), the energy dissipated through the leaks reduces the decay energy available for ex-vessel heating. As a result, ex-vessel failures do not occur. However, leak rates as high as 480 gpm per RCP are unlikely, and if they developed, there is a reasonable likelihood that HPME will not occur since the RCS pressures could be sufficiently reduced by seal leakage before lower head failure.

SCDAP/RELAP5/MOD3 results indicate that the potential for HPME during a station blackout without operator action in the Zion PWR is low for sequences that progress at full system pressures (without RCS leaks). Like Surry, natural circulation of steam and steam flow through the PORVs leads to surge line and hot leg failures with RCS depressurization before failure of the lower head. A potential for HPME exists, however, for sequences that progress at reduced pressures as a result of RCP seal leaks. Ex-vessel failures do not occur before failure of the lower head in those cases. The most important factors that contribute to that result include the bypass configuration and the decay power density of the Zion PWR. Those factors lead to core blockages that are not readily cooled by accumulators, which contributes to relatively early core melting and relocation into the lower head. RCS pressures at the time of lower head failures in sequences that progress at reduced pressures are expected to be between 1.38 and 6.89 MPa, providing the conditions that could lead to a potential for HPME.

PEER REVIEW OF RELAP5/MOD3 DOCUMENTATION

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A peer review was performed on a portion of the documentation of the RELAP5/MOD3 computer code. The review was performed in two phases. The first phase was a review of Volume III, *Developmental Assessment Problems*, and Volume IV, *Models and Correlations*. The reviewers for this phase were Dr. Peter Griffith, Dr. Yassin Hassan, Dr. Gerald S. Lellouche, Dr. Marino di Marzo and Mr. Mark Wendel. The second phase was a review of Volume VI, *Quality Assurance of Numerical Techniques in RELAP5/MOD3*. The reviewers for the second phase were Mr. Mark Wendel and Dr. Paul T. Williams. Both phases used the NRC's "Charter for Evaluation of RES Code Documentation" as a guide for the reviews. Some additional review criteria for Volume VI were included that addressed adequacy of the documentation of the numerical techniques.

While not unanimous in this regard, most of the reviewers felt that Volume III was well written and organized. However, the document has several significant deficiencies when compared to the criteria for acceptance defined in NUREG-1230 for documentation to be used to support the code scaling, applicability and uncertainty (CSAU) evaluation process. Modifications in several key areas would be required before the document could meet those criteria. A summary of the reviewers' major recommendations is provided below:

1. All code assessment activities should be performed with a frozen version of the code.
2. A validation plan should be completed. This plan would set forth the logical framework for testing the code. This would lead to a comprehensive set of assessment cases which would demonstrate comprehensive adequacy.
3. Where code results do not match experimental data, more discussion should be offered that details the reasons for the discrepancy. Identified code deficiencies should be evaluated and their impact on the code results assessed.
4. The description of code limitations should be expanded and scaling effects should be addressed.
5. Whenever code features are disabled, the impact on accuracy and code applicability should be discussed.
6. Guidelines for users for performing similar analyses should be included in the report, particularly where difficulties are encountered with code models.

The reviewers' reactions to Volume IV varied from strongly positive (Griffith) to rather negative (Lellouche). The majority felt that the description of what was in the code was fairly clear and understandable, though there is room for improvement. Certainly correction of numerous typographical errors is needed. There were definite differences in the reviewers' reactions to limitations in the description of the applicability and justification of the codes' models and correlations, some judging these to be clear deficiencies in the documentation and others more inclined to attribute them to limitations in the code itself or in our knowledge of the physical phenomena. A summary of the reviewers' major recommendations is provided below:

1. Adopt a consistent set of symbols and nomenclature throughout the volume.
2. Provide additional supporting references, justification and explanation for flow regime maps, for applications of correlations and models beyond their original data bases and for modifications made in implementing correlations and models.
3. Provide an explanation for the limits placed on variables and coefficients, particularly in Chapter 4, Section 1.
4. Enhance the readability of Chapters 6 and 7, either by better defining the FORTRAN used or by adopting an alternate presentation strategy.

The major conclusions reached in the review of Volume VI are:

1. Generally speaking, while all criteria are addressed, specific areas require revision and elaboration to meet documentation requirements.
2. Specifically, Chapters 4 and 5 do not meet the requirement of being "sufficiently detailed," and there is insufficient linkage between the theoretical studies presented in Chapter 4 and the computational experiments presented in Chapter 5.
3. Although Volume VI is organized in a logical fashion, significant problems exist with regard to readability due to awkward sentence structure, grammatical and typographical errors, and nomenclature inconsistency.
4. Consideration should be given to retitling the volume or including sections to address the formal requirements of quality assurance.

Following from these conclusions, the set of recommendations summarized below was identified:

1. Include more detailed information in Chapters 4 and 5; specifically, (i) address two theoretic issues when applying Lax's Equivalence Theorem to algorithms for two-phase flow, (ii) provide a linkage between Chapters 4 and 5, and (iii) include geometry, and boundary and initial conditions (or at least a brief summary and appropriate reference) for the computational experiments in Chapter 5.
2. Adopt a consistent nomenclature throughout the volume.
3. Enhance the readability of the volume by correcting numerous grammatical and typographical errors and revising awkward sentence structure.

BENCHMARK ANALYSES WITH RELAP5 FOR USNRC SIMULATORS

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The U.S. Nuclear Regulatory Commission adopted Kemeny Commission recommendations that all nuclear plants have a plant-specific simulator for operator training. In support of this requirement a project was initiated to examine the capabilities of the current generation of simulators using advanced thermal hydraulic systems codes such as RELAP5 and TRAC-B.

Using the advanced systems codes as a baseline in the assessment of a simulator code is a unique role for such codes. While these advanced systems codes play an integral part in the safety analysis of nuclear power plant systems, their inherent uncertainty and limits must be qualified before meaningful conclusions can be deduced. One of the difficulties inherent in this type of procedure is that some models in simulator codes are capable of better performance than the best-estimate codes because they have been specifically designed for a given process or system. Since the advanced systems codes involve building mathematical models from a set of "building blocks", some detail may be lost from complex subsystems.

As part of the project, RELAP5 models of Pressurized Water Reactor simulators at the U.S. Nuclear Regulatory Commission's Technical Training Center have been developed and sets of transients preformed for comparison with simulator predictions. One such model was for the Washington Nuclear Project Unit 1 Simulator. Thermal-hydraulic analyses of five hypothetical accident scenarios were performed with the RELAP5/MOD3 computer code, then the same scenarios performed on the simulator, prior to a scheduled upgrade with S3 Technology's RETACT simulator code. The five transients performed were (1) Loss of AC power, (2) Small Break Loss of Coolant Accident with Loss of AC Power, (3) Stuck Open Pressurizer Safety Valve, (4) Main Steamline Break with Steam Generator Tube Rupture, and (5) Loss of Main Feedwater with Delayed Scram.

Comparison of code and simulator data was performed by reviewing each transient with a team of plant analysts and experienced reactor operators. The initial findings show that both the simulators and the system codes' modelling needs improvement. Comparisons show that the simulator does not model natural circulation and that leak rate guidelines are in error. The comparisons also showed that the RELAP5/MOD3 model's Integrated Control System modeling did not follow load reduction as it should have. Finally, comparisons also found some phenomenon for which there was no immediate "right or wrong" answer; additional analysis is required. The conclusion drawn from this preliminary study is that simulator benchmarking is and should be a dynamic, iterative task with benefits provided to both simulator engineers and to plant systems analysts.

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Depressurization As An Accident Management Strategy To Minimize Direct Containment Heating[a]

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During certain accident sequences, the potential exists for ejection of molten corium from a high pressure reactor coolant system (RCS) and for the dispersal of this corium into the containment atmosphere causing direct containment heating (DCH). High pressure melt ejections account for 13% of the core damage frequency in the NUREG-1150 risk study of the Surry nuclear power plant (NPP). Accident management strategies have been identified that have the potential to either prevent or mitigate the high pressure melt ejection that could result in DCH. These strategies include intentionally depressurizing the RCS prior to lower head failure.

Intentional depressurization of the RCS relies on the operator to latch open the power operated relief valves (PORVs) at some point during a transient. A station blackout transient was selected for analysis because station blackout contributes 95% to high pressure melt ejection occurrences in NUREG-1150. Previous analyses of the Surry NPP during station blackout considered two intentional depressurization strategies, referred to as early and late depressurization. Early depressurization depends on the operator latching the PORVs open at the time of steam generator dryout. Late depressurization depends on the operator latching the PORVs open at the time of a core exit thermocouple reading of 922 K (1200^{o}F). A temperature of 922 K ensures that the core is in the process of uncovering and that fuel damage is imminent. Results indicated that late depressurization permitted more time for the operator to restore ac power or water sources and also led to less core damage during the depressurization process. Late depressurization was therefore the preferred strategy over early depressurization.

An approach was developed for extending the Surry late depressurization results to other pressurized water reactors (PWRs). Based upon this approach, the PWRs in the U.S. were categorized into four groups according to their perceived capability to employ the intentional

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

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depressurization strategy. A representative PWR of each of these groups was chosen for a systematic evaluation during a station blackout sequence using the thermal-hydraulic/core damage progression computer code package SCDAP/RELAP5/MOD3. The four PWRs chosen for analysis were the Babcock & Wilcox Oconee NPP, the Combustion Engineering Calvert Cliffs NPP, the Westinghouse Surry NPP, and the Westinghouse Sequoyah NPP. (The Surry NPP was re-evaluated to take advantage of improvements to the SCDAP/RELAP5 computer code package.) The results of these evaluations have then been extended to the remaining PWRs in each group.

Results of these analyses indicate that the intentional depressurization strategy to minimize DCH may be successfully employed at all Westinghouse PWRs. For these PWRs, it is likely that the RCS can be depressurized sufficiently prior to lower head failure either through the PORVs or surge line failure to prevent high pressure melt ejection. Babcock and Wilcox PWRs may be able to reduce RCS pressure sufficiently prior to lower head failure, but only if surge line failure occurs as predicted. These PWRs have insufficient PORV capacity to reduce RCS pressure sufficiently to prevent the likelihood of high pressure melt ejection. Because surge line failure is not predicted and RCS pressure cannot be reduced sufficiently prior to lower head failure through the PORVs, high pressure melt ejection and possible DCH will likely occur for Combustion Engineering PWRs.

For all PWRs, some limited modifications and enhancements in plant equipment and operating procedures may be necessary prior to the implementation of the intentional depressurization strategy. A cost benefit analysis of these modifications and enhancements would be needed before taking any regulatory action.

Three Mile Island Unit 2 Vessel Investigation Project: Overview*

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In October 1988, the NRC, in cooperation with 10 foreign countries under the auspices of the Organization for Economic Cooperation and Development's (OECD) Nuclear Energy Agency, began a joint research program to examine and analyze material samples from the lower head of the TMI-2 reactor pressure vessel. The objectives of this program, called the TMI-2 Vessel Investigation Project (VIP), were to: (1) investigate the condition and properties of materials extracted from the lower head of the TMI-2 reactor pressure vessel, (2) determine the extent of damage to the lower head by chemical and thermal attack, and (3) determine the margin of structural integrity that remained in the pressure vessel. The three major elements of the TMI-2 VIP were: (1) the extraction of vessel steel, and nozzle and guide tube samples from the lower head region, (2) examinations of the samples, and (3) analytical studies to better understand the vessel's margin-to-failure.

The steel samples, along with the nozzles and guide tubes, were examined to determine the condition and properties of the samples and the extent of damage to the lower head during the accident. The results of these examinations were used to assist in quantifying potential reactor vessel failure modes, to estimate the vessel steel temperatures in the lower head during the accident, and to develop physical and mechanical property data to support the analysis effort. Scoping calculations and sensitivity studies were performed in an effort to quantify the margin-to-failure for different reactor failure modes and to identify which modes had the smallest margin-to-failure during the accident.

Results of the TMI-2 VIP contributed significantly to increased understanding of the extent of damage to the reactor vessel lower head and the margin of structural integrity that remained in the vessel during the TMI-2 accident, as well as lower vessel head behavior during severe accidents in general.

The steel extraction was very successful, and 15 reactor vessel steel specimens, 14 incore nozzles, and 2 incore guide tubes were extracted from the lower head over a 30-day period ending March 1, 1990. The prism-shaped vessel steel samples extended approximately one-half way through the 13.7-cm thick reactor vessel wall.

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* This work was done as part of the OECD/Nuclear Energy Agency's TMI-2 Vessel Investigation Project.

Overall, the vessel steel examinations revealed that a localized hot spot formed in an elliptical region on the lower head that was approximately 1 m x 0.8 m. The hot spot was located in the area where the most severe nozzle damage had occurred. Metallographic examinations of samples taken from this region indicated that the inner surface of the vessel steel reached temperatures between 1075 and 1100°C during the accident and remained at that temperature for approximately 30 minutes prior to being cooled. Temperatures 4.5 cm into the vessel wall were estimated to be 100 ± 50°C lower than the peak inner surface temperature. Away from the vicinity of the hot spot, lower head temperatures did not exceed 727°C.

Results of analyses of potential reactor vessel failure modes showed that ex-vessel instrument tube penetration rupture and tube ejection failure have large margins to failure. Analyses of global or local creep rupture failure are strongly dependent on the background vessel temperature outside the hot spot. Analyses were performed using the lower head temperature distribution and duration based on information from the vessel steel examination and companion sample examination data. Without modeling enhanced cooling of the debris on the lower head, the margin-to-failure scoping calculations indicated that the reactor pressure vessel would have failed via creep rupture when the reactor system was repressurized by plant operators at about 300 minutes. These results indicate that an additional safety margin, not currently considered within severe accident analyses, may occur due to coolant circulating within channels or cracks within the debris and within channels or gaps between the debris and the vessel. If relocated debris contains these channels, debris coolability is greatly enhanced, and lower head vessel failure is less likely.

Results of Examinations of Pressure Vessel Samples and Instrument Nozzles from the TMI-2 Lower Head

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The accident at the Three Mile Island, Unit 2 (TMI-2) reactor in March 1979 resulted in the relocation of approximately 19,000 kg of molten core material to the lower head of the reactor pressure vessel. This material produced significant metallurgical changes in portions of the lower head, and caused extensive damage to many of the instrument tube nozzles projecting through the lower head. The results of extensive examinations and characterizations of the lower head material and selected instrument tube nozzles are summarized here.

Fifteen steel samples were removed from the lower head of the TMI-2 pressure vessel for determination of metallurgical condition and mechanical properties following the accident. The samples were in the shape of inverted prisms, with a typical length of ≈165 mm, a width at the top of ≈70 mm, and a depth of ≈65 mm into the 127-mm-thick pressure vessel wall. The samples included the Type 308L stainless steel cladding (≈5 mm thick) weld deposit on the inner surface of the pressure vessel over the A533, Grade B, Class 1 steel base metal.

The fifteen lower head samples were sent to Argonne National Laboratory (ANL) for initial examination, decontamination, and sectioning. Full triangular cross sections from thirteen of the lower head samples, including the stainless steel cladding, were sent to the Idaho National Engineering Laboratory (INEL) for metallographic examination and hardness measurements. Because of difficulties encountered in sample decontamination, only partial cross sections that did not include the cladding layer were provided to INEL for the remaining two lower head samples.

Four of the fifteen samples examined at INEL showed evidence of exposure to temperatures in excess of the ferrite/austenite transformation temperature (727°C) during the accident. The maximum temperature at the base metal/cladding interface for two of these samples was estimated to have been in the range of 1075-1100°C for about 30 minutes; the other two samples were determined to have reached 1040-1060°C for a similar time. At a depth of 45 mm below the interface, the maximum temperature was determined to have been approximately 100°C lower. These four samples defined the location of an elliptical "hot spot" approximately 1 m x 0.8 m in size near the bottom of the lower head. The microstructures and hardnesses of the remaining eleven samples were essentially unaltered from the as-fabricated condition, indicating that the maximum temperature attained for those locations did not exceed the ferrite-to-austenite transformation temperature (727°C) during the accident. Results from independent examination conducted by several European laboratories were generally in good agreement with the INEL results.

During the initial examination of the lower head samples at ANL before decontamination, cracks were found in the stainless steel cladding of three samples, all of which were from the hot spot. The cracks appeared to be the result of hot-tearing, probably assisted by intergranular penetration of liquid Ag-Cd. Crack propagation into the A533 vessel steel was a maximum of ≈6 mm. Materials in the cracks suggest the presence of control-assembly debris (Ag, Cd, Zr, Fe, and Cr without U) on the lower head before the massive fuel debris flow arrived.

Tensile tests were conducted at ANL on the lower head material at room temperature and at temperatures of 600-1200°C. A strong dependence of yield and tensile strengths on temperature was observed, and the data generally merged well with available literature data on A533, Grade B, Class 1 steel. However, the strengths of material from the hot spot in the as-received condition, when tested at 600°C or below, lay well above the remaining data, reflecting the heat treatment received during the accident.

Creep tests were conducted on the lower head material over the temperature range of 600-1200°C at stress levels resulting in failure times of 1-100 h. The data from the lower head material compared well with similar data obtained earlier on archive material at 600°C from the Midland reactor. However, at higher temperatures, the TMI-2 lower head data fell increasingly above Midland material. The TMI-2 data were fit using both Larson-Miller and Manson-Haferd time-temperature parameters.

Charpy V-notch impact tests were conducted on four groups of test specimens. Specimens from the hot spot showed significantly lower upper-shelf energies and higher transition temperatures than specimens from regions that did not exceed 727°C during the accident.

The results of the instrument tube nozzle examinations conducted at ANL indicated that some nozzles were melted off by interaction with molten core debris, whereas others were only thermally affected by contact with core debris, some of which attached itself to nozzle surfaces. The elevations at which the nozzles were melted off suggested that the liquid core debris was atop a crust of solidified material that apparently generally insulated the reactor vessel from the hottest debris. The pattern of nozzle degradation was consistent with the location of the vessel hot spot, i.e., the nozzles in the hot spot region were melted off closer to the vessel. Based upon the observed severe damage to some nozzles and not to others in relatively close proximity, it can be concluded that the flow of material across the initial debris bed was in the form of individual rivulets as opposed to massive unified flow.

RESULTS FROM THE TMI-2 VESSEL RESPONSE ANALYSIS*

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The TMI-2 Vessel Investigation Project included vessel response calculations for mechanisms, which have the potential to threaten the integrity of the vessel. These calculations were performed to improve understanding of events occurring during the TMI-2 accident and to determine for which failure mechanism which there existed the least margin during the accident. Analyses considered penetration tube and vessel failure mechanisms using input based upon data from Vessel Investigation Project sample examinations.

Results from these calculations allowed the elimination of tube rupture and tube ejection as potential failure mechanisms during the TMI-2 accident. Global vessel failure analyses indicated that significant debris cooling must have occurred within the first two hours after debris relocation to the lower head. Although examinations of companion debris samples did not provide any supporting evidence of this cooling, metallurgical examinations did provide evidence that this cooling occurred. Localized vessel failure analyses indicated that it is possible for the vessel to withstand the hot spot temperatures for time periods inferred from VIP metallurgical examinations provided that the balance of the vessel is relatively cool.

Analyses were performed to quantify the magnitude of additional cooling needed to obtain thermal and structural analysis results consistent with metallurgical examination data indicating that the peak vessel temperatures outside the hot spot region remained below 1000 K and that the vessel steel in the hot spot region cooled at a rate between 10 and 100 K per minute. Although there is insufficient TMI-2 data to determine the exact mechanisms that caused the debris to cool, scoping calculation results indicate that a minimal volume of cooling channels within the debris and a minimal size gap between the debris and the vessel could supply the cooling needed to obtain vessel temperatures and cooling rates determined in metallurgical examinations.

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LESSONS LEARNED FROM THE CORA PROGRAM

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A better understanding of the in-vessel degraded core phenomena is important

- for risk assessments to be able to characterize the conditions of the core debris as it relates to the various accident sequences that are to be considered,
- for the purposes of accident management to understand the progression of the accident and the potential impact of any proposed strategies,
- for the design of advanced or next generation reactors.

In addition, in-vessel core-melt-progression investigations provide important input for all containment failure issues. For example, the quantity and rate, temperature, physical and chemical state of the melt released from the reactor pressure vessel and the amount of hydrogen generated as well as the steam availability, all can effect the integrity of the containment and thus impact risk.

The research work which has been performed so far contributed substantially to the understanding

- how a severe accident progresses,
- how to mitigate its consequences, and
- how to terminate it.

Important questions concerning the high-temperature core material behavior during an early-phase severe reactor accident have been:

- when and how the core loses its original geometry,
- what configurations are formed,
- how much hydrogen is generated by the steam oxidation of core materials and
- how the rate of oxidation (H_2 generation) is affected by changes in geometry,
- which influence of core degradation exists on the release of fission products (source term),
- by what processes solid and liquid core materials are transported within the core and to the lower plenum of the reactor pressure vessel,
- coolability of partially blocked cores, and
- influence of flooding (quenching) of degraded cores.

In the frame of the CORA program the chemical interactions among fuel element materials that may occur with increasing temperature up to the complete melting have been examined in out-of-pile experiments. The materials behavior of PWR, BWR and VVER-1000 fuel rod bundles have been studied in large-scale integral experiments and extensive separate-effects tests.

Oxidation processes of core components by steam determine temperature escalation, hydrogen generation and rubble formation, contribute to the core heatup rate (decay heat), diminish the importance of cladding preoxidation during service life, and counteract against melting processes, melt release and relocation. All of the CORA experiments performed with an initial heatup rate of 1 K/s showed a temperature escalation above about 1100 °C, driven by the exothermic Zircaloy cladding oxidation by steam. Maximum temperatures of approximately 2000 °C were reached within minutes. The available time for any accident management measure is therefore too short during this accident period.

Chemical interactions between core components determine liquid phase formation mainly as a result of eutectic interactions. In most cases, the reaction products have lower melting points than the original core components. This results in a relocation of liquefied components, often far below their melting temperatures. Control rod materials can separate from fuel materials by low-temperature liquefaction and melting processes that relocate different materials at different temperatures and times. This non-coherent stage-by-stage material relocation processes may cause recriticality problems during flooding of a partially degraded core by unborated water. UO_2 can be liquefied by molten Zircaloy about 1000 K below its melting point which may result in an increased fission product release and a redistribution of decay heat sources in the core.

As a result of the various experimental studies three distinct temperature regimes can be defined in which liquid phases form in the reactor core in different large quantities causing different types of core damage

- 1200 - 1400 °C, localized core damage,
- 1800 - 2000 °C, extended core damage, and
- 2600 - 2850 °C, total core destruction.

The extent of damage depends in addition on the initial heatup rate and the extent of Zircaloy cladding preoxidation.

The experimental results of the CORA program have a) contributed substantially to the understanding of the high-temperature core material behaviour in severe reactor accidents (damage initiation and progression), and b) provided a unique data base for the development, improvement and verification of models and code systems.

One of the important material behaviour phenomena which needs further experimental examination is the impact of water injection or flooding of uncovered core regions to understand the various causal mechanisms of renewed accelerated Zircaloy oxidation and correspondingly fast heatup, melting and strong H_2 generation during quenching.

Early phase core degradation is relatively well understood. However, code systems predicting core damage propagation still need further development and verification to bring them up to date with the experimental data base. The experimental data base on late phase melt progression is much poorer than for early phase, leading to far greater uncertainties in calculation of late phase core melt progression. Therefore, more well designed late phase melt progression experiments are needed.

Interpretation of the Results of the CORA-33 Dry Core BWR Test

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SUMMARY

Boiling Water Reactor (BWR) degraded core experiments have been performed in-pile in the Annular Core Research Reactor at the Sandia National Laboratory (the DF-4 test conducted in 1986) and have been planned for the Canadian National Research Universal (NRU) reactor (the FLHT-6 test, to be conducted in late 1993). Out-of-pile BWR core degradation experiments have been conducted exclusively in the German CORA test facility at the Kernforschungszentrum Karlsruhe (KfK) as part of the German Severe Fuel Damage (SFD) Program, which is coordinated by the Light Water Reactor (LWR) Safety Project Group (PRS). The CORA Program performs out-of-reactor experiments to provide information on the behavior of LWR fuel elements under severe fuel damage conditions. Six BWR experiments have been conducted in CORA during the period from November 1988 to October 1992.

All BWR degraded core experiments performed prior to CORA-33 were conducted under "wet" core degradation conditions for which water remains within the core with continuous steaming that can feed metal/steam oxidation reactions on the in-core metallic surfaces. However, one dominant set of accident scenarios would occur under "dry" core degradation conditions and, prior to CORA-33, had been neglected experimentally.

In American nuclear power plants (NPPs), Emergency Procedure Guidelines (EPGs) have been established whose basic functional goal is to provide the prudent actions to be taken by the plant operators in response to the symptoms observed by them at any point in time. Once entry into the EPGs has occurred, the operators are expected to take the specified actions regardless of equipment design bases, limitations, or licensing commitments. The guidelines use multiple mitigation strategies wherever possible so that recovery from an abnormal situation does not require successful operation of any one system or component.

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If partial uncovering of the core occurs in an American BWR and no injection is available, the operators, per the EPGs, will initiate the "steam cooling" maneuver. The purpose of this maneuver is to delay fuel heatup by cooling the uncovered upper regions of the core by a rapid flow of steam. Since the source of the steam is the remaining inventory of water in the reactor vessel, however, the steam cooling maneuver provides only a temporary delay (20 to 30 minutes) in core heatup and concludes with a vessel water level below the core plate. (The core plate provides lateral support for the core assemblies and is located approximately 23 cm below the active fissile region of the core.) The characteristics of a BWR severe accident sequence, following a steam cooling maneuver, are (1) no injection, (2) vessel depressurized, (3) boiloff with flashing during depressurization, and (4) steam-starved core degradation. This is termed a "dry" core degradation scenario because core degradation occurs under minimal steaming conditions with essentially a stagnant, steam-starved core atmosphere.

The CORA-33 experiment was designed specifically to address this dominant set of BWR "dry" core severe accident scenarios and to partially resolve phenomenological uncertainties concerning the behavior of relocating metallic melts which are draining into the lower regions of a "dry" BWR core (the ex-reactor experiments at Sandia National Laboratories will further address these uncertainties). CORA-33 was conducted on October 1,1992, in the CORA test facility at KfK. Initial review of the CORA-33 data indicates that the test objectives were achieved; that is, core degradation occurred at a core heatup rate (characterized by the absence of any temperature escalation due to oxidation) and a test section axial temperature profile (at incipient structural melting) that are prototypic of full-core NPP simulations at "dry" core conditions.

In all BWR degraded core experiments conducted under "wet" core conditions, the metallic melts formed partial blockages above the bottom of the active fuel, that is, within the core. CORA 33, however, was conducted under near-prototypic "dry" core conditions. A preliminary assessment of the posttest CORA 33 bundle condition and the experimental thermocouple data indicates that more metallic melt relocated further than was observed in any previous test. In fact, in CORA 33, a massive partial blockage occurred <u>below</u> the bottom of the active fuel; that is, in a reactor the melt could have drained into the core plate region.

Simulations of the CORA 33 test at the Oak Ridge National Laboratory (ORNL) have required modification of existing control blade/canister materials interaction models to include the eutectic melting of the stainless steel/Zircaloy interaction products and the heat of mixing of stainless steel and Zircaloy. The timing and location of canister failure and melt intrusion into the fuel assembly appear to be adequately simulated by the ORNL models.

This paper will present the results of the posttest analyses, based upon the experimental data and the posttest examination of the test bundle at KfK, which were carried out at the Oak Ridge National Laboratory. The implications of these results with respect to degraded core modeling and the associated safety issues will also be discussed.

The MP-2 Late Phase Melt Progression Experiment in ACRR

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The MP-2 melt progression experiment was conducted in the Annular Core Research Reactor (ACRR) at Sandia National Laboratories on November 7, 1992. This experiment addresses the later stages in commercial light water reactor core melt progression and provides unique information on these processes for use in the development and validation of models and codes predicting the response of nuclear reactor meltdown accidents. The experiment investigated the melt progression of a debris bed of UO_2 and ZrO_2 overlying a blockage of metallic control and structural materials (Ag, In, Zr and stainless steel) formed around zircaloy-clad UO_2 fuel rods. This geometry approximates conditions at the interface of a molten pool/crust region that is migrating through relatively intact rod structures, moving toward a core boundary. Nuclear heating provided by the ACRR driver core simulated the internal decay heat source in the fuel debris, and led to the formation of a molten UO_2/ZrO_2 pool region that attacked and migrated into the underlying crust/fuel rod geometry. The metallic crust and much of the fuel rod cladding melted and ran away from the advancing ceramic crust which contained the molten fuel. Peak fuel temperatures in the pool are estimated to have reached 3400 K and were sustained for a period of about an hour. The experiment package has been disassembled and examined metallurgically. Details on this examination are presented in this paper, as well as the results of post-test analyses performed with the DEBRIS porous media melt relocation model.

Turbulence Model for Melt Pool Natural Convection Heat Transfer

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ABSTRACT

In severe reactor accident scenarios, pools of molten core material may form in the reactor core or in the hemispherically shaped lower plenum of the reactor vessel. Such molten pools are internally heated due to the radioactive decay heat and are actively mixed due to the resulting turbulent buoyant flow. The variation of the local heat flux over the boundaries of the molten pools is important in determining the subsequent melt progression behavior. This study presents the first phase results of an ongoing effort towards development of a well-validated mathematical model for the prediction of buoyant flow and heat transfer in internally heated pools under conditions expected in severe accident scenarios.

In this study, a low-Reynolds number version of the k-ε turbulence model is utilized as the basis due to its generality, computational economy, and its successful use for a wide variety of problems. This basic turbulence model is refined by incorporating the effect of buoyancy on turbulence using the Eddy viscosity model or the Generalized Gradient Diffusion Hypothesis. This is important in accurate prediction of turbulent mixing in buoyancy driven flows. In addition, a near-wall correction term has been incorporated in the turbulence dissipation equation, as proposed by Yap (1987), to properly predict the length scales in the near wall region. Five different permutations of the Jones and Launder low-Reynolds number turbulence model are validated against the experimental data for three buoyant flow problems available in the literature. These three test problems are the buoyant flow in a square cavity with hot left wall and cold right wall, internally heated fluid layer, and an internally heated semicircular cavity. The results indicate that the low-Reynolds number k-ε turbulence model accurately predicts the experimentally observed heat transfer characteristics for the first two test problems considered when the effect of buoyancy on turbulence and the Yap correction are included. For natural convection in a semicircular cavity, computations show that, as the Rayleigh number is increased, the heat transfer rates on the flat and the curved surfaces approach each other due to stronger mixing throughout the cavity – a behavior that is different from that at low Rayleigh numbers.

In subsequent phases of the work, the turbulence models will be further tested against experimental data available for other geometries and refined further. It will then be applied for the prediction of natural convection in internally heated pools of shapes and heating conditions expected under severe accident conditions.

Detection and Effects of Pump Low-Flow Operation*

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ABSTRACT

Operating experience and previous studies performed for the Nuclear Plant Aging Research Program have shown that a leading cause of pump problems and failures is associated with hydraulic instability phenomena induced by low-flow operation. Operation at low-flow rates can create unsteady flows within the pump impeller and casing. This condition can result in an increased radial and axial thrust on the rotor, which in turn causes higher shaft stresses, increased shaft deflection, and potential bearing and mechanical seal problems.

Two of the more serious maladies of low-flow pump operation are cavitation and recirculation. Both of these conditions can be characterized by crackling sounds that accompany a substantial increase in vibration and noise level, and a reduction in total head and output capacity. Cavitation is the formation and subsequent collapse of vapor bubbles in any flow that is at an ambient pressure less than the vapor pressure of the liquid medium. It is the collapse of these vapor bubbles against the metal surfaces of the impeller or casing that causes surface pitting, erosion, and deterioration. Pump recirculation, the reversal of a portion of the flow back through the impeller, can be potentially more damaging than cavitation. If located at the impeller eye, recirculation damages the inlet areas of the casing. At the impeller tips, recirculation alters the outside diameter of the impeller. If recirculation occurs around impeller shrouds, it damages thrust bearings. Recirculation also erodes impellers, diffusers, and volutes and causes failure of mechanical seals and bearings.

ORNL has continued to investigate low-flow pump phenomena by evaluating the types of measurements and diagnostic techniques that are currently used by licensees to evaluate pump degradation. Two independent analysis techniques have been identified that could potentially lead to improved ways to dynamically determine the onset of hydraulic instability. A new, enhanced application of motor current signature analysis has been developed that uses a signal comparison technique to produce an instability ratio indicative of normal or unstable flow conditions. Secondly, applications of deterministic chaos theory using mutual information analysis techniques have been shown to also provide insight in detecting aperiodic or chaotic behavior in stable and unstable pump and flow system operations.

Examples of both types of low-flow detection techniques are presented in this paper along with a brief discussion of the various types of techniques currently being used by licensees to evaluate pump operation and determine possible degradation.

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Understanding Aging in Containment Cooling Systems

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An aging assessment of containment cooling systems in commercial nuclear power plants has been performed as part of the Nuclear Plant Aging Research (NPAR) program. The containment cooling function is performed by several different systems, depending on the type and design of the plant. The two systems focused on in this study are the containment spray system, which is used in pressurized water reactors (PWRs) and boiling water reactors (BWRs), and the fan cooler system, which is used in PWRs. These systems were selected since they are the primary means of removing containment heat during accident conditions.

One of the national databases used for this study is the Nuclear Plant Reliability Data System (NPRDS). Over 50% of the approximately 2200 NPRDS records reviewed (data for all U.S. PWRs and BWRs from 1986 to 1991) were related to degradation caused by aging. These failures typically result in a degraded operating state for the system, or a loss of redundancy. Other findings from the data analysis are summarized in Table S.1. The results of this work show that aging is a concern for the containment cooling system and should be addressed in plant programs.

Table S.1 Summary of Data Analysis Results

Analysis Finding	Containment Spray System	Fan Cooler System
Sample size	1368 records	808 records
Percentage of failures related to aging	59%	52%
Most frequently failed components	Valves (47%)	Circuit Breakers (32%)
Predominant failure cause	Normal Service (74%)	Normal Service (60%)
Predominant effect of failure	Degraded Operation (60%)	Loss of Redundancy (57%)
System status during failure detection	Test (57%)	In Service (64%)
Predominant failure detection method	Test Results (58%)	Abnormal Operation (31%)

From the unavailability analysis performed on one common PWR containment spray system design, the dominant contributor to system unavailability was found to be a non-aging related event; namely a human error involving failure to reposition manual valves following surveillance testing. For components that could be affected by aging, pumps and MOVs were found to be important to system unavailability. Increases in their failure rate produce a noticeable increase in system unavailability. For example, a ten fold increase in pump failure rate increases system unavailability by a factor of three, and the pump contribution to system unavailability exceeds that of the human error.

The unavailability analysis of one common PWR fan cooler system design showed no dominant, single contributor to system unavailability. The largest single contributor was a common mode failure of the fan motors. Based on cumulative contributions from all potential failure scenarios in which it appears, unavailability due to maintenance was the largest contributor to system unavailability, followed by dampers failing to open, circuit breaker malfunction, and fan motor failures. The parametric analyses showed that for a ten fold increase in damper failure rate, system unavailability increases by a factor of approximately 66. The exponential increase in unavailability is due to the redundancy of the components in the system design. When circuit breaker failure rate increases by a factor of 10, system unavailability increases by a factor of approximately 13. Therefore, proper aging management of dampers and circuit breakers in this specific system design is important and should be addressed in plant programs.

This phase I aging analysis has provided a basis for understanding the effects of aging in containment cooling systems. Conclusions and recommendations resulting from this study are summarized below:

- Aging degradation exists in containment cooling systems and is a significant contributor to failures. Since these systems play an important role in accident mitigation, plant programs should specifically address the proper management of aging in containment cooling systems. Each of the aging mechanisms identified in this study should be addressed by at least one monitoring technique.

- The failure data show that most containment spray system failures are detected by surveillance tests and inspections. This is significant since it shows the importance of performing tests and inspections on standby systems to detect degradation before it results in an operational failure. It should also be noted that for standby systems, testing can be a significant contributor to aging degradation.

- The failure data show that the most common human error type failure occurs during or as a result of maintenance activities. It is recommended that, if efforts to reduce human errors are made, they should be concentrated in the area of maintenance. In addition to improving human performance, one possibility that should be considered is the elimination of unnecessary maintenance activities.

- The review of industry and plant specific data has shown that the failures occurring in the containment cooling systems were not severe enough to result in a complete loss of system function. Typically, the most severe failure will result in a loss of redundancy, however, the system is still able to perform its design function.

- Failure trends identified from NPRDS for most of the major system components show a trend for increasing failures with age. This increasing trend will result in a corresponding increase in system unavailability with age, if the trend is not properly controlled. Therefore, plant programs should include a similar plant specific analysis to identify any time-dependent trends in component failures so they can be properly managed.

PHASE I AGING ASSESSMENT OF NUCLEAR AIR-TREATMENT SYSTEM HEPA FILTERS AND ADSORBERS

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SUMMARY

A Phase I aging assessment of high-efficiency particulate air (HEPA) filters and activated carbon gas adsorption units (adsorbers) was performed by the Pacific Northwest Laboratory (PNL)[a] as part of the U.S. Nuclear Regulatory Commission's (NRC) Nuclear Plant Aging Research (NPAR) Program. The filters and adsorbers are key components of nuclear air-treatment systems, systems whose failure can impact both plant and public safety. The two components are designed to capture radioactive gaseous and particulate contaminants and, following an accident, can serve as the last barrier between a radioactive release and the public. As part of the aging assessment of these two key components, information was compiled concerning design features; failure experience; aging mechanisms, effects, and stressors; and surveillance and monitoring methods.

U.S. Department of Energy (DOE) sites had earlier been surveyed to investigate the reasons for filter changeouts. Based on the survey, Carbaugh (1983) reported the occurrence of over 1100 filter failures or 12% of the installed filters. Investigators from other national laboratories have suggested that aging effects could have contributed to over 80% of these filter failures (Johnson et al. 1989). The report by Johnson et al. (1989) also included the results of an experimental evaluation of the tensile breaking strength of aged filter media specimens: 42% of the samples did not meet specifications for new filter material.

Although less than 1% of over 60,000 licensee event reports (LERs) appeared to be related to filters and adsorbers, several instances have been described of the premature aging of carbon from error-induced conditions (Moeller and Kotra 1985). Low radioiodine decontamination factors associated with the Three Mile Island (TMI) accident were attributed to the premature aging of the carbon in the adsorbers (Rogovin and Frampton 1980). Inspection, surveillance, and monitoring methods (ISMMs) have been established to observe filter pressure drop buildup, check HEPA filters and adsorbers for bypass, and determine the retention effectiveness of aged carbon. Exemptions to TMI technical specifications postponed the surveillance test, (laboratory tests of adsorbent) that could have revealed the extent of carbon aging.

Aging mechanisms associated with filters range from particle loading of the media to corrosion of metal members, and physicochemical reactions that alter properties of sealants, gaskets, and water repellents. Also, filter media may become embrittled from prolonged exposure to air. In the case of

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(a) Work supported by the U.S. Nuclear Regulatory Commission under U.S. Department of Energy Contract DE-AC06-76RLO 1830.

adsorbers, aging mechanisms lead to impaired performance in terms of the capture of volatile iodine species. The deterioration in performance is caused by oxidation as well as the competitive loading of other airborne constituents. Many airborne constituents, including moisture, can react with or be adsorbed by carbon beds, reducing the number of active "sites" on the carbon bed that otherwise would be available for radioiodine adsorption. Stressors include heat, humidity, radiation, and airborne contaminants and pollution.

While adequate ISMMs exist for normal filter and adsorber service conditions, it is recognized that, excluding local ruptures and tears, neither pressure drop monitoring nor surveillance leak testing of installed HEPA filters indicate aging in terms of reduced filter media strength. However, aged, intact filters can continue to effectively remove particles under normal conditions. On the other hand, this lack of indication may be important when considering reactor accident conditions. HEPA filters and adsorbers are considered to have a long service life, especially the filters. Thus, if a severe accident happens, it is likely to occur when these two final confinement barriers have been in use for an extended period, even years. Even with existing ISMMs, aged or possibly degraded components could fail to provide the radiation protection needed for safe shutdown or could be the weak link that allows the release of radionuclides to the environment.

The Phase I aging assessment revealed that better evaluation of filter and adsorber effectiveness during severe accidents will require an improved definition of accident conditions, and possibly, additional information to evaluate the performance of aged components under such conditions. Although beyond the scope of the NPAR program, additional insights into accident conditions should become available as a result of recent regulatory efforts to provide improved source term definitions. Use of these improved definitions should provide better estimates of accident conditions and the associated challenges to filters and adsorbers.

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PRIORITIZATION OF MOTOR OPERATED VALVES BASED ON RISK IMPORTANCES

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In response to Generic Letter 89-10 requiring testing of motor operated valves (MOVs), the use of probabilistic risk assessments is proposed as one means for prioritizing the specific MOVs to be tested. PRA-based prioritization of MOVs could potentially identify those MOVs which are important to risk and those MOVs which are not important to risk. In previous research carried out under the Nuclear Plant Aging Research (NPAR) Program, approaches were developed to prioritize components when multiple degradations and common cause failures could exist. These scenarios are similar to those described in Generic Letter 89-10. Using the NPAR work as a basis, different prioritization approaches are evaluated and are compared. Validation and checking procedures are described. Criteria for prioritization procedures to achieve given risk levels are also described. Specific demonstrations are provided to illustrate the findings.

AGING MANAGEMENT OF LIGHT WATER REACTOR CONCRETE CONTAINMENTS[a]

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Summary

The paper presents insights for effective aging management of light water reactor (LWR) concrete containments and foundation elements to maintain their integrity. These insights have been gained from a comprehensive review of the technical literature related to containment designs, to degradation of containment materials including concrete and steel, and also to inspection, testing, monitoring, and mitigation of aging damage.

The operating experience with LWR concrete containments in the United States is less than 25 years and has not revealed any significant or generic aging concern. A review of documented history of the U.S. containment structures supports this observation; it has revealed only a few instances of age-related degradation. However, there are some degradation mechanisms associated with reinforced concrete containments that could cause degradation after years of satisfactory performance. Three such mechanisms are sulfate attack on concrete, alkali-aggregate reactions in concrete, and corrosion of reinforcing steel. The paper focuses on these mechanisms and briefly discusses other potential degradation mechanisms.

Sulfate attack produces an expansive product called ettringite that leads to concrete cracking. Sulfate attack also forms gypsum, which reduces concrete stiffness and strength and could lead to concrete cracking. The extent of damage depends on concrete permeability, cement type, and sulfate type and concentration. Because the concrete in U.S. LWR containments is made with Type II cement, which has low tricalcium aluminate content, it resists the cracking caused by ettringite formation. However, even in the absence of tricalcium aluminate, sulfate attack can form gypsum, and in the case of magnesium sulfate attack, concrete loses its cementitious characteristics after prolonged exposure, which could lead to disintegration of the concrete. The specific consequences of gypsum formation are not yet well established. Monitoring of physical and mechanical properties could identify any delayed activities, like expansion and strength drop, that are common in sulfate attack. Chemical and microstructural studies could assist in identifying the events associated with sulfate attack that trigger these delayed activities.

Alkali-aggregate reactions are the internal reactions between the alkali hydroxides in the concrete and certain minerals in aggregates that form expansive products in the presence of moisture. These expansive products could produce concrete cracking and significantly increase the concrete permeability. The alkali-silica reaction has been the most disruptive of the reactions observed in general industry. It can cause degradation after years of satisfactory performance. Certain siliceous aggregates have exhibited slow reactivity with

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alkalies that is not detected until after about 20 years. These aggregates passed the ASTM test methods in effect at the time they were selected. The slow-reacting aggregates, which appear to be most common in the eastern states, pose a potential problem for the concrete containments that used such aggregates in their construction and that have sufficient alkalies present in the concrete to sustain the alkali-silica reaction. The ultraviolet radiation technique can be used to detect the presence of any alkali-aggregate reaction.

Corrosion of the reinforcing steel is unlikely unless the steel is depassivated as a result of one of the following three conditions: (a) chlorides in the pore solution in the hydrated cement paste, (b) reduced alkalinity of the pore solution caused by carbonation of the concrete or by leaching of calcium hydroxide from the concrete, or (c) presence of stray electric current. Corrosion of reinforcing steel eventually causes the concrete cover to expand, crack, delaminate, and spall because the corrosion products occupy a larger volume than the original steel. Corrosion associated with chloride ions or carbonation will not be initiated until these corrosion-inducing agents have penetrated through the concrete to the reinforcement. Once corrosion is initiated by chloride ions, concrete cracking may occur within about 5 to 10 years. Carbonation will require about 100 years to reach the reinforcement in concrete containments constructed with a low permeability concrete and a concrete cover of 50 mm or more thickness. After that, cracking will likely occur within about 10 to 20 years. The half-cell potential method can be used to detect corrosion of reinforcing steel. Cathodic protection could be used to protect the reinforcing steel.

Other nondestructive testing techniques can be used for estimating the condition of concrete. Some examples include the rebound hammer test (to estimate the compressive strength of concrete), infrared thermography (to locate subsurface voids and delaminations in concrete), the impulse radar technique (to estimate the depth of the defect once it is located), and the capacitance test (to measure the moisture content of concrete). Condition monitoring of the containments using periodic visual inspection and testing, and the use of proven repair and maintenance techniques can help in maintaining long-term containment integrity.

Assessing Functional Diversity by Program Slicing

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One of the most commonly invoked safeguards against design basis events in nuclear power plants is the use of functional diversity. When auditors for the United States Nuclear Regulatory Commission (NRC) examine safety systems for nuclear power plants, they also inspect the software embedded within the safety systems to ensure functional diversity. For computer software, this means that there is no common code shared among diverse computations. The steps the auditor must take include identifying the types of system hazards that could occur, using techniques, like fault tree analysis, to locate software that should be examined further, and then inspecting the code for diversity. This paper describes an approach using a concept called program slicing to support auditors.

Program slicing is a family of program decomposition techniques based on extracting statements relevant to a computation in a program. Program slicing as originally defined produces a smaller program that reproduces a subset of the original program's behavior. This is advantageous since the slice can collect an algorithm for a given calculation that may be scattered throughout a program, excluding irrelevant statements. It should be easier for a programmer interested in a subset of the program's behavior to understand the corresponding slice than to deal with the entire program. The utility and power of program slicing comes from the potential automation of tedious and error prone tasks. Program slicing has applications in program debugging, program testing, program integration, parallel program execution and software maintenance. Program slices fit in with the way programmers understand programs since after trying to understand an unfamiliar program, programmers recognize slices from the program better than random chunks of code from a program.

Once the system hazards have been identified, the objective is to mitigate the risk that they will occur. One approach to achieving this objective is to use system fault tree analysis. Under the assumption that there are relatively few unacceptable system states and that each of these hazards has been determined, the analysis procedure is as follows. The auditor assumes that a hazard has occurred and constructs a tree with the hazardous condition as the root. The next level of the tree is an enumeration of all the necessary preconditions for the hazard to occur. These conditions are combined with logical *and* and *or* as appropriate. Then each new node is expanded as far as possible.

After system fault tree analysis gives the auditor the sub-components of the system that must be carefully examined, then software fault tree analysis can refine the identification of components further. Part of this examination is the validation that there are no interactions with non-critical functions. The determination of the specific components that will be examined is up to the auditor. This information should be obtainable from the design documentation.

At NIST we are developing a program slicing tool called **unravel**, adapted for NRC auditor use in software safety audits of vendor software. **Unravel** is being developed on a POSIX compatible lap top system to allow an NRC auditor to examine vendor source code written in ANSI C.

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SOFTWARE RELIABILITY ASSESSMENT

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The increased usage and sophistication of computers applied to real time safety- related systems in the United Kingdom has spurred on the desire to provide a standard framework within which to design, build, and assess dependable computing systems. Recent accidents and ensuing investigations and legislation have acted as a catalyst in this area. The approach to producing dependable computing systems within the UK varies widely across the different user sectors. Various significant organisations have recognised this shortcoming and have taken initial steps to overcome the problem. This paper relates some of the author's experience of the approaches to software reliability and safety achievement, since this is the most onerous area.

In particular, the paper focuses on the assessment of software reliability, based on the experiences gained from commercial and research projects undertaken by the Programmable Electronic Systems (PES) Department of AEA Technology, Consultancy Services (SRD).

The topic of software reliability assessment is discussed here in two contexts:

(i) that for safety-critical software where there is a requirement for the software to be licensed for high-integrity applications, such as nuclear reactor protection systems;

(ii) that for commercial grade software (eg proprietary software, where the manufacturer is reluctant to provide information about its development), which might or might not need to be licensed, depending upon its application, and which might be used in safety related applications, or where system availability is crucial.

1 THE ASSESSMENT OF SAFETY CRITICAL SOFTWARE

Assessment is a necessary pre-cursor to licensing; the objective of licensing software for safety critical applications is to provide a judgement on the adequacy of the safety aspects based on documentary information submitted by the developers. This judgement will be influenced by approaches to licensing in other industries and emerging and established standards/guidelines.

The Assessor is seeking by objective argument to convince, or otherwise, the Licensor that:

- the correct safety functions have been specified in the software requirements
- these functions have been correctly implemented in the design and development
- safety will continue to be maintained in operational life, via the integrity of the software maintenance and change mechanisms

Hence the requirements specification is a vital document to the licensing and assessment functions.

Numerous problems encountered in the assessment and licensing process arise from interaction problems between the assessor, licensor, and developer. These can be minimised by providing a generic, structured framework to ensure that the relationships and responsibilities of the assessor, licensor, customer and developer are well-defined. The framework also requires that the development, justification and licensing activities are planned and scheduled together so that problem areas are rapidly detected and licensing delays are minimised.

From the above, it is obvious that the assessment and licensing task should be considered as an exercise in communication, but more specifically, it should be considered as an exercise in formal written communication. This latter aspect cannot be over-emphasized, and has caused major problems in the experience of AEA Consultancy Services.

2 THE ASSESSMENT OF COMMERCIAL GRADE SOFTWARE

The success of the assessment of commercial grade software depends heavily on the software manufacturer to provide information on:

- the software development process, eg the standards and procedures used, the software QA applied, verification activities, and configuration management

- and the software product, eg: dynamic testing and static analysis, and the extent of validation undertaken.

This conflicts with the desire of the manufacturer to protect commercial secrets. Hence, the manufacturer is very careful as to whom information is released to. Our experience shows that a manufacturer is much more amenable to releasing information to a 3rd party independent consultant, than to the end user, and even then signed agreements of non-disclosure are required.

But what if the manufacturer does not wish to release such information? There are ways of establishing (limited) confidence in the software based upon high level information, eg pedigree of the company, stability of the software product, and use of the software in similar applications.

The SCOPE (Software Certification Programme in Europe) Project (an ESPRIT II project) was triggered by an EEC resolution to aim for a global approach for conformity and assessment, and it has addressed the above and other issues for a range of software categories. The evaluation methodology developed in this project has been accepted by the ISO IEC 9126, and it will soon be voted upon, to become a "DIS" - Draft International Standard.

Evaluation "bricks" have been applied to a number of case studies, and the results from these case studies have been fed back into the methodology. It is thus now possible to carry out an objective product assessment of commercial grade software in accordance with emerging European standards. This could be carried out at the lowest level if solely the documentation were available, or if the code were available, it could also be carried out as a much more detailed study via analysis tools, eg metrics.

Class 1E Software V&V: Past, Present and Future

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What is the problem?

Nuclear reactors, like any complex industrial plant, routinely experience equipment or operational failures, some of which could lead to serious consequences. Unlike many industrial plants, one potential consequence of reactor accidents is the release of radiation or radioactive material into the environment. Until the mid 1970s, software was not used in Class 1E systems, but a rapid transition into an era of computer and software control of these Class 1E systems is now occurring. This movement has been accelerated by the President's Commission's report on Three Mile Island, which recommends that using computers could improve plant safety. The problem specifically addressed in this paper is how to gain the trust necessary to license Class 1E software.

What is Class 1E Software?

IEEE Standard 379 defines Class 1E as, "The safety classification of the electrical equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling and containment and reactor heat removal, or are otherwise essential in preventing significant release of radioactive material to the environment." In this paper, software used in Class 1E systems is referred to as Class 1E software. Class 1E software currently is being used in reactor trip systems, ESFAS, and in some emergency generator load sequencers.

What Makes Class 1E Software Different?

Considering the use of computers and software during the last 20 years, why is there concern over the application of this technology to Class 1E systems? First, software does not wear out and hence fail as hardware does; but it can contain manufacturing or design defects which will produce hazards and failures in operational use. Second, software development is a labor-intensive, intellectual activity. A noted software expert, E. Dikstra, has stated that software development is one of the most intellectually challenging activities in which humans can engage. Errors can be very subtle and completely overlooked during implementation but can cause catastrophic failure after years of apparently successful operation. A large body of opinion, however, believes that with proper use of modern software engineering techniques, the number of residual defects in delivered Class 1E software can be reduced to an acceptable minimum, and these remaining defects will not have severe consequences.

This paper compares current electro-mechanical Class 1E systems to software-based Class 1E systems and identifies new issues related to the use of software in these systems, including: lack of experience in developing safety-critical Class 1E software, inability to measure the required ultra-high reliability of software, immature mathematical theory or basis for software construction, difficulty in proving software correctness, new potentials for common-mode failures, lack of operational data, unproved development methods, and an environment in which small errors may have significant consequences. In spite of this, software-based systems may be the only reasonable alternative for replacing aging protection system components. Traditional analog/relay equipment is becoming much more expensive and, in some cases, is totally unavailable. If substitute equipment is available, it often contains embedded digital hardware and software.

What is Software Verification and Validation (V&V)?

This paper addresses the process of software verification and validation (V&V) which must be employed over the full software life cycle. It describes the various activities to be performed, specifies the inputs and outputs of these activities, and addresses the planning and management of the V&V effort. Software V&V is a primary quality control activity and is one of the state-of-the-practice techniques that enables confidence and trust to be placed in Class 1E software systems. IEEE Standard 610.12-1990 defines software verification and validation as, "The process of determining whether the requirements for a system or component are complete and correct, the products of each development phase fulfill the requirements or conditions imposed by the previous phase, and the final system or component complies with specified requirements."

Verification and Validation Audits—What Are They?

The goal of the audit process is to protect the health and safety of the public by ensuring that the Class 1E system being audited will in fact bring the reactor to a safe shutdown condition when appropriate. An objective of any V&V audit is to provide an unbiased compliance confirmation of software processes used and software products created by these processes, in order to certify adherence to required standards, guidelines, specifications and procedures. It is very difficult for a single audit to provide adequate assurance that Class 1E software will reliably perform its required functions. Ideally, the audit process should begin when the project begins and continue with audits scheduled throughout the design, implementation, testing, and installation stages of the project.

This paper addresses the various V&V audits called for in ANSI/IEEE-ANS-7-4.3.2. The standard requires software verification audits to be performed on both the software development processes and interim software products at several points in the development life cycle, and requires execution of both a validation audit and an installation audit. This paper describes which processes and products should be evaluated in each audit and discusses evaluation criteria to be used in establishing confidence in the Class 1E system being audited. Finally, this paper lists several references and standards that can be useful during audits of Class 1E systems.

EVALUATION OF THE COMPUTERIZED PROCEDURES MANUAL (COPMA-II)

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Abstract

The Computerized Procedure Manual (COPMA-II) is an advanced computer system that can be used to select and execute procedures in nuclear power plants, and to display system parameters and graphics. The effectiveness of COPMA-II was evaluated in a study that was conducted at the Scaled Pressurized Water Reactor Facility (SPWRF) at North Carolina State University. Sixteen operators (eight SROs, eight ROs) were trained to operate the SPWRF with traditional procedures and with the COPMA-II system. Each team then performed one change of power with each procedure system. Teams then performed one of two accident scenarios (small break loss of cooling accident and steam generator tube rupture) with traditional procedures, and the remaining scenario with the COPMA-II system. Error rates, performance times, and subjective estimates of workload were collected for each scenario, and were calculated for each combination of procedure type (Paper vs. COPMA-II) and trial type (Normal vs. LOCA vs. Steam Tube Rupture). Study results revealed that change of power errors were lowest for COPMA-II procedures. For the accident scenarios, performance was significantly faster for paper (M = 21.65) than for COPMA-II procedures (M = 28.23) for the first trial of both accident scenarios. However, for the second accident scenario, COPMA-II performance was the most rapid. Thus, it appears that operators employed COPMA-II with relatively little training to increase their performance speed and reduce their error rates. These results are discussed in relation to implementation of the COPMA_II system into operational control rooms.

VALIDATION OF THE USE OF NETWORK MODELING OF NUCLEAR OPERATOR PERFORMANCE

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This effort is an attempt to explore the feasibility of using task network modeling to study human factors issues in a nuclear power plant control room. Task network modeling extends human factors engineering task analysis into a computer model. This technology has been used increasingly by human factors engineers over the past decade to predict human and system performance. However, its efficacy has never been proven in the nuclear operations environment. This study seeks to determine the practicality as well as validity of task network modeling as a tool for the analysis of human factors in nuclear systems.

Previous applications of task network modeling support the notion that individual tasks within a model can be varied in such a manner as to predict the effect of workstation or control room changes on human performance. Such variations may include control room design changes, task automation, and the consideration of human performance stressors. Task network modeling permits researchers to vary attributes of the human operator and/or control room design without the expense of physical modifications and without intrusion on operations.

In this study, nuclear operators performed various tasks with standard paper procedures and with a computerized version, the Computerized Procedures Manual (COPMA -II). Task network models were used to predict operator performance during critical tasks with the COPMA-II condition and then compared to actual COPMA-II performance. The effectiveness of the task network model to predict operator performance using computerized procedures will be discussed.

ACCOMPLISHMENTS IN NRC-SPONSORED FISSION PRODUCT RELEASE RESEARCH AT ORNL

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The U.S. Nuclear Regulatory Commission (NRC) has sponsored high-temperature fission product release research at Oak Ridge National Laboratory (ORNL) since 1981. Fuel used in this testing program came from full burnup bundles of standard fuel used in light-water reactors: H. B. Robinson, Peach Bottom-2, Oconee, Monticello, and a Belgian reactor, BR3. Atmospheres have been steam, limited steam, or hydrogen. Test temperatures ranged from 1675 to 2700 K.

Recent advances in our knowledge about fission product release and behavior under severe accident conditions have centered on the less volatile elements, especially their response to different atmospheres. Most previous research has concentrated on the volatile fission products: the fission gases xenon and krypton, cesium, and iodine. The releases of these volatiles are generally agreed to be about equal on a percentage basis, with some evidence that the release of iodine may be slightly lower. Tests at ORNL have shown that the effect of atmosphere (steam, limited steam, or hydrogen) is relatively small. The possibility of the reaction between Zircaloy and fuel causing an enhancement of fission product release was first suggested at a meeting of the President's Commission at ORNL 2 months after the TMI-2 accident. This reaction is often referred to as "liquefaction," but the ORNL tests have shown that volatile fission product releases in the reducing atmosphere that permits this reaction are not different from those in steam atmospheres within the experimental error band. The small effect on volatile fission product release is partly the result of a large part of the Zircaloy running off the fuel and the remainder being quickly oxidized by the excess of UO_2.

The atmosphere does have a strong effect on the release rates of many of the less-volatile fission products. Several of the important effects of atmosphere were first observed in the early 1960s and were reported in ORNL-3981. These were that oxidizing atmospheres enhance the release of ruthenium, and that melted unoxidized Zircaloy inhibits the release of tellurium while simultaneously increasing the releases of barium and strontium. It is only recently that these and other atmospheric effects on the less-volatile fission products have been quantified using commercial-type Zircaloy-clad UO_2 fuel under severe accident conditions. The capture of tellurium by unoxidized Zircaloy cladding results in an effective release rate from the fuel-cladding combination 1/40 of that which occurs when the cladding is completely oxidized. Several laboratories have investigated the trapping mechanism. The recent ORNL tests have shown that fission product antimony is also trapped by unoxidized cladding. This effect has not been included in the various fission product release models.

In many reactor accidents, part of the core may become steam starved. The resulting hydrogen atmosphere allows Zircaloy cladding to melt and react with UO_2 and chemically reduce many of the fission products. The effect on barium, strontium, and europium is to form chemical species that are more volatile than those that exist in a neutral or steam atmosphere. These differences have been quantified, with europium showing the greatest change in volatility.

Oxidizing atmospheres such as steam or air form oxides of fission products ruthenium and molybdenum that are more volatile than those that normally exist in the fuel. With small pieces of unclad fuel, the oxidation effect on ruthenium occurs rapidly, but the ORNL tests with 15-cm-long segments of Zircaloy clad fuel do not show this effect. Either kinetic restriction or trapping of the volatile ruthenium oxide by the oxidized cladding or other ceramic materials is believed to occur.

Release rates that include the atmosphere/cladding effects have been quantified for all of the above-mentioned fission products. Preliminary results from similar tests performed in France and the results from two in-reactor tests at Sandia National Laboratory confirm these behaviors.

Fractional release rate models, such as CORSOR-M, do not correlate experimental results as well as simple diffusion-based models. Examples of the latter are the ORNL Diffusion Release Model and a similar but expanded model, CORSOR-BOOTH. Both CORSOR-M and CORSOR-BOOTH include release expressions for most of the above-mentioned fission products and for some others based on similarity of chemical properties. Minor or major corrections are needed for most fission product elements in both models based on recent release data.

Fission product release occurs through many simultaneous mechanisms, but CORSOR-BOOTH provides surprisingly good predictive ability for a simple diffusion model. Diffusion models can also be used to predict the vaporization of structural, cladding, and control rod components. A suitable artificial grain size must be used in the diffusion equations for these materials.

Corium Vessel Interaction Studies - Status of the CORVIS Project

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Although they are extremely unlikely to occur, reactor core melting accidents, whether in presently operating or in future plants with nuclear reactors, will always rank at the top of the public interest in reactor safety, since by the laws of physics they produce the largest source terms. Despite possible severe accident management provisions, the course of the core melting accident in which the reactor pressure vessel (RPV) fails will have the worst impact on the environment around nuclear reactors. The safety of present and future reactor plants will always be assessed on the basis of the source terms produced by accidents of this type, even if the estimated probability of occurrence is further reduced by technical improvements and in new constructions.

It is important to be able to show that a RPV failure can be prevented even in the last minute, if only the cooling becomes available again (see the TMI-2 case). Possible interventions (whether automatic or manual) during various core melting processes can significantly reduce the radiotoxic effects on the reactor environment. The OECD project RASPLAV will obtain an experimental data base and will provide recommendations on practical measures to prevent the melt-through of a RPV during a severe accident by water cooling from outside. Nevertheless, it remains an open problem, how a core melting accident would develop, if all possible measures to prevent the melt-through of the RPV fail or are not executed at all. One can envisage both passive (inherent to the construction) and active measures to reduce the source term, which would be effective even after the destruction of the RPV. The integrity of the safety containment (SC) will always be of the utmost importance, since it usually is the last barrier against the release of fission products to the exterior. The SC can fail due to the pressure or to other mechanical loads, static or dynamic; or else it might melt through (erosion of the concrete base material). Therefore, a basic point concerning the determination of the source term is if and for how long the integrity of the SC can be maintained. It depends on the measures actually executed if the SC fulfills its purpose notwithstanding an RPV failure, e.g. cooling of the molten material outside the RPV and removal of the excess heat from the SC. Furthermore, the SC integrity depends on the loading which might occur if the RPV fails under high pressure. In this case, the cross-sectional area of the leakage is an important measure, since it determines the speed at which the pressure builds up inside the SC and governs the duration of shear and pressure forces on the RPV itself. The leakage size also determines if and how molten material might also be ejected, besides the steam.

According to present-day knowledge, core melting under high pressure is more probable than under low pressure. Since severe accident management steps comprise a timely pressure reduction, before the RPV failure, the problem posed by a RPV failure under low pressure remains relevant, if no auxiliary cooling can be provided. The destruction of the core in the interior of the RPV follows the same pattern both in the high as in the low pressure case, up to the evaporation of the remaining water. Thus, due to the pressure, the possible states before the RPV failure only differ by the additional stresses in the RPV material and - if there are penetrations in the bottom - in the weldings for example. The failure pattern is determined by the temperature and stress distributions. After the RPV failure the pressure governs the efflux of material out of the RPV interior. It depends essentially on the failure pattern whether it is possible to cool down the molten material beneath the RPV without endangering the SC. In some core melting scenarios there is pool water under the RPV. It depends on the consistency of the molten material, on its temperature and on its efflux rate whether the material flowing out of the RPV can be cooled; in particular, the efflux rate dominates the ensuing development of the accident. The RPV failure pattern thus also governs the pressure loading of the SC due to evaporation or in extreme cases to possible steam explosions. The same parameters determine success or failure of the cooling even if no pool water is available immediately, but a subsequent flooding is provided for. This is intensively investigated in the MACE project.

It can thus be stated that the prediction of the failure pattern of an RPV at core melting is the crux in the field of source term studies. Models and computations for its validation are direly necessary.

The purpose of the CORVIS (Corium Reactor Vessel Interaction Studies) project is both to prepare experimental data and to develop computational models. With the knowledge obtained from these efforts it will be possible to develop the phenomenology which is necessary to support the current research for a quantitative description of the failure mode of the RPV lower head carrying penetrations. The experiments will be performed under atmospheric pressure in a cylindrical steel container, the bottom of which represents a piece of the RPV bottom carrying typical penetrating tubes to a scale of 1:1. The substitute of the core melt is either the metallic or the oxidic part of a thermite melt or aluminum oxide on top of iron (up to 2000 kg at 2400 °C). A sustained heating with a submerged electric arc prevents the melt from resolidification in the test section. The computational model will simulate the observed phenomena with respect to the heat transfer and the flow pattern in the melt as well as to stresses and deformations of the structure. The purpose of this effort is to produce a model of RPV-failure which can be applied for predictive analyses. Four modes of lower head failure can be distinguished: heat-up and failure of penetration tubes, tube ejection, lower head global failure and jet impingement. The initial conditions of the melt-through experiments simulate a pool of molten corium in the lower plenum of the RPV. The following phenomena which characterize the process of heat-up and failure of the lower head will be quantitatively or qualitatively registered:

Heat-up of the vessel wall and temperature dependent deformations due to the weight of the structures and of the melt; the location of a first melt release from the test vessel, i.e. melt-through of a penetration tube above the welded joints to the vessel, failure of the welded joints or failure of the lower head itself; the type of vessel leakage, i.e. growth of a single hole, formation of further holes or a large break in the vessel wall; the rate and the constitution of the melt released from the vessel; a possible solidification of the melt inside the tubes and tube blockage; a possible solidification of the melt outside the tubes below the vessel which would partially inhibit a fast release into the SC.

The finite element model under present development should be able to take internal pressure into account. This is necessary for the extrapolation of the experimental results to high pressure core melt scenarios.

The paper will describe the CORVIS test facility and will report about the experiments carried out so far. It will also summarize the work done by the international CORVIS Task Force.

CURRENT STATUS AND VALIDATION OF RASPLAV CODE.

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Computer code RASPLAV which was initially developed to model molten core-concrete interactions is being developed to consider the interactions of melt with design materials including the interaction of molten pool with the reactor pressure vessel (RPV). This paper presents some results obtained for the heat transfer problem in heat generating fluids as well as comparison of the results against different types of experimental data.

Introduction. RASPLAV code to model MCCI[1] was successfully used for analysis of several large scale tests[2]. Initially this code allowed to model heat conductivity problem in complex geometry including different design materials. Further development of RASPLAV code to model LHF problem leads to the necessity of the implementation of several new models namely hydrodynamic behavior of the molten pool for applications of the code to analyze heat transfer due to natural convection and hydrodynamic spreading of molten materials on the concrete basemate. Thus to day this code consists of three parts that allow to analyze several phenomena in course of severe accidents namely MCCI, LHF and core spreading[3].

Description of the code. To model interactions of molten fuel with vessel steel the model based on the solution of the 2D set of hydrodynamic equations was developed. Governing equations may be written in the dimensionless form:

$$\frac{\partial \vartheta}{\partial t}+\frac{1}{r}\frac{\partial(ru\vartheta)}{\partial r}+\frac{\partial(v\vartheta)}{\partial z}=\frac{1}{Pr}\left[\ \frac{1}{r}\frac{\partial}{\partial r}\left(r\frac{\partial \vartheta}{\partial r}\right)+\frac{\partial^2}{\partial z^2}\vartheta+1\ \right]$$

$$\frac{\partial \omega}{\partial \tau}+\frac{1}{r}\frac{\partial(u\omega)}{\partial r}+\frac{\partial(v\omega)}{\partial z}=\left[\ \frac{\partial}{\partial r}\frac{1}{r}\left(r\frac{\partial r\omega}{\partial r}\right)+\frac{\partial^2}{\partial z^2}\omega\ \right]-\frac{Ra}{Pr}\frac{\partial \vartheta}{\partial r}$$

$$\frac{\partial}{\partial r}\frac{1}{r}\frac{\partial \psi}{\partial r}+\frac{1}{r}\frac{\partial^2}{\partial z^2}\psi=-\omega$$

$$u=-\frac{1}{r}\frac{\partial \psi}{\partial z}\qquad v=\frac{1}{r}\frac{\partial \psi}{\partial r}$$

Two demensionless parameters Rayleigh and Prandtl numbers characterize heat transfer of circulating fluid. Other parameters which may be introduce present the shape of the cavity and heat exchange from melt boundaries. Typical values of these parameters for reactor case are as follows: $Ra = 10^{14}$-10^{15}, Pr=0.5 for the temperature of about 3100 K. The flow pattern for this value of Ra number is strongly turbulent so special models are needed to analyze flow. Standard k-ε turbulence model was considered for this case.

Different boundary conditions may be defined to complete the problem definition. In the case of crust formation taking into account that the thickness of crust is small compared with the characteristic dimensions of the molten pool we consider heat transfer problem with isothermal boundary conditions assuming that heat flux can be removed from the interface boundaries. In this case only aspect ratio influence heat transfer.

Combination the models of natural convection together with the heat conductivity model in the vessel steel allows to specify and solve the thermal aspects of the lower head failure problem.

Validation of code. There exists relatively large amount of data concerning the problem of natural convection in different geometry and flow regimes. To validate numeric models as well as to check the possibilities it is necessary to describe precisely the boundary layers in different flow regimes as well as bulk convection of molten materials including both laminar and turbulent convection regimes. Thus the following tests were chosen for analysis :

- To check the consistency of the boundary layers description the standard "cold-hot" tests[4] were considered. In this case the heat transfer is mainly due to the properties of the boundary layers thus use of this kind of tests allows to validate the boundary layers description.

- To validate the code in the case of natural convection due to the internal heat generation in the laminar case Kulacki[5] and Mayinger[6] experiments were considered.

- Applications to the reactor case were checked by the use of the results of COPO[7] tests for Rayleigh number about 10^{15}.

Detailed analysis of the natural convection modeling in different situations allows to make a conclusion that reasonable agreement was obtained for all considered cases.

Further development. RASPLAV code seems to be relatively adequate for modeling of the thermal aspects of LHF problem. On the other side the following items should be investigated and corresponding models should be implemented:

- mechanical behavior of the vessel due to high thermal and pressure loads;
- external cooling of the vessel in the case of reflooding of reactor concrete cavity;
- the role of chemical interactions of vessel steel and core materials especially in the case of long term confinement of the melt.

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An Overview of the Ex-Vessel Debris Coolability Issue

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One of the unresolved issues in severe accidents research is whether sufficient heat can be removed from the molten core debris on the containment floor by adding water and thus bringing the debris to a coolable state. The progression of severe accidents that results in core debris on the floor involve in-vessel core melting followed by relocation of molten core into lower head, subsequent failure of reactor vessel pressure boundary, and subsequent discharge of molten debris from the failed vessel. Ultimately, the molten debris spreads on the containment floor and poses a threat to containment integrity by basemat penetration and containment overpressurization. Adding water to molten debris under the circumstances is considered the most practical accident management strategy, the intent being to cool the molten debris sufficiently such that the basemat ablation is arrested. The issue of such long term debris coolability is particularly significant in the context of U.S. Nuclear Regulatory Commission's certification program for advanced light water reactors (ALWR).

NRC sponsored a limited number of small-scale coolability experiments in the mid 1980s at the Sandia National Laboratories. In two tests in the FRAG series, the long term interactions of hot stainless steel shots (simulating initially quenched and fragmented debris) with concrete were examined in the presence of water. In two other tests in the SWISS series, the effect of the timing of water addition on the long term interactions of molten steel (simulating unquenched debris) with concrete was examined. More recently, NRC sponsored the WETCOR program at Sandia under which the effect of an overlying water pool on an oxidic debris interacting with concrete was investigated. The Electric Power Research Institute (EPRI), through the cooperative Advanced Containment Experiments (ACE) program with participation from NRC, U.S. Department of Energy (DOE) and a score of other international agencies, has also sponsored in recent years a series of Melt Attack and Coolability Experiments (MACE) involving prototypic debris materials. The objectives of these experiments are to investigate the conditions for debris coolability and to develop a data base for analytical modeling. Most recently, the Japan Atomic Energy Research Institute (JAERI) has performed a series of coolability tests with thermite simulant on a magnesium oxide basemat (simulating a non-reactive containment floor) at a scale smaller than that of WETCOR and MACE tests. This paper gives an overview of the debris coolability research (both past and ongoing activities) mentioned above, an account of progress made in the understanding of the ex-vessel debris coolability phenomenon, knowledge gained, and recommendations for future research.

ABSTRACT

Simulator Benchmarking Studies for ATWS Scenarios.

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There is much interest in the nuclear industry concerning the ability of training simulators to adequately model severe accident conditions, specifically ATWS events. The Pennsylvania Power and Light Co. has recently installed a new simulator which was provided by S3 Technologies. As part of the licensed operator training program, PP&L provides training on Emergency Operating Procedures (EOPs). Since the ATWS event is challenging from both a computational and operational point of view, the Engineering Department was asked to benchmark the new simulator performance. The purpose of this benchmark was to ensure simulator fidelity with EOP basis calculations which are numerically more rigorous. Once acceptable simulator fidelity had been demonstrated, EOPs were evaluated to ensure they could be implemented by the operators.

This paper examines the details of the new simulator response for ATWS events, and exposes the PP&L ATWS procedures to further examination. The simulator benchmark was carried out using the PP&L developed SABRE[1] code. For many ATWS scenarios, the new simulator, which is based upon first principles, provides predictions consistent with SABRE. Reactor power levels, consistent with SABRE results, are significantly higher than predicted by the old simulator, and containment pressurization occurs much more rapidly than previously simulated. Additionally, the new simulated reactor water level, pressure and power are far more responsive to perturbations than predicted by the old simulator. This responsiveness is consistent with SABRE predictions and has helped to define modifications to the ATWS emergency operating procedures. The modified procedures enhance the operators ability to respond to ATWS given the much more realistic reactor model. In other ATWS scenarios, inconsistencies between SABRE and the new simulator were discovered. As a result of these inconsistencies, areas of additional investigation for benchmarking are identified in order to provide a more complete assessment of simulator capability for ATWS events.

This effort was very beneficial to the company. It brought together the expertise of operators, trainers and engineers. As a result of this cooperative effort, we are confident that the new simulator provides realistic representation of the ATWS event and thereby provides an optimal platform to test PP&L's ATWS strategy and operator proficiency.

[1]The SABRE code has been benchmarked to RETRAN and TRACG results.

Irradiated Fuel Behaviour During Reactivity Initiated Accidents in LWR's: Status of Research and Development Studies in France

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A new emphasis has been given in FRANCE to the study of reactivity initiated accidents in Light Water Reactors after the Chernobyl accident and in the frame of the future increase of mean fuel burn-up to 52 GWd/t linked to the implementation of a new fuel management system.

In particular, in the case of the design basis accident of control rod ejection, new safety evaluation is needed to check the validity of the criteria presently used for irradiated fuel and which are the absence of fuel dispersion and mechanical energy release for a maximum fuel enthalpy of 200 cal/g together with a molten fuel volume lower than 10% and a maximum clad temperature of 1482°C.

Indeed, these criteria have been established based on SPERT experiments using fresh and low burn-up irradiated fuel (only two tests at 30 GWd/t) and in the range of such high burn-up as foreseen, experimental basis and knowledge are still lacking.

We discuss in this paper the specific aspects of high irradiation level on the behaviour of fuel rods submitted to a reactivity initiated accident (RIA) taking into account the main outcomes from the experiments already performed and we present the status of the current French Research and Development studies.

PHEBUS-FP: Analysis Program and Results of Thermal Hydraulic Tests

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The Phebus-FP experimental programme is an ambitious one. It aims to degrade a bundle of 20 short but otherwise representative PWR fuel rods plus a control rod in the Phebus reactor, and to pass the fission products and structural materials released in the process through a model circuit into a volume representing the reactor containment on a volumetric scale of 1/5000, for the study of aerosol physics and iodine chemistry. The first experiment, FPTO, will take place in Autumn this year. The analytical preparation of the experiments is also an ambitious undertaking. Led by teams from CEA and the CEC-JRC, participants from all countries taking part in Phebus (including USNRC) have contributed to calculational benchmarks, analytical investigations of the behavior of the various components of the Phebus experimental system and scaling and dimensioning studies to define the overall layout, the geometry of FPTO, and the detailed boundary conditions (including neutronic power and steam flow histories, containment component temperatures, sump pH etc) for that test. Intensive work is now in progress to define in the same way the geometry and conditions of the remaining five tests, stimulated in part by a recent proposed revision of the test matrix which puts more emphasis on the conditions and releases associated with the later phases of a severe accident. This paper describes the current status of Phebus preparation, emphasizing the problems arising with the later tests and how they are being addressed.

In preparation for the tests a series of thermal-hydraulics tests have been performed in which steam is injected into the containment vessel in a series of steady states. Before these tests a series of calculations were performed by various teams using codes that are currently used for reactor calculations (CONTAIN, MELCOR, JERICHO etc.) and some that have been specially written for the Phebus tests. A preliminary comparison of these double blind calculations with the experimental results in presented in this paper as an example of the process of test preparation through calculation.

A REVIEW OF POTENTIAL USES FOR FIBER OPTIC SENSORS IN NUCLEAR POWER PLANTS, WITH ATTENDANT BENEFITS IN PLANT SAFETY AND OPERATIONAL EFFICIENCY*

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Fiber-optic-based sensing has a wide range of potential applications in nuclear power plants, and a fiber optic analog now exists for virtually every conventional nuclear power plant sensing system. Fiber-optic-based sensors are likely to eventually supplant many conventional sensors due to their inherent advantages (reduced mass and size, ruggedness to vibration and shock, physical flexibility, high sensitivity, electrical isolation, EMI immunity, resistance to high temperatures, reduced calibration requirements, passive operation, and resistance to radiation). In addition, fiber-optic-based sensors exist which are capable of measuring reactor and performance parameters that cannot be measured by conventional means (in high EM fields, in-core, and distributed measurements).

However, fiber optic sensors remain at too low a level of development for immediate application in safety-critical systems. Moreover, fiber optic sensors have different failure modes and mechanisms than conventional sensors and therefore considerable regulatory research will be necessary to establish the technical basis for the use of fiber optic sensors in safety-critical systems.

Since fiber optic sensors rely upon developed fiber optic communication components, they have only been introduced as the fiber optic communication industry has matured. These advanced fiber optic sensors range in state of development from conceptualization to small-scale commercialization, with none yet having been ruggedized sufficiently for use in safety-critical systems.

Temperature is a key process parameter throughout nuclear power plants. While conventional temperature sensors have been extensively studied and in general perform quite well, they suffer from a variety of limitations: need for calibration, unwanted sensitivity to EMI and radiation, single-point measurement, slow response time, and need for isolation from the process fluid. A fiber optic temperature sensor exists in some state of development based upon virtually every known thermo-optical property. The temperature-dependent optical properties that have been seriously considered to date include changes in: fluorescent properties; optical absorption, reflection, and scattering; optical path length (resulting in interference); birefringence; thermally generated radiation (blackbody radiation); and fiber light loss due to thermally induced microbending.

Pressure is likewise a fundamental process parameter that requires accurate and reliable measurement throughout nuclear power plants. Conventional pressure sensors have been studied extensively, yet significant areas remain for performance enhancement. Many conventional pressure sensors use oil to buffer the gauge mechanism from the process fluids and are subject to insidious failures if an oil leak occurs. Also, conventional pressure sensor performance verification and calibration require significant periodic effort. Moreover, the accuracy provided by conventional pressure sensors during accident conditions is limited, and they provide only a single-point pressure measurement. Fiber optic-based pressure sensors have the potential for overcoming all of these limitations. Fiber optic pressure sensors being seriously considered for industrial applications are based on: diaphragm deflection (both intrinsic and extrinsic), Fabry-Perot interferometry, Mach-Zehnder interferometry, and piezoluminescence.

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Radiation measurements are another area of potential usefulness for fiber-optic-based sensors. Optical fibers are well suited to transmit scintillator light back from a harsh process environment to the control room environment, where sensitive detection electronics can survive. Fiber-optic-based radiation measurements can also be made in a spatially distributed manner. One possible arrangement would be to segment an optical fiber by interleaving small scintillator pieces within an otherwise continuous fiber.

Fluid level measurement is required throughout the primary and secondary coolant systems, with several of the measurements being safety-critical. One method to determine fluid presence is to couple the output light from a fiber segment into a prism. Using the proper prism angle and material, the light is totally internally reflected when the prism is in air and lost into the liquid when the prism is submerged. Other, more advanced, continuous liquid level monitors have also been proposed.

The status of high-voltage electrical components is difficult to determine electrically, due to the high levels of electromagnetic interference present. Fiber optic current and voltage measurements have been under investigation for over a decade and a wide variety of electro-optic and magneto-optic based devices have been proposed.

A wide spectrum of *chemical measurements* are made at nuclear power plants (water chemistry, effluent monitoring, hydrogen in containment, etc.). Fiber optic chemical sensors are commercially available from several manufacturers. In these sensors fibers are used to enable remote fluorescence, absorption, index-of-refraction, and spectroscopy measurements. Fiber optic chemical sensing is winning rapid acceptance in the medical, industrial process, and environmental fields.

Vibration measurements can be implemented optically and non-invasively. Pointing an optical fiber bundle at a reflective surface on the monitored equipment, illuminating the surface with fibers from the bundle center, and measuring the time and spatial location of the light back-coupled into the surrounding receiver fibers yields an equipment vibration spectrum.

Improved *strain measurements* on high-pressure components and welds in nuclear power plants would be of significant benefit since strain increase may provide an indication of incipient cracking and failure. Three different optical interferometric approaches to strain measurement appear to be particularly promising. Sagnac, Mach-Zehnder, and Fabry-Perot based strain measurement show great promise; however, all still require considerable development effort before being ready to install in safety-critical applications.

Several different fiber-optic-based measurements schemes have also been proposed for *flow measurements, position measurements, and leak detection.* These advanced fiber optic sensors range in state of development from conceptualization to small-scale commercialization,with none yet having been ruggedized sufficiently for use in safety-critical systems.

Engineering the Development of Optical Fiber Sensors for Adverse Environments

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Optical fiber sensors offer immunity to electromagnetic interference and inherent electrical isolation which give them many advantages over their electromechanical counterparts where noise, high voltage, and ground loops are problems. Moreover, these sensors may be installed in previously inaccessible areas because of their relatively small size and small, flexible connecting fiber. Thus they have significant potential for use in harsh environments, high–speed rotating machinery, biomedicine and other applications that require remote sensing.

Over the last decade, many sensors have been developed for particular applications in these areas with limited success. Off–the–shelf optical fiber sensors and measurement systems are not available, partly because they have not been engineered to meet the environmental requirements necessary for applications outside the laboratory. Furthermore, no generalized computer–aided tools exist to help advance their development, design, and use.

Such computer–aided tools are currently being developed. Structural finite element analyses have been coupled with optoelastic analyses of both all–fiber interferometers and serial microbend sensors for distributed measurement of various physical quantities. The combined analyses have been parameterized and implemented on microcomputers for use as design/development tools that can be used to determine the performance of different sensor configurations in various environments. Potentially, these computer–aided design/development tools could be used for failure diagnosis and redesign of existing optical fiber sensors. Performances predicted by the computer simulations are verified with experimental data and numerical analyses from the literature. In addition, current experimental work to evaluate performance predictions, determine appropriate signal processing, and identify key issues for application in high temperature, radiation environments is presented. The long–term goal is to develop user–friendly software packages for both sensor manufacturers and end users.

On-Line Calibration Monitoring for Instrumentation Channels in Nuclear Power Plants

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ABSTRACT

This paper reports on the progress of a research and development (R&D) project which has been underway since October 1991 to determine the validity and establish the accuracy of on-line monitoring techniques for detection of any significant calibration drift in the instrumentation channels of nuclear power plants. A feasibility study has been completed under a Phase I project and the results have been documented in NUREG/CR-5903 published in January 1993. A comprehensive Phase II project is underway and is due for completion in the fall of 1994.

The Phase II project includes both laboratory and in-plant validation work on typical nuclear plant sensor and signal conversion and signal conditioning equipment. The laboratory validation tests are being performed in a test loop in which a number of nuclear grade temperature and pressure sensors have been installed. They are connected to a Westinghouse Model 7300 instrumentation system of the type used in Pressurized Water Reactors (PWRs). The study includes the comparison of data from empirical and physical models developed as a part of this project with data measured in the test loop. The loop is used to determine if simulated drift in the sensors can be effectively detected by on-line monitoring methods. It is also used to perform verification and validation of the on-line monitoring software packages being developed for the commercial aspects of this project.

The in-plant validation work is a joint effort with Duke Power Company and is being performed at the McGuire Nuclear Power Station where 170 process signals are continuously being monitored. These signals include the primary coolant RTDs, core exit thermocouples, neutron flux detectors, the reactor vessel level indicating system (RVLIS) and pressure, level and flow transmitters. The signals are analyzed to identify any channels that may have drifted out of tolerance. After the completion of data acquisition for a complete fuel cycle, the results of this work will be compared with the results of the conventional hands-on calibrations that are normally performed during refueling outages. This comparison will help determine if on-line monitoring techniques can identify the same drift that is detected by conventional calibration methods.

The results of the laboratory and in-plant research completed to date have shown that on-line drift monitoring techniques will be successful in identifying the instrumentation channels that must be calibrated. This will help limit the calibration activities to those channels which need to be calibrated, as opposed to calibrating all safety-related channels, as is presently being done.

RELIABILITY ISSUES ASSOCIATED WITH THE USE OF MICROPROCESSOR-BASED PROTECTION SYSTEM HARDWARE IN NUCLEAR POWER PLANTS*

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In some countries, digital technology has been utilized in nuclear power plant control and protection systems for more than a decade. However, extensive use of microprocessor-based technology, multiplexing, and fiber optic transmission—as exemplified in proposed protection systems for light-water reactors of advanced design—has fostered renewed interest in the reliability of such systems and technologies when applied to reactor protection systems.

The U.S. Nuclear Regulatory Commission (NRC) initiated the Qualification of Advanced Instrumentation and Control Systems Program in late 1991 at Oak Ridge National Laboratory to develop an understanding of the technical issues involved in evaluating the long-term performance of such "advanced" technologies when applied to the protection systems of commercial nuclear power plants. Initial studies focused on developing evaluation templates for proposed advanced light-water reactor (ALWR) designs, which included a comparison of the impact of environmental stressors on protection system strings in present-day vs proposed reactor designs.

In this paper, generic analog and microprocessor-based reactor protection system designs are examined with respect to reliability and fault-tolerance issues. Both the nuclear industry and

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regulatory bodies accept the analog hardware implementations of protection systems employed in existing LWRs. Over the years, considerable experience with their use has accumulated and a reasonable understanding of both normal operation and failure modes for such systems has developed. In contrast, microprocessor-based systems are expected to have failure modes that are distinctly different from functionally equivalent analog counterparts. This paper critically examines such issues as transient failures, reliability of communications among protection system channels, and the potential for common-mode failures. Also, reliability assessment methods currently available for application to microprocessor-based hardware will be discussed. The results of the study are one element of a proposed methodology for the qualification of microprocessor-based protection systems being developed under this program.

Requirements for Dependable and Cost-Effective Implementation of Digital I&C Systems

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Instrumentation and Control (I&C) systems in nuclear power plants need to be upgraded in a reliable and cost-effective manner. The primary impetus is to gain maximum advantage from the application of digital technology in support of the utilities' overall drive to improve competitiveness. The majority of nuclear plants in the United States are operating with hardware that is no longer fully supported by the original equipment manufacturer. Thus, the procurement of replacement modules and spares under current requirements is costly, time consuming, and, in some cases, not even possible. Replacing analog with digital systems provides the opportunity to decrease operations and maintenance costs, while enhancing safety.

The EPRI Instrumentation and Control Upgrade Initiative is designed to help utilities upgrade the I&C systems in their plants. The goals of this initiative are to: (i) Develop a methodology to formulate and subsequently implement a plant-specific, integrated I&C upgrade plan; (ii) demonstrate the methodology through utility application to at least ten key systems by the year 2000; and (iii) document that the upgrades lead to some or all of the followings: reduced vulnerability to I&C obsolescence; increased plant performance; reduction in operating and maintenance costs; lower radiation exposure; and enhanced safety. To support these goals, EPRI is taking a three pronged approach. The three prongs consist of research and development activities, demonstration plant activities, and licensing stabilization activities. The research and development activities support the development and implementation of digital systems as well as providing a technical basis for qualification and licensing questions. It also provides part of the bases for the requirements and methodologies needed to design, develop, qualify, implement, and operate digital systems. The demonstration plant activities identify utility's needs, provide part of the bases for requirements and methodologies, provide a testbed for and feedback on requirements and methodologies for upgrade systems, and capture experience from implementing new digital systems. The licensing stabilization activities provide technical support, as requested, for the industry licensing positions on digital systems which are being developed by utility working groups and committees.

The existing hardware technology in the plants reflects 20 to 30 year-old designs, components, and manufacturing techniques. Technological improvements, particularly the availability of digital systems, offer improved functionality, performance, and reliability, solutions to obsolescence of equipment, and reduction in operation and maintenance costs. Reliable and cost-effective methodologies for designing, qualifying, and implementing digital systems in nuclear power plants are urgently needed. These methodologies should utilize, as much as possible, relevant information and experience from other process industries. Commercial-grade hardware and software have proven reliable in other process industries for applications including safety systems. Cost-effective approaches are needed to implement and qualify commercial-grade hardware and software for nuclear power plant applications.

The test for future digital system upgrades will be whether they are cost beneficial. Digital systems with their inherent advantages will be adopted in nuclear power plants only if they to support reduced power production costs. This can be done if (i) dependable and cost-effective implementation and (ii) licensing acceptance are achieved.

A DYNAMIC FAIL-SAFE APPROACH TO THE DESIGN OF COMPUTER-BASED SAFETY SYSTEMS

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ABSTRACT

For over 30 years AEA Technology has carried out research and development in the UK in the field of nuclear instrumentation and protection systems. Throughout the course of this extensive period of research and development the dominant theme has been the achievement of fully fail-safe designs. These are defined as designs in which the failure of any single component within a unit will result in the unit output reverting to a demand for trip action status.

At an early stage it was recognized that the use of dynamic rather than static logic could ease the difficulties inherent in achieving a fail-safe design protection system. A dynamic logic system is one in which an output signal alternating between logic 1 and logic 0 is a healthy state and a static output either logic 1 or logic 0 is a tripped state. The alternating state is a higher energy system than the static one. Since faults tend to move the system to a lower energy state the use of dynamic logic is preferred when designing fail-safe systems.

The first dynamic logic systems coupled logic elements magnetically. Early examples used magnetic amplifiers, later systems used coils magnetically linked through a common core. Such a configuration can be used as a 2 out of 3 voting element. Further developments have led to the application of the approach to solid-state hardware systems. By using coded pulse strings passing through the solid-state logic elements, systems have been constructed which are inherently self-testing and fail-safe in operation. An added feature of using coded pulse strings is that it provides a powerful diagnostic capability.

This proven and successful approach has also been adapted to computer-based systems. It has been shown that the input data stream to the safety system can be modified to produce at the output a unique pattern of dynamically alternating logic 1s and 0s. This unique pattern can then be confirmed to be correct by a hard-wire verifier. Detection of an incorrect pattern or failure of the verifier itself results in a demand for trip action.

Currently, AEA Technology is collaborating with Ohio State University in an EPRI R&D programme on Dynamic Safety Systems. One aim of this work is to research whether the dynamic safety principles can be applied to input sensors.

AEA Technology and Duke Power Co. are also collaborating on an ISAT™ demonstration at Duke's Oconee Nuclear Station. Duke and AEA plan to install the ISAT™ dynamic safety system approach in the control interface portion of the Oconee Reactor Protection System (RPS). ISAT™ will replicate a complete protection channel to demonstrate the dynamic safety approach in an operating Pressurized Water Reactor (PWR) environment. This installation is planned for the Spring 1994 refuelling outage for Oconee Unit 1.

AN EXAMINATION OF HUMAN FACTORS IN EXTERNAL BEAM RADIATION THERAPY: FINDINGS AND IMPLICATIONS

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External beam radiation therapy (or teletherapy) is a multi-disciplinary, multi-phased treatment methodology for treating cancerous and other tissue through selective exposure to a beam of ionizing radiation delivered from a source external to the patient. A radioactive isotope, typically cobalt-60, or a linear accelerator capable of producing very high energy x-ray and electron beams are the principal sources of radiation. Treatment typically takes place on a daily basis in fractional doses over a period of weeks and is planned and administered by a team of specialists, including a radiation oncologist, radiation physicist, dosimetrist, and radiation therapy technologists. Effective treatment requires a concern for precision and consistency of human-human and human-machine interactions throughout the duration of therapy. Records maintained by the U.S. Nuclear Regulatory Commission (NRC) have identified instances of teletherapy misadministration where the delivered radiation dose has differed from the radiation prescription (e.g., instances where fractions were delivered to the wrong patient, to the wrong body part, or were too great or too little with respect to the defined treatment volume). Both human error and machine malfunction have led to misadministrations. Misadministration above the prescribed dose runs the risk of destroying healthy tissue and organs; misadministration below the prescribed level can result in ineffective treatment. Either way, the consequences of misadministration can be life threatening.

The present paper reports on a series of human factor evaluations sponsored by NRC's Office of Nuclear Regulatory Research to identify the factors that contribute to misadministration in the radiation therapy environment. The six major parts of the overall study include: 1) a function and task analysis of the teletherapy activities, 2) evaluation of human-system interfaces 3) evaluation of the procedures used by teletherapy staff, 4) evaluation of the qualifications and training of teletherapy staff, 5) evaluation of organizational practices and policies, and 6) identification of human factors priority areas for NRC and industry attention.

With respect to the study's methodology, a site sampling strategy for visiting departments of radiation oncology within the continental U S. was developed to insure geographic dispersion and representation of different types of facilities (e.g., university-based centers, large community hospitals, and smaller free standing or satellite facilities) to accommodate differences in treatment practices, management style, personnel, patient load as well as other factors likely to vary among facility sites. Efforts also were made to focus on centers with Cobalt-60 units because of NRC's by-product regulatory responsibilities. Interviews were conducted at 26 sites throughout the U.S. with radiation physicists, dosimetrists, radiation ocologists, chief technologists, staff technologists, training coordinators, and administrative personnel. The interviews were supplemented with observations of on-going treatments, examination of equipment controls and displays, and a review of the radiation oncology literature.

To summarize and make sense of the findings, the paper introduces a model for identifying and showing the relationships among the major contributing factors and for identifying the individual factors in each category that are likely to influence the occurrence of misadministrations. Findings are discussed within the context of the major factors of the model, which includes individual variables, task variables or the nature of the work itself, the physical environment, the user-system interfaces, the organizational/social environment, and managerial factors.

Individual variables include knowledge, skill level, adequacy of training, and even organismic states such as alertness, motivation, physical capabilities and fatigue, while task variables refer to the characteristics of work itself such as patient load or scheduling, presence/absence of a co-worker, perceptual-motor requirements, and competing tasks. The physical environment pertains to a host of factors including the physical layout of equipment and workstations, the flow of essential information, and ambient factors such as temperature, illumination, noise, hazards and upkeep/attractiveness of the facilities. The likely impact that these factors have on human performance in the teletherapy environment is discussed. The extent to which the design of various user-system interfaces (e.g., equipment controls and displays, software control features, patient charts) impede or facilitate human performance also is addressed. Recent improvements in treatment delivery are examined. For example, how successful are computerized verify and record systems in preventing errors that might otherwise go undetected? Given that the current direction towards software control of treatment machines and treatment planning systems is dramatically changing the nature of the user-system interface, the benefits and challenges that face users of computer-controlled systems warrant our close attention. Also receiving close scrutiny are human performance problems that can be traced to organizational/social factors such as poor communication practices and organizational climates which tend to inhibit the self-reporting of errors. The pervasive influence of managerial policy and practices in relation to human performance and human error is becoming better understood in recent years. The paper makes the case that managerial factors are frequently less easy to discern because of their delayed and dormant consequences; however, managerial dictum regarding staffing, workload, patient scheduling, accessibility of personnel, and quality assurance procedures still have their impact. Finally, potential interventions for correcting deficiencies where they exist are identified.

Human Error in Remote Afterloading Brachytherapy

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Introduction

Remote Afterloading Brachytherapy (RAB) is a medical process used in the treatment of cancer which uses a computer-controlled device to remotely insert and remove radioactive sources close to a target area (or tumor) in the body. RAB allows precise and repeatable control of source placement with minimum radiation exposure to medical staff. However, some RAB treatment planning and treatment delivery problems affecting the radiation dose to the patient have been reported and have been attributed to human error. The consequences of these events can be severe. The U.S. Nuclear Regulatory Commission has sponsored this study to determine the root cause of human error in the RAB system.

Approach

As part of the study, a human factors team visited 23 RAB treatment sites in the U.S. The team performed an analysis of the RAB system to identify the functions and tasks involved in RAB and analyzed the performance requirements placed on users in carrying out RAB functions and tasks. The function and task Analysis also identified information, control capabilities, step-by-step directions, knowledge, skills, ergonomic requirements and environmental conditions needed to support the required level of task performance. The results of this analysis provided a framework for follow-on evaluations to determine availability and suitability (based on human factors standards and guidelines) of human-system interfaces, procedures and practices, training, and organizational policies in RAB. These evaluations included on-site observations of RAB treatment planning and delivery, interviews with RAB personnel, and walk-throughs, during which staff demonstrated their interfaces with the equipment and the procedures and practices used in performing RAB tasks. Results of these evaluations, in conjunction with the function and task analysis, were used to identify human factors problems, i.e., the factors which lead to human error in the RAB system. The research analysts then evaluated the impact of those factors on the performance of functions and tasks and prioritized them in terms of safety significance. Finally, the project identified and evaluated alternative approaches for resolving the safety significant problems related to human error.

Results

The study identified 55 opportunities for error in performing tasks and transmitting information between the humans and the machines in this system. Many of these error opportunities were compounded by poor feedback between the machines and the users so that errors were difficult to correct, and in some cases impossible to detect. Significant error opportunities identified included:

- Patient misidentification
- Radioactive source positioning and movement error
- Patient record-keeping entry error
- Treatment plan entry error
- Source transport or guide tube connection error
- Quality assurance performance error
- Radioactive source calibration and / or decay calculation error

A primary objective of the project was to identify mismatches between RAB human performance requirements and what people can reasonably be expected to do. Therefore, the project developed twenty-six alternative approaches for resolving safety significant problems and reducing human error. These approaches addressed changes in RAB hardware, software, human-system interfaces, procedures, training, organization and communications processes and included:

- Modified identification labels on patients and records
- Improved methods for tagging of equipment and supplies
- Interface changes to increase feedback to task performers
- Standardized communication paths between staff
- Changes in task procedures and in the allocation of tasks
- Improvement in intramural training provided to staff
- Changes in calibration and Q/A procedures

Conclusion

The systematic approach to evaluating RAB systems allowed the analysts to identify human factors problems and root causes of human error. Evaluations of the system's HSI, procedures, training, and organization against human factors standards and guidelines, combined with the function and tasks analysis, enabled the analysts to identify and evaluate alternative approaches for resolving the human factors problems and thereby reduce human error in the RAB system.

Human Factors Issues in Severe Accident Management: Training for Decision Making under Stress

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Summary

Training for operator and other technical positions in the nuclear power industry traditionally has focused on mastery of the formal procedures used to control plant systems and processes. However, there is a growing awareness that the decision-making tasks required for nuclear power plant control also involve cognitive skills (e.g., situation assessment). These cognitive skills can make nuclear power plant control more efficient and reduce the potential for error. The need for cognitive skills is even more clear in situations where formal procedures may not exist or may not be as prescriptive, as is the case in severe accident management (SAM).

Westinghouse STC, under contract to the NRC, investigated the potential cognitive demands of SAM on the control room operators and Technical Support Center staff who will be most involved in the selection and execution of severe accident control actions. This analysis was followed by analysis of the cognitive skills that could be developed through training and an identification of effective training approaches.

Initially, a model of decision making was developed to identify the types of cognitive skills that may be required for effective performance. The model is organized around six general cognitive processes:

- Monitor / Detect - passive (detect) and active (monitor) means for acquiring data about plant state.
- Interpret Current State - the development of a mental representation of plant status.
- Determine Implications - the determination of how the current plant state will progress (e.g., potential consequences, side effects). Also, a set of goals is defined in which more important goals are given higher priority and complex goals may be broken down into subgoals.
- Plan - the selection of a response plan, which could be a high-level action or formal procedure, that addresses the goal(s) with the highest priority.
- Control - the coordination and execution of a specific sequence of control actions.
- Feedback - the information gained from control actions is used to update understanding.

This model identifies, at a high level of description, specific cognitive skills that may be required for SAM. This model also aided the identification of potential human factors issues in SAM, including:

- Decision-making authority may be shared by the control room and Technical Support Center requiring greater reliance on communication.
- Reliable and accurate plant state data may be more difficult to obtain.
- SAM guidance documents may not be prescriptive to the same degree as current emergency procedures.

- Uncertainty in decision making may be increased.
- Stress levels on decision makers may be increased.

The analysis supported by the model was supplemented by an analysis of specific SAM situations. Specifically, 12 SAM scenarios were developed to reveal more specific decision-making difficulties. These scenarios were created to be presented to panels of plant personnel in order to examine their potential response strategies (i.e., How would crews deal with the types of difficulties that are likely to occur in SAM?).

After a set of cognitive skills was identified, approaches for training cognitive skills or facilitating the training of cognitive skills were identified. Training approaches that applied both to individuals and teams of individuals were identified. Training approaches were identified by reviewing the cognitive psychology and training literature to find approaches for training similar cognitive skills. For example, training for complex electronics troubleshooting skills addresses many of the same types of cognitive skill required for SAM. Specifically, training approaches were identified that address the following elements of cognitive skill:

- Extensive knowledge (e.g., validity of plant state indications, likely SAM phenomena).
- An accurate representation of the plant, including the interconnections between systems.
- An accurate representation of likely physical phenomena and their progressions.
- An understanding of SAM goals, subgoals, and the strategies or HLAs that can be used to achieve goals.
- Metacognitive skills to manage not only the diagnosis and selection of appropriate actions, but also to manage the response of distributed personnel.

Review of these approaches revealed that the following general characteristics have been important in effective training of cognitive skills:

1. A model of skilled or expert performance is used as a model and as a diagnostic aid in training.
2. Trainees become involved in evaluating their performance (or the performance of others) using as a standard the model of skilled performance.
3. Trainees actively engage in the task as a setting for instruction.
4. Trainee performance is supported by the instructor to make it possible to perform the complete task and to see how individual skills must be integrated.
5. Trainees are aided in managing mental workload throughout training.

In a separate analysis, the likely effects of stress on the performance of cognitive tasks were identified through a literature review. The types of performance impairments due to stress included:

- A narrowing and shift in attentional focus.
- A reduced working memory capacity.
- Speed-accuracy trade-offs in some decision-making tasks.
- Crew members are less likely to influence the crew leader.

As a follow-on to this analysis, training techniques were identified for eliminating or mitigating the effects of stress on the performance of cognitive skills. The primary techniques that have the potential to be effective are the following:

- Expose the crew to realistic emergency and severe accident events through simulation.
- Reduce the need for mental resources and make processing more efficient.
- Enhance crew communication and coordination skills.

Organization and Management
Activities in the Nuclear Power Industry

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The purpose of organization and management activities in the commercial nuclear power industry is to foster high levels of power plant performance and safety through improved human performance. The NRC has been working to develop assessment tools to assay the effects of organizational factors on plant safety. Using their own evaluation methods, the utility industry has been working on ways to improve individual accountability and other factors important to human performance.

Organization and management activities do not focus on industry organizational charts, but on the personnel processes and dimensions (factors) that affect safety and economic performance. As individual terms these activities are often combined and referred to as organizational factors. As an area of study, organizational factors has become more prominent as the industry emphasis has switched in recent years from hardware issues related to safety and economics, to personnel issues related to the improvement of human performance.

Activities to improve human performance are carried out by the Institute of Nuclear Power Operations (INPO) and individual companies. The INPO activities in the form of programs include the Human Performance Enhancement System (HPES), the development of good practices (which amounts to the sharing of techniques), operating experience sharing, and the Organizational and Administration (O&A) assessments that are carried out as part of INPO's periodic evaluations of individual site performance.

Individual utility activities, which target organizational factors to enhance human performance, include the following:

- <u>Bench Marking</u> - Utilities bench mark their processes and performance against other utilities and industries with proven successful operations.

- <u>Process Mapping</u> - The mapping out, in logic diagram form, a complex written procedure to enhance its visualization, simplification and process improvement.

- <u>Monitoring human error rate</u> - An activity to trend human performance problems.

- Root Cause Analysis - Root cause analysis is fundamental to eliminating recurring human performance problems.

- Organizational Development - Organizational development is a company-wide process of data collection, diagnosis, action planning, intervention and evaluation aimed at:

  - re-aligning organizational components
  - developing new solutions to old problems
  - developing the organization's ability to renew itself

- Quality Management - Examples include Total Quality Management (TQM), teamwork and leadership, professionalism, empowerment, selection, assessment, training and development, performance evaluation and succession planning.

Analysis of the TMI-2 accident demonstrated the need to develop highly effective organizations. Industry programs in support of effective organizations began in earnest in the mid-1980's. The increase in programs which affect the quality of organizational performance in the nuclear industry over the past seven years indicates that this process is a rapidly developing, evolutionary activity.

The nuclear utility industry in the U.S., like the manufacturing industry internationally, recognizes that product or performance quality cannot be inspected in. Quality in personnel performance or product development is achieved most effectively, and in a more timely and productive manner, when it is built into day-to-day operations. The challenge for each nuclear organization has been to establish and cultivate principles that integrate quality objectives into daily work activities at the organization and individual levels. Line organization components are viewed as the key to quality performance. This accounts for the increase in industry programs geared toward enhancing work group effectiveness. Strong industrywide support for these quality-enhancing programs is essential to ensure the nuclear utility industry maintains its viability in the nations' energy mix as it produces electricity safety and cost effectively.

Potential Human Factors Research
Relating to Modern Technology in Nuclear Power Plants

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Summary

The technology shift in the '70s and '80s away from analog systems toward digital instrumentation, computing, and display offers substantial opportunities for designing reliable, economical, usable, and effective human-machine interface (HMI) applications for nuclear power plants. Some of the equipment in existing plants has become difficult and costly to maintain for want of replacement parts. New equipment and new plant design concepts are based solidly on modern digital technology. If we effectively pursue the opportunity that this provides, we should be able to give control room and workstation operators, supervisors, maintainers, engineers, planners, and schedulers all of the information needed in the appropriate form to effectively manage their workloads and direct plant activities. There can be greater awareness of the state of plant processes; and all users can be fully supported in decision-making and problem-solving tasks.

There is a clear opportunity to improve crew and system performance. The technology is available. The need is to understand how best to use it. This includes gaining a clear understanding of the specific benefits we hope to obtain and problems we expect to resolve by applying it. It also includes avoiding new problems that may be introduced, and determining the means by which we can assess system performance and effectiveness.

Since TMI, EPRI has provided a wealth of human factors guidelines on topics ranging from control room and display system design to current work on annunciator system specifications and alarm minimization. Requirements have been developed for designing new systems in the Advanced Light Water Reactor (ALWR) program and for updating existing systems in the Integrated Instrumentation and Control Upgrade Initiative, both of which rely heavily on modern technology. In the human factors area the next step is to determine the specific research needed to support the advanced system designs and to establish a rational, prioritized plan for so doing. This is the objective of the subject HMI research plan.

EPRI's contractor, MPR Associates, and its subcontractors are developing the first iteration of a prioritized HMI research plan that will indicate needs, risks, benefits, and where feasible, performance criteria and potential performance measures. An important part of this work is to avoid unnecessary duplication by determining lessons learned from completed research and from implementations of advanced technology and design concepts.

The following are examples of suggested high priority research needs that have been identified. As general topics most of these have a familiar ring. Virtually all have been mentioned in a variety of forums as genuine research issues.

- Improved support for knowledge-based behavior
- Information access and display navigation
- Organization and structure of information to support users and tasks
- Intelligent display - control coupling
- Integration of information displays with procedures
- Integration of automation with procedures
- Operator aiding

The differences here are twofold. The first obviously centers on the above shift in technology which provides a clear opportunity, if not demand for change. Secondly, and more subtly, it involves depth and scope considerations. This includes recognized needs for improvement that are based on years of related research that has not reached full fruition. For example, EPRI's continuing work on annunciator systems has identified problems with existing systems that need attention; as well as opportunities for growth that involve proven concepts in alarm diagnostics and minimization. The culmination of this line of research will be to integrate displayed procedures and operator aiding with alarm conditions for each mode of plant operation. Once the components are defined, an integrated display system can provide operators and others with situation awareness, and can provide automatic or semiautomatic techniques to aid in task and workload demands.

Each of the above general topics subsumes many difficult research questions and issues, some of which are basic, if not altogether new. For example - How can large data bases be organized to support the cognitive styles and information processing needs of users? How many displays are needed and of what types? How do we avoid overburdening operators in accessing large quantities of information? What is the best use of automation and means of keeping operators in the loop? And what should be done with the control room environment to facilitate both CRT usage and operator alertness?

EPRI intends to explore the first iteration of the plan with the research community in order to better identify high priority issues needing immediate attention and to identify appropriate lessons learned.

An Assessment of Human Factors Regulatory Research Facilities and Capabilities for the U.S. NRC

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The Human Factors Branch of the Nuclear Regulatory Commission's (NRC) Office of Nuclear Regulatory Research is sponsoring a study to determine the need for (if any) and availability of additional facilities for supporting human factors regulatory research. Human factors research has been performed to support the NRC's regulatory mission for more than ten years. This research has been conducted at diverse facilities, including private sector research institutions, universities, Department of Energy national laboratories and international cooperatives. From time to time researchers have experienced constraints on their access to appropriate research facilities, personnel and to other resources important to support the resolution of regulatory research needs. The constraints include:

* Lack of available licensed nuclear power plant operators or other cognizant personnel to serve as test subjects for research involving control room design, operations, maintenance practices, etc.

* Limited access to realistic control room simulator environments

* Research settings that represent plants quite different in design and operation from those in the United States

* Lack of sufficient laboratory time to support needed longitudinal studies (such as vigilance, shift work, slowly evolving events)

* Difficulty achieving industry participation/cooperation because of real or perceived regulatory exposure or conflict of interest.

The purpose of this study is to identify and evaluate alternative methods for addressing these limitations. The objectives of the study are to: (1) determine the availability and capabilities of existing research facilities to support the current and expected human factors regulatory research needs of the NRC; (2) determine the need, if any, for an enhancement of, or supplement to the present human factors research facilities, by detailing those regulatory research needs, current and expected, that cannot be met with existing facilities, or that cannot be performed at these facilities; (3) specify the characteristics of facilities that would be required to support these needs; and (4) perform a cost-benefit study of possible alternatives. This paper will describe the methods and findings of this effort to date.

Probabilistic Based Design Rules for Components Affected by Intersystem LOCAs[a]

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Low-pressure piping systems that branch to the reactor coolant systems of light water reactors (LWRs) are normally protected by multiple valves from the reactor coolant. Failure or misalignment of these isolation valves leads to an intersystem (sometimes called interfacing system) loss-of-coolant accident (ISLOCA) and to the potential leak or rupture of low-pressure systems. SECY-90-016 and the associated Staff Requirements Memorandum from the Nuclear Regulatory Commission (NRC) Commissioners require evolutionary advanced light water reactor (ALWR) plant designers to include ISLOCA in the design basis; this requirement has been extended to passive plant designers as well. However, the NRC has not provided design stress allowables or uniform criteria for ISLOCA design. Subsequently, it was decided to use ASME Code, Section III, Service Level A, design stresses and a fraction of the actual design pressure to achieve an overall system failure probability not exceeding 10%.

Four ALWRs are presently under review by the NRC staff: ABWR, SBWR, AP600, and System 80+. The piping systems of these plants that may be affected by ISLOCAs include, for example, the residual heat removal system, the safety injection system, the makeup and purification system, and other low pressure lines that branch to the reactor coolant system. While the chief safety concern deals with low pressure lines that bypass containment, piping that remains entirely within the containment is of concern as well.

To date, no full ISLOCAs have occurred in U.S. nuclear plants although there have been several instances of leakage past valves from the reactor coolant system to connecting systems. These instances have been benign and correctable by the plant staff. The probability of a full ISLOCA for existing LWRs is estimated to be on the order of 10^{-5} to 10^{-11} events per year, depending on the assumptions. Thus it is a low probability event. Since the maintenance procedures and operator training on ALWRs are expected to be similar to that at present LWRs, the ISLOCA probability for ALWRs is expected to be in the same range as for current LWRs. The NRC position is that for purposes of design, an ISLOCA can be treated as a severe accident with very low frequency of occurrence, thereby allowing relatively high conditional failure probabilities.

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

To support the NRC objective, the Idaho National Engineering Laboratory initiated a study to develop and execute a methodology (based on NUREG/CR-5603) for evaluating ISLOCA design rules on a probabilistic basis. The methodology is intended to include piping elements, flange connections, on-line pumps and valves, heat exchangers, and tanks and vessels. The initial pilot execution treated only piping elements and was applied to Advanced Boiling Water Reactor (ABWR) piping. The methodology was later extended to other PWR and ABWR components.

The basic methodology for piping and vessels is to compute failure stresses that can be used with an ISLOCA survival curve to calculate the probability of component survival at a given system pressure. The curves were developed by estimating the median and the uncertainty of the failure stress and are material and temperature dependent. The methodology for flange connections is to require a sufficiently large flange and bolt torque so that the leakage during ISLOCA conditions will be limited to a minimal amount.

The overall conclusion is that designing to ASME Code, Section III, Service Level A, rules for a pressure of 40% of the primary system pressure will in general meet the NRC goal of achieving an overall system failure probability not exceeding 10%. However, additional requirements may be needed to supplement this criterion for some components.

Structural Aging Program Approach to Providing an Improved Basis for Aging Management of Safety-Related Concrete Structures*

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ABSTRACT

Concrete structures play a vital role in the safe operation of all light-water reactor plants. In general, the performance of concrete structures in nuclear power plants has been good. However, there have been several instances where the capability of concrete structures to meet future functional and performance requirements has been challenged due to problems arising from either improper material selection, construction and design deficiencies, or environmental effects. Examples of some of the potentially more serious incidences include post-tensioning anchor head failures, leaching of concrete in tendon galleries, voids under vertical tendon bearing plates, containment dome delaminations, corrosion of steel tendons and rebars, water intrusion through basemat cracks, and leakage of corrosion inhibitor from tendon sheaths. Such incidents indicate that there is a need for improved surveillance, inspection and testing, and maintenance to enhance the technical bases for assurance of continued safe operation of nuclear power plants

The Structural Aging (SAG) Program is addressing the aging management of safety-related concrete structures in nuclear power plants for the purpose of providing improved technical bases for their continued service. The SAG Program objective is to prepare documentation providing USNRC license reviewers with (1) identification and evaluation of the structural degradation processes; (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 in-service inspection or structural assessment programs; and (4) methodologies required to perform current assessments and reliability-based life-predictions of safety-related concrete structures. To accomplish this objective, the SAG Program has conducted activities under three major technical task areas: (1) Materials Property Data Base, (2) Structural Component Assessment/Repair Technologies, and (3) Quantitative Methodology for Continued Service Determinations.

The objective of the Materials Property Data Base task is to develop a reference source which contains data and information on the time variation of material properties

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under the influence of pertinent environmental stressors and aging factors. This source 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. Primary activities under this task include development of the Structural Materials Information Center (SMIC), assemblage of materials property data, and formulation of material behavior models. The SMIC consists of the *Structural Materials Handbook* and the *Structural Materials Electronic Data Base*. The *Structural Materials Handbook* is an expandable, hard-copy reference document containing complete sets of data and information for each material, e.g., material composition, constituent material properties, and performance and analysis information useful for structural assessments and safety margins determinations. The *Structural Materials Electronic Data Base* is an electronically accessible version of the *Structural Materials Handbook* providing an efficient means for searching the various data base files to locate materials with similar characteristics or properties. Reference sources and testing of prototypical concrete samples obtained from nuclear power facilities have been used to develop over 120 material data bases for the SMIC. Also, a review has been completed of models for each of the degradation processes which could potentially impact the performance of nuclear power plant concrete structures.

The objectives of the Structural Component Assessment/Repair Technologies task are to (1) develop 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 adversely impact the durability of safety-related concrete structures in nuclear power plants; and (2) provide recommended in-service inspection or sampling procedures which can be utilized to develop the data required both for evaluating the current condition of concrete structures and for trending the performance of these components. Associated activities include an assessment of techniques for repair of concrete components which have experienced an unacceptable degree of deterioration, and the identification and evaluation of techniques for mitigation of any environmental stressors or aging factors which may act on critical concrete components. Recent activities under this task have included reviews of North American and European repair practices for degraded reinforced concrete structures. U.S. utility experience with degradation and repair of concrete structures has been assembled through responses to a questionnaire. Also, a state-of-the-art report has been prepared addressing corrosion of reinforced concrete structures.

The overall objective of the Quantitative Methodology for Continued Service Determinations task is to develop a procedure which can be used for performing condition assessments and making reliability-based life predictions of critical safety-related concrete structures in nuclear power plants. The methodology integrates information on degradation and damage accumulation, environmental factors, and load history into a decision tool that provides a quantitative measure of structural reliability and performance under projected future service conditions based on an assessment of a new or existing structure. A probabilistic methodology for condition assessment and reliability-based life prediction of concrete structures has been developed. The methodology has been applied to structures subjected to combinations of structural load processes and to structural systems. The effect of degradation in component strength on component and system reliability function has been investigated using simple parametric representations of time-dependent strength. The methodology has also been used to investigate optimization of inspection and maintenance strategies to maintain the failure probability below a specified target value.

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EXPERIMENTS TO EVALUATE BEHAVIOR OF CONTAINMENT PIPING BELLOWS UNDER SEVERE ACCIDENT CONDITIONS[1]

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Summary

Th's paper describes an ongoing test program whose goal is to determine the ultimate leak tight capability of containment piping bellows when subjected to beyond design basis loadings -- the so-called 'severe' accident. The tests are part of the Containment Integrity Programs, which are being conducted at Sandia under the sponsorship of the Nuclear Regulatory Commission (NRC).

The final goal of the Containment Integrity Programs is to generate a complete set of validated methods that can be used to predict containment behavior when subjected to severe accident conditions. In pursuit of this goal, a series of scale model containment buildings have been tested to failure. The models were subjected to static internal overpressurization at ambient temperatures, with the response being monitored by a large number of sensors. The measured response was then compared with analytical results that were compiled both before and after the test, in order to verify the analytical methods.

Because of the limited number and scale of the containment models, separate programs have been conducted to further investigate the severe accident behavior of containment penetrations. Electrical penetration assemblies, compression seals and gaskets, inflatable seals, personnel airlocks, and equipment hatches have been tested. The ongoing bellows experiments are a part of the containment penetration test series.

The bellows test program was initiated as a result of concerns that bellows could be a possible source of containment leakage during a severe accident. Bellows are used at the piping penetrations of steel containments to minimize the loadings imposed on the containment shell that are caused by differential movement between the pipe and the containment wall. Since these bellows are an integral part of the containment pressure boundary, they are subjected to the same conditions as the containment building. During a severe accident, those conditions would involve combinations of axial and lateral displacements, internal pressure, and elevated temperatures. The objective of this program is to determine if bellows are a possible source of containment failure during a severe accident, and if so, to develop methods to estimate the conditions that would likely cause such a failure.

[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.

After determining that no data on the performance of bellows subjected to severe accident conditions existed, a test series was designed that examines the behavior of various configurations of bellows geometries under severe accident loadings. The test series contains twenty tests of ten different bellows geometries that are representative of those found in actual containments.

To date, six tests have been completed. These tests have subjected bellows to extreme axial and lateral deformations such as those that could be experienced during a severe accident. The bellows have demonstrated the ability to withstand large deformation and remain leak tight. Most of the test specimens were able to endure one complete cycle of full compression, or full extension, with 2 inches of lateral displacement and remain leak tight. However, these tests have not included the effects of temperature and pressure that would be experienced during a severe accident.

The stainless steel that is used in the construction of bellows experiences a loss of ductility at elevated temperatures. Future tests in this series will investigate that effect as well as the effect of internal pressure on the loading conditions required to produce bellows leakage. Other tests in the series will examine the performance of test specimens that are representative of some full-size process piping bellows such as those found in a Pressurized Water Reactor and approximately 1:4-scale vent line bellows found in a Boiling Water Reactor Mark-I.

GUIDELINES FOR SEISMIC QUALIFICATION BY EXPERIENCE IN ALWRs[1]

Kamal K. Bandyopadhyay[2]

Abstract

The methodologies and acceptance criteria for seismic qualification of equipment are provided in IEEE Std. 344 (Reference 1) endorsed by the Nuclear Regulatory Commission (NRC) in the Standard Review Plan (Reference 2). The IEEE Standard allows seismic qualification by use of the similarity analysis method. Similarity is demonstrated between the equipment item that is to be qualified with the item that has been qualified. If similarity is established, then the equipment item is qualified up to the demonstrated vibration level. Data available from past earthquake events or vibration tests can be used in this regard. The nuclear industry has collected a vast amount of earthquake experience and test data in the last decade, and is planning to use it for seismic qualification of equipment in advanced light water reactor (ALWR) plants. In order to provide recommendations on the use of experience data for equipment qualification in ALWR plants, the NRC has appointed an Expert Panel. The Panel prepared a set of recommendations and acceptance criteria for seismic qualification of equipment by use of experience data. This paper discusses these recommendations and the respective technical bases[3].

For the purpose of qualification by use of experience data, equipment classes can be divided into three groups according to their design complexities, inherent strength and our ability to identify the potential malfunction mechanisms. A review of equipment characteristics and the experience data demonstrates that certain equipment categories can successfully withstand a high level of vibration since either they are designed to carry large operational loads, such as, mechanical vibration, or their functionalities are not sensitive to earthquake motion. These equipment classes belong to Group 1 and can be qualified with the least effort. The equipment classes that are not fit for belonging to Group 1 and whose malfunction mechanisms can be identified with a high degree of confidence belong to Group 2. The remaining

[1]The program is sponsored by the United States Nuclear Regulatory Commission (NRC), Office of Nuclear Research.

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[3]The Expert Panel consists of the author and Drs. Daniel Kana, Robert Kennedy and Anshel Schiff. The views expressed in this paper are not necessarily supported by other Panel Members or the NRC.

equipment classes, including those for which no experience data exist belong to Group 3 and require qualification by shake table testing or a combination of shake table testing and analysis as recommended by the IEEE Standard. For Groups 1 and 2, the respective experience-based response spectra can be used as the qualification levels, and the limitation of the data base and possible variability of future products can be recognized by defining the inclusion and exclusion rules.

REFERENCES

1. IEEE Standard 344-1987, "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."

2. U.S. Nuclear Regulatory Commission Standard Review Plan, Section 3.10.

Large-Scale Seismic Test Program at Hualien, Taiwan

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Abstract

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 Electric Power Research Institute (EPRI) and the Taiwan Power Company (Taipower) organized the program and took the lead in planning and managing the program. Other organizations cofunding, cost-sharing and providing technical as well as management input to the LSST Program are the U.S. Nuclear Regulatory Commission (NRC), the Central Research Institute of Electric Power Industry (CRIEPI), the Tokyo Electric Power Company (TEPCO), the Commissariat A L'Energie Atomique (CEA), the Electricite de France (EdF), Framatome, the Korea Electric Power Corporation (KEPCO), the Korea Institute of Nuclear Safety (KINS) and the Korea Power Engineering Company (KOPEC).

The LSST was initiated in January 1990, and is envisioned to be five years in duration. The goal of this program is to collect real earthquake induced soil structure (SSI) data in order to evaluate computer codes used in SSI analysis of nuclear power plant structures. A major task completed in 1992 was the construction of a ¼ scale model of a reinforced concrete containment with instrumentation located on the structure and in the field along a three dimensional strong ground motion array. Also to date two forced vibration tests (FVT) have been conducted, one without backfill and one with backfill and the design of a vertical flat bottom tank has been completed. This paper describes progress of the Hualien project with emphasis on site investigation and the FVT.

INTEGRAL SEISMIC TESTING OF CIRCUIT BREAKERS AND RELAYS MOUNTED IN SWITCHGEAR[1]

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Abstract

Among all equipment types considered for seismic qualification, relays have been most extensively studied through testing due to a wide variation of their designs and seismic capacities. A temporary electrical discontinuity or "chatter" is the common concern for relays. A chatter duration of 2 milliseconds is typically used as an acceptance criterion to determine the seismic capability of a relay. Many electrical devices, on the other hand, receiving input signals from relays can safely tolerate a chatter level much greater than 2 ms.

In Phase I of a test program, Brookhaven National Laboratory performed testing of many relay models using the 2 ms chatter criterion (Reference 1). In Phase II of the program, the factors influencing the relay chatter criterion, and impacts of relay chatter on medium and low voltage circuit breakers and lockout relays were investigated. This paper briefly describes the Phase II tests and presents the important observations.

The first part of the Phase II test program concentrated on obtaining the response of circuit breakers and lockout relays subjected to electrical pulses. Both single and multiple pulses were used to determine the required tripping time. The time required to initiate unlatching of the tripping mechanisms due to buildup of the electromagnetic force was significantly less than the total tripping time that includes the time required by the latching mechanism to complete its motion until tripping occurs. The electrical test data were used to plan the vibration tests.

For the shake table testing, a set of source relays were used to provide input signals to a set of load devices that included lockout relays and a medium voltage circuit breaker mounted in a switchgear cabinet. In one test set up, the source relays were vibrated on the shake table and the load devices were placed on a stationary stand. Subsequently, both the source relays and load devices were placed on the shake table. The load devices withstood chatter durations much greater than 2 ms indicating greater seismic capabilities of the source relays. Most trippings occurred for vibration in the vertical direction.

[1]The program is sponsored by the United States Nuclear Regulatory Commission (NRC), Office of Nuclear Research.

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Development of an Improved Methodology for Probabilistic Seismic Hazard Analysis

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

The NRC, EPRI, and the U.S. Department of Energy (DOE) are jointly supporting an 18-month project that began in the spring of 1993, with the goal of developing a recommended methodology, including implementation guidelines, suitable for performing PSHA. The final product of the project will be a methodology that can be used in seismic regulation of nuclear power plants and other critical facilities.

To accomplish this objective, an independent committee of technical experts, the Senior Seismic Hazard Analysis Committee, has been established under joint NRC, DOE, and EPRI sponsorship.* This Committee is developing the desired implementation guidelines for PSHA. In the course of this work, it is expected that the Committee will evaluate the PSHA methodologies already developed by EPRI and under NRC sponsorship by LNLL.

Technical Support Panels sponsored by NRC, DOE, and EPRI will perform analyses and studies as defined by the Committee. A final Committee report will summarize the work performed in this project and provide the desired implementation guidelines, including a recommended methodology, suitable for the performance of PSHA.

This paper will discuss the progress to date for this important project, including the approach being taken and some of the key technical issues that are being confronted. Among these are how to provide definitive guidance on seismic zonation, seismicity modeling, ground-motion modeling, and expert elicitation.

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

Thermal-Hydraulic Computer Code
Development and Assessment Process for ALWRs
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In September 1988, the NRC issued a revised ECCS rule (10 CFR 50.46) for light water nuclear power reactors to allow the use of best-estimate computer codes in safety analysis as an option. A key feature of this option requires the licensee to quantify the uncertainty of the calculations and include that uncertainty when comparing the calculated results with acceptance limits provided in 10 CFR 50.46. To support the revised ECCS rule and illustrate its application, the NRC and its contractors and consultants developed and demonstrated an uncertainty evaluation methodology called code scaling, applicability, and uncertainty (CSAU). See NUREG/CR-5249 for details.

The CSAU methodology as described in NUREG/CR-5249 is the culmination of 20 years of ECCS research on current LWR designs involving extensive iteration of experiments and analysis in which the developmental process was essentially completed. This allowed establishment of a structured top-down process of determining code capabilities and adequacy of code assessment and a bottom-up process of code sensitivity and uncertainty analysis.

Using the original CSAU methodology it is necessary that six conditions be met:

1) The computer code is "frozen."
2) The code documentation is complete.
3) A single transient or closely related set of transients are the subject of assessment.
4) The safety evaluation criteria are established.
5) The experimental assessment matrix is complete.
6) Sufficient information and a cadre of experts exist to construct a valid Phenomena Identification and Ranking Table (PIRT).

However, at present, the above six conditions cannot be met for the passive ALWRs. This indicates that to some degree the computer codes are in a developmental state for ALWR safety analysis and that sufficient experimental data does not yet exist for code assessment.

The NRC Research staff and contractors have embarked on a code development and assessment process which utilizes the principles of the top-down portion of the CSAU methodology. The code development process involves modification to existing codes (principally RELAP5) to assure that ALWR processes and phenomena are adequately addressed. The assessment process involves analysis of plant transients and vendor and NRC-sponsored ALWR experiments.

When the development and assessment process is complete it should be possible to meet the six conditions listed above. At that time sensitivity studies can be reliably performed and the nature of any bottom-up uncertainty analysis can be determined in the context of the regulatory design certification process.

NRC Confirmatory Safety System Testing in
Support of AP600 Design Review

SUMMARY

Westinghouse Electric Corporation has submitted the Advanced Passive 600 MWe (AP600) nuclear power plant design to the NRC for design certification. The Office of Nuclear Regulatory Research is proceeding to conduct confirmatory testing of AP600 safety systems to help the NRC staff evaluate the safety of the AP600 reactor systems.

In contrast to the current generation of reactors, this new design features passive safety systems for mitigating accidents and operational transients. Since these passive safety systems rely on gravity-driven flow, the driving forces for the safety functions are small compared to those available under conventional pumped systems. Thus, the performance of these new safety systems may be adversely affected by small variations in thermal hydraulic conditions. Also, the computer analyses of the passive safety systems pose a challenge for current thermal-hydraulic system analysis codes in that the current codes were not sufficiently assessed for conditions of low pressure and low driving heads and for the system interactions that may occur among the multiple flow paths used in the AP600 design. Therefore, integral effects test data are being obtained for evaluation of AP600 safety system performance and for independent assessment and validation of computer analysis codes. Westinghouse is sponsoring integral test programs in the SPES-2 (Simulatore Per Esperienze di Sicurezza-2) and OSU (Oregon State University) test facilities. SPES-2 is a full-pressure, full-height test facility in Italy but much smaller in scale (1/395 by volume) than ROSA which represents a 1/48 volume-scale for current reactors and a 1/30 volume-scale for AP600. OSU facility is a low pressure, reduced height facility with a considerably smaller volumetric scale (1/200 by volume) as compared to ROSA. NRC confirmatory safety system testing is not required for design certification but would provide additional technical bases for the NRC licensing decisions.

For confirmatory testing, it was determined that the most cost-effective route was to modify an existing full-height, full-pressure test facility rather than build a new one. Thus, all the existing integral effects test facilities, both in the United States and abroad, were screened to select the best candidate. The criteria for the initial screening included the size, facility configuration similarity, availability schedule, willingness to share the cost, and the ability to enter into a confidential agreement with Westinghouse for handling proprietary information. This screening revealed that the best candidate was the Rig of Safety Assessment (ROSA) Large Scale Test Facility in Japan Atomic Energy Research Institute (JAERI). To confirm these initial results and to determine the extent of modification necessary to simulate the AP600, the Idaho National Engineering Laboratory (INEL) was contracted to perform a comparative study between ROSA and AP600 using the RELAP5/MOD2.5 code.

A comparison between the existing ROSA facility and the AP600 design showed that ROSA did not contain the key components important for safety response of the AP600. It was not obvious how much hardware modification to the ROSA facility would be needed to simulate the AP600. The fidelity of simulation

must be balanced against the associated cost. The fidelity should be high enough to result in a facility capable of producing data for code assessment covering the major AP600 phenomena in the correct sequence. At the same time, the cost and the schedule have to be affordable. To make an optimum choice, four levels of modifications in progressively more extensive stages were considered. The first level of modifications was the absolute minimum, and the fourth level was the most inclusive among the four levels.

In evaluating each level of modification, the RELAP5/MOD2.5 code was used as a primary tool for comparing the predicted behavior of ROSA with that of AP600 for selected accident scenarios. This approach is based on the assumption that RELAP/MOD2.5, although not assessed against AP600 systems test data, will show major trends in overall behavior in such global parameters as depressurization rate, mass inventory, and energy distribution. The validity of this assumption is partially supported by the fact that the RELAP5 code reasonably matched experimental data from many different facilities, of different sizes, which were designed to simulate current PWRs. Since the thermal-hydraulic processes involved in current reactors and passive reactors are fundamentally the same, it is likely that the RELAP5 code will also show the major trends in AP600 and ROSA, even though the predictions may not be as accurate until further improvements are made in such areas as mathematical modelling of condensation in the presence of noncondensible gases, boron transport, and the computation of level tracking and thermal stratification in a tank.

The comparative analyses of ROSA and AP600 with the RELAP5 code for the accident scenarios as indicated above lead us to conclude that the optimum choice was the fourth level modification. This included two core makeup tanks (CMTs) with appropriate pressure balance lines, a passive residual heat removal (PRHR) system with stimulated secondary cooling, automatic depressurization system (ADS) with stages 1 through 3 on top of the pressurizer and stage 4 on the hot leg, minimization of the pump loop seal lengths, a properly scaled AP600 pressurizer with a surge line, installment of appropriate upper head flow paths, and an incontainment refueling water storage tank (IRWST). Since there is only one cold leg in each loop, CMT cold leg pressure balance lines are connected to the same cold leg for most transients when asymmetry between the two CMTs is not expected, but connected to a different cold leg for non-symmetric transients, such as a break in a pressure balance line or a break in a direct vessel injection (DVI) line.

The chosen level of facility modification is being implemented by Sumitomo Heavy Industries (SHI) which constructed the ROSA facility and has been maintaining and operating it for the past several years as a contractor to JAERI. The facility modification will be completed by January 1994, and a series of tests will be performed in 1994.

As a confirmatory testing program, the ROSA/AP600 testing will cover not only design basis accidents but also beyond-design basis accidents which may not be required for vendors to address. There will be some counterpart tests among ROSA, SPES-2, and OSU tests. Test data from different scale facilities will help predict what will occur in a full-scale AP600 reactor.

NRC CONFIRMATORY TESTING PROGRAM FOR SBWR

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Introduction

General Electric Company (GE) has submitted for design certification an advanced boiling water reactor (BWR) called the Simplified BWR (SBWR). The SBWR design is largely based on the proven BWR technology with many years of successful experience. However, there are some differences. First of all, unlike most of the operating BWRs, SBWR is a natural circulation reactor with core flow driven by the hydrostatic head difference between the downcomer and the core. There is a long chimney region above the core to enhance the natural circulation flow. Second and more importantly, the SBWR uses passive (not pump-driven), safety systems to provide emergency core and containment cooling. These passive systems rely on gravity, natural convection, and condensation to provide driving forces to maintain their operation without the use of any pumps. In addition to the passive systems, the SBWR also has operator-activated, pump-driven, non-safety systems as the first line of defense to prevent and mitigate accidents.

There are three passive safety systems of importance in a SBWR: (1) the low-pressure Gravity-Driven Cooling System (GDCS) for providing emergency cooling and makeup water to the vessel, (2) the low-pressure Passive Containment Cooling System (PCCS) for maintaining containment cooling and integrity, and (3) the high-pressure Isolation Condenser System (ICS) for removing decay heat. Both the GDCS and PCCS are unique to the SBWR and do not exist in any of the operating BWRs, while the ICS is similar to those on some of the earlier BWRs but with a different condenser design.

Because of the unique features in the SBWR design, GE has established several testing programs to demonstrate their performance as well as to provide a sufficient data base for assessing analytical tools, in compliance with the requirements of 10 CFR 52.47 for design certification. These testing programs assess the performance of GDCS, PCCS, ICS, depressurization valves (DPVs) of the automatic depressurization system (ADS), and flow stability of natural circulation core cooling.

To confirm GE test results and to broaden the data base for code assessment, the NRC has determined that it is highly desirable to establish a Confirmatory Testing Program for SBWR to assess the "integral" performance of GDCS and PCCS under various accident and transient conditions.

NRC Confirmatory Testing Program

The NRC Confirmatory Testing Program includes experiments and supportive analyses. A test facility at a university is under design and will be built by September 1994. It will become fully operational and begin to produce data around January 1995. All tests will be completed by July 1996 so that sufficient data will be available to support the NRC design certification review of the SBWR.

The NRC-sponsored test facility will have all of the key components and systems required for investigating the integral performance of GDCS and PCCS.

This includes a vessel with electrically-heated fuel rods, an upper drywell and a lower drywell, the suppression pool, GDCS, PCCS, ICS, non-safety drywell and wetwell sprays, connecting pipes and valves, and sufficient instrumentation to collect data for code assessment. Before the facility design is finalized, a detailed scaling analysis including a PIRT (Phenomena Identification and Ranking Table) exercise will be completed to identify the important vessel and containment phenomena/processes expected to occur in a SBWR and to reproduce them in the facility. A test matrix under development will cover the range of phenomena/processes for a spectrum of the design basis accidents (DBAs) and transients.

Since both GDCS and PCCS are designed to operate at a pressure much lower than the SBWR operating pressure (at 1040 psia), a low-pressure test facility is adequate for assessing the GDCS and PCCS performance. The NRC-sponsored facility is a low-pressure (150 psia), reduced-height facility. Its volume scale is complementary to GE's GIST and GIRAFFE facilities but smaller than the PANDA facility. GIST, GIRAFFE, and PANDA are three low-pressure, full-height facilities. GIST has a volume scale of 1/508 of an earlier SBWR design, and its data are for short-term GDCS performance. GIRAFFE has a volume scale of 1/400 of an earlier SBWR design, and its data are for long-term PCCS performance and containment phenomena. PANDA has a volume scale of 1/25 of the current SBWR design, and it will provide data of long-term PCCS performance and containment phenomena.

There are several ways that the NRC Confirmatory Testing Program will provide additional and valuable data for code assessment beyond the data of GIST, GIRAFFE, and PANDA. First, there is a gap in GIST, GIRAFFE, and PANDA data. GIST is focused on short-term GDCS injection to the vessel. Its tests were initiated after vessel was nearly depressurized and ended when GDCS injection into the vessel was completed at about half a hour or less after accident initiation. On the other hand, GIRAFFE and PANDA are focused on the long-term PCCS performance with tests started at about one hour after accident initiation. To fill this data gap, the NRC Confirmatory Testing Program will provide data starting at about the same time as GIST tests and continuing beyond the initiation of PANDA and GIRAFFE tests. Second, the impact of PCCS and ICS on GDCS performance was not investigated in GIST due to its lack of PCCS and ICS. Since there are PCCS and ICS in the NRC facility, the GDCS performance will be experimentally assessed with and without the operation of PCCS and ICS. Third, the NRC facility can probably better reproduce two or three dimensional phenomena as expected in the SBWR than GIST or GIRAFFE. This is because the reduced-height NRC facility has a width-to-height ratio much closer to a SBWR than that of the full-height GIST or GIRAFFE. As a result, data from the NRC facility will provide additional information on multi-dimensional phenomena for code assessment. Fourth, the NRC facility is based on the current SBWR, while both GIST and GIRAFFE are based on an early SBWR design with somewhat different vessel and containment dimensions than the current design.

Analytical support is provided by BNL using the linked RELAP5/CONTAIN code. BNL will first predict the test facility behavior under a selected accident or transient, and it will then compare the results against a similar calculation for an SBWR under the same conditions. If the comparison is reasonable and understandable, this will provide some confidence that the scaled test facility adequately reproduces the important phenomena/processes as in a SBWR and provides a good simulation.

ASSESSMENT OF RELAP5/MOD3 WITH GIST DATA

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SUMMARY

General Electric Company (GE) has proposed an advanced light water reactor design, the Simplified Boiling Water Reactor (SBWR) that relies on passive, gravity-driven safety systems to provide emergency core coolant injection under postulated accident conditions. A unique element of the SBWR emergency core cooling system that has not been used in previous BWR designs is the gravity-driven cooling system (GDCS).

A GDCS test program conducted by GE, and sponsored by the U. S. Department of Energy was completed in October of 1989. The test program involved construction of a full-height GDCS Integrated Systems Test (GIST) facility and performance of a series of GDCS tests simulating a wide range of conditions. The primary objectives of the GIST test program were to demonstrate the technical feasibility of the GDCS concept and to provide a sufficient database to qualify the thermal-hydraulic analysis code TRACG for use in SBWR accident analysis with respect to GDCS phenomena. Although the GIST facility design was based on an earlier SBWR conceptual design, the short-term gravity drain function of the GDCS in the current design, utilizing a passive gravity-driven coolant injection system, is similar in nature to the GIST facility design.

Four major categories of design basis accidents were simulated with the GIST facility. These categories include:

- Main steam line break (MSLBs)
- Bottom drain line break (BDLBs)
- GDCS drain line break LOCA (GDLB)
- Loss of feedwater or no break (NB).

A total of 24 unique experiments falling into the above categories were performed. For each class of accidents, parametric variations in test conditions were examined to ensure that GDCS response would maintain the core in a cooled state.

GIST is a low pressure test facility, designed to simulate post-LOCA conditions during the recovery period. Initial conditions for each test were obtained from TRACG simulations used to calculate the transition period between nominal operating pressure of 7171 kPa (1025 psig) and the initial test pressure of 791 kPa (100 psig). The TRACG output was then used to establish the initial conditions for the GIST facility at 140 psig, and to confirm the proper depressurization rate for the facility between 1067 kPA

(140 psig) and 791 kPa (100 psig). Actual test measurements started when the vessel pressure reached 791 kPa (100 psig), at which time the vessel discharge was switched from the atmosphere to the containment mock-up.

The NRC Office of Nuclear Regulatory Research has sponsored a project at the INEL to assess the adequacy of the GIST Test Program and perform an assessment of the RELAP5/MOD3 code against the GIST facility experimental results. The adequacy of the facility was reviewed in terms of the scaling basis of the facility, adequacy of instrumentation and recorded data, and the accidents and transients simulated. The applicability of the GIST experimental results to the current SBWR design was also addressed. The RELAP5/MOD3 assessment effort included a set of five calculations representing the full range of GIST facility experiments. These calculations include one experiment from each of the four categories mentioned above and an additional main steamline break experiment performed with a reduced initial reactor water level.

Coupling of RELAP5/MOD3 to CONTAIN for ALWR Analyses

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The motivation for the union of these two computer codes stems from the unique safety analysis challenge presented by the new Advanced Light Water Reactor (ALWR) conceptual designs. Incorporated into many of these designs are requirements for long term passive cooling systems integrating both mechanisms in the main reactor coolant system and in the containment. Westinghouse's AP600 and General Electric's SBWR are two examples of designs that meet this description.

The prospect of coupling RELAP5/MOD3 and CONTAIN introduces a new dimension in best-estimate nuclear power plant systems analysis. Both RELAP5/MOD3 and CONTAIN are tools based on "first principles physics" addressing transient thermal-hydraulic and containment response issues, respectively. The integration of these two computer codes advances the state-of-the-art of best-estimate systems analysis of nuclear power plants by linking the two-fluid, six equation, non-equilibrium, non-homogeneous thermal-hydraulic models of RELAP5 with the unique containment models of CONTAIN.

A new feature is being implemented into RELAP5/MOD3 that creates a data communication port for the transmission of data to and from RELAP5/MOD3 and CONTAIN. The implementation of this feature into RELAP5/MOD3 has been designed to be general to accomodate future coupling links to other codes. This feature relies on the data communication function of the Parallel Virtual Machine (PVM) software. New coding in RELAP5/MOD3 reads information concerning with what code to communicate (i.e., CONTAIN), data to send, data to receive, and how often to execute the data transmission.

The new RELAP5/MOD3 input for this feature requires a time dependent table of the communication frequency for both send from RELAP5 and receiving by RELAP5, a table of the RELAP5 variables that will serve as sources to CONTAIN, and a table of RELAP5 variables that will provide boundary conditions that have been defined by CONTAIN. CONTAIN does not require new input, it receives any information it needs from RELAP5/MOD3.

Data received by RELAP5/MOD3 is introduced into source tables utilized by the TMDPVOL and TMDPJUN components and heat structures. This limits the information that can be introduced into a RELAP5 problem to those variables that can be defined as a source by those components (i.e., pressure, temperature, internal energy, flow rates, etc.). Any variable in RELAP5/MOD3 is available to send out to an external code. Data received by CONTAIN is introduced by a mechanism similar to that for introducing sources through input. New subroutines mimic the functions of SORATM, SORSRV, and SORPL for introducing sources directly to an atmosphere cell, to a cell via a safety/relief valve, or to a pool cell.

RAMONA-4B DEVELOPMENT FOR SBWR SAFETY STUDIES

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1. INTRODUCTION

The Simplified Boiling Water Reactor (SBWR) is a revolutionary design of a boiling-water reactor. The reactor relies on passive safety phenomena such as natural circulation, gravity flow, pressurized gas, and condensation. SBWR has no active systems, and the flow in the vessel is by natural circulation. There is a large chimney section above the core to provide the buoyancy head for natural circulation. The reactor can be shutdown by four systems:

(1) Scram
(2) Fine-Motion Control Rod Drive (FMCRD)
(3) Alternate Rod Insertion (ARI)
(4) Standby Liquid Control system (SLCS).

The safety injection is by gravity drain from Gravity Driven Cooling System (GDCS) and Suppression Pool (SP). The heat sink is through two types of heat exchangers submerged in the tank of water. These heat exchangers are Isolation Condenser (IC), and Passive Containment Cooling System (PCCS).

The unique design of SBWR imposes new requirements on the analytic methods. The close coupling between the power and flow, and the flow distribution among the parallel channels require a multidimensional power prediction capability. The startup of the reactor has vapor generation and condensation taking place in the core requiring a model of non-homogeneous, nonequilibrium two-phase formulation. The instability at low-flow/high-power conditions requires modelling of the control systems and balance of plant, which has significant impact on the amplitude of the instability-induced power and flow oscillations.

RAMONA-4B code has been developed to simulate the normal operation and reactivity transients, and to address the instability issues for SBWR. The code has a three-dimensional neutron kinetics coupled to multiple-parallel-channel thermal hydraulics. The two-phase thermal hydraulics is based on a nonhomogeneous nonequilibrium drift-flux formulation. It employs an explicit integration to solve all state equations (except for neutron kinetics) in order to predict the instability without numerical damping.

The objective of this project is to develop SUN SPARC and IBM RISC 6000 based RAMONA-4B code for applications to SBWR safety analyses, in particular for stability and Anticipated Transient Without Scram (ATWS) studies.

2. CODE IMPROVEMENTS

The earlier version of RAMONA-3B code used slip formulation for two-phase flow, and modelled components specific to current BWRs, but had no balance-of-plant models. The natural-circulation capability existed in the transient section, but not in the steady-state section of the code. The RAMONA-4B code has been obtained from RAMONA-3B code by incorporating the following improvements:

1. Natural Circulation Capability in Steady Section
2. Chimney Component
3. Flow Dependent Loss Coefficients
4. Isolation Condenser Component
5. Balance of Plant (BOP)
6. Boron Circulation in the Vessel
7. Standby Liquid Control System (SLCS)

3. DEVELOPMENTAL ASSESSMENTS

A set of sample problems have been setup to assess the code. These sample problems involve various types of ATWS events: Main Steamline Isolation Valve (MSIV) closure, turbine trip with or without bypass, Loss of Feedwater Heaters (LOFH), Recirculation Pump Trip (RPT for current BWRs), and a start-up transient.

The MSIV closure ATWS event was used to assess the IC model, the pressure control system, and the boron transport model. The LOFH event was used to study the reactivity transient due to loss of feedwater heating, and the SBWR shutdown by the FMCRD run-in. The RPT event was used to assess the capability for predicting thermal-hydraulic instability. The startup transient is being used to assess the natural-circulation prediction capability, and to study the SBWR instability during startup.

4. CONCLUSIONS

The RAMONA-4B code has been upgraded to include the balance of plant and control systems along with components specific to SBWR. The code has also been made operational on workstations. This code is now available to investigate stability issues not only for the current BWRs but also for ABWR and SBWR. RAMONA-4B can also be used for reactivity transients such as a rod-drop accident, as well as ATWS events.

HYDROGEN MIXING EXPERIMENTS IN THE HDR-CONTAINMENT UNDER SEVERE ACCIDENT CONDITIONS

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Whereas experimental results of the HDR-H_2-mixing tests were primarily presented for E11.2 and E11.4 at the 19th WRSM, this presentation will focus on the results of a detailed assessment of the other H_2-mixing experiments of Test Group E11. These results will be compared to those of E11.2 and E11.4 where applicable.

Among the parameters which have been examined and will be presented are:

- blockage of flow path in the global convective loop
- effect of boiling sump
- effects of internal and external spray periods (condensation, pressure decrease, thermal homogenization)
- containment atmosphere behaviour after spray actions (re-stratification)
- temperature distribution in the annular gap
- containment venting position on containment depressurization
- updated energy balances for E11.2 and E11.4

Whereas all previous presentations concerning data versus code predictions concentrated on the blind post-test efforts, this presentation will focus on the results of the comparisons with parametric, best-estimate, open post-test predictions for experiments E11.2 and E11.4 with the containment analysis computer codes RALOC, WAVCO, CONTAIN, MELCOR and GOTHIC.

The final results of these comparisons show (after correcting a number of deficient input parameter previously supplied by KfK/PHDR as input):

E11.4:

- **Standard lumped-parameter** codes are able to predict H_2-mixing and distribution phenomena when H_2 is injected into a well-mixed atmosphere in lower zones of the containment with excellent agreements in most of the important quantities
- **A few** discrepancies remain, dependent upon the codes' modeling methodologies

E11.2:

- **Accounting** for the corrections improves the agreements substantially compared to the blind post-test predictions
- **However,** more or less large discrepancies still remain concerning the predictions of the thermal stratification pattern and the H_2-distribution
- **Parametric** changes of input parameters lead to improvements of agreements in some quantities but at the same time worsen others
- "Innovative" concepts of changing certain input parameters beyond present practice improves the quality of the predicted H_2-concentrations

HYDROGEN DEFLAGRATION EXPERIMENTS IN MULTI-COMPARTMENT GEOMETRIES

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Recent hydrogen deflagration experiments in the Battelle Model Containment (BMC) resulted in unexpectly high transient pressure loads in multi-compartment geometry, demonstrating physical processes substantially different from those rather benign phenomena previously observed in single compartment tests for the same hydrogen concentration. This high pressure build-up is associated with the accelerated jet ignition induced by the vents connecting the subcompartments. After a hydrogen deflagration event is initiated in the first subcompartment, the resultant gas expansion leads to high velocities vent flows which induce high level turbulence in the subsequent subcompartments. The local combination of characteristic temperature, turbulence and hydrogen-concentration can result in an instantaneous ignition along the whole jet penetration path leading to an instantaneous pressure increase up to close to the adiabatic-isochoric limit curve. The phenomenon of accelerated jet ignition has been observed at hydrogen concentrations as low as 9-10 % vol % in air. As the maximum permissible differential pressure loads for most separating walls between sub-compartments are about 0.1 MPa, even moderate, local hydrogen-concentrations may result in localized damage and the resultant structural debris may potentially threaten the integrity of the steel containment shell.

Major parameters affecting the maximum pressure are:

- Position of ignition relative to vent position
- Size of primary vent flow openings between ignition and subsequent compartments
- Size of secondary pressure relief vents in subsequent compartments
- Direction of jet from primary vent
- Hydrogen concentration
- Steam content
- Flame propagation history in ignition chamber
- Subcompartment internals

For nuclear power plant severe accident safety analyses, potentially adverse effects by deflagrative pressure increases have to be considered especially for the following circumstances:

- High hydrogen release rates leading to high local hydrogen concentrations
- Development of ignitible mixtures by condensation-induced decrease of steam inerting potential
- Ignition in low-concentration compartment followed by jet ignition of high-concentration compartment

The contribution by the HDR Safety Program, Phase III, to the German hydrogen research activities were two-fold:

(1) to confirm the findings of the experiments in the BMC in volumes of typically about 100 m^3 by similar ones at larger scale with a total volume of 500 m^3
(2) to broaden the data base for assessing the emerging modelling strategy for larger scales towards more realistic subcompartment sizes.

In order to supplement the BMC-results in a proper, controlled manner for additional model development and computer code verification a total of 7 experiments was performed and the following positions for hydrogen ignition were examined:

Test Group E12.1:	vertical hydrogen deflagration in a vertically-oriented sub-compartment
Test Group E12.2:	ignition close to the vent
Test Group E12.3:	accelerated jet ignition

For these test subgroups the following phenomena will be reported:

- flame propagation in the ignition and subsequent subcompartments
- flame front acceleration at the vent and its effect on the pressure increase
- pressure and thermal loads on structures
- effect of lateral vent on the deflagration process for a vertical flame front
- examination of accelerated jet ignition phenomena in HDR-geometry (E12.3 only)
- interactions between flame, gas distribution, completeness of burn, burn duration and rate.

A number of computational activities were initiated prior to the E12-experiments, involving the CONTAIN 1.10/1.11 and BASSIM-codes. A few comparisons between data and code results will be shown.

RESULTS OF RECENT NUPEC HYDROGEN RELATED TESTS

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NUPEC has started NUPEC Containment Integrity project entitled "Proving Test on the Reliability for Reactor Containment Vessel" since June. 1987. This is the project for the term of twelve years sponsored by MITI (Ministry of International Trade and Industry. Japanese Government). Based on the test results. computer codes are verified and as the results of analysis and evaluation by the computer codes. containment integrity is to be confirmed.

This paper indicates the results of hydrogen mixing and distribution test and hydrogen burning test (Small scale).

The hydrogen mixing and distribution tests are to investigate their behaviors in the containment vessel with multiple compartments representing a typical large dry containment of a PWR. The test vessel has a volume of 1,600m^3 that is about 1/4th scale of an actual PWR containment vessel. Compartment number 25 in the test vessel is the same as that of actual plants. Helium gas is used for this test instead of hydrogen to avoid unexpected explosion. Main test items are efect of natural circulation with helium injection. effect of density difference between helium and air. heat sink effect of containment vessel wall and compartments wall etc.

The NUPEC tests conducted so far suggest that hydrogen is well mixed in a containment vessel and the prediction by the computer code is in good agreement with the data.

Hydrogen burning tests are conducted at NUPEC with the objectives to investigate hydrogen burning phenomena including mitigation effect of steam. spray. and nitrogen inerting in a containment vessel. and to confirm containment integrity against hydrogen burning. The hydrogen burning tests are conducted by using a small scale cylindrical vessel with 5m^3 and a large scale spherical vessel with 270m^3. In the small scale test. the effects of temperature. pressure. turbulence. spraying. distribution and concentration of gases have been investigated in detail prior to the large scale test.

A comparison of the NUPEC data with previously performed FITS test data at SNL is presented in terms of the peak combustion pressure normalized with respect to the initial pressure. The NUPEC data are in good agreement with the FITS data which were obtained at the lower hydrogen concentration condition. New data bases have been added in the higher hydrogen concentration by the NUPEC data.

HIGH-TEMPERATURE HYDROGEN-AIR-STEAM DETONATION EXPERIMENTS IN THE BNL SMALL-SCALE DEVELOPMENT APPARATUS*

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The Small-Scale Development Apparatus (SSDA) was constructed to provide a preliminary set of experimental data to characterize the effect of temperature on the ability of hydrogen-air-steam mixtures to undergo detonations and, equally important, to support design of the larger-scale High-Temperature Combustion Facility (HTCF) by providing a test bed for solution of a number of high-temperature design and operational problems. The SSDA, the central element of which is a 10-cm inside diameter, 6.1-m long tubular test vessel designed to permit detonation experiments at temperatures up to 700K, was employed to study self-sustained detonations in gaseous mixtures of hydrogen, air, and steam at temperatures between 300K and 650K at a fixed pressure of 0.1 MPa. Hydrogen-air mixtures with hydrogen composition from 9 to 60 percent by volume and steam fractions up to 35 percent by volume were studied for stoichiometric hydrogen-air-steam mixtures.

Detonation cell size measurements provide clear evidence that the effect of hydrogen-air gas mixture temperature, in the range 300K to 650K, is to decrease cell size and, hence, to increase the sensitivity of the mixture to undergo detonations. The effect of steam content, at any given temperature, is to increase the cell size and, thereby, to decrease the sensitivity of stoichiometric hydrogen-air mixtures. The hydrogen-air detonability limits for the 10-cm inside diameter SSDA test vessel, based upon the onset of single-head spin, decreased from 15 percent hydrogen at 300K down to about 9 percent hydrogen at 650K. The one-dimensional ZND model does a very good job at predicting the overall trends in the cell size data over the range of hydrogen-air-steam mixture compositions and temperature studied in the experiments. The experimentally measured detonation velocity generally agrees within 2 to 3 percent with predictions based upon Chapman-Jouget theory over the temperature range considered, and measured peak detonation pressure agrees within 10 percent of the calculated Chapman-Jouget pressure. The peak pressure is found, both experimentally and according to the theory, to decrease with increasing temperature.

Preliminary experiments indicated that the maximum temperature for which it was found possible to load combustible gases into the test vessel without an immediate burn was 650K. Experiments were conducted to measure the rate of hydrogen oxidation in the absence of ignition sources at temperatures of 500K and

*This work was performed under the auspices of the U. S. Nuclear Regulatory Commission.

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650K, for hydrogen-air mixtures of 15 percent and 50 percent, and for a mixture of equimolar hydrogen-air and 30 percent steam at 650K. The rate of hydrogen oxidation was found to be significant at 650K. Reduction of hydrogen concentration by chemical reaction from 50 to 44 percent hydrogen, and from 15 to 11 percent hydrogen, were observed on a time frame of minutes. The DeSoete rate equation predicts the 50 percent experiment very well, but greatly underestimates the reaction rate of the lean mixtures.

Experiments planned for the High-Temperature Combustion Facility will provide additional data for the detonation cell size as a function of hydrogen concentration, temperature, steam fraction, and pressure. Experiments will also address the effect of temperature on the conditions for deflagration-to-detonation transition and hot jet initiation mechanisms of detonation initiation. A more complete assessment of the effect of temperature on the likelihood of detonations in containment will be possible following availability of this data from the HTCF test program.

IGNITION OF HYDROGEN-AIR-STEAM MIXTURES BY A HOT GAS JET

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The combustion of hydrogen-air mixtures is of a great interest in the safety aspect of industrial hazards and can be the cause of serious accidents. The present study describes the conditions and the limits for the ignition of hydrogen-air-steam mixtures by a hot hydrogen-argon jet diluted or not by carbon dioxide or steam.

The test facility has been built-up in the laboratory and consists of a stainless steel cylindrical shock tube (4.40 m long, 52 mm i.d.) connected to a stainless steel cylindrical combustion chamber (1.20 m long, 102 mm i.d.). The driven section of the shock tube is filled with a H_2-Ar mixture diluted or not by carbon dioxide or steam and the driver section with helium. These two sections are separated by a double diaphragm. The driven section is coupled to the combustion chamber, filled with an H_2-air-steam mixture, by means of an injection system obturated by a very thin terphane foil. This foil is ruptured about 200 µs after the shock is reflected at the tube end. The gas mixture, in the driven part of the shock tube, is heated up to 1000 - 2800 K by the shock wave just before being spurted through a 2.5 mm diameter orifice into the combustible mixture contained in the combustion chamber. The temperature behind the reflected shock wave is maintained constant in the shock tube for several milliseconds. The driven section of the shock tube is equipped with a series of piezo-electric transducers for the measurement of shock pressure and velocity and the determination of the shock parameters. Diagnostics in the combustion chamber include optical windows coupled to photomultipliers for the detection of the flame propagation and piezo-electric transducers for the measurement of the induced overpressure. The concentration of hydrogen in the combustible mixture has been varied over a wide range, from the lower flammability limit to the upper flammability limit at an initial pressure ranging from 100 to 300 kPa and for two initial temperatures, respectively 373 and 403 K. The hot gas jet is constituted of 60%H_2-40%Ar in which the hydrogen is substituted progressively by carbon dioxide or steam. The initial jet temperature range is from 1000 to 2800 K.

The ignition of H_2-air-steam mixtures by a hot gas jet has been extensively studied and the effect of many parameters has been determined. First of all, the effect of the initial jet temperature on the ignition area: it has been demonstrated that the minimum concentration of steam to be added to 18%H_2-82%air in order to suppress combustion depends on the initial jet temperature.

However, above a critical value of the initial jet temperature, the steam concentration limit tends to be constant. For an initial jet temperature close to this critical value, called "asymptotic temperature", it is possible to define an "ignition limit" corresponding to the ignition by a hot gas jet, which is compared to the classical flammability limit. The ignition limit of the 18%H_2-82%air mixture by a hot 60%H_2-40%Ar jet has been determined also at different initial pressures and temperatures ranging respectively from 100 to 300 kPa and from 373 to 403 K. As the initial pressure is increased, the ignition limit of hydrogen-air-steam mixtures enlarges and the same trend is observed as the initial temperature becomes higher. The ignition ternary diagram of H_2-air-steam has been drawn for an initial temperature and pressure respectively of 403 K and 100 kPa. Finally, the influence of carbon dioxide or steam added to the hot gas jet on the ignition limit is presented. As the diluent concentration increases in the jet, a reduction of the ignition area is observed.

FUEL-COOLANT INTERACTION RESEARCH at the UNIVERSITY OF WISCONSIN

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Over the last two years a research program was begun at the University of Wisconsin to investigate fuel-coolant interaction phenomena in two areas: adding water to a degraded core and vapor explosion benchmark experiments.

The issue of adding water to a degraded core was a natural outgrowth of accident management concerns where the objective is to reach a coolable state and mitigate further core degradation. Our objective was to review the current knowledge. Adding water to degraded core materials could occur within four time intervals during the severe accident: 1) when the core heatup has just begun ($T_f < 1500$ K), 2) after degradation has begun within the core region ($T_f > 1500$), 3) the time of core relation into the vessel lower plenum, and 4) ex-vessel water addition. The former time intervals particularly the second were the major focus of this work where specific emphasis was placed on hydrogen and steam generation during water addition. The presentation will focus on insights from a review of past experimental data and models.

The vapor explosion is a physical event which involves molten fuel mixing with coolant as it pours into a water pool, explosion triggering by vapor film collapse and subsequent escalation and propagation of the explosion. This process can produce dynamic pressures and a mechanical energy release of such a magnitude that may be a hazard during a severe accident. Past experimental studies have focused on the necessary threshold conditions for triggering of an explosion, or integral measurements of the explosion process. Only recently has there been experiments which concentrate on the details of the explosion propagation; i.e., KROTOS. The vapor explosion facility constructed in the last couple of years (WFCI) is designed to continue the study of the details of the vapor explosion propagation as it is affected by mixing conditions. This work will describe the WFCI vapor explosion facility along the initial test series performed to investigate explosion reproducibility, the effect of the artificial trigger and the system inertial constraint. The test results also indicate explosion behavior which is unique and has not been previously reported in controlled vapor explosion experiments. The quantitative results from the tests will also be discussed; i.e., dynamics pressures, expansion work and post-debris analyses. Finally, a technique is proposed which utilizes the explosion data to estimate the premixture conditions.

Large-scale Testing of In-Vessel Debris Cooling Through External Flooding of the Reactor Pressure Vessel in the CYBL Facility

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ABSTRACT

The possibility of achieving in-vessel core retention by flooding the reactor cavity or the "flooded cavity" is an accident management concept currently under consideration for advanced light water reactors (ALWR) as well as for existing light water reactors (LWR). The CYBL (CYlindrical BoiLing) facility is a facility specifically designed to perform large-scale confirmatory testing of the flooded cavity concept. CYBL has a tank-within-a-tank design; the inner 3.7 m diameter tank simulates the reactor vessel and the outer tank simulates the reactor cavity. The energy deposition on the bottom head is simulated with an array of radiant heaters. The array can deliver a tailored heat flux distribution corresponding to that resulting from core melt convection. The present paper provides a detailed description of the capabilities of the facility as well as results of recent experiments with heat flux in the range of interest to those required for in-vessel retention in typical ALWR's. The paper concludes with a discussion of current needs and experiments to be performed for the confirmatory testing of the flooded cavity design for ALWR's under design.

Implications of an HRA Framework for Quantifying Human Acts of Commission and Dependency: Development of a Methodology for Conducting an Integrated HRA/PRA

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As part of an NRC sponsored program evolving from an assessment of human reliability issues in Low Power and Shutdown (LP&S) operations in nuclear power plants (NPPs) an improved approach to human reliability analysis (HRA) is currently being developed. This approach will be consistent with and reflect human behavior based on detailed analysis of actual events that have been encoded into the Human Action Classification Scheme (HACS). It is intended to be fully integrated with probabilistic risk assessment (PRA) methodology and enable a better assessment of the human contribution to plant risk, both during LP&S and at-power operations.

Weaknesses in existing HRA methods and specific areas for concentrated development were identified, based on the insights gained from the relatively short but intense study of human reliability issues in LP&S operations and the experience from applications of existing HRA methods. A detailed program plan outline for producing an integrated HRA/PRA methodology that addresses these weakness, has been developed.

This outline identifies tasks for an assessment of user needs, a refinement of an existing HRA framework, the characterization and representation of errors of commission (EOCs), and the development of an approach to deal with dependency. The outline also identifies anticipated follow-on tasks including the development of a quantification process and implementation guidelines as well as, a demonstration of the guidelines and methodology. NUREG/CR-6093 provides details on the human reliability issues and the program plan outline. The following provides an overview of progress to date beyond that presented at the 20th Water Reactor Safety Meeting with respect to each outline task.

Through the assessment of user needs several findings that the integrated HRA/PRA methodology should address were identified. These findings included the need for:

- A more realistic representation of the dynamic nature of the human-system interaction, especially during response to accidents.

- Facilitating realistic evaluation of multiple factors influencing human performance.

The purpose of the HRA framework is to provide a logical and explicit basis for the development of rules for incorporating human failure events into PRAs that are consistent with knowledge about the consequences and rates of occurrence of different types of human errors. In order for the framework to best describe the relationships between human errors as considered in the behavioral sciences and human failure events as considered in the PRA systems-analysis tasks, an existing framework was selected and refined.

Once refined, this framework provided a basis for incorporating different kinds of human errors into the evaluation of various human failure events. It further provided an indication of the kinds of data relationships that will be required to produce a working HRA/PRA methodology. This framework, therefore, is essential for tasks involving the representation of EOCs and dependency as well as the quantification process.

EOCs have been identified as a critical area for HRA development in this project. The principal reason for this identification is that the state-of-the-art in HRA does not address EOC modeling and that data analyses have shown EOCs to be dominant contributors. The fundamental characteristics of EOCs are being examined in order to develop EOC modeling methods. In support of this development effort an EOC has been defined as "an overt human action that leads to a change in plant configuration with the consequence of a worsened plant state."

Dependency can be viewed as the property of two or more basic PRA events (a, b) involving human actions that causes the following probabilistic relationship $\{P(a,b) \neq P(a) \times P(b)\}$ to be true. There are several different kinds of dependence mechanism that can cause this relationship. For this project the dependency mechanisms of concern are those that influence multiple human actions. Examples being examined include:

- direct dependence on some common external process (e.g., procedure-writing or planning);
- multiple actions in response to a single rule-based mistake (e.g., misdiagnosis);
- task-sequential, single-person (or group) dependencies - errors in performing task A influences reliability of subsequent task B;
- multiple tasks involving common factors that influence human performance (e.g., common supervision).

Once the examination of EOCs and Dependency is completed, the effort for quantification process development will commence followed by the development of implementation guidelines. Finally, a demonstration of the methodology using the guidelines will be conducted by PRA/HRA analysts on appropriately selected events for a BWR and PWR. This demonstration will be used to assess the usefulness and understandability of the guidelines including, their ease of implementation and consistency with expectations and other PRA/HRA results.

RESULTS AND INSIGHTS OF A LEVEL-1 PRA FOR A PWR DURING MID-LOOP OPERATIONS*

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SUMMARY

Traditionally, probabilistic risk assessments (PRA) of severe accidents in nuclear power plants have considered initiating events potentially occurring only during full power operation. Some previous screening analysis that were performed for other modes of operation suggested that risks during those modes were small relative to full power operation. However, more recent studies and operational experience have implied that accidents during low power and shutdown could be significant contributors to risk.

During 1989, the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. The program includes two parallel projects being performed by Brookhaven National Laboratory(BNL) and Sandia National Laboratories(SNL). Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied.

The objectives of the program are to assess the risks of severe accidents initiated during plant operational states other than full power operation and to compare the estimated core damage frequencies, important accident sequences and other qualitative and quantitative results with those accidents initiated during full power operation as assessed in NUREG-1150[1]. The scope of the program includes that of a level-3 PRA.

The objective of this paper is to present the approach utilized in the level-1 PRA for the Surry plant, and discuss the results obtained. A phased approach was used in the level-1 program. In phase 1 which was completed in Fall 1991, a coarse screening analysis including internal fire and flood was performed[2]. The objective of the phase 1 study was to identify potential vulnerable plant configurations, to characterize (on a high, medium, or low basis) the potential core damage accident scenarios, and to provide a foundation for a detailed phase 2 analysis.

In phase 2, mid-loop operation was selected as the plant configuration to be analyzed based on the results of the phase 1 study. The objective of the phase 2 study is to perform a detailed analysis of the potential accident scenarios that may occur during mid-loop

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operation, and compare the results with those of NUREG-1150. The scope of the level-1 study includes plant damage state analysis, uncertainty and sensitivity analysis. Internal fire and internal flood analysis are also included. A separate study on seismic analysis is being performed for the NRC by Future Resources Associated, Inc. and PRD Consulting.

In the phase 2 study[3,4], system models applicable for shutdown conditions were developed and supporting thermal hydraulic analysis were performed to determine the timing of the accidents and success criteria for systems. Initiating events that may occur during mid-loop operations were identified and accident sequence event trees were developed and quantified. In the preliminary quantification of the mid-loop accident sequences[3], it was found that the decay heat at which the accident initiating event occurs is an important parameter that determines the success criteria for the mitigating functions, and time available for operator actions. In order to better account for the decay heat, a "time window" approach was developed. In this approach, time windows after shutdown were defined based on the success criteria established for the various methods that can be used to mitigate the accident. Within each time window, the decay heat and accident sequence timing are more accurately defined and new event trees developed and quantified accordingly. Statistical analysis of the past outage data was performed to determine the time at which a mid-loop condition is reached, and the duration of the mid-loop operation. Past outage data are used to determine the probability that an accident initiating event occurs in each of the time windows. This probability is used in the quantification of the accident sequences.

The final quantification of the phase 2 study is currently underway. It is expected to be completed before the Twenty-first Water Reactor Safety Information meeting. The objective of the paper is to present the approach used and the final results of the phase 2 study. A comparison of the results with those of other shutdown studies will be provided. Relevant safety issues such as plant and hardware configurations, operator training, and instrumentation and control will be discussed.

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Handbook of Methods for Risk-Based Analysis of Technical Specification Requirements[*]

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Summary

Technical Specifications (TS) requirements for nuclear power plants define the limiting conditions for operations (LCOs) and Surveillance Requirements (SRs) to assure safety during operation of these plants. These requirements were, in general, developed based on deterministic analysis and engineering judgments. Experiences with plant operation indicate that some elements of the requirements are unnecessarily restrictive, and some others may not be conducive to safety. Desires to improve these requirements are facilitated by the availability of plant-specific Probabilistic Risk Assessments (PRA).

Use of risk and reliability-based methods to improve TS requirements has gained wide interest because of the capability of these methods to

- quantitatively evaluate the risk impact and to justify changes based on objective risk arguments
- provide a defensible basis for regulatory analysis

USNRC is sponsoring research to develop systematic risk-based methods to address various aspects of TS requirements. The handbook of methods, which is being developed, summarizes methods to apply risk-based methods to improve TS requirements.

The scope of the handbook includes reliability and risk-based methods for evaluating: allowed outage times (AOTs), action statements requiring shutdown where shutdown risk may be substantial, surveillance test intervals (STIs), defenses against common cause failures, managing plant configurations, and scheduling maintenances. For each of these topics, the handbook summarizes analysis methods and data needs, outlines insights to be gained and lists references for more detailed information and presents example evaluations.

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

THE CAPABILITIES AND APPLICATIONS OF THE SAPHIRE 5.0 SAFETY ASSESSMENT SOFTWARE[a]

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ABSTRACT

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

This paper provides an overview of the features and capabilities provided in version 5.0 of this software system. Some of the major new features include the ability to store unlimited cut sets, the ability to perform location transformations, the ability to perform seismic analysis, the ability to perform automated rule based recovery analysis and endstate cut set partitioning, the ability to perform endstate analysis, a new alphanumeric fault tree editor, and a new alphanumeric event tree editor. Many enhancements and improvements to the user interface as well as a significant reduction in the time required to perform an analysis are included in version 5.0. These new features and capabilities provide a powerful set of PC based PRA analysis tools.

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Predicting Earthquake Ground Motions in Western and Eastern North America

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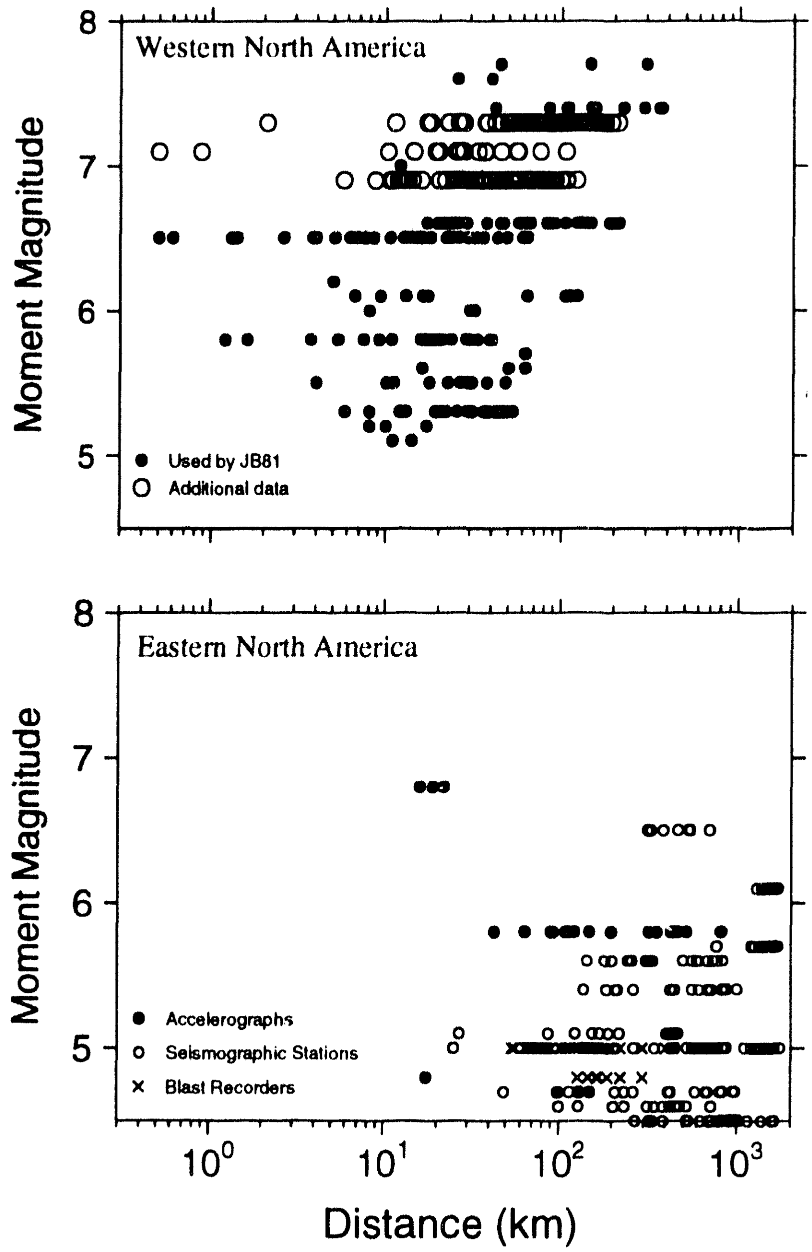
Estimates of ground shaking as a function of distance for earthquakes of specified magnitude are an essential ingredient in the seismic design of nuclear reactors. This estimation is best done by using recordings from past earthquakes. For western North America such data are relatively complete for earthquakes of magnitude less than about 7.5 at distances less than approximately 100 km. This is shown in the figure, where each point represents a single, 3-component (usually) recording. In contrast, as shown in the second part of the figure, data in eastern North America are largely lacking for earthquakes of particular interest to reactor safety. Because of the fundamental differences in the data available, different approaches are used for ground-motion estimation in eastern and western North America.

In western North America, ground motions are usually estimated from equations whose coefficients have been determined from regression analysis. The usual form of the equation contains a few coefficients describing the distance attenuation of ground motions and the magnitude scaling of the motions at a fixed distance. In addition, coefficients describing the dependence on site geology are included. I and my colleague William Joyner are currently revising equations that we published more than ten years ago. The most important differences, relative to our previous work, are the addition of data from three large earthquakes (shown by circles in the figure) and a new scheme of site classification, based on the average shear-wave velocity in the upper 30m of the earth. We find that the site geology has an important effect on the motions at all periods, although the effect is most pronounced for periods greater than about 0.3 secs. In all cases the motions on hard rock sites (with an average shear velocity greater than 750 m/s) are lower than those on softer materials. As we found before, the shapes of response spectra are dependent on earthquake size, and for this reason the derivation of a design spectrum through a scaling of a fixed spectral shape is not advised.

In eastern North America, the preferred way for estimating motions makes use of a simple model that combines seismological information that can be obtained from small earthquakes with engineering notions regarding the randomness of the ground shaking. The idea is simply stated: the ground motions are constructed by assuming that the spectral shape and amplitudes of the motion given by a seismological model are distributed with random phase over a duration controlled by the size of the earthquake and the distance from the source. The model has been applied with good success to predicting motions in many parts of the world from earthquakes whose magnitudes range from less than 0 to greater than 9. I and my colleague Gail Atkinson have recently updated our equations for predicting ground motions in eastern North America. We have used an attenuation model that is somewhat more complex than the simple 1/r model used previously. Most importantly, we have used a source-scaling model for which the spectral amplitudes near 1 Hz are smaller than those used in our previous study. The result is that our motions are lower at periods near 1 sec than before; at higher frequencies our motions are increased slightly over our previous results.



Western North America
Used by JB81
Additional data
Eastern North America
Accelerographs
Seismographic Stations
Blast Recorders
Moment Magnitude
Distance (km)
8
7
6
5
10^0
10^1
10^2
10^3

Liquefaction evidence for strong prehistoric earthquakes in southern Indiana and Illinois

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Features that I interpret to have an earthquake-induced liquefaction origin have been discovered throughout large portions of southern Indiana and southern Illinois. The features are mainly steeply dipping sand- or sand-and-gravel-filled fissures (dikes), which are tabular and dip steeply. The dikes generally narrow upward, and many pinch to a point. However, many of the larger dikes (>0.3 m in width) have served as conduits for venting of sand and gravel onto an ancient ground surface. The vented deposits can exceed 0.3 m in thickness and 10 m in width. The largest sand-filled fissure is 2.5 m in width. Horizontal separation of the ground accompanied the formation of the largest fissures; these lateral movements are interpreted to have been caused by earthquake-induced lateral spreads.

Detailed descriptions of the dikes, the physical settings in which the dikes formed, and the areal distribution and sizes of the dikes discovered as of January 1993 are described in USGS Professional Paper 1536 (Liquefaction Evidence for One or More Strong Holocene Earthquakes in the Wabash Valley of Southern Indiana and Illinois, with a Preliminary Estimate of Magnitude). The paper also contains discussion for the interpretation of an earthquake-induced liquefaction origin to the dikes. Since publication of Professional Paper 1536, many more data have been collected to support the hypothesis of an earthquake origin. Basis for this interpretation are the following:

1. The largest dikes lie within a regional core area that is surrounded by smaller dikes. This core area is in the Wabash River valley along the border of Indiana and Illinois, near Vincennes, Indiana.

2. The largest dikes all seem to have formed at the same time. Radiocarbon and archeological data show that they formed between 5,000 and 6,500 years ago.

3. The only dikes that are unweathered were discovered in areas where there are accounts of liquefaction during the 1811-12 New Madrid earthquakes. These likely 1811-12 features are present near the confluence of the Wabash and Ohio Rivers, and near St Louis, Missouri.

4. Source mechanisms other than earthquake-induced liquefaction are implausible. The dikes have a morphology that is not consistent with an artesian spring origin. Dikes at many

widespread places have formed in physical settings where a landslide origin is thought to be impossible.

5. No dikes have been discovered in the Ohio Valley, upstream from the confluence with the Wabash River. Physical setting of the Ohio Valley is the same as that of other valleys of other rivers where dikes were discovered.

6. The dikes of Indiana and Illinois have a morphology and characteristics that are generally the same as dikes formed during 1811-12 in the epicentral region of the New Madrid earthquakes. Physical setting of alluvial sediments in much of the New Madrid epicentral region is generally similar to that in the valleys of large rivers in southern Indiana and southern Illinois.

Almost all bracketing of ages of dike formation has been in Indiana. The data show that, in addition to the large earthquake event between 5,000 and 6,500 years ago, it is possible that dikes at some sites formed between 10,000 and 14,000 years ago. Less equivocal evidence of other geologically recent strong shaking in the Wabash Valley comes from four (possible six) sites in the central Wabash, lower White, and lower West Fork valleys (all sites within a radius of 30 km to the south, east, and north of Vincennes), where small to medium dikes (1-10 cm wide) penetrate late Holocene sediments. Two radiocarbon dates, combined with limited archeological dating, suggest that the dikes at these sites formed sometime between 4,400 and 1,500 years ago. In Illinois, most of the effort toward bracketing the ages of the dike is being conducted by archeologists and geologists of the Illinois State Museum.

In both Indiana and Illinois, archeological data have proven invaluable as an efficient and effective means of bracketing the time period of dike formation. Without archeological input, bracketing the dike ages over a large region would be almost impossible.

The magnitude of the large earthquake between 5,000 and 6,500 years ago is estimated to be on the order of moment magnitude 7.5 (Professional Paper 1536). An ongoing geotechnical engineering study has the purpose of back-calculating the range of possible accelerations from this ancient earthquake, over a large region. The geotechnical study may provide insight into the estimate of magnitude.

EVIDENCE FOR REPEATED STRONG GROUND SHAKING IN THE NEW MADRID SEISMIC ZONE

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We recognize two, possibly three, episodes of prehistoric liquefaction in the central area of the New Madrid seismic zone (NMSZ), based upon close examination of liquefaction features, their cross-cutting relations, soil characteristics, and associations with Native American occupation layers. Prehistoric liquefaction features indicate that this region has experienced recurrent strong ground shaking during the past 5,000 years. The NMSZ is known for three great earthquakes that occurred near New Madrid, Missouri in 1811 and 1812. A repeat of a great earthquake in the NMSZ today would threaten unprepared structures throughout the central United States (US).

Our interdisciplinary approach, involving archaeology, geology, and pedology, makes it possible, not only to recognize and date prehistoric liquefaction features, but also to correlate liquefaction features across the region and to develop a NMSZ paleoearthquake chronology. To date, our study has been focussed in a 200 km^2 area in northeastern Arkansas and southeastern Missouri, where we have examined both historic (1811 and 1812) and prehistoric liquefaction features at ten sites. Like the historic features, the prehistoric liquefaction features include sand dikes, sand sills, and sand blows characterized by flow structure and clasts of soil and host sediment. Also, the prehistoric features are similar in size to historic features, suggesting that prehistoric events were similar in magnitude to the 1811 and 1812 earthquakes. Sand dikes range from 1 to 80 cm in width; sills range from 1 to 100 cm in thickness; and sand blows reach 1 m in thickness.

At some of our study sites, there is evidence for only a single liquefaction event. At other sites, complex liquefaction structures are indicative of multiple events. In the latter cases, cross-cutting relations of the liquefaction features, as well as the degree of soil development within the features, help to decipher their relative ages. At one site, four ages of liquefaction features occur above a wood-bearing clay layer. Compared with the older features, the youngest liquefaction features exhibit relatively little soil development suggesting that they formed during the 1811 and 1812 earthquakes. The radiocarbon age of wood collected from the underlying clay layer is 4,930$\pm$160 years B.P. Therefore, all four ages of features, three of which pre-date the 1811 and 1812 events, must have formed in the past 5,000 years. At three other sites, Native American occupation layers occur in the upper sections of sand blows. At another site, Native American pits are present in a sand blow crater. In these four cases, artifacts within the occupation layers and pits are typical of the Early period Mississippian culture, indicating that these sand blows formed prior to 800 A.D. to 1,000 A.D. At one of the sites, two sand blows are sandwiched between a Mississippian occupation layer above and a Woodland (500 B.C. to 800 A.D.) occupation layer below. Therefore, both these sand blows pre-date the 1811 and 1812 earthquakes but are no more than 2,500 years old. More advanced soil development within the lower sand blow indicates that it is considerably older (perhaps 500 to 1,000 years) than the upper sand blow. This suggests that the two causative events are widely separated in time. Radiocarbon dating, currently underway, will help to narrow the age ranges of prehistoric features observed at our study sites.

It is well established that soil horizons (e.g., A horizon, B_t horizon) and soil properties (e.g., rubification, consistence, structure, and acidity) develop systematically with age. Evidence of soil development in sand blows and dikes first alerted us that some of the features in our study area are prehistoric in age. An historic (1811 and 1812) sand blow exhibits only minimal soil development in the upper few centimeters, including an increase in soil hue and the development of crumb structure and slightly hard consistence. In addition, soil acidity decreases by only 0.25 of a pH unit over a depth range of 1 m. In contrast, soil development is evident to a depth of 1.2 m within a sand blow that formed during the Early period Mississippian (800 A.D. to 1, 000 A.D.) culture. Soil in this prehistoric sand blow exhibits subangular blocky structure and friable consistence, as well as an increase in soil hue and chroma (rubification). In addition, soil acidity decreases 0.5 of a pH unit over a depth range of 25 cm. By measuring soil properties of liquefaction features of different ages, we will be able to construct local soil development curves that may be useful for estimating the age of liquefaction features in those situations where more conventional methods can not be applied.

Much remains to be resolved about the timing, magnitude, and location of prehistoric earthquakes in the NMSZ. However, our ongoing, interdisciplinary study in the central area of the zone is proving fruitful and we are making progress toward developing a paleoearthquake chronology for the region. Such a chronology would provide insight into the repeat time of damaging earthquakes. Also, by mapping the distribution of contemporaneous liquefaction features, it may be possible to define the source areas and minimum magnitudes of the prehistoric earthquakes.

Geologically Recent Near-Surface Folding and Faulting in the Valley and Ridge Province: New Exposures of Extensional and Apparent Reverse Faults in Alluvial Sediments, Giles County, SW Virginia.

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New excavations for land-fill material along the north side of the New River Valley between Pembroke and Pearisburg in Giles County, Virginia, have revealed a series of extensional and apparent reverse faults cutting alluvial terrace deposits composed of stratified, but unconsolidated, silts, sands and boulder-pebble size gravels. The sediments are situated at 55.0 m above the current level of the New River. Assuming that the sediments are related to the New River, then previously published estimates of erosion rates for this part of the New River (Houser 1980, Mills 1986) would indicate that they are between 190,000 and 3.5 million years old. Detailed sampling and laboratory analysis, using standard sieving and chemical separation techniques, has so far failed to reveal any fossils or microfossils which might facilitate the more precise dating of these highly oxidized sediments.

The stratified sediments are arched into a broad ENE-WSW trending antiform measuring a horizontal distance of at least 95 m from limb to limb, with dips of up to 30° on each fold limb. The calculated fold axis plunges at 7° towards N64°E. A graben structure (Graben 1), measuring 11.0 m in width at its highest currently exposed structural level, is present in the hinge zone of this antiform. The graben is a downward-narrowing structure, defined by two faults with orientations: strike N51°E, dip 71°SE (Fault 1) and N80°E, 56°N (Fault 2). A minimum apparent dip-slip off-set of 8.5 m is indicated by lithological correlation across these graben-defining faults. A second graben structure (Graben 2) is present on the SSE dipping limb of the antiform and is located 6.0 m to the ESE of Graben 1. Graben 2 is a downward-narrowing structure, measuring 5.0 m in width at its highest exposed structural level, and defined by two faults with orientations: strike N46°E, dip 71°SE (Fault 3) and N70°E, 57°N (Fault 4). Faults 3 and 4 display apparent dip-slip off-sets of 1.37 m and 1.0 m respectively. Fault 4, defining the SE margin of Graben 2, displays an apparent dip-slip off-set of 1.0 m. A further extensional fault (Fault 5) is located 6.0 m to the SSE of Fault 4. Fault 5 strikes N56°E, dips at 54° NW and displays an apparent dip-slip off-set of 2.8 m.

Faults 2, 4 and 5 dip away from the New River valley and, together with Faults 1 and 3, are marked by 10-20 cm wide zones of clay-rich fill. Within these zones, the basal planes of individual clay grains are oriented parallel to the planar fault zone margins and define a macroscopic foliation. Lineation on the foliation surfaces is constantly oriented within individual fault zones and, measured downward in the foliation from the SW-W strike direction, pitches at 48°, 37°, 48°, 65° and 70° on Faults 1-5 respectively. Assuming these lineations formed parallel to the total slip vector, then total displacements of approximately 11.4 m, 11.14 m, 1.84 m, 1.1 m and 3.0 m are indicated for faults 1, 2, 3, 4 and 5 respectively. The close geometrical relationships between the observed fold and fault geometries suggests that folding and faulting may be contemporaneous.

In addition to the clay-rich extensional faults described above, three other fault-types are recognized at the Pembroke site. i) Discreet extensional fault surfaces, without any clay fill, typically displaying off-sets of less than 30 cm. Many hundreds of these barren extensional faults have been recorded, and at least some appear to be of syn-sedimentary origin. ii) Barren faults on the limbs of the antiform with apparent reverse-sense off-sets of less than 30 cm. The orientation of these faults with respect to bedding indicates that, if they formed before folding, they could be extensional faults which have been tilted through the vertical during folding to produce apparent reverse-sense off-sets. iii) Reverse faults with 5-10 cm of clay-rich fill located near the hinge-zone of the antiform, whose geometries indicate a reverse-sense of motion even if they formed before folding. These reverse faults display a top to-the N thrust sense on the northern limb of the antiform (with minimum off-sets of c. 60 cm) and a top to-the S thrust sense (off-sets of c. 30 cm) on the southern limb.

Many of these deformation features may post-date sedimentation. However, earlier syn-sedimentary deformation is also indicated by the presence of angular unconformities and small growth faults in the exposed section. This suggests that tilting and faulting of the sediments may have occured over a considerable time interval. We currently interpret these deformation features as indicating that the faults formed in response to tectonic rather than surficial processes, although the possibility that they may be associated with either solution collapse or landsliding must also be borne in mind. An integrated borehole and geophysics program is currently being planned to test these alternative driving-force models for faulting and folding of the New River terrace sediments.

Seismic monitoring studies over the last twenty years in this part of the Valley & Ridge Province have indicated that earthquake foci are located within crystalline basement at depths greater than 5 km. No seismic or geological evidence for near-surface faulting appears to have previously been recorded in this area. Therefore, these faults are of considerable importance to both the assessment of seismic hazard and to our understanding of fault development in this area as they indicate, at least locally, the occurence of geologically recent near-surface faulting.

STATUS OF PALEOSEISMIC INVESTIGATIONS IN THE SOUTHEASTERN AND NORTHEASTERN UNITED STATES

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A systematic search for paleoliquefaction evidence of large prehistoric earthquakes both within the meizoseismal area of the 1886 Charleston, SC earthquake and elsewhere along the southern to mid-Atlantic seaboard indicates that six other liquefaction episodes may have occurred in coastal South Carolina during the mid- to late Holocene. Importantly, no evidence of large pre-historic earthquakes along the Atlantic seaboard outside of South Carolina has been found. As an extension of these studies we implemented neotectonic/paleoliquefaction studies in non-coastal areas of the Southeast and in several areas of the northeastern United States.

Within the Southeast field studies were implemented in four inland areas, each the locale of historic or instrumental seismicity. They include: (1) Bowman, SC, (2) Union, SC, (3) eastern Tennessee, and 4) Giles County, VA. Investigations in the Bowman and Union, SC areas, focused on fluvial deposits located proximal to historic seismicity. Although, field studies confirm that potentially liquefiable sediments are present, to date no seismically induced liquefaction (SIL) features have been found.

Field investigations within eastern Tennessee and Giles County, VA include the search for evidence of SIL features, as well as the use of more region specific neotectonic techniques such as the identification and analysis of Holocene landslides and deformation in caves. Some of the largest prehistoric landslides in the eastern United States are present in the Giles County seismic zone. Preliminary stability analyses suggest a possible seismic origin for these failures. Studies are underway to better determine the age and origin of these slides. For example, we are conducting age-dating studies of bogs, which have formed in pull-apart basins on the updip side of many of these landslides. In addition, we are also conducting studies at Mountain Lake, VA; this is the only natural lake in the Southern Appalachians and was apparently formed by a prehistoric landslide.

Our Giles County investigations have also searched alluvial floodplain and terrace deposits for evidence of seismically induced deformation. Although some faulting of these geologically young materials was observed, it is most probably related to karst development in underlying rocks. Common sinkhole occurrence in the underlying Valley and Ridge rocks necessitate that any hypothesis purporting a seismic origin, first discount the effect of karst.

One new neotectonic method we are investigating is based on the assumption that speleothem (cave) deposits in the Southern Appalachian region possibly recorded the effects of very large prehistoric earthquakes and, thus, provide insights to the region's Holocene seismic characteristics. We are in the process of investigating Tuckaleechee and Indian

Caves in Eastern Tennessee and Tawny's and New River Caves in Giles County, VA. The temporal and spatial trends linking breakdown episodes, such as fractured and offset columns or stalactite breakdown with new speleothem growth, will be critical in determining the genesis of these features.

In the northeastern United States our investigations focused on the Moodus, CT, Newbury, MA, and the Ossipee, NH areas, each a locale of historic seismicity. In the Moodus area several investigators have identified and suggested a seismic origin for wedge-shaped sediment deformation structures present in late Wisconsinan glacial outwash deposits. Our studies of these features show that no grain size, mineralogical, chemical, or morphological relationship exists between the wedge sediments and the underlying "hypothesized" source bed. We, thus, rule out a liquefaction origin for the features. Near the location of the 1727 Newbury, MA event, we are presently searching tidal marshes, which are underlain by potentially liquefiable sands, for the presence of injected sand dikes and lateral spreading features.

The 1940 Ossipee earthquake was not large enough to cause seismically induced liquefaction. However, we have trenched potentially liquefiable materials in the epicentral area to determine if larger prehistoric earthquakes may have occurred. To date no SIL features have been found. Radiocarbon dating of these sediments will provide an indication of the length of time that a large earthquake **has not** occurred in this area. We are also investigating the area for seismically triggered landslides. As a final task in the Ossipee area, overwater geophysical surveys at Winnipesaukee, Silver, and Ossipee Lakes are being conducted. Our reasoning is that a large prehistoric seismic event possible induced subaqueous slope failures that would be preserved as scars, mounds, or boulder piles on the lacustrine slopes or bottoms; thus, sub-bottom and sonar profiles should reveal the presence of these features. Preliminary results show no obvious features that might be related to scar or mound areas associated with slope failure of subaqueous sediments.

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This report contains summaries of papers on reactor safety research to be presented at the 21st Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel, Bethesda, Maryland, October 25-27, 1993. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, U.S. NRC. Summaries of invited papers concerning nuclear safety issues from U.S. government laboratories, the electric utilities, the Electric Power Research Institute (EPRI), the nuclear industry, and from foreign governments and industry are also included. The summaries have been compiled in one report to provide a basis for meaningful discussion and information exchange during the course of the meeting and are given in the order of their presentation in each session.

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PLENARY SESSION IN CONGRESSIONAL BALLROOM			
	GRAND BALLROOM-B/C	GRAND BALLROOM-D/E	CHEVY CHASE ROOM
Mon. AM	1 Severe Accident Research I N. Grossman	2 Primary System Integrity I C.Z. Serpan, Jr.	3 Advanced Reactor Research R. Meyer
Mon. PM	4 Severe Accident Research II R. Foulds	5 Primary System Integrity II C.Z. Serpan, Jr.	6 Thermal Hydraulics D. Bessette
Tues. AM	7 Severe Accident Research III R. Wright	8 Aging Research, Products & Applications G. Weidenhamer	9 Advanced Control System Technology J. Kramer
Tues. PM	10 Severe Accident Research IV R. Lee	11 Advanced Instrumentation and Control Hardware C. Antonescu	12 Human Factors Research J. Persensky
Wed. AM	13 Structural & Seismic Engineering J. Costello	14 Thermal Hydraulic Research for Advanced Passive LWRs L. Shotkin	15 Severe Accident Research V Y. Chen
Wed. PM	16 Probabilistic Risk Assessment Topics M. Drouin	17 Seismology & Geology R. McMullen	
LUNCH AND RECEPTION IN CONGRESSIONAL BALLROOM			

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