Transactions of the Nineteenth Water Reactor Safety Information Meeting

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Compiled by: Allen J. Weiss, Meeting Coordinator

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PREFACE

This report contains summaries of papers on reactor safety research to be presented at the 19th Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel in Bethesda, Maryland, October 28-30, 1991. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, USNRC. Summaries of invited papers concerning nuclear safety issues from U.S. government laboratories, the electric utilities, the Electric Power Research Institute (EPRI), the nuclear industry, and from the governments and industry in Europe and Japan are also included. The summaries have been compiled in one report to provide a basis for meaningful discussion and information exchange during the course of the meeting, and are given in the order of their presentation in each session.

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

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The coding developed within the BWRSAR code framework for calculating the behavior of a BWR lower plenum debris bed after dryout and the associated vessel bottom head response is being made operational within the MELCOR code at Oak Ridge. After successful testing of these lower plenum debris bed and bottom head models within the structure of a local version of MELCOR, recommendations for formal adoption of these models will be made to the NRC and to the MELCOR code development staff at Sandia National Laboratories (SNL).

The purpose of these models is to provide a means for calculating the time-dependent release of core and structural materials from the reactor vessel for accident sequences in which a dry debris bed is formed in the vessel lower plenum. Decay heating, energy release by zirconium oxidation, conduction, and radiative heat transport are represented within the bed as well as the effects of material melting, relocation, and freezing at a lower level. The melting and freezing processes include consideration of the formation of eutectic mixtures within the debris, as defined by user input.

If bottom head penetration failures occur, the release pathway is by overflow of molten materials within the debris bed into the instrument housing guide tubes and subsequent passage through the reactor vessel wall. Consideration is given to the potential for this flowing liquid to ablate the material surrounding the original instrument housing guide tube locations in both the lower portions of the debris bed and in the vessel wall.

The vessel bottom head is divided into a series of calculational nodes for the wall sections adjacent to the debris bed, between the upper surface of the bed and the shroud baffle, and adjacent to the downcomer region. For the purpose of calculating the bottom head wall temperatures, each node is divided into three equal-volume radial segments. Heat is transferred from the debris bed into the adjacent wall nodes and along and across the wall from segment to segment by conduction. Wall nodes above the elevation of the upper bed surface are heated by radiation from the bed. Any water remaining within the downcomer region is treated as a heat sink for the upper bottom head wall. Heat transfer from the outer segment of each wall node to the drywell atmosphere is represented.

In addition to the requirement for provision of information exchange at the lower plenum model interface, some local modification of MELCOR has been necessary to identify the mass, energy, and decay heat associated with the debris within the lower plenum at the time of dryout so that the initial bed configuration can be established. Any materials subsequently predicted by MELCOR to be relocated downward from the core region are added to the upper bed with their appropriate energies and any associated decay heat. All of these changes will be reviewed independently by the MELCOR development staff at SNL prior to incorporation into the master code.

THE MELCOR PEER REVIEW
PROCESS AND FINDINGS

by

B. E. Boyack, Los Alamos National Laboratory
V. K. Dhir, University of California at Los Angeles
T. J. Haste, UK Atomic Energy Authority, Winfrith
J. A. Gieseke, Battelle Columbus
M. A. Kenton, Gabor, Kenton, and Associates
M. Khatib-Habbar, Energy Research, Incorporated
M. T. Leonard, Science Applications International Corporation
R. Viskanta, Purdue University

Overview: MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants. MELCOR is being developed at Sandia National Laboratories for the US Nuclear Regulatory Commission (NRC) as a second-generation plant risk assessment tool and as the successor to the Source Term Code Package.

MELCOR has been under development since 1982. The newest version of MELCOR, MELCOR 1.8.1, was released in June 1991. The code has now reached sufficient maturity that a number of organizations inside and outside the NRC are using or are planning to use the code. Although the quality control and validation efforts are in progress, the NRC identified the need to have a broad technical review by recognized experts to determine or confirm the technical adequacy of the code for the serious and complex analyses it is expected to perform. A peer review committee was organized using recognized experts from the national laboratories, universities, MELCOR user community, and independent contractors to perform this assessment.

The objective of this paper is to summarize the peer review process and to summarize the findings of the MELCOR Peer Review Committee formed to conduct the MELCOR peer review.

Committee Charter: The charter of the MELCOR Peer Review Committee was to (1) provide an independent assessment of the MELCOR code through a peer review process (2) determine the technical adequacy of MELCOR relative to the code design objectives and the code's targeted applications, and (3) issue a final report describing the technical findings of the Committee.

Peer Review Process: The Committee developed and followed a multi-step process for the the MELCOR Peer Review. The steps in the process were as follows.

1. Identify design objectives for the MELCOR code.
2. Identify targeted applications for the MELCOR code.
3. Identify the MELCOR code version to be reviewed.
4. Identify and distribute the MELCOR "Document Data Base" to Committee members.
5. Select one boiling water reactor (BWR) plant and one pressurized water reactor (PWR) plant for subsequent review steps.
6. Select severe accident scenarios for the selected plants.
7. Develop a common Committee perspective regarding technical adequacy.
8. Identify dominant phenomena for the plants and scenarios.
9. Define "Standard of Technical Adequacy" to be used in developing findings.
10. Define a process for reviewing technical adequacy.
11. Assess technical adequacy of individual models and/or correlations within the MELCOR phenomenological packages (bottom-up review).
12. Assess technical adequacy of the integral code against the MELCOR design objective and the MELCOR targeted applications (top-down review).

**Major Findings**

1. The Committee found that considerable progress has been made in developing the MELCOR code. The component parts of MELCOR have been developed and assembled such that integrated calculations of some severe accident sequences in both BWRs and PWRs can be completed. A limited number of benchmarks have been prepared for some of the individual models and correlations and a limited set of benchmarks have been completed for the integrated code. An extensive set of documentation has been prepared, including a code manual, reference manuals for the phenomenological packages, and users' guides.

2. The Committee determined that MELCOR is not a completed code. The Committee also determined that additional development is needed before MELCOR can reasonably satisfy its design objectives and can be applied with confidence to its targeted applications.

3. The NRC has identified six MELCOR [design] objectives. They are that (1) MELCOR should appropriately and consistently model phenomena essential to the description of severe light water reactor accidents, (2) MELCOR should provide best-estimate predictions of the progression and consequences of severe accidents, (3) MELCOR should permit estimates of the uncertainties associated with its predictions of severe core damage accidents to be made without requiring modifications to the code, (4) Models incorporated in MELCOR should be in adequate detail to address the phenomenology but permit a practical running time for accident sequence analysis, (5) MELCOR should be applicable for severe accident studies and related plant applications for both PWRs and BWRs, and (6) MELCOR should be portable, maintainable, and have a structure that facilitates incorporation of new or alternative phenomenological models. The Committee findings relative to these code objectives are too extensive to present in this brief summary. One key Committee finding was that code numerics are the source of a primary concern regarding the technical adequacy of the code. A more complete listing of findings relative to the code objectives are provided in the full paper.

4. The Committee reviewed the technical adequacy of MELCOR relative to its primary targeted applications of probabilistic risk assessment (PRA) and accident management (AM). Of these two applications, the PRA usage is deemed more important by the NRC. For either PRA or AM applications, the Committee concluded that several important phenomena are not currently modeled in MELCOR. The absence of these key models limits the number of event sequences that can be studied. The Committee also identified some MELCOR models for which the physics are generally understood but are not correctly represented within the code.

5. The Committee recommended that a more extensive program of MELCOR sensitivity studies and code validation using test data be conducted.

6. Finally, the Committee recommended that documentation of the pedigree, applicability, and fidelity of the MELCOR models and correlations be improved and that documentation of user guidelines be expanded.
Planned MELCOR Improvements and Assessment*

Randall M. Summers
Lubomyra N. Kmetryk
Sandia National Laboratories

MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants. 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 and pressurized water reactors. MELCOR has been especially designed to facilitate sensitivity and uncertainty analyses. Its current uses include estimation of severe accident source terms and their sensitivities and uncertainties in a variety of applications.

Version 1.8.1 of MELCOR, frozen in March 1991 and distributed in July, included several significant error corrections, improved numerics to increase computational efficiency, and improvements in input and output processing. The capability to define control volumes with fluid conditions specified by tables or user-defined control functions was added. Several problem areas in the modeling of radionuclide behavior were addressed, including fission product removal by sprays, the transport of fission products with bulk fluids, aerosol mass conservation, and beta decay energy absorption in control volumes. The thermal coupling of heat structures to the hydrodynamics was modified to mitigate oscillatory heat transfer behavior, and the water condensation/evaporation model was altered to eliminate discontinuous behavior with the introduction of noncondensibles to a pure steam environment. More flexibility and user control were provided for the failure of structures such as the core plate. Modeling of the effects of conglomerate debris on convective and radiative heat transfer rates was added, and limits were placed on heat transfer rates from particulate debris beds by applying a dryout heat flux correlation. Modeling of transport properties for pure fluids and fluid mixtures was substantially enhanced. New correlations to treat debris-concrete and interlayer heat transfer were incorporated and new material properties were added in the MELCOR implementation of CORCON-MOD2.

MELCOR 1.8.1 has recently undergone a comprehensive technical review by recognized experts to determine the technical adequacy of the code for the serious and complex analyses it is expected to perform. The review process and findings are documented in a previous paper for this session. These findings corroborated those of a prior review conducted internally at Sandia, and work had already begun in several areas recognized as deficient. Modeling of ice condensers has recently been incorporated by modifying the heat structure degassing model to simulate the melting of ice and adding additional parameters to properly treat deposition and drainage of fission products. Improvement in the treatment of natural circulation is underway to model fine-scale circulation patterns within the core and between the core and upper plenum, and to

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treat single-phase counter-current flow between the reactor vessel and hot leg piping or steam generator. Adaptation of direct containment heating models from CONTAIN has begun, with additional parametric capabilities and user flexibility planned. Implementation of CORCON-MOD3 into MELCOR is also underway, which will include improved bubble behavior, interlayer mixing, improved chemistry, and integration of VANESA fission product release modeling, as well as the addition of a time-dependent melt radius option. The addition of models to treat interactions of core materials (e.g., Zr-UO₂) is ongoing, and studies are underway at ORNL to investigate the effects of more sophisticated modeling on lower plenum debris behavior and reactor vessel failure.

Additional improvements are planned for the near future. Several users have encountered severe numerical sensitivities to computer type, time step size, and small changes in modeling parameters. This was a high-priority concern of the peer review committee, and efforts will be made to identify the causes of such behavior and eliminate them. Improvements are planned for the treatment of interfacial momentum exchange to include flow path geometry effects, to treat more appropriately the draining of liquid pools (e.g., in the pressurizer), and to correct difficulties in modeling two-phase natural circulation and reflux cooling. New CORSOR models for fission product release developed at Battelle Columbus that include mass transport limitations, Booth diffusion modeling, and updated release coefficients are planned for implementation. Modeling of water condensation and evaporation is to be upgraded to include high mass transfer effects and the thermal resistance of condensate films on heat structure surfaces and to properly treat the thermal resistance of a stable stratified saturated liquid layer in a subcooled water pool. New models are planned to treat the full or partial quenching of debris as it pours from the core to the lower plenum and to treat ex-vessel coolable debris beds prior to the initiation of core-concrete interactions. Several fission product model additions are also planned to treat vapor scrubbing in water pools, turbulent flow deposition and inertial (flow-directed) impaction mechanisms, surface reactions, and aqueous chemistry.

One of the key findings of the peer review was the need for expanded MELCOR assessment. A comprehensive multi-year assessment plan has been developed and activities are underway to meet this need. Only a small portion of the plan has been accomplished so far, but it is a high priority to obtain assessment results for each of the major phenomena treated by the code as soon as possible, in particular to provide input for user guidelines. The plan anticipates the participation of a number of organizations, including universities and foreign institutions. Assessment calculations have recently been completed and documented for the LACE LA4 aerosol transport experiment, the FLECHT SEASET natural circulation tests, the HDR V44 steam blowdown experiment and T31.5 hydrogen mixing standard problem (ISP-23), and the PHEBUS B9+ core damage standard problem (ISP-28). Work has also begun on calculations for the LOFT LP-FP-2 integral in-vessel core degradation test and the ACRR ST-2 fission product release test. Additional assessment calculations planned for the near future include the CORA 13 core degradation standard problem (ISP-31); the ACRR DF-4 BWR fuel bundle damage and relocation test; the Semiscale S-SG-7 integral PWR steam generator tube rupture test; the Marviken-V ATT-2b and ATT-4 aerosol transport tests; the SURC-2 urania-concrete interaction test; PNL ice condenser tests 11-6 and 16-11; and the ACRR MP-1 late-phase melt progression test.
NEW CONTAINMENT MODELING FEATURES OF THE CONTAIN CODE

K. K. Murata, D. C. Williams, R. G. Gido, R. O. Griffith,
K. E. Washington, and D. Y. L. Louie
Sandia National Laboratories
Albuquerque, NM 87185

SUMMARY

The CONTAIN code is the United States Nuclear Regulatory Commission’s (USNRC) best-estimate code for the integrated analysis of phenomena in reactor containments during severe accidents. While the most recent complete set of documentation applies to the CONTAIN 1.10 code version,[1] two major revisions beyond CONTAIN 1.10 have recently been released. The purpose of this paper is to highlight the new features of the recent revisions, as well as discuss the new features currently under development. Discussion will be limited to those features that address containment issues either through new or significantly improved modeling or by providing the user with a means of conveniently assessing uncertainties for those modeling areas in which the phenomena are poorly understood. The variants of CONTAIN that model reactors with heavy water or liquid metal coolant are not discussed here but are documented elsewhere.[2,3]

The recently released code versions are CONTAIN 1.11 and 1.12. The new features of CONTAIN 1.11 that are discussed here include (1) a quasi-mechanistic concrete outgassing model, (2) the connected structure option for heat transfer structures, and (3) a model for forced convective heat transfer. The recently revised outgassing model addresses a potentially important source of steam that could significantly affect pressures, the distribution of heat loads, and the steam inerting of hydrogen burns. The connected structure option allows heat conduction between compartments through a common wall to be modeled, while allowing a full range of processes to be modeled on the exposed faces. The forced convective heat transfer model calculates gas velocities in a compartment from flow path velocities for use in heat transfer correlations and in the debris trapping models discussed below.

The principal new feature of CONTAIN 1.12 is the direct containment heating (DCH) models. These models represent a major extension of code capabilities. The single field modeling of the suspended core debris droplets is similar to the interim model used in earlier analyses.[4] However, the trapping models have been made more mechanistic. Also, cavity dispersal models, which were not part of the interim model, have now been incorporated into CONTAIN.

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**Los Alamos Technical Associates, Albuquerque, NM.
New code features currently under development include a revised model for hydrogen burns. The revised model includes autoignition criteria and treats continuous hydrogen burns. The need for such modeling was clearly indicated in earlier DCH analyses[4] and is also recognized as important for hydrogen burns when igniters are operating. To demonstrate the revised model, CONTAIN calculations comparing the old and the revised models are presented. The effects of the burn modeling on the predicted pressures and thermal loads within a containment are discussed.

New DCH modeling is also under development to remove significant modeling limitations identified in earlier analyses. A multi-field formulation for suspended debris drops has been developed to replace the single field model in CONTAIN 1.12, and a chromium chemistry model has been added. The multi-field formulation is designed to track debris droplets with different debris composition, temperature, and size, whereas only average properties can be tracked in the single field model. Thus, the effects of distributions in droplet composition, temperature, and size and the correlations between these quantities cannot be readily evaluated in the latter model. Illustrative calculations will be presented to illustrate the types of differences found between the single field and multi-field approaches, for postulated distributions in the droplet properties at injection.

REFERENCES


Technology for the analysis of crack initiation and arrest is central to the reactor pressure vessel fracture-margin-assessment process. Regulatory procedures for nuclear plants utilize this technology to assure the retention of adequate fracture-prevention margins throughout the plant operating license period. As nuclear plants age and regulatory procedures dictate that fracture-margin assessments be performed, interest in the fracture-mechanics technology incorporated into those procedures has heightened. This has led to proposals from a number of sources for development and refinement of the underlying crack-initiation and arrest-analysis technology. An important element of the Heavy-Section Steel Technology (HSST) Program is devoted to the investigation and evaluation of these proposals. This paper presents the technological bases and fracture-margin assessment objectives for some of the recently proposed crack-initiation and arrest-technology developments. The HSST Program approach to the evaluation of the proposals is described and the results and conclusions obtained to date are presented.

Shallow Flaw Fracture Toughness

Pressurized thermal-shock loading produces maximum stresses adjacent to the inner surface of a reactor vessel. Irradiation embrittlement effects are also most severe at this same location. These effects combine to produce a preponderance of crack initiations in a probabilistic PTS analysis from shallow, inner-surface flaws. An accurate characterization of fracture toughness for shallow flaws is therefore of primary importance in the analysis of the conditional probability of vessel failure under PTS loading.

Fracture toughness for shallow flaws has been observed to be higher than that for deep flaws. This result is attributed to relaxation of constraint at the tip of a shallow flaw due to the proximity of the free surface. Shallow-flaw fracture-toughness tests of A533B material are currently in progress within the HSST Program. Large-scale specimens are used in these tests in order to permit the fracture-toughness results to be interpreted in terms of both the relative crack depth a/W and the absolute crack depth a. Preliminary results indicate an elevated fracture toughness for A533B in the lower-transition region consistent with that reported by other investigators for A36 and A517 steels.

Preliminary analyses have shown that the observed increase in fracture toughness for shallow flaws can have a significant impact on the predicted vessel failure probability. Equally important is the demonstration of a reduced fracture potential for a class of flaws which is particularly difficult to detect using currently available nondestructive examination techniques.
Dynamic Crack Arrest

Prior HSST Program research has demonstrated crack arrest behavior at $K_{\text{Ia}}$ levels substantially higher than the 220 MPa\(\sqrt{\text{m}}\) (200 ksi\(\sqrt{\text{in}}\)) upper-bound value given in the ASME Section XI Appendix A, rules for the analysis of flaws. Application of an extended $K_{\text{Ia}}$ curve to the analysis of vessel failure probability under PTS transient loading, however, produced disappointing results. Cleavage cracks were found to arrest at depths where the onset of ductile tearing rendered them unstable. This result has motivated proposals for a refinement of crack-arrest analysis models which would have the effect of arresting the crack at a shallower depth. One such model has been termed the Static Equivalence Model (SEM). The basic premise behind the SEM is that the crack propagation is so fast that the reactor vessel inertia will prevent its movement up to the time of arrest. The basic question associated with such a model relates to the stability of the arrested crack.

Thermoelastic dynamic analyses were performed using a 2-D plane-strain finite-element model of the flawed vessel. The model was analyzed using an ORNL modified version of the ADINA (Automatic Dynamic Incremental Nonlinear Analysis) general purpose finite-element code. Dynamic crack-initiation toughness has been estimated using strain rates obtained from the dynamic solutions together with the available dynamic fracture-toughness data for A 533 B. The dynamic analysis predicted re-initiation after an initial arrest. Crack-initiation depths predicted by the dynamic model were significantly greater than those predicted by the SEM.

Circumferential Flaws

Pressure loading in a reactor vessel induces maximum membrane stresses and tensile strains in the hoop direction. Any flaws embedded in a circumferential weld will experience positive strains parallel to the crack front. This condition is not duplicated in any of the standard fracture-toughness test configurations. A concern exists that the positive stresses acting parallel to the crack front will increase the hydrostatic to von Mises effective stress ratio at the crack tip and, thereby, increase maximum stresses by inhibiting crack-tip yielding. The net result of this condition could be a reduction in fracture toughness for circumferential flaws relative to that measured in plane-strain fracture-toughness tests.

The ultimate determination of the effect of net-section positive out-of-plane stresses and strains on fracture toughness will require biaxial fracture tests conducted using realistic, reactor vessel stress levels and stress ratios. The HSST Program is evaluating the utility of an exploratory series of such tests. An analytical study of crack-tip constraint in circumferential welds has been undertaken as a part of that evaluation.

A 2-D ABAQUS elastic-plastic finite-element model was used in the analysis. The same model cross section was analyzed in both plane-strain, end-loaded, and axisymmetric pressure-loaded configurations. Plastic zone size, hydrostatic to von Mises effective stress ratio ($\sigma_p/\sigma_e$) and the O'Dowd-Shih triaxiality parameter, Q, were all evaluated as indicators of the effect of constraint. Preliminary results show a small influence of positive out-of-plane strain on crack-tip constraint. A larger influence is obtained from the radial compressive stresses in the pressure-loading case. These stresses act to increase the von Mises equivalent stress and decrease the hydrostatic stress, thereby decreasing the crack-tip constraint. This may be an important factor to be considered when interpreting results from pressure loaded fracture tests where high pressures are used to compensate for a reduced vessel r/t ratio.

Results presented in this report were generated by the HSST Program staff at ORNL and by the University of Maryland under an HSST subcontract.
Pipe Fracture Behavior under High-Rate (Seismic) Loading
-- the IPIRG Program

by

R. A. Schmidt, G. M. Wilkowski,
P. M. Scott, and R. J. Olson

Battelle
Columbus, Ohio

The International Piping Integrity Research Group (IPIRG) Program was an international group program managed by the USNRC and funded by a consortium of organizations from nine nations: Canada, France, Italy, Japan, Sweden, Switzerland, Taiwan, the United Kingdom, and the United States. A Technical Advisory Group, consisting of experts from all member countries, met formally twice a year to exchange ideas, data, and analysis and to provide input on broad program direction. Active participation from the members provided significant benefits by establishing an international consensus on issues of common interest. The five-year program was conducted at Battelle in Columbus, Ohio, and was completed in July 1991.

The objective of the program was to develop data that are needed to verify engineering methods for assessing the integrity of nuclear power plant piping that contains circumferential defects. The program encompassed numerous tasks including material characterization studies, updates of a pipe fracture data base, seminars and workshops, and a leak-rate investigation that involves experiments, analysis, and computer code development, but the primary focus was an experimental task designed to investigate the behavior of circumferentially flawed piping and piping systems subjected to high rate loading typical of seismic events. These data are considered essential to verify the validity and degree of conservatism of current LBB analyses and in-service flaw assessment methods.

The behavior of flawed piping and piping systems subjected to high rate loading was investigated by conducting both "separate effects" experiments on simple pipe specimens and full-scale experiments on a large-diameter piping system tested at PWR conditions. The separate effects experiments provided an evaluation of the effects of loading rate and cyclic loading on the fracture behavior of flawed pipe subjected to displacement-controlled loads and inertial loads. This paper will briefly describe results from these experiments which were conducted on 6-inch diameter pipe at 288 C; some also contained water pressurized to 15.5 MPa. The test matrix included various combinations of (1) carbon steel and stainless steel pipe, (2) through-wall cracks, external surface cracks, and an internal surface crack, (3) quasi-static loading and high strain-rates typical of seismic events, and (4) monotonic and cyclic loading with load ratios of 0 and -1.

To obtain critically needed data on a large-diameter representative piping system under combined inertial and displacement-controlled stresses, a unique test facility was designed and constructed. A pipe loop was fabricated that consisted of 30 meters of 16-inch diameter Schedule 100 pipe, five long-radius elbows, and a large mass that simulated a swing
check valve. Pipe supports were specially designed and constructed to provide well-defined boundary conditions and were not intended to simulate supports found in actual plants. These supports consisted of hangers with spherical bearings, vertical supports with hydrostatic bearings, and fixed ends set in massive steel and concrete supports. These well-defined boundary conditions provided a valuable test bed for evaluating analytical tools and models. Five cracked specimens were cyclically loaded to failure at approximately 4 Hz in this facility under PWR conditions. Specimens included circumferential surface cracks placed in welds and base metal of carbon steel, wrought stainless steel, and a section of artificially aged cast stainless steel pipe. The paper will briefly describe the results of these pipe system experiments and will summarize the key conclusions reached for the overall program.

Various stress analyses were conducted to design the test facility, design the experiments, and to evaluate the results. Linear-elastic finite element analyses with 0.5 percent damping were shown to give an excellent description of the response of the uncracked piping system. The dynamic behavior of the pipe system containing a section of surface-cracked pipe was accurately modeled with a nonlinear, time-dependent finite element analysis that included a nonlinear spring model for the cracked section. Fracture behavior of the experiments were analyzed using predictive fracture methods (Net-Section Collapse, Dimensionless Plastic Zone Parameter, and J-estimation) and Code analyses (R6 and ASME Section XI).

Key conclusions include:

(1) Dynamic strain aging in some carbon steel pipes can lower the fracture resistance at loading rates of interest in seismic applications. There was little effect of the high-rate loading on the tensile and fracture behavior of the stainless steels and stainless steel welds evaluated.

(2) Cyclic loading with negative load ratios and small incremental crack growth can lower the load-carrying capacity of flawed steel pipe.

(3) Inertial loading can contribute to crack instability and should usually be treated as a load-controlled component of stress.

(4) Data from C(T) tests can overestimate the effective fracture resistance observed in a surface-cracked pipe system experiment by as much as a factor of seven.

(5) A double-ended guillotine break (DEGB) is probably a very unlikely occurrence unless a very long surface crack exists in the pipe. A DEGB was achieved in two of the five pipe system experiments, but only after extensive cyclic through-wall crack growth occurred. This degree of stability was unexpected and may be due to the reduction in bending moment caused by the thrust forces produced by the steam jetting from the crack. The time required to produce a DEGB, even in the most severe circumstances, is probably much longer than the design basis of 1 millisecond.
This program started on March 23, 1990, and has a duration of 4 years. The objective of the program is to develop and verify analyses by using existing and new experimental data for circumferentially cracked pipes, so modifications and improvements can be made to LBB and in-service flaw evaluation criteria. There are 7 technical tasks with the following specific objectives. In general, they deal with circumferentially cracked straight pipe under quasi-static loading.

Task 1, Short TWC Pipe Evaluations, has the objective to modify and verify analyses for short through-wall-cracked (TWC) pipe using existing and new data on large diameter pipe. The regulatory impact of these results can help to refine analyses that have been used for LBB crack sizes in large diameter pipe. FY91 efforts involved:

- conducting two large diameter and one small diameter pipe experiments,
- development of a J-estimation scheme capable of accounting for the strength of the weld metal in its analysis,
- deriving improved functions for the GE/EPRI elastic-plastic handbook for short through-wall circumferential cracked pipe in bending, and
- assessment of ovalization effects for short cracks by improving the analyses to be consistent with maximum load predictions for uncracked pipe.

Task 2, Short Surface-Cracked Pipe Evaluations, has the objective to modify and verify analyses for short surface-cracked (SC) pipe using existing and new data on large diameter pipe. These results have the regulatory impact to verify and possibly refine analyses that have been used for in-service flaw evaluations such as those in ASME Section XI. FY91 efforts involved: designing experiments with the smallest circumferential flaw that would fail by fracture rather than by buckling. Two 6-inch-diameter and one 16-inch-diameter stainless steel pipe experiments with short surface cracks were conducted. Additionally, one 28-inch-diameter stainless steel recirculation line pipe test was conducted with a short surface crack in an SAW. Improvements were also made in the SC.TNP and SC.TKP surface-cracked pipe J-estima-
tion analyses to extend these analyses to external as well as internal finite-length circumferential surface-cracks in pipes.

Task 3, Bi-metallic Cracked Pipe Evaluations, has the objective to develop experimental data and analyses for cracks in bi-metallic welded pipe. No efforts were conducted in FY91.

Task 4, Dynamic Strain Aging and Crack Jump Evaluations, has the objective to develop a screening criterion to assess which ferritic steels are susceptible to dynamic strain aging and unstable crack growth at LWR temperatures. This methodology is applicable to both LBB and in-service flaw evaluations for ferritic pipe. The FY91 efforts involved conducting tensile, hardness, and C(T) tests on five different ferritic steels at a variety of temperatures to see if high temperature hardness tests can be used as a screening criteria for determining if a carbon steel is susceptible to dynamic strain aging.

Task 5, Anisotropic Fracture Evaluations, has the objective to assess if anisotropic fracture toughness properties can cause failure stresses to be less than those typically calculated by current LBB analyses. This has a direct impact on LBB analysis procedures. The FY91 efforts involved conducting FEM analysis to determine the crack driving force relations for a crack that starts growing in a helical angle from a circumferential crack, metallurgical evaluations of several nuclear ferritic pipes, and conducting tensile and C(T) tests with different angled crack orientations.

Task 6, Crack-Opening-Area Evaluations, has the objective to improve the crack-opening-area predictions used in LBB leak-rate analyses. These results will help to refine analyses that have been used for leak-rate analyses in LBB evaluations. In FY91 the crack-opening-displacement prediction capability in several J-estimation methods was extended to include tension as well as combined tension and bending.

Task 7, NRCPIPE Improvements, has the objective to improve the NRCPIPE PC code for circumferential TWC pipe fracture analysis, and to create a surface crack version. In FY91 an initial version of the surface crack J-estimation scheme code was developed. In addition, a separate code for Ramberg-Osgood fitting of stress-strain data was made with options for fitting the data over various strain ranges.

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

W. R. Corwin
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831

Maintaining the integrity of the reactor pressure vessel (RPV) in a light-water-cooled nuclear power plant is crucial in preventing and controlling severe accidents and the potential for major contamination releases. It is imperative to understand and predict the capabilities and limitations of its integrity. In particular, it is vital to fully understand the degree of irradiation-induced degradation of the RPV's fracture resistance which occurs during service, since without that radiation damage it is virtually impossible to postulate a realistic scenario which would result in RPV failure. For this reason, the Heavy-Section Steel Irradiation (HSSI) Program has been established by the U.S. Nuclear Regulatory Commission (USNRC) to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior, and in particular the fracture toughness properties, of typical pressure vessel steels as they relate to light-water reactor pressure-vessel integrity. Results from the HSSI studies provide information needed to aid in resolving major regulatory issues facing the USNRC which involve RPV irradiation embrittlement such as pressurized-thermal shock, operating pressure-temperature limits, low-temperature overpressurization, and the specialized problems associated with low upper-shelf (LUS) welds. Taken together the results of these studies also provide guidance and bases for evaluating both the aging behavior and the potential for plant life extension of light-water reactor pressure vessels.

The principal materials examined within the HSSI program are high-copper welds since their postirradiation properties are most frequently limiting in the continued safe operation of commercial RPVs. In the HSSI Fifth and Sixth Irradiation Series, designed to examine the shifts and possible changes in shape in the ASME $K_{IC}$ and $K_{IA}$ curves for two irradiated high-copper welds, it was seen that both the lower bound and mean fracture toughness shifts were greater than those of the associated Charpy-impact energies, whereas the shifts in crack arrest toughness were comparable. The irradiation-shifted fracture toughness data fell slightly below the appropriately indexed ASME $K_{IC}$ curve even when it was shifted according to

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Revision 2 of Regulatory Guide 1.99 including its margins. Statistical studies of the fracture data and the implications of brittle pop-ins to vessel integrity evaluation methods were evaluated. The high-copper beltline weld removed from the Midland reactor is being examined in the Tenth Irradiation Series to establish the effects of irradiation on a commercial LUS weld. The wide variation previously found in the unirradiated fracture properties of the Midland weld, with values of RTNDT ranging from -23 to 20°C and copper contents from 0.16 to 0.46 wt %, were investigated more fully. The low values of ductile fracture toughness initiation and tearing modulus of irradiated stainless steel cladding revealed by the Seventh Irradiation Series were further investigated and compared with effects of thermal aging.

Embrittlement modelling studies have shown that the time or dose required for the point defect concentrations, which ultimately contribute to irradiation embrittlement, to reach their steady state values can be comparable to the component lifetime or to the duration of an irradiation experiment. Thus, embrittlement models which rely on the assumption of steady state at these conditions are not valid. Consequently, simple direct comparisons of embrittlement generated under different rates of exposure (eg. test reactors vs power reactors) may be misleading because the relative state within the initial defect production transient will likely be different. Since the effect of displacement rate is different in the transient and steady state regimes, data extrapolation from test reactor irradiations may cross mechanism boundaries and lead to poor estimates of material response at low displacement rates. The detailed examination of irradiation mechanics which exist in low-temperature irradiations has led to the tentative conclusion that the cause of the accelerated embrittlement of the HFIR pressure vessel is the high fraction of very low-energy thermal neutrons rather than the low exposure rate. This conclusion is being investigated in two experiments. Specimens are being irradiated at low temperatures and high-fluence rates in spectra with and without large components of thermal neutrons. Other specimens are being exposed in the cavity of a pressurized water reactor at low temperatures and low fluence rates. As a result of the two experiments it should be possible to establish the mechanism primarily responsible for the accelerated low-temperature embrittlement.

Additional activities have been initiated which will provide for more detailed examination of the irradiation, annealing, and reirradiation response of LUS weldmetal, establish an improved facility for the examination of materials irradiated in service, and to assure a continuing source of supply for correlation monitor materials for use in light-water reactor surveillance programs.
TESTS IN THE ATLE LOOP ON THE PIUS DESIGN

Dusan Babala, Ulf Bredolt, and John Kemppainen
ABB Atom AB
S-721 63 Vasteras Sweden
Telex: 40629 Atom Va S

The self-protective thermal hydraulic design of the primary system that prevents reactor power from exceeding the coolant capability of the submerging water independent of outside surveillance and supervision is a fundamental feature of the PIUS design.

The experimental demonstration of the self-protective features of the primary system in a large scale test loop (The ATLE Loop) in the ABB Atom Engineering Laboratories will be described in the paper.

The volume scale of the ATLE Loop is approximately 1:300. The height of the test loop is the same as in the full scale design to ensure similarity in the driving buoyancer force during a transient. The test loop includes the following: one electrically heated, simulated fuel assembly, reactor pool, primary systems, upper and lower density locks and heat exchangers.

System response to various severe transients and normal operations was studied. Comparisons were made with the predictions of the RIGEL code, which has been developed by ABB Atom specifically for study of the PIUS design and its predecessor designs. Selected results of these studies and comparison of these results with predictions obtained using RIGEL are presented in this paper.

The following types of tests were performed and the results will be summarized in this paper: power control, grid voltage disturbance, recirculation pump trip, partial loss of heat sinks, and uncontrolled boron dilution.

Comparisons between test results and calculated results were made for main state variables such as pressures, temperatures, concentrations, heat fluxes, and mass flow rates. These comparisons will also be shown in this paper.

The test results have demonstrated the self-protective thermohydraulics of primary systems designed according to the PIUS principle and verified the capability of the RIGEL code to predict the dynamics of such systems. These studies are an integral part of the initial design phase of PIUS.
Exploiting Digital Systems Technology to Improve Nuclear Safety

R.A. Olmstead, N.M. Ichiyen, J. Pauksens
Atomic Energy of Canada Limited, AECL CANDU

Nuclear plant designers in the 1990's have exceptional opportunities to exploit rapidly evolving computer and information system technology to make significant improvements in public safety. The Canadian experience with digital systems indicates that these improvements will be realized in two areas.

1. Reliability, functionality, and simplicity in the design of Safety Protection Systems and Engineered Safety Features Systems (ESFS).

2. Improved performance and fewer errors from the operations staff. Benefits achieved by automation, superior information management, and simplified maintenance.

CANDU reactors have utilized extensive computer automation for reactor control for 20 years. A direct consequence is that the frequencies for forced outages and spurious protection system trips have been among the lowest for all reactor types.

Historically, CANDU was among the first commercial power reactor to utilize computers to implement protection system. System functions with the PDC's (Programmable Digital Comparators) used in the CANDU 600 reactors (early 1980's). PDC's were used to implement the trip decision logic for the process trip parameters.

By definition, failures in the safety systems which either reduce the redundancy of protection, or affect the designed operation, are called "unsafe" failures. Failures that result in unnecessary operation of safety systems (i.e. spurious trips) are called "safe" failures. To date, we have had excellent operating statistics. The three CANDU 600 stations (Wolsong 1, Gentilly 2, and Point Lepreau) have a total of 288 PDC-years of operating history without a single unsafe failure reported. All PDC failures have been safe failures which can be contrasted with the experience with the conventional portions of the system where about 1/4 of the failures are potentially unsafe. This is due largely to the design that employs features such as self-checks, "continuous" testing, hardware watchdog timers, etc. which convert detected unsafe faults into safe failures (i.e. trip the channel). From the production reliability viewpoint, there have been no spurious reactor trips attributed to PDC related failures.

This experience has confirmed our original reasons for using computers, that they enhance safety availability (convert unsafe failures into safe ones), and also improve production reliability.

This concept evolved further in the Darlington fully computerized shutdown systems. Here the functionality has increased to include not only the safety trip decision logic but also channelized displays to the Operator and increased automated testing and monitoring. The safety critical portions have been localized (to the channelized trip computers only) and the trip functions kept as simple as possible. This is a key component of the "Canadian" approach.
The full paper provides more detail on other safety benefits that have been realized from the use of digital automation for control and protection. The paper will thus describe how accident risk reduction can be achieved in other ways with digital systems. The following areas of Canadian experience will be described:

1. Elimination of human error potential by automation of the protection system testing in the Darlington Generating Station.

2. Lessons learned from the Darlington licensing process; hence the development of the design standards and design changes which resolved the issues.

3. How AECL has put CANDU operators in touch with critical safety variables by giving them surveillance facilities like the margin to trip indicators.

4. How selective application of computer automation has relieved the operator of stressful safety critical decisions, particularly those where safety goals conflict with production or economic goals.

5. How automation has been applied to give nuclear plant operators much more time to think and plan in the first few hours of a design basis event or an undefined serious accident.

6. How the CANDU control room is being designed to bring the relevant information together to suit the particular context of the situation being attended to in the station.

7. The use of comprehensive real time equipment status monitoring to achieve superior operational configuration control in the Darlington station.

Some of these safety benefits can be quantified in reduced accident risk numbers in the PRA analysis. Some of the safety impact is intuitively obvious. Taken together, the safety impact is significant.
ANALYSIS OF POSTULATED EVENTS FOR THE REVISED ALMR/PRISM DESIGN*

G. C. Slovik and G. J. Van Tuyle
Brookhaven National Laboratory

The PRISM reactor is presently under pre-application licensing review by the NRC, with Brookhaven National Laboratory (BNL) providing technical assistance. In this paper, we'll describe the latest set of results from our SSC and MINET code calculations. A series of postulated events were analyzed, for the most current PRISM design, to evaluate the system performance under unscrammed conditions.

The PRISM reactor utilizes a metal fuel composed of 27% Pu, 10% Zr, and 63% U, based on the fuel developed by Argonne National Laboratory as part of the Integral Fast Reactor program. The fuel has a small power and temperature defect as a result of the high fuel thermal conductivity of metal and a hard neutron spectrum. The reactivity feedbacks of the core are designed to provide a negative response to off-normal, high temperature conditions, by invoking negative responses from the thermal expansion of the fuel, control rods, and core radial dimensions.

The core restraint system incorporates the limited free bowing feature, which generates an outward bow of the in-core portion of the fuel assembly when a temperature gradient exits. The core also has three Gas Expansion Modules (GEMs) placed around the periphery, which will remove a combined worth of -69 cents of reactivity when the pumps are tripped from full power conditions.

In evaluating this passive shutdown response, in which the PRISM reactor power should decrease significantly in response to overheated conditions, one considers three classes of unscrambled events. These include the unscrambled loss of flow (ULOF), loss of heat sink (ULOHS), and transient over power (UTOP) events. For the ULOF case, it is assumed that all four electromagnetic (EM) pumps trip and coast down, using the coastdown energy from four synchronous machines (motor-generator sets). In the ULOHS event, it is assumed that heat removal through the IHXs is stopped suddenly, possibly due to the dumping of the intermediate loop sodium (e.g., due to a steam generator tube rupture). The UTOP covers the maximum credible reactivity addition resulting from the withdrawal of all six control rods.

For the initial version of PRISM (1989), both GE's analysis using the ARIES codes and BNL's independent analyses using SSC and MINET, showed very similar trends. For the ULOF, the power reduction was almost as fast as the flow reduction, and no hazardous

*This work was performed under the auspices of the U.S. Nuclear Regulatory Commission.
conditions were predicted (however, the failure of one synchronous machine did cut very deeply into the safety margins). In the ULOHS event, the initial part of the event was slow and benign, and there were few concerns. Because the UTOP was for a maximum of 35 $c$, that event did not seem to be a major concern, although a protracted event might lead to some cladding damage.

In response to a draft safety evaluation report (SER) (1989), GE modified the PRISM design and requested a revision of the draft SER. Most of the design changes addressed NRC concerns, although GE's inclusion of some changes requested by DOE (increasing the reactor power, for instance) raised some new concerns. GE included new ARIES calculations for the postulated unscrammed events, and also factored in some ANL analytical support using the SASSYS Code.

Because of the extent of GE's design revisions, BNL had to repeat all of the unscrammed analyses. In addition, some new modeling was used to represent the GEMs and some newly recognized fuel behavior. BNL's calculations were continued until we could effectively match the GE/ANL analysis, when we chose to do so. In some cases, we then made more conservative assumptions regarding the system performance or the fuel behavior.

For the revised ULOF analyses, the addition of the GEMs improved the margins for the failure of one or more synchronous machines events. For the ULOHS, which is not affected by the GEMs, the margins were little changed. It is now the UTOP event that poses the biggest concern, due to increased concerns about the metal fuel behavior.

The ternary (U, Pu, and Zr) metal fuel is a fairly recent development, and the initial high-burnup data is just now becoming available. There is evidence of considerable migration of the uranium and zirconium fuel components, and there may be some relocation of the plutonium, as well. This component relocation changes local thermal conductivities, melting temperatures, and possibly power densities. For the postulated UOTP events, which result in increased power production, there are now concerns about localized melting and possible fuel relocation. It is noted, however, that the designer has some options for reducing the maximum reactivity insertion (currently 40$^c$), and may choose to do so if the expected data does not resolve our concerns.
This paper summarizes research performed at Oak Ridge National Laboratory (ORNL) to assist the Nuclear Regulatory Commission (NRC) in preliminary determinations of licensability of the U.S. Department of Energy (DOE) reference design of a standard modular high-temperature gas-cooled reactor (MHTGR). The work described includes independent analyses of core heatup and steam ingress accidents and the reviews and analyses of fuel performance and fission product transport technology.

Reactor Description The MHTGR consists of four tall cylindrical ceramic core reactor modules each with a thermal power rating of 350 MW and a single once-through steam generator (OTSG) with a superheater to provide high-temperature ($538^\circ C, 1000^\circ F$) steam. High-pressure helium coolant is driven downward through the core by a single motor-driven circulator. A smaller capacity circulator/heat exchanger loop, the shutdown cooling system (SCS), is located within the reactor vessel. In cases for which neither the main nor the SCS loop is available, afterheat is removed by the passive, safety-grade air-cooled reactor cavity cooling system (RCCS), which is in operation at all times and which does not require operator or automatic actuation. There is no conventional containment building, because the multilayered silicon carbide coatings on the microscopic fuel particles are considered by DOE to be a sufficient containment barrier.

Heatup Accident Analyses A wide variety of accident scenarios are studied interactively by using a workstation version of the ORNL MORECA code. Analyses have been made of postulated long-term core heatup scenarios for which active cooling systems used to remove afterheat are either unavailable or available only intermittently in degraded states. Long-term loss-of-forced-convection (LOFC) accidents are simulated both with and without depressurization of the primary coolant. Because computations are fast (up to 1400 times faster than real time), sensitivity studies can be run readily. Recent upgrades allow the assumption of no scram at the time of the LOFC. The analyses show fuel temperatures approaching or slightly exceeding the nominal limit ($1600^\circ C$) only for anticipated transients without scram (ATWS) cases and those in which the RCCS is assumed to fail catastrophically. In some LOFC scenarios, predicted vessel temperatures exceed the extended ASME code limits.

Steam Ingress Accident Analyses The inadvertent admission of secondary-side steam into the primary circuit results in a positive reactivity insertion and hydrolytic fission product release from the graphite-fuel matrix. To quantify these phenomena, a model of steam ingress has been developed, and a model of fission product release into the coolant is being formulated. The ingress model includes neutronics, thermohydraulics, and control system formulations which permit ingress to be coupled with important reactivity and control feedback effects. Initial studies show
that, for the large breaks postulated (five OTSG tubes), the rate of ingress and accumulation of steam in the primary is a comparatively slow process, requiring typically 10 minutes to reach maximum values, so power excursions are relatively mild. In the previous scoping studies in which helium displacement was not accounted for, very large power excursions resulted. The submodel being added to treat physical chemistry processes includes the spatial representation of diffusion of steam through and adsorption on graphite, graphite oxidation, fuel-particle hydrolysis, and diffusion of fission products out to the coolant channels.

Fuel Reliability Study NRC has taken a cautious position with respect to the suitability of housing MHTGRs in non-sealed concrete buildings as opposed to conventional containment vessels. Caution is required because consequence predictions are singularly dependent on the integrity of the fuel particles. NRC has adopted the concept of "weak fuel" and is exploring the sensitivity of predicted consequences to fuel performance. "Weak fuel" is a qualitative concept in which an arbitrary fraction of the fuel particles are assumed to fail during accident sequences.

No indication of "weak fuel" has been evident in any operating HTGR or in any capsule irradiation test involving Triso fuel particles. Mechanisms have been identified, however, that could introduce faults during the manufacturing process which lead to early failures during normal operation or to extensive failures under accident conditions. The complexities of fuel failure mechanisms need to be better understood. There needs to be sufficient confidence that the data provided by quality assurance/quality control (QA/QC) procedures for the manufacturing process, when combined with reactor service experience and fuel test data, can adequately account for all significant failure mechanisms.

The "weak fuel" study includes (1) a review of current fuel design requirements, (2) a review of known fuel failure mechanisms, (3) a review of fuel manufacturing procedures to determine whether weaknesses can go undetected by QC methods, (4) an outline of preliminary QA/QC options identified by DOE, and (5) trends in light-water reactor fuel quality and manufacture. Judgments are to be drawn on the advisability of retaining the conservatism inherent in the "weak fuel" concept for MHTGRs with no sealed containment.

Review of Fission Product Plateout and Liftoff A review of the technical basis for predicting radioactivity release resulting from primary system depressurization was completed. The study concluded that iodine releases from dry depressurization events are likely to be extremely low because of a predictably low degree of chemical desorption, a low degree of dust liftoff, and a low affinity of iodine for dust particles.
CORA EXPERIMENTS ON THE MATERIALS BEHAVIOR OF LWR FUEL ROD BUNDLES AT HIGH TEMPERATURES

P. Hofmann, S. Hagen*, G. Schanz, L. Sepold*

Kernforschungszentrum Karlsruhe
Institut für Materialforschung
*Hauptabteilung Ingenieurbtechnik
Postfach 3640, W-7500 Karlsruhe 1
Federal Republic of Germany

ABSTRACT

The chemical interactions that may occur in a fuel rod bundle with increasing temperature up to the complete melting of the components are described. The materials behavior of BWR and PWR fuel rod bundles has been studied in integral experiments (CORA program) and extensive separate-effects tests. The kinetic results of the most important chemical interactions are represented. In most cases the reaction products have lower melting points or ranges than the original components. This results in a relocation of liquefied components, often far below their melting points. In addition, the influence of thin oxide layers, which form on Zircaloy surfaces during normal reactor operation on the chemical interactions is given. As a result of the various studies three distinct temperature regimes can be defined in which liquid phases form in the fuel rod bundles in different large quantities. Their influence on damage progression and on possible accident management measures to avoid an uncontrolled core melt down accident are described in detail.
Analysis of the EJET Boiling Jet Mixing Experiments Using the Integrated Fuel-Coolant Interaction Code, IFCI

M.J. Rightley
M.F. Young
D.F. Beck

Sandia National Laboratories

SUMMARY

In the event of a severe reactor accident leading to core melt, it is likely that molten fuel materials will come into contact with water, producing a molten fuel-coolant interaction (FCI). FCIs can occur for a variety of conditions in the core, the lower plenum, or in the reactor cavity. The nature of the FCIs that could occur ranges from benign static boiling, possibly including melt dispersion when the coherent melt mass is broken up on a time scale of 100's of milliseconds, to energetic steam explosions when the melt is finely fragmented on a time scale of milliseconds. Experimentation has revealed that scale-dependent processes occur in FCIs and that these dependencies are not understood. Attempts to model the process have generated several competing models. Unfortunately, the limited size and nature of the experimental database have made the choice of the correct model difficult. The integrated fuel-coolant interaction code, IFCI, was developed to provide a best estimate tool for FCIs, based on known physical laws and available experiments.

The process of assessing the performance of IFCI involves comparing it to the different stages of FCI phenomena such as boiling jet breakup, detonation and products expansion. The NRC Program "Molten Fuel-Coolant Interactions" was initiated to perform this assessment against the current experimental data and other codes that have been developed to model FCIs. Upon completion of the assessment of the code, IFCI will be applied to reactor-scale simulations of lower plenum coarse mixing, steam and hydrogen production rates and steam explosion probabilities and their intensities.

The first phase of the assessment of IFCI is to compare predicted boiling jet breakup behavior to the experimental data obtained in the EJET series conducted at SNL. These tests involved introducing high temperature thermite generated melt jets into saturated and subcooled water tanks and photographing the resultant behavior. Data, digitized from the high speed records of the tests, were compared to predicted melt and steam volume fraction profiles from the code. Direct comparisons with the data were obtained through data post-processing techniques that allowed overplotting of the transient melt profiles onto IFCI-generated contours.
The IFCI predictions of the leading edge penetration rates and the initial spreading of the jet were satisfactory when both the melt and steam were considered in the analysis. Improvements to the bulk boiling model in the code were required to adequately address the transition of the jet through a rapid expansion phase thought to be associated with boiling regime transitions.
DCH Experiments in the CWTI Facility
at
Argonne National Laboratory

by

A. Sharon, J.L. Binder, L.M. McUmber, I. Baumgarten and B.W. Spencer

Argonne National Laboratory

The NRC is sponsoring an experimental program investigating the Direct Containment Heating Issue in PWR's. Integral Effect Tests (IET) are planned at Argonne National Laboratory (ANL) in the Corium Water Thermal Interaction (CWTI) facility at a nominal 1/30 linear scale to a reactor case. These tests are designed to be counterpart to the nominal 1/10 scale to be performed at Sandia National Laboratory (SNL). The main goal of the early tests is to identify scale distortions so as to better understand how to scale up the experimental data obtained in the program to a full scale plant.

The Initial DCH tests at ANL are planned to mock up the containment, cavity and the primary system of the Zion power plant. An accumulator with steam represents the scaled steam volume in the primary system. The accumulator is connected to a melt generator that contains corium simulant material. The melt is ejected out of the vessel into the cavity which is again a geometrically scaled model of the actual reactor cavity in the Zion plant. Entrained debris that exits the cavity enters the lower subcompartment at the bottom of a large expansion vessel. All major walls and structures in this subcompartment are represented in the scaled model of the CWTI facility. These walls and structure would tend to deetrain debris by trapping it and removing it from the gas stream. The rest of the expansion vessel represents the upper volume of the containment.

In these experiments the pressures, gas temperatures and hydrogen concentration in the different components are measured along with the debris dispersal, particle size distribution and oxidation. These results will be compared to similar measurements at SNL to deduct scale distortions and provide predictions of a full scale containment to DCH.

The paper summarizes the main experimental results of the tests and provides for some thoughts on the controlling phenomena and their scaling as deducted from the ANL results.
Adiabatic Equilibrium Models
For Direct Containment Heating

M. Pilch
M. D. Allen

Severe Accident Phenomenology
Sandia National Laboratories
Albuquerque, New Mexico

PRA studies are being extended to include a wider spectrum of reactor plants than was considered in NUREG-1150. There is a need for simple Direct Containment Heating Models (DCH) models that can be used for screening studies aimed at identifying potentially significant contributors to overall risk in individual nuclear power plants. This paper presents two adiabatic equilibrium models suitable for the task. The first, a single-cell model, places a true upper bound on DCH loads. This upper bound, however, often far exceeds reasonable expectations of containment loads based on CONTAIN calculations and experiment observations. In this paper, a two cell model is developed that captures the major mitigating feature of containment compartmentalization, thus providing more reasonable estimates of the containment load.

The single-cell equilibrium model assumes that the containment can be treated as a single adiabatic control volume. Reactor coolant system (RCS) blowdown, debris/gas heat transfer, debris oxidation, and hydrogen combustion all contribute to containment pressurization. The pressure rise in the containment due to these energy sources can be written as

\[
\frac{\Delta P}{P^0} = \frac{\sum \Delta E_i}{U^0(1 + \Psi)}
\]

The heat capacity ratio ($\Psi$) represents the total heat capacity of all airborne debris divided by the total heat capacity of the containment atmosphere. The heat capacity ratio appears because, at thermal equilibrium between debris and the atmosphere, the debris still carries sensible or latent heat that is not available for containment pressurization. The heat capacity ratio can have a strong effect on containment pressurization when the gas mass is small.

Real containments are compartmentalized such that the bulk of DCH interactions occur with only a limited portion of the entire containment volume. The two-cell equilibrium model partitions DCH interactions between the subcompartment and the upper dome of the containment. Locally in the subcompartment, the heat capacity ratio can dramatically limit the energy actually available to heat the atmosphere. This thermal saturation can result in greatly
reduced estimates of containment pressurization, which can be represented by

$$\frac{\Delta P}{P_0} = \eta \left( \frac{\Delta P}{P_0} \right)_{\text{single cell}}$$

where \( \eta \) is an efficiency that can be expressed in terms of the relative volumes of the subcompartment and upper dome.

The adiabatic equilibrium models have been assessed against DCH experiments from the Limited Flight Path (LFP) test series. Containment compartmentalization was simulated by partitioning the Surtsey vessel into an upper and lower volume with a concrete slab. Debris dispersed from a scaled reactor cavity was confined to the volume below the slab, but gas could flow freely between the volumes through a large gap around the outside of the slab. The relative volumes of the upper and lower cells were varied by suspending the concrete slab at various elevations within the vessel.

The single-cell equilibrium model overpredicted observed pressures by factors ranging up to 5-6 with the greatest margins occurring in the most compartmentalized geometry (i.e. debris sourced into a small lower volume). Although conservative, margins this large have limited utility in DCH screening studies. Predicted pressures using the two-cell equilibrium model consistently overpredicted containment pressures by factors less than 2, regardless of the relative volumes, which is a more useful margin for DCH screening studies. Additional improvements await a better understanding, representation and validation of the kinetic processes controlling DCH loads.
RESULTS OF RECENT NUPEC HYDROGEN RELATED TESTS

K. Takumi and A. Nonaka, Nuclear Power Engineering Center
K. Moriya, Hitachi, Ltd.
J. Ogata, Mitsubishi Heavy Industries, Ltd.

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 eleven 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.

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 1,600m³ 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 effect 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 will be well mixed in a containment vessel and the prediction by the computer code is in excellent 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 5m³ and a large scale spherical vessel with 270m³. 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.
COMPARISONS BETWEEN HDR-H\textsubscript{2}-DISTRIBUTION EXPERIMENTS E11.2 AND E11.4

Project HDR, Kernforschungszentrum Karlsruhe GmbH, Karlsruhe, FRG
** HDR, Karlstein, FRG
* Battelle-Europe, Frankfurt/Main, FRG

The combination of issues important for the physical phenomena governing the H\textsubscript{2}-mixing and distribution mechanisms in post-severe accident containment atmospheres, namely:

- large scale of the experimental facility
- high H\textsubscript{2}-release rates
- superheated steam injection into the containment
- multi-compartment geometry with sufficiently large dome volume representation
- representative internal concrete and metal structures and surfaces
- energy transfer across the containment steel shell into a very high annular ring space of 50 m height
- multiple steam and H\textsubscript{2}-injection phases
- different axial positions for H\textsubscript{2}-releases
- examination of the efficiency of H\textsubscript{2}-mitigating features including external dome spray as well as venting

has not been investigated yet in any experimental facility worldwide.

The HDR with its subsystems has the features cited above. Therefore, Test Group E11 was designed and performed in 1989 to provide a first data set for severe accident containment atmospheres in close to real multi-compartment geometry. The test group consisted of the total of eight different experiments addressing a wide spectrum of H\textsubscript{2}-distribution and mitigation issues.

The paper will focus upon experiments E11.2 and E11.4 which both cover small break scenarios and a variety of mitigating measures. A gas mixture consisting of 85 vol\% He and 15 vol\% H\textsubscript{2} was used to examine the distribution and mixing behavior. The major differences between the experiments E11.2 and E11.4 concern primarily the axial break and gas release positions (E11.2: high; E11.4: low), the test duration and total energy input.
Major experimental findings will be presented by comparing the data of both experiments. It will be shown that the break and gas release position has a decisive influence upon the course of the atmospheric conditions as well as upon the effect of mitigative measures such as sprays and venting. Whereas top break position (E11.2) always leads to axially stratified temperature and concentration profiles; bottom break positions (E11.4) result in more or less homogeneous atmospheric conditions.

External spray on top of the dome decreases the containment dome temperatures and steam concentrations as well as the containment pressure by about 0.5 bar and 0.25 bar for E11.2 and E11.4, respectively. However, vigorous condensation results in drastic gas concentration spiking during the initial spray period of E11.2.

Special highlights of individual test phases of E11.2 and E11.4 will be emphasized which are of interest and potential importance for assessing the containment behavior of ALWR-designs.

Both, E11.2 and E11.4 were used as PHDR blind post-test benchmark exercises. In the meantime, E11.2 has been additionally selected as an international open post-test standard problem, No. 29, by the OECD/CSNI.

Qualitative results of the comparisons between data and blind predictions of the PHDR Benchmark Exercise will be shown. They reveal substantial deviations; especially pressures and temperatures have been consistently overpredicted. This led to speculations of erroneous measurements and the evidence supporting these speculations mounted during the spring of 1991.

In order to remedy this uncertain situation, special efforts were undertaken by virtue of recalibration tests at conditions originally used during the E11-experiments. They confirmed measurement errors in the external steam mass flow which had consequently entered into the predictions. Findings of the recalibration tests will be reported and informations provided for the error bands of major experimental data as a result of a subsequent error analysis.

The recently corrected experimental quantities constitute the basis of the continuation of the ISP 29 which had been halted in view of the prevailing uncertainties.

Despite of erroneous data which equally affected all of the blind computations of the PHDR Benchmark Exercises, substantial learning effects by participants in these exercises are noticeable. An overview of these will be discussed.
OVERVIEW OF BATTELLE-EUROPE EXPERIMENTS
AND MODEL DEVELOPMENT ON H₂-DEFLAGRATIONS,
IGNITER AND CATALYTIC DEVICE VERIFICATIONS

L. Wolf, T. Kanzleiter, U. Behrens, G. Langer, K. Fischer, H. Holzbauer
Battelle-Europe, Frankfurt/Main, FRG
M. Seidler
NIS, Hanau, FRG
U. Wolff
RWE-Energie AG, Essen, FRG

The design and implementation of an optimal mitigation strategy against the potential
threat by H₂-deflagrations/detonations in the aftermath of least probable severe
accidents necessitates a knowledge base about these phenomena for representative
conditions and realistic geometries.

A review of relevant research in the past indicates that the important phenomena were
studied primarily in single volume geometries at different scales, results of which
entered into computer codes such as HECTR and CONTAIN in form of correlations
and parameters.

In order to extend this existing know-how into the area of multi-compartment
geometries connected by vents, the German Ministry of Research and Technology
(BMFT) initiated in 1987 and fully sponsored since 1988 a multi-faceted experimental
research program on H₂-deflagrations and related issues in multi-compartment
geometries in the Battelle Model Containment (BMC).

The following test groups constitute the Battelle-program: Test Group H: basic tests in
H₂-air and H₂-steam air-mixtures; Test Group I: (sponsored by industry): supplemental
tests in H₂-air and H₂-steam-air atmospheres; Test Group G: efficiency of H₂-mitigative
measures such as igniters, catalyst modules and combinations thereof.

Thus far, a total of 70 experiments have been performed and evaluated.

The paper gives an overview of the Battelle-program by reviewing the examined
geometries, bandwidths of tested H₂-concentrations, steam concentrations, ignition
locations, specific atmospheric conditions (homogeneous, stratified) and other
important parameters varied during and among the different test groups H, G, and I.
Some experimental findings about accelerated flames and free jet ignition effects will be reported and data compared with the predictions following a Battelle-developed methodology of combining multi-dimensional BASSIM-computations with a lumped-parameter code such as CONTAIN to handle complex multi-compartment situations.

In addition, efforts and results will be reported about the experimental qualification and verification program at Battelle for the passive catalyst module design by NIS/RWE. The rationale and experimental results of different test stages in various test scales under relevant accident and adverse conditions (fires, iodine etc.) will be presented. Comparisons will be shown between data obtained in BMC multi-compartment geometries during catalyst module operations and blind pre-test predictions with the EPRI/NAI code GOTHIC which has been extended for this purpose.

Both, igniter and deflagration experiments and the catalyst module tests constitute an unique data base to develop a H₂-mitigation strategy based on a balanced choice for a proper combination (dual concept) of igniters and passive catalyst modules which both work on diverse physical principles.
Annealing of Soviet VVER-440 Reactor Pressure Vessels

C. Z. Serpan, Jr.
U. S. NRC

Operators of VVER-440 pressurized water reactors in the Soviet Union and Eastern Europe have had to anneal a number of their pressure vessels in the recent past because the embrittlement has become too high. The continued operation of these plants was not considered allowable by regulatory authorities, so a vigorous research and development program was undertaken by the Soviets to determine the time and temperature criteria for annealing, to produce the equipment and hardware necessary for annealing, and to develop the in-plant procedures for its accomplishment.

The Soviets have now accomplished ten annealings of reactor vessels in the Soviet Union, East Germany and Bulgaria. They have done the research necessary to understand the reasons why their steels embrittle so rapidly; they have determined that annealing at 460 C (475 + 15 C) for 100 to 150 hours will yield essentially 100% recovery, and that subsequent reembrittlement will occur at a rate no faster than the original rate (and possibly even at a slower rate than originally).

The Novovoronezh Unit 3 (NV3) reactor vessel was the first one annealed by the Soviets in 1987. The mid-core circumferential weld of NV3 was annealed at 420 C for 150 hours; the lower temperature of 420 C was chosen largely for conservatism. The NV3 reactor mid-core circumferential vessel weld was again annealed in February 1991, but this time at 460 C, since it was felt that too much residual embrittlement remained after the original anneal, and the Soviets wanted all reactor vessels to be annealed on the same basis. The annealing, witnessed by two different teams of US specialists, was carried out in a very business-like manner on a schedule that required a total time of less than 24 days of operations, although they were spread out over a longer time period.

The sequence of operations for annealing of NV3 began with the taking of "template" samples from the carbon steel wall of the pressure vessel. An electric discharge cutting machine with a flat-bottom scoop was used to take three weld and three base metal samples from peak azimuthal flux locations. Sub-size Charpy V-notch specimens were then to be made from the "templates" as a check on the pre-anneal properties. (Two more weld metal "templates" were taken following the anneal to provide a measure of the anneal recovery). This consumed about nine days, but was not done on a round-the-clock basis. Pre and post-anneal hardness measurements were to be made for the NV3 reanneal, as had been done for every other previous annealing, but the equipment had a broken part which could not be repaired in time. Neither the property measurements from the "template" samples nor the hardness measurements, however, are taken by Soviet regulatory authorities as definitive measures of recovery. Instead, the Soviets have relied on laboratory measurements of surveillance specimens.
irradiated in power reactors as their primary reference, and in test reactors as their secondary reference.

The next step is check-out of the annealing rig. It has three levels of 18 heaters each, with a sector of six heaters on a level being controlled by a pair of thermocouples that reach out and contact the vessel wall as the rig is lowered into the vessel. Heater controls are simple and rugged; a microprocessor controls the annealing operation, but a multiply-tiered backup system is also available which ends in simple manual control. Next, a large inspection rig is lowered into the reactor vessel as the water is pumped out. A man enters the working spaces of the inspection rig and exits into the bottom of the vessel head to mop up the last bits of water and any debris prior to heat-up. The inspection rig is removed, and the annealing rig is lower into position. Radiation levels are sufficiently low in soviet VVER-440 reactors that that all working and observing personnel could remain inside the operating hall during this step. Heatup of the rig required less than two days, and following the 100 hour anneal, cooldown was achieved in about four days.

Annealing of Soviet VVER-440 reactor vessels is now accomplished as a relatively routine operation. Because of the common design, their annealing rig can be reused many times effecting considerable cost savings. The personnel operating the annealing rig are a very capable, well-oiled unit who can come into a plant and get their job done quickly and efficiently. Transfer of this technology to application in the USA is being made. The most important aspect for our consideration is that US reactors will require both a circumferential and an axial weld to be annealed, so a high heat level will have to applied near the nozzle ring of a vessel. This may cause plastic strains in the piping and the vessel flange area. Another aspect of concern is the higher level of radiation in US plants, that will make removal of internals and operations near the dried-out vessel much more difficult. In principal however, there does not seem to be any insurmountable roadblock to use of annealing for embrittlement recovery of US reactor vessels in the future.
Mechanisms and Models of Irradiation Embrittlement of Nuclear Reactor Structural Steels

G. R. Odette and G. E. Lucas
University of California, Santa Barbara

Introduction

The objective of this work is to develop a fundamental, physically based understanding of embrittlement phenomena which occur in pressure vessel and other structural steels used in nuclear power plants. The strategy is to use carefully selected experiments to guide the development and calibration of quantitative embrittlement models. These models aim to incorporate the separate and interactive effects on embrittlement of metallurgical (i.e. composition, microstructure, microchemistry, and the initial mechanical properties) and environmental (temperature, flux, fluence, spectrum and post irradiation annealing/reirradiation) variables. The physically based models will be used in statistical analysis of the surveillance data base to provide more accurate embrittlement predictions.

Both experiments and modeling are facilitated by dividing embrittlement into three component processes: 1) the effect of the irradiation environment on microstructure; 2) the effect of microstructure on fundamental properties; and 3) the relationship between the fundamental properties and the various measures of fracture toughness. These component processes are integrated into composite models of embrittlement. The remaining comments will be directed at models for transition temperature shifts measured in Charpy-v-notch impact tests.

Fundamental Property - Fracture Parameter Relations

Increases in transition temperature can be directly related to the increases in the yield stress induced by irradiation. However, the relation is not strictly proportional and the shift/yield stress coefficient K, depends on the initial Charpy properties and the magnitude of the yield stress increase. Typical values of K are in the range of 0.6±0.2°C/MPa. The most significant fundamental observation is that irradiation does not appear to change the microcleavage fracture stress of notched specimens.

Microstructural Evolution

Irradiation conditions encountered in light water reactor environments result in the evolution of microstructures on a very fine scale (on the order of a nanometer or even less). Hence advances in high resolution characterization methods, including small angle neutron scattering, atom-probe and advanced electron microscopy techniques have revealed many of the key features of fine scale microstructural changes. Coarser scale structures, such as carbides, grain boundaries, dislocations and lath structures are largely unaffected. Perhaps the most important feature of the fine scale microstructure in sensitive steels are
coherent copper rich phases alloyed with manganese and nickel. At typical concentrations, copper is highly supersaturated, hence, would thermally precipitate given sufficient time. The major effect of irradiation is to greatly accelerate the kinetics of precipitation, by a radiation enhanced diffusion mechanism: the excess vacancies and interstitials generated during irradiation result in constituent diffusion coefficients that are many orders of magnitude higher than thermal values in the range of vessel operating temperatures. The irradiation enhanced precipitation, however, does differ from purely thermal behavior in that it is primarily associated with the growth (rather than coarsening) of a very high density of precipitates which form early in irradiation. The alloying of precipitates with nickel and manganese can be understood and predicted based on non-equilibrium thermodynamic principles. The precipitation mechanism explains many observations of the variable dependence of the copper dependent component of embrittlement, including: the effects of the copper content itself, preirradiation heat treatment, fluence, flux, irradiation temperature and post irradiation annealing treatments.

At least one (and probably more than one) other microstructural feature also contributes to embrittlement. The existence of such feature(s) is necessary to explain, for example, embrittlement observed in low copper steels and some aspects of post-irradiation annealing response. However, the specific character of such features have yet-to-be identified, and hence, they have generically been classified as so-called matrix defects. Candidates include: interstitial loops - interstitial cluster complexes; microvoid - microvoid-solute complexes; phosphides and carbonitrides. These features respond differently to the metallurgical and irradiation variables and, for example, some appear to be promoted by nickel (and also possibly copper), phosphorous as well as lower temperatures and higher fluxes.

**Microstructure-Yield Stress Relations**

The relation between the the irradiation induced precipitates and yield stress increases are fairly well-understood for simple alloy systems. However, hardening mechanisms and laws are not as well characterized for more complex microstructures characteristic of commercial steels. The main issues center around the laws governing the superposition of multiple components, including the contributions from the yet-to-be identified features, of the overall alloy strength.

**Composite Models**

Composite models can qualitatively, and in some cases quantitatively, predict many embrittlement observations. However, some components of these models (e.g. the matrix defect contributions) remain essentially empirical, and some aspects of the physically based components (e.g. precipitation) need further mechanistic justification and quantification (e.g. flux effects).
Fatigue and environmentally assisted cracking of piping and pressure vessels in light water reactors (LWRs) are important concerns as extended reactor lifetimes are envisaged. Another concern is failure of reactor-core internal components after accumulation of relatively high fluence, which has occurred in boiling water reactors (BWRs) and pressurized water reactors (PWRs). During this year the work in this program has addressed these concerns.

Fatigue of Ferritic Piping and Pressure Vessel Steels

The fatigue life of A533-Gr B pressure vessel steel in high-purity (HP) deoxygenated water, in simulated PWR water, and in air was studied. The material for the tests was a medium-sulfur-content (0.016% S) steel obtained from the lower head of the Midland reactor. The fatigue data collected to date lie above the ASME design curve. Because fatigue crack growth rates (CGRs) in this class of material can show significant environmental enhancement in simulated PWR water under certain loading conditions, the effects of load shape and loading rate will be further explored in subsequent testing.

Stress Corrosion Cracking of Ferritic Steels

Fracture-mechanics CGR tests have been performed on composite specimens of A533-Gr B/Inconel-182/Inconel-600 plated with nickel, and on homogeneous specimens of A533-Gr B material plated with chrome and nickel. Conventional unplated specimens have also been tested to provide baseline data.

The tests on the composite specimen examined both the effects of dissolved-oxygen content and load history. The CGRs were markedly accelerated by small amplitude cyclic loading (R = 0.95). Under the cyclic loading, crack growth was observed at $K_{\text{max}}$ values that produced no crack growth under constant $R = 1$ loading. Under $R = 0.95$ loading, relatively high CGRs ($2.3 \times 10^{-9} \text{ m.s}^{-1}$) were observed in HP water with 200-300 ppb dissolved oxygen. However, crack growth was erratic at this oxygen level. A period of relatively high CGR was followed by periods of very low or unmeasurable crack growth. Increasing the dissolved-oxygen level to 6 ppm or higher initiated a return to the high CGR regime, at least under $R = 0.95$ loading.

Tests on the homogeneous specimens are being conducted under lower R loading (0.2-0.7) to compare "corrosion fatigue" CGRs in plated and conventional unplated specimens. Initial tests were performed with plated and unplated specimens in a single "daisy" chain. The CGR in the plated specimen was higher than in the unplated specimen so testing under comparable loading conditions was not possible after the initial test period. Additional tests are being performed to more accurately determine CGRs in the unplated specimens and thus more accurately measure the degree of acceleration of the crack growth.

Effects of Water Chemistry on Stress Corrosion Cracking of Austenitic Stainless Steels

Current BWR operating practices have reduced to very low levels the ingress of ionic impurities into the coolant system. In most cases soluble corrosion products from system
materials such as chromate, which arises from corrosion of stainless steel (SS) piping, weld cladding on the interior of the reactor vessel, and internal components fabricated from this material, are now the major ionic species present. Because only a small fraction of the recirculation water in BWRs passes through the reactor water cleanup system, the concentration of corrosion-product ions in the reactor water can be much greater than in the feedwater (e.g., ~30–40 ppb versus <1 ppb, respectively).

The effect of chromate on stress corrosion cracking (SCC) of Type 304 SS has been investigated by fracture-mechanics CGR tests. Possible synergistic reactions with sulfates were also investigated. The test results suggest the presence of competitive chemisorption process involving inhibitive chromate ions (at low concentrations) and aggressive sulfate ions on the specimen surface and crack walls. Thus, the low chromate concentration in BWR water (25–35 ppb) arising from corrosion of the piping system may actually have a beneficial effect on SCC, provided that the sulfate concentration is below a critical level, which may depend on the degree of sensitization of the steel.

Irradiation-Assisted Stress Corrosion Cracking of Type 304 SS

Failures of austenitic SS after accumulation of high fluence have been attributed to radiation-induced segregation (RIS) or depletion of elements such as Si, P, S, Ni, and Cr. However, the exact identity of the elements that segregate and the degree to which RIS produces susceptibility of the core-internal components of LWRs to irradiation-assisted SCC are unclear.

High- and commercial-purity (CP) Type 304 SS specimens were obtained from control blade absorber tubes from two operating BWRs. Microchemical and microstructural changes in the steels were studied by Auger electron spectroscopy. Significant RIS of Si, P, Ni, and an unidentified element or compound associated with an Auger energy peak at 59 eV was observed in the CP material. Except for Ni, such segregation was negligible in the HP material. No evidence of S segregation was observed in either material. Chromium depletion from grain boundaries was more pronounced in the HP than in the CP material. Stress corrosion cracking has been observed in slow strain rate tests conducted on CP Type 304 SS with a fluence of $2.0 \times 10^{21}$ n·cm$^{-2}$ (E >1 MeV). Slow strain rate tests on the medium- and low-fluence CP material and on high-fluence HP material are in progress.
Estimation of Mechanical Properties of Cast Stainless Steels during Thermal Aging in LWR Systems

O. K. Chopra

Materials and Components Technology Division
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439

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

Thermal aging of cast stainless steels at <500°C (<932°F) leads to precipitation of additional phases in the ferrite, e.g., formation of a Cr-rich α' phase by spinodal decomposition; nucleation and growth of α'; precipitation of an Ni- and Si-rich G phase, M23C6, and γ2 (austenite); and additional precipitation and/or growth of existing carbides at the ferrite/austenite phase boundaries. Formation of α' phase provides the strengthening mechanisms that increase strain-hardening and local tensile stress. Consequently, the critical stress level for brittle fracture is achieved at higher temperatures.

Work at Argonne National Laboratory and elsewhere has shown that thermal embrittlement of cast stainless steel components can occur during the reactor lifetime of 40 y. Such embrittlement results in a brittle fracture associated with either cleavage of the ferrite or separation of the ferrite/austenite phase boundary. A predominantly brittle failure occurs when either the ferrite phase is continuous, e.g., in cast material with a large ferrite content, or the ferrite/austenite phase boundary provides an easy path for crack propagation, e.g., in high-carbon grades of cast steels that contain phase-boundary carbides. Consequently, the amount, size, and distribution of the ferrite in the duplex structure and phase-boundary carbides are important parameters that control the extent of thermal embrittlement. In general, the low-C CF-3 steels are the most resistant, and the Mo-bearing high-C CF-8M steels are the least resistant to thermal aging.

The kinetics of thermal embrittlement of cast stainless steels are controlled primarily by the kinetics of ferrite strengthening, i.e., the size and spacing of Cr fluctuations produced by spinodal decomposition of ferrite. Small changes in the constituent elements of the material can cause the kinetics of thermal embrittlement to vary significantly. Activation energies for thermal embrittlement can range from ~65 to 230 kJ/mole (~15 to 55 kcal/mole). Also, the aging behavior at 400°C (752°F) shows significant heat-to-heat variation. Production heat treatment, and possibly the casting process, influence aging behavior at 400°C and, therefore, the kinetics of thermal embrittlement. The log of the
aging time at 400°C for a 50% reduction in Charpy-impact energy has been shown to be a useful parameter for characterizing the kinetics of thermal embrittlement. Activation energy for thermal embrittlement is high for those steels that show fast embrittlement at 400°C and low for those that show slow embrittlement at 400°C. Precipitation of G phase in the ferrite has little or no effect on the kinetics of thermal embrittlement. Material parameters that influence the kinetics of thermal embrittlement, however, affect G-phase precipitation. However, activation energy for thermal embrittlement is generally low for cast stainless steels that contain G phase.

A procedure and correlations are presented for predicting mechanical properties of cast stainless steel components during thermal aging in LWRs at 280–330°C (535–625°F). These correlations are updates of those presented at the last meeting. The fracture toughness J–R curve and Charpy–impact energy are estimated from material information that can be determined from the certified material test record. Fracture toughness of a specific cast stainless steel is estimated from the extent and kinetics of thermal embrittlement. The extent of embrittlement is characterized by the room-temperature Charpy–impact energy. A correlation for the extent of embrittlement at "saturation," i.e., the minimum impact energy that can be achieved for the material after long-term aging, is given in terms of the chemical composition. The results indicate that Charpy–impact energy can be <85 J/cm² (<50 ft-lb) for cast stainless steels with ferrite contents as low as 10%.

Extent of thermal embrittlement as a function of time and temperature of reactor service is then estimated from the extent of embrittlement at saturation and from the correlations describing the kinetics of embrittlement, which are given in terms of chemical composition and the aging behavior at 400°C. The fracture toughness J–R curve for the material is then obtained from the correlation between fracture toughness parameters and room-temperature Charpy–impact energy used to characterize the extent of thermal embrittlement. A common lower-bound J–R curve for cast stainless steels of unknown chemical composition is also defined for a given material grade and temperature. Correlations are also presented for estimating the increase in tensile flow stress from data on the kinetics of thermal embrittlement; initial tensile properties are required to determine the flow stress of the aged material. Fracture toughness parameters, e.g., $J_{IC}$ and tearing modulus, are obtained from the estimated J–R curve and tensile flow stress. Examples for estimating mechanical properties of cast stainless steel components during reactor service are described.
Mechanical-Property Degradation of Cast Stainless Steel Components from the Shippingport Reactor

O. K. Chopra

Materials and Components Technology Division
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439

Cast duplex stainless steels used in LWR systems for primary pressure-boundary components such as valve bodies, pump casings, and primary coolant piping are susceptible to thermal embrittlement after extended service at reactor operating temperatures, i.e., 280-320°C (536-608°F). Aging of cast stainless steels at these temperatures causes an increase in hardness and tensile strength and a decrease in ductility, impact strength, and fracture toughness. Most studies pertaining to thermal embrittlement of cast stainless steels involve simulation of end-of-life reactor conditions by accelerated aging at higher temperatures, viz., 400°C (752°F), because the time period for operation of power plants (=40 y) is far longer than can generally be considered for laboratory studies. Thus, estimates of the loss of fracture toughness suffered by cast stainless steel components are based on an Arrhenius extrapolation of high-temperature data to reactor operating conditions.

Several laboratory studies have been conducted to investigate thermal embrittlement of cast stainless steels under LWR operating conditions. A procedure and correlations have been developed for estimating the fracture toughness, tensile, and Charpy-impact properties of cast stainless steel components during thermal aging. In view of the complexity of the embrittlement mechanisms and kinetics, there has been a need for microstructural studies and mechanical testing of actual component materials that have completed long in-reactor service, to ensure that the mechanisms observed in accelerated aging experiments are the same as those occurring in-reactor. Cast stainless steel materials from the decommissioned Shippingport reactor offer a unique opportunity to validate and benchmark the laboratory studies.

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

Microstructural examination of the cast materials indicates that the mechanism of low-temperature embrittlement is the same as that of the laboratory-aged materials. All materials showed spinodal decomposition of the ferrite to form a chromium-rich ale phase. In addition, the check-valve materials contained the Ni- and Si-rich G phase in the ferrite, together with M23C6 carbides at the austenite/ferrite phase boundary.
The Charpy-impact, tensile, and fracture toughness properties of Shippingport materials have been characterized. Baseline mechanical properties for as-cast material were determined from tests on either recovery-annealed material, i.e., annealed for 1 h at 550°C and then water-quenched, or material from the cooler region of the component. The materials exhibit a modest decrease in impact energy. Room-temperature impact energy is relatively high, i.e., >120 J/cm² (>70 ft-lb). The check-valve materials are weaker than the main-valve or pump-volute materials because of the presence of phase-boundary carbides. Fracture toughness J-R curves, J_{IC} values, tearing modulus, tensile flow stress, and Charpy-impact energy show very good agreement with estimations based on accelerated laboratory-aging studies. The correlations for estimating thermal aging degradation of cast stainless steels indicate that the degree of thermal embrittlement of the Shippingport materials is relatively low. After long-term aging of the materials, the minimum room-temperature impact energy is estimated to be >75 J/cm², while fracture toughness J_{IC} is estimated at >300 kJ/m². The materials were aged further in the laboratory at temperatures between 400–320°C to determine the kinetics of thermal embrittlement and degree of embrittlement at saturation, i.e., the minimum impact energy that would be achieved by the specific material after long-term aging. The results show good agreement with the estimates; activation energies for thermal embrittlement range from 150 to 230 kJ/mole.
SUMMARY

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

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

Participation in ASME Section XI and Section V Code activities continued toward achieving industry acceptance of NRC-funded PNL research and development work. PNL personnel are maintaining formal liaison with industry activities to implement the new Appendix VIII requirements. These activities include development of design specification documents and conceptual drawings, and acquisition of specimens for reactor piping, pressure vessels, and bolting. Progress has been achieved in developing performance demonstration requirements for Section XI eddy current applications.

The equipment interaction matrix study was expanded to include component curvature effects. This expanded study focused on equipment center frequency tolerances for both narrow and wide band systems when inspecting curved surfaces. This year the modelling program was modified to handle component curvature and model validation was performed.

Re-analysis of PISC-II round robin test data was completed and compiled in a draft formal report for dissemination to the international technical community. The objective of a new task, initiated at mid-year, is to determine ISI requirements for license renewal. Initial efforts have focused

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1Work supported by the U.S. Nuclear Regulatory Commission under Contract DE-AC06-76RLO 1830; FIN B2289 and L1099; Dr. J. Muscara, NRC Program Manager.
on reviewing the technical bases for existing reactor pressure vessel flaw acceptance criteria. The next action will be to determine if the flaw acceptance criteria are adversely impacted by including degraded material properties in the analysis. The goal is to determine if adequate NDE reliability exists for flaw detection and sizing after considering the effects of material property degradation on flaw location, type, and size from the license renewal perspective.

A new inspection criteria task is developing methodology and criteria for improved ISI (type, extent, and frequency) to achieve acceptable component failure probabilities as related to associated goals for core damage frequency. This task includes active participation on an ASME task force on risk-based inspection guidelines. Plant-specific pilot calculations have addressed risk-based inspection priorities for the Surry Unit 1 Nuclear Power Station. Data obtained during an expert judgement elicitation were analyzed to estimate rupture probabilities for components in four critical systems at the Surry-1 plant (reactor pressure vessel, reactor coolant system, low pressure injection system, and auxiliary feedwater system). A ranking of components in these four systems for ISI prioritization was completed.

To familiarize NRC Region personnel with the problems of inspecting cast (coarse-grained) material, a one-day workshop was conducted to describe wave behavior with anisotropic material and provide hands-on experience. A technique less sensitive to microstructural changes was documented in a white paper which proposed employing adaptive ultrasonics to actively compensate for phase distortion during an inspection. A cooperative agreement between the NRC and UKAEA is being established to apply theoretical modelling to this problem.

Support continued for the Program for the Inspection of Steel Components (PISC-III) through coordinating the U.S. teams in Action 5 Steam Generator Tests; coordinating dates for U.S. teams in the Action 3 Nozzles and Dissimilar Metal Welds; coordinating the design, development, and conduct of the Action 4 Austenitic Steel Tests; and supporting the Action 7 studies on Human Reliability.

The second NRC program reported on is entitled "Characterization of Fabrication Defects in U.S. Reactor Pressure Vessels. A technical data base is being developed to quantify the fabrication flaws that exist in nuclear pressure vessels under an separate program for characterizing defects in U.S. reactor pressure vessels (RPVs) with respect to density, location, type, and size distribution. Accomplishments during the past year include 1) refining the program plan for this activity to assure relevant results for the U.S. RPV population, 2) completion of the analysis of SAFT results obtained on blocks removed from the Midland RPV, and 3) initiation of SAFT inspection of the PVRUF RPV. Four Midland blocks, each containing about 4 feet of circumferential weld, were inspected. Many indications were found throughout the thickness of the blocks, but none exhibited tip signals usable for accurate flaw sizing. By insonifying the cladding from the unclad side, some unusual properties were observed that were probably due to unbonds, inclusions, or lack of fusion.
VALIDATION AND TRANSFER OF ADVANCED NDE TECHNOLOGIES\(^1\)

S. R. Doctor, R. E. Bowey, P. H. Hutton, R. J. Kurtz, L. D. Reid, G. J. Schuster
Pacific Northwest Laboratory
Operated by Battelle Memorial Institute
Richland, Washington 99352

SUMMARY

This paper reports on progress achieved under the NRC program entitled "Field Validation, Acceptance and Training for Advanced NDE Technology." Work under this program in FY91 has concentrated on validation testing and the transfer of technologies previously developed under NRC supported programs. The objective of these efforts is to provide new beneficial NDE techniques which have been validated in their intended application mode on a reactor(s) and which to transfer technologies to the nuclear industry. Program efforts are focused on acoustic emission (AE) for continuous monitoring and the synthetic aperture focusing technique for ultrasonic testing (SAFT-UT). These technologies will benefit nuclear plant regulation by providing timely indication of flaw initiation and growth as it occurs and the reliable detection of flaws with an accurate measure of the flaw size and shape.

Validation of the AE technology has been in progress at Philadelphia Electric Company’s Limerick Unit 1 Generating Station in Pennsylvania since May 1989. A flaw indication was detected in a nozzle-to-safe end weld during normal ISI. The weld has been monitored for indications of growth using AE methods during one fuel cycle - May 1989 to September 1990 - and monitoring for a second cycle started in December 1990. Correlation between AE results and ultrasonic testing (UT) results at the end of the first fuel cycle were mixed. In some locations, the two methods agreed quite well on flaw growth; however, the AE indicated limited growth in regions not detected by UT. This is not necessarily an inconsistency because the AE indicated growth was small and the UT must detect the cracking by a far-side insonification, which means that substantial crack size is needed before reliable UT detection can be assured. The work has been continued for a second fuel cycle with the intent of resolving the difference between AE and UT results. ASME Code Case N-471 directed to the application of AE monitoring to detect flaw growth in reactor components such as that at Limerick Unit 1 is in place having been approved in FY90.

In December 1990 the SAFT-UT system was taken to the Staatliche Materialprüfungsanstalt (MPA) in Stuttgart, Germany to inspect a full-scale boiling-water-reactor pressure vessel plus modular full-scale PWR components. The objective of this work was to pursue validation field testing of the SAFT-UT technology through participation in Action No. 2 of the PISC-III Program.

\(^1\)Work supported by the U.S. Nuclear Regulatory Commission under Contract DE-AC06-76RLO 1830; FIN B2913; Dr. J. Muscara, NRC Program Manager.
The MPA vessel and test assemblies contain a number of implanted defects that are useful for establishing SAFT-UT flaw sizing capability.

Eleven days were spent in Stuttgart inspecting the vessel and test assemblies. A total of eight areas were inspected containing a variety of defects such as cracks, inclusions, and an EDM notch. More than 600 Mbytes of SAFT data were gathered. All eight areas were scanned from the inside surface, and all but one area were scanned from the outside surface. In general, scans from the inside surface were performed with 45°, 60°, and 70° longitudinal beam transducers. In addition, 45° and 60° shear tests and a normal beam scan were also performed from the inside surface. Scanning from the outside surface was usually conducted with a normal beam test, 45° and 60° shear tests, and in some cases tandem SAFT was utilized.

All of the raw data was SAFT processed and an analysis performed. Comparison of the SAFT results with true conditions must await destructive sectioning of the MPA vessel and test assemblies. However, this paper will contain a comparison of SAFT data with the intended sizes of the implanted flaws.

Participation in ASME Section XI and Section V Code activities continued toward achieving industry acceptance of NRC-funded PNL research and development work. The Section V Subcommittee endorsed the PNL-proposed concept of including rules for computerized UT imaging techniques (e.g., SAFT) in ASME Section V, and new Code requirements for this application were developed and submitted to the Subgroup on Ultrasonic Testing (SGUT).
Part 100 of Title 10 of the Code of Federal Regulations (10 CFR 100) describes criteria which guide the commission in its evaluation of the suitability of proposed sites for nuclear power plants. Appendix A of this part describes seismic and geologic siting criteria for nuclear power plants. Some engineering design aspects of nuclear power plants are also briefly addressed in Appendix A. These include: definitions of the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE), definitions of safety related structures, systems and components, ratio of OBE to SSE, identification of acceptable analytical methods, and definition of vibratory ground motion. In fact, the seismic design requirements for nuclear power plants in the U.S. were established with the publication of Appendix A to 10 CFR 100, pertinent Regulatory Guides, and the Standard Review Plan.

One of the interpretations of the regulations on the selection of the two earthquake design levels, the SSE and OBE, was that the SSE would control the design of all safety related systems while the OBE would be applied to the remaining systems required for continued power operation. In practice, however, with the load factors, damping, and service limits, the OBE, rather than the SSE, has controlled the design for some systems.

The work in progress at LLNL and at the NRC will develop the technical basis to support engineering related changes to Appendix A. Included in this effort are the following: investigation of the impact of removing the OBE from design considerations with regard to safety options and review of possible strategies regarding the definition and requirements of the OBE for future reactors.

This paper will report on the status and results of studies performed to date in support of this project. These include a study to identify components that are important to plant safety and also are affected by OBE designs; a study to evaluate the impact on ASME BPVC Section III Division 1 of eliminating the OBE; and a study to evaluate the impact on design margins if the OBE is eliminated from design. This paper addresses the engineering design aspects of Appendix A only.
Enhancing the Seismic Margin Review Methodology to Obtain Risk Insights

Robert J. Budnitz
Future Resources Associates, Inc.
2000 Center Street, # 418, Berkeley, California 94704

This paper will discuss methods for obtaining risk insights from the seismic margin review (SMR) methodology. The SMR methodology was originally developed in 1984-1987 with the objective of analyzing an individual nuclear power plant to ascertain whether the plant has the ability to withstand earthquakes substantially beyond the design-basis earthquake without suffering a core-damage accident. Recently, in the context of NRC's IPEEE program, the SMR methodology has been developed further by NRC to allow plants to identify plant-specific "vulnerabilities" (in the IPEEE sense) to seismic events. The objective of these enhancements has been to provide a methodology for IPEEE seismic reviews that is substantially less expensive than a full-scope seismic PRA, but that achieves the IPEEE's vulnerability-search objectives.

Among the important enhancements of the SMR methodology is a method to obtain risk insights. As originally developed, the SMR approach stops with analysis up to the onset of core damage --- to achieve this it uses systems-analysis methods roughly analogous to those for a "level-I" PRA, with the emphasis on assuring that core-damage events do not occur or identifying combinations of failures that might lead to such a core-damage event.

A recent report (NUREG/CR-5679)* describes the enhanced methodology, which consists of several "steps" that an analyst should carry out, beyond those already specified in either the NRC-type or the EPRI-type SMR methodology. It is estimated that carrying out these additional steps will cost only a few percent extra above the cost of a typical seismic-margin review.

In this paper, the steps involved in the enhanced methodology will be discussed, with emphasis on their rationale. Also, insights will be presented that were obtained by the retrospective reanalysis of two early trial applications (to Maine Yankee and to Hatch) of the SMR methodology. In this retrospective reanalysis, the approach proposed for the enhanced methodology was used to demonstrate that risk-type insights can be readily obtained.

The most important insight is that, by applying the enhanced methodology, it is fully feasible to identify those "vulnerabilities" associated with a large-radioactive-release scenario and to differentiate them from vulnerabilities associated with small-release or no-release scenarios. This and other more plant-specific insights will be discussed.

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* The authors of NUREG/CR-5679, besides the author of this short paper, are David L. Moore and Jeffry A. Julius, both of NUS Corporation, Kent, Washington. The work has been sponsored by the Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission.
The Large-Scale Seismic Test (LSST) Program at Hualien, Taiwan, is a follow-on to the soil-structure interaction (SSI) experiments at Lotung, Taiwan. The planned SSI studies will be performed at a stiff soil site in Hualien, Taiwan, that historically has had slightly more destructive earthquakes in the past than Lotung. The LSST is a joint effort among many interested parties. EPRI and Taipower are the organizers of the program and have the lead in planning and managing the program. Other organizations participating in the LSST program are U.S. Nuclear Regulatory Commission (NRC), the Central Research Institute of Electric Power Industry (CRIEPI), the Tokyo Electric Power Company (TEPCO), the Commissariat A L’Energie Atomique (CEA), Electricite de France (EdF) and Framatome.

The LSST was initiated in January 1990, and is envisioned to be five years in duration. Based on the assumption of stiff soil and confirmed by soil boring and geophysical results the test model was designed to provide data needed for SSI studies covering: free-field input, nonlinear soil response, non-rigid body SSI, torsional response, kinematic interaction, spatial incoherency and other effects. Taipower had the lead in design of the test model and received significant input from other LSST members. Questions raised by LSST members were on embedment effects, model stiffness, base shear, and openings for equipment. This paper describes progress in site preparation, design and construction of the model and development of an instrumentation plan.
EXPERIMENTS ON CONTAINMENT PERFORMANCE UNDER SEVERE ACCIDENT LOADINGS

M. B. Parks
B. L. Spletzer
L. D. Lambert

Sandia National Laboratories
Albuquerque, NM 87185

Summary

The primary purpose of this paper is to provide a status report on two ongoing research programs to examine the behavior of components of the containment pressure boundary under severe accident conditions. The first program to be presented is a series of "separate effects" tests to investigate possible tearing of the steel liner in concrete (reinforced and prestressed) containments due to overpressurization of the containment building. An experimental study to determine the conditions that would cause leakage past penetration bellows will also be described. The research work described herein is a part of the Containment Integrity Programs. These Programs are being conducted at Sandia National Laboratories under the sponsorship of the U.S. Nuclear Regulatory Commission (NRC).

A complete set of validated methods to predict containment behavior when subjected to severe accident conditions is the final goal of the Containment Integrity Programs. To accomplish this goal, a series of scale model containment buildings have been tested to failure by internal overpressurization at ambient temperature. The models were heavily instrumented so that the test results could be compared to pre- and posttest analyses. In this way, the adequacy of existing analytical methods to predict containment behavior has been assessed and improvements have been made.

The number and size of penetrations included in the containment models were limited by their scale and thus, an adequate representation of the wide variety of penetration designs could not be included in the containment models alone. Therefore, several other research programs have been performed to better understand the behavior of penetrations under severe accident conditions. These programs have included tests of a full-size personnel airlock and electrical penetration assemblies (EPAs), a 1:6-scale pressure unseating equipment hatch, compression seals and gaskets, and inflatable seals. The results of the penetration programs, as well as the model tests, are documented in numerous reports to the NRC and in technical papers.

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1This 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.
As a continuation of the Containment Integrity Programs, there are two ongoing research activities. The first to be presented is a series of tests to better understand the effects of various parameters on tearing of steel liners in reinforced and prestressed containments. This test series was motivated by the failure mode of an internally pressurized 1:6-scale reinforced concrete containment model. In this model, failure occurred as a result of excessive leakage through a tear in the liner near a thicker and thus, stiffer insert plate. Posttest examination and analyses have indicated that the tear was primarily caused by point loadings induced on the liner by the stud anchors. The point loadings occur as the studs resist the slippage between the liner and the reinforcement that is caused by the thickness transition.

The liner tearing test series consists of two phases. The primary purpose of the Phase 1 tests is to determine the effect of preload in the liner on liner tearing. For these tests, an initial preload will be imposed on the liner followed by application of shear loading to the studs. At constant liner preload, the stud loading will be increased until either the studs fail in shear or the liner tears. The main objective of the Phase 2 tests is to determine if the mechanism that caused liner tearing in the reinforced concrete containment model can be reproduced using a uniaxial test specimen. The Phase 2 specimen represents a circumferential slice of the model at the location where the main liner tear occurred.

The other program that will be described is a series of tests of containment penetration bellows. 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 wall of the containment. These types of bellows are an integral part of the containment pressure boundary. During accident conditions, the bellows may be subjected to combinations of axial compression, lateral offset, internal pressure, and elevated temperatures. The purpose of the test program is to determine the types of load combinations necessary to cause leakage past typical containment penetration bellows.
High Level Seismic/Vibrational Tests at the HDR – An Overview

C. A. Kot, M. G. Srinivasan, B. J. Hsieh
Argonne National Laboratory

D. Schrammel, L. Malcher
Kernforschungszentrum Karlsruhe, FRG

H. Steinhilber
Fachhochschule Giessen, FRG

J. F. Costello
U.S. Nuclear Regulatory Commission

In the Phase II testing conducted by the Kernforschungszentrum Karlsruhe (KfK), FRG, at the HDR (Heissdampfreaktor) Test Facility in Kahl/Main, FRG, high-level seismic/vibrational experiments were performed on the reactor building, structures, components and piping. These experiments were conducted as a cooperative venture among a number of organizations in Europe and the USA, including the U.S. NRC Office of Research. In the first series of tests called SHAG, performed in 1986, the entire reactor building was subjected to high-level vibrational excitation by means of a large eccentric mass shaker, capable of generating loads in excess of $10^4$ kN (1000 tonne) and located on the reactor operating floor. The purpose of these experiments was to study soil–structure interaction (SSI), load transmission, and response of the as-built structure, components, vessels and piping under strong vibrational excitation equivalent to the levels experienced in design basis earthquakes (SSE and OBE). Specifically the nonlinear behavior of the structural systems was of interest. Other objectives were to validate/verify nonlinear calculational procedures and to evaluate the performance of various dynamic pipe support configurations.

The results of SHAG show that the response of the reactor building, surrounding soil, and mechanical systems was highly nonlinear. This is evidenced by soil cracking and the strong reduction in modal frequencies of the soil–structure system with a simultaneous increase in damping. Impacts and other nonlinearities in the load transmission in supports lead to significant response in mechanical systems (e.g., vessels, piping), at higher frequencies than those generated by the coast–down shaker. The strains and bending moments observed in the outer concrete shield structure were nearly five times higher than those that would result from a typical nearfield earthquake for this location. Thus the structure was tested to levels of incipient failure. The highest accelerations in the building were 0.4 g. Pre- and post-test calculations demonstrated that nonlinear modeling is essential to capture the actual soil–structure response and the structural behavior in the highly stressed regions of the concrete shield building. Nonlinearities such as increase in material damping and reduction in soil stiffness, and soil layering need be modeled to get reasonable agreement between calculation and measured results.

The VKL (Versuchskreislauf) piping system was tested in SHAG with up to seven different support configurations, ranging from very rigid systems using snubbers and struts to very soft configurations with mainly spring and constant force hangers. In other configurations, energy absorbers, seismic stops and viscous damper supports were used to replace the snubbers. It was
found that a stiff hanger configuration with snubbers offers no advantages over reasonably compliant hanger systems.

A second series of tests called SHAM was conducted at the HDR in 1988. The objectives of these experiments were to (i) study the response of piping subjected to seismic excitations that exceed design levels manifold and which may result in failure/plastification of pipe supports and pipe elements; (ii) provide data for the validation of linear and nonlinear pipe response analysis; (iii) compare and evaluate, under identical loading conditions, the performance of various dynamic pipe support systems, ranging from very flexible to very stiff configurations; and (iv) establish seismic margins for piping and dynamic pipe supports.

The test object in SHAM was again the VKL piping system. Excitation was applied directly to the piping by two servo-hydraulic actuators, each capable of generating up to 400 kN of force. Six different dynamic piping support configurations were developed by the participants in SHAM. Besides the typical US nuclear plant system using snubbers and struts, there were snubber replacement devices, i.e., energy absorbers and seismic stops; configurations with only rigid struts; and configurations with no seismic support at all. All support systems (except one) were designed for a common prescribed floor spectrum with 0.6 g ZPA, corresponding to 100% of an SSE. All configurations were tested up to excitation levels just short of producing plastic pipe deformation, i.e., about three times the design SSE. In addition, two dynamic support configurations (called the NRC and KWU systems) were subjected to seismic loads up to eight times the design SSE.

At the low and intermediate levels of seismic loading, up to 300% SSE, the highest observed local strains were only of the order of 0.3%. Strain records at two elbows indicate that some strain ratcheting occurred for both the NRC and KWU configurations at the highest level (800% SSE) test. The highest local strains observed in the seismic tests were of the order of 1%.

The first two snubber failures in the NRC support configuration occurred at a load level of 300% SSE. These were replaced. At 600% SSE snubbers failed at three locations. These were not replaced for the 800% SSE run when an additional snubber failed as did an expansion anchor at another location. No failure of struts or other supports occurred in the tests and while local plastification occurred no pipe failures were observed.

Using the test data for the NRC and KWU configuration, seismic margins for the piping and supports were estimated based both on design loading and design capacity. Accounting for some of the overdesign of the systems and using Service Level C allowables, it is estimated that the margin for snubbers is at least two. The lower bound of margin for struts is estimated to be five and that for piping about four.

Extensive pretest and posttest calculations were carried out by the participants in the SHAM tests. These included linear calculations for the design and lower level (up to 300% SSE) test predictions, as well as nonlinear calculations for the high level, 800% SSE, tests. The results are quite variable depending on location with a general tendency to somewhat underpredict the piping response and grossly underpredict the peak support forces.
The Structural Aging Program has the overall objective of preparing a report which provides United States Nuclear Regulatory Commission (USNRC) license reviewers and licensees with: (1) identification and evaluation of the degradation processes that affect the performance of structural components; (2) issues to be addressed under nuclear power plant continued service reviews, as well as criteria, and their bases, for resolution of these issues; (3) identification and evaluation of relevant in-service inspection (or structural assessment) and repair programs in use, or needed; and (4) quantitative methodologies for assessing current, or predicting future, structural safety margins. The results of this program will provide an improved basis for the USNRC staff to permit continued operation near, at, or beyond the nominal 40-year design life of a nuclear power plant.

In meeting the above objective, the program has three primary activities presently under way: (1) development of a structural materials information center (handbook and electronic database) containing information on the time variation of concrete and concrete-related material properties; (2) establishment of procedures to make quantitative evaluations of the presence, magnitude and significance or environmental stressors or aging factors that can impact critical component performance, as well as techniques which can be utilized for repair of degraded concrete structures; and (3) formulation of a quantitative methodology for use in performing current condition assessments and making reliability-based life predictions of critical concrete structures in nuclear power plants. Accomplishments to date under these activities include: formulation of a structural materials information center containing data and information on over 30 structural materials; development of a structural aging assessment methodology for ranking Category I concrete structures in terms of importance, safety significance and degradation effects; assessment of nondestructive evaluation...
AGING STUDY OF THE IE POWER AND REACTOR PROTECTOR SYSTEMS:
AN EVALUATION OF MONITORING AND SURVEILLANCE EFFECTIVENESS

V. Sharma and J. L. Edson
Idaho National Engineering Laboratory
E.G.&G. Idaho, Inc.
Idaho Falls, ID 83415-2406

SUMMARY

The Idaho National Engineering Laboratory (INEL) performed an aging study of the Class IE Power (IE) and Reactor Protection Systems (RPS) in support of the United States Nuclear Regulatory Commission's (USNRC) Nuclear Plant Aging Research (NPAR) Program. This research program consisted of: 1) an examination of operational data to identify those failing components that were not being detected by routine inspection, surveillance and monitoring methods (IS&MM); 2) an investigation of improved IS&MM practices applicable to the Class IE and Reactor Protection Systems; and 3) an identification of the risk significant components in the Class IE Power system and the effects of aging upon their failure probabilities.

The operational data review showed that over 51% of all Class IE Power System failures were not detected by routine IS&MM. Batteries, circuit breakers, inverters, and relays accounted for over 80% of all IE System failures. The DC Power subsystem had the most IE failures followed by the Instrument AC subsystem, the Emergency Power subsystem, and the Plant AC subsystem. The data also showed that, for batteries, routine IS&MM were able to detect failure caused by wear and cyclic fatigue while not detecting failures caused by short/ground circuits and open circuits. For circuit breakers routine methods detected failures caused by wear, binding, and open circuits. These same methods failed to detect events caused by weld faults and mechanical adjustment problems; also there was a higher percentage of open circuit failures not detected by routine IS&MM. Defective connection and open circuit failures were detected by routine IS&MM for inverters, while short/ground circuit events were not being detected; and again there was a higher percentage of open circuit failures not detected by routine methods. For relays, routine methods caught failures caused by set-point drift and out-of-calibration. The routine IS&MM missed open circuit and wear failures.

The operational data review also revealed that approximately 40% of all Reactor Protection System failures were not detected by routine IS&MM. Over 70% of all RPS events were accountable to transmitters, integrators, and bistables. For transmitters routine methods were able to detect failures caused by out-of-calibration and set-point drift, while they missed events...
caused by wear and cyclic fatigue. Also, there was a lower percentage of set-
point drift and out-of-calibration failures not detected by routine methods.
Set-point drift, out-of-calibration, and wear were causes of failures detected
by routine IS&MM for bistables. Open and dirty circuit failures were not
detected by routine methods; but again, there was a lower percentage of set-
point drift and out-of-calibration failures not detected by routine methods.
Set-point drift and out-of-calibration failures, again, were the two major
routinely detected events for integrators. Cyclic fatigue and defective
connections caused the failures not detected by these routine methods. As in
the case of the bistables and transmitters, there was a lower percentage of
set-point drift and out-of-calibration failures not detected by routine
methods.

The failure causes not detected by routine methods for the components in
the Class IE and the Reactor Protection Systems are similar. Open and
short/ground circuit failures are predominant in the IE system; they account
for over 60% of all undetected IE events. Weld faults, wear, and mechanical
adjustment are the other leading undetected IE failure causes. Cyclic
fatigue, over 30% of all undetected failures, is the leading cause in the RPS.
Wear, open circuits, dirty circuits, and defective connections are the other
undetected failure causes.

There are ways to improve current IS&MM practices for both the Class IE
and Reactor Protection Systems. One method uses electromagnetic signals to
measures various parameters of electrical equipment for use in trending. This
system collects baseline data and uses it in a comparison on later tests of
the same piece of equipment; this has proven especially effective for cables.
Other improved methods include the use of infra-red technology; this methods
also collects data for trending. The basic principle of this device is to
measure the heat generated by electrical equipment; the test data is compared
to baseline criteria to determine whether that component is within a certain
operating range. The INEL is performing research on these and other improved
IS&MM methods.

A standard Probabalistic Risk Assessment (PRA) was used to determine the
risk significant Class IE components. The fault trees for the PRA included
circuit breakers, transformers, bus work, inverters, rectifiers and batteries;
relays were not included explicitly in the PRA. Truncation of the cut sets
left only the circuit breakers, transformers and bus work as potentially risk
significant components. The effects of aging were determined by simply
applying aging rates, as determined by expert opinion, to the initial failure
rates used in the PRA to determine if significant changes occurred as a
function of time. The failure rate for the circuit breaker increased the most
at 20%, followed by the transformer and the bus work. Requantification of the
PRA, using the new failure rates, revealed a less than 1% increase in the core
damage frequency. This was verified by using operational data and statistical
methods to formulate an actual aging failure rate for the circuit breakers,
since they had the highest theoretical failure rate increase. The results of
the statistical analysis confirmed the results of the PRA using the
theoretical aging rates; the risk increase associated with the aging of the
circuit breakers, transformers, and the bus work is not significant.
AN AGING ASSESSMENT OF TRANSFORMERS IN 1E POWER SYSTEMS*

J.L. Edson and E.W. Roberts
Idaho National Engineering Laboratory
Idaho Falls, Idaho

SUMMARY

This paper presents the findings of the INEL Phase 1 study of the effects of age on nuclear power plant Class 1E power transformers, the significance of transformer aging on plant safety, and the capabilities to mitigate the effects of transformer aging to prevent risk significant failures. The following areas were included in the INEL study:

- The characteristics of Class 1E power transformers used in nuclear power plants were determined. The characteristics include size (KVA), voltages, materials of construction, insulation mediums, and the cooling methods.

- All known transformer stressors were identified, their effect on transformer materials determined, and the effects of these stressors with time (aging) assessed. The stressors identified include electrical loading, humidity, system electrical transients, internal partial discharge faults, high temperature, and corrosives.

- On and off line techniques to detect the degradation of all types of transformers were identified and their effectiveness examined. Although all known techniques were examined, the focus was on industry proven methods and instrumentation.

- The information provided in NPE, LER, and NPRDS data bases was evaluated to determine the types and age of the 1E transformers in use, the problems experienced, the numbers of unexpected failures, and the cause of these problems and failures.

• The INEL reviewed the monitoring, inspection, maintenance and replacement recommendations included in NRC documents, IEEE and ANSI Standards, and manufacturers manuals. The capability of these recommendations to mitigate the effects of transformer aging and prevent unexpected transformer failures was evaluated. A sampling of the nuclear industry practices was made to determine if their practices were in agreement with recommendations.

• The risk significance of transformer aging was determined using risk analysis information from the INEL Class 1E aging study of a typical plant (Surry 1).

• Conclusions have been made on the effects of aging on transformer performance, the capability to prevent/detect/mitigate aging effects prior to transformer failure, and the effectiveness of current maintenance and monitoring methods.
NEW METHOD FOR DETECTING DEGRADATION IN INSTALLED CABLES IN NUCLEAR POWER PLANTS

Yasuo KUSAMA, Toshiaki YAGI, Yousuke MORITA, Seiji KAMIMURA*, Hideki YAGYU*

Japan Atomic Energy Research Institute
*Hitachi Cable

Simple and precise methods for detecting degradation in installed cables in nuclear power plants are expected to develop from the viewpoint of the safety operation and the life extension of the plants. Generally, the degradation of cable insulation by thermal and/or radiation aging has been examined by tensile tests, and the degree of degradation was evaluated from the ultimate strength and elongation. The degradation of cables installed in the facility is likely to be detected by non-destructive techniques. For this purpose, two kinds of detecting methods i.e., a) torque-strain response and b) thermal gravimetry were studied in order to apply to the low voltage cables, for which effective detecting methods have not been completed. Fig. 1 shows a basic idea to determine degradation of the low voltage CV-cables (polyethylene insulation, polyvinylchloride jacket). Elongation at break was used as a primary standard to evaluate availability of the methods.

a) Torque-strain response

A prototype apparatus was produced for detecting degradation of installed cables by focusing on the mechanical properties' change due to the degradation. A fixed length of the cable was held by two chucks and a small angle torsion (usually 5 to 10 degrees) of various period was added through one of the chucks. Stress (torque) yielded by torsion was detected at the other side of the chucks. The relation between the torque value and the degradation of the cables (elongation at break) was investigated.

Followings are the results obtained for the torque-strain response:

- Appropriate period of torsion added to the cable was found to be in a range 0.4 to 1 Hz.
- Torque value increases linearly with torsion angle up to 10 degrees and levels off.
- 50 mm is the most suitable length for accurate measurement for eliminating any kind of effect by constituents other than jacket.
- Good corelation is observed between the torque value and the elongation at break of the CV cables degraded by accelerated aging with simultaneous and sequential exposure to radiation and heat.

It is concluded that the torque-strain response is available to detect degradation of the installed cables non-destructively with high accuracy. Development of the more compact and light apparatus will be necessary for the convenient measurement of the installed cable degradation.
b) Thermal gravimetry

Degradation of the polyvinylchloride by radiation and heat is supposed to be mainly due to the decrease in molecular weight of the polymer by main chain scission. A small amount of the sample (4-5 mmg) was picked out from the cable jacket and the thermal gravimetric measurement was carried out in an inert gas environment under a constant temperature rise. To ascertain the degradation mechanism, molecular weight and molecular weight distribution were measured by using gel permeation chromatography.

Relations between temperature at 5% decrease in weight (expressed as T5%) and the elongation at break were investigated.

- Degradation of the PVC jacket material by radiation and heat exposure is mainly due to the decrease in molecular weight of the polymer.

- T5% decreases linearly with radiation dose and with heat exposure time.

- Definite relationship is observed between the elongation at break and the T5%.

From above results, it is concluded that the thermal gravimetry will be effective and applicable as one of the non-destructive methods to estimate the installed cable degradation. Accumulation of experimental data will be necessary to advance the reliability of this method.

Fig.1 Basic idea for deciding degradation of the CV cables.
DETECTING AND MITIGATING AGING IN COMPONENT COOLING WATER SYSTEMS*

Robert J. Lofaro
Brookhaven National Laboratory
Upton, New York 11973

SUMMARY

The component cooling water (CCW) system is one of many systems that is important for safe operation of nuclear power plants. In a research program sponsored by the U.S. Nuclear Regulatory Commission (NRC), the CCW system has been studied to determine how aging affects its performance and reliability. The study was performed in two phases and included extensive analyses of data obtained from national data bases, as well as data obtained from actual plant visits. This paper discusses how the results of those analyses can be used to help detect and mitigate the affects of aging in CCW systems.

The function of the CCW system is to remove heat from various loads throughout the plant and discard it to an open loop cooling system, such as the service water system. The loads serviced by the CCW system can be safety-related or non-safety-related, such as the reactor coolant pump seals, the shutdown heat exchangers, the residual heat removal heat exchangers, and the safety injection pumps. Due to the diversity of the loads dependent on it, the CCW system is continuously operating, and is required during normal, as well as off-normal plant operation. Therefore, the affects of aging must be properly managed to ensure safe plant operation in later years.

In phase I of the CCW system aging study\(^1\) an analysis of past operating experience showed that the CCW components are susceptible to aging degradation, and that this degradation leads to an increase in failure rate as the components age. Of the failures reviewed, over 70% were related to aging degradation, with the dominant cause of failure being "normal service." The dominant failure mechanism was "wear," which is consistent with the high percentage of failures attributed to aging. The components having the largest number of failures were valves, followed by pumps, instrumentation, and heat exchangers.

Using time-dependent failure rates calculated from the data, a probabilistic risk assessment (PRA) was done for a typical CCW system design. The results showed that if component failure rates increase linearly with age, the unavailability of the system can increase exponentially. This is due to the redundancy of the components. Since the CCW system is important to safety, this could lead to an increase in plant risk. These findings clearly show that proper detection and mitigation of aging degradation should be an important part of daily plant operation.

To determine the most effective methods of managing aging, a second phase of the CCW system aging study was performed\(^2\). In this part of the study inspection, surveillance, monitoring, and maintenance (ISM&M) practices were investigated. Information on ISM&M practices currently used at plants was

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* Work performed under the auspices of the U.S. Nuclear Regulatory Commission.
obtained from a survey, along with actual plant visits and personnel interviews. In addition, various advanced practices were identified through literature searches and discussions with component manufacturers. The findings provided an excellent overview of what methods are available to properly control aging degradation.

The information on currently used ISM&M practices showed that there are two categories; basic practices, which are typically required by codes or plant technical specifications, and supplemental practices, which are selected based on particular plant operating characteristics and environment. An effective ISM&M program requires a combination of basic and supplemental practices to ensure that at least one method is in place to detect and mitigate each of the common aging mechanisms that may lead to component failure. As an aid in evaluating a plant’s ISM&M programs, the various practices identified in this study were correlated with the aging mechanisms they can detect and/or mitigate, and the results were tabulated for each of the major components. These tables are included in the full paper.

REFERENCES


THE EFFECTS OF AGING ON FRICTION OF MOVs

T. H. Hunt and U. P. Sinha

Idaho National Engineering Laboratory
E.G.&G. Idaho, Inc.
Idaho Falls, ID 83415-2406

SUMMARY

This report studied the effect of three aging mechanisms: corrosion, erosion, and deposition on the friction coefficients of the sliding surfaces of motor-operated valves (MOVs) used in nuclear plant systems. The U. S. Nuclear Regulatory Commission (NRC) sponsored the study and it was performed following the guidelines of their Nuclear Plant Aging Research (NPAR) program.

The operating force prediction equation describes the force components typically considered by the nuclear valve industry to specify a motor-operator for a valve. Of the loads accounted for, the disk load generally is the largest fraction of the total load and it contains the friction term of concern to this study. The disk load is a function of the surface area of the face of the disk, the differential pressure across the disk, and the friction coefficients of the sliding surfaces of the valve.

The corrosion rates of materials used in MOVs were evaluated to determine their impact on valve operation. Under PWR conditions, the maximum corrosion rate was for carbon steel at 1.1 mil/year, for BWRs conditions it was for carbon steel at 2.2 mil/year. These rates should not cause valve failure due to obstruction of the mechanical tolerances of the MOVs. Limited deposition studies were available, but the deposition rates for activation products on Type 304 stainless steel in PWRs ranged from 0.003 to 0.02 mil/year. No information was available for deposition rates on carbon steel materials. Erosion rates could not be bounded due to the wide diversity of valve materials and their interaction with the various environmental conditions that can occur in reactor systems.

Operating history data identified the types of failure modes and their causative mechanism for valve failures occurring because of an internal component failing in the MOV. The predominant failure mode was internal leakage, accounting for between 57% and 89% of the failures depending on the data source. The failure modes with the greatest safety significance were those where the MOV failed to function, such as failing to close when required. These failures accounted for 10% to 42% of the failures.

The distribution of failure causes showed corrosion causing only a small percentage (1% to 5%) of the failures. Erosion and deposition appeared in

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several cause categories without a specific category of their own. These categories contain erosion and deposition events but the bulk of the failures in these categories are not necessarily aging related phenomena. The erosion categories caused 45% to 66% of the failures, and the deposition categories caused 2% to 10% of the failures.

Since failure to function failure modes are significant, the failure cause distribution for these failures was examined separately. Categories indicative of increased operating force caused 49% to 83% of these failures. This is quite significant because the aging mechanisms may cause frictional increases that may appear as an increase in operating force. Corrosion caused 1% to 7% of these types of failures. Erosion categories caused 6% to 15% and deposition categories caused 3% to 15% of the failure to function mode failures.

The ASME codes for MOVs generally only require stroke time testing at least once per quarter year. This testing program fails to identify the changes occurring in the friction coefficients because it is normally done under low stress conditions where the motor-operator has a large margin to the required force and is not sensitive enough to detect these changes.

The NRC's Generic Letter 89-10, requires diagnostic testing of safety system MOVs. This diagnostic testing has a greater potential to identify changes in friction coefficients because it uses more sensitive measurement devices. But, because it is performed under low stress conditions where the motor-operator has large reserve capability it is also unlikely to detect subtle changes.

The authors reached three general conclusions:

- Corrosion and deposition should not prevent valve operation by obstructing the mechanical tolerances of the MOVs. The maximum corrosion and deposition rates expected for the MOV is quite low, and the sliding surface tolerances are much larger than this value. Operating experience data shows a small number of failures caused by corrosion induced valve seizure.

- The aging mechanisms may increase the friction coefficients due to roughening the surfaces with age. The oxide layer created by corrosion and deposition increases surface roughness and increases the friction coefficient. Operating data shows an increase in valve operating force causes most of the failure to function mode failures. This force increase may be caused by an increase in valve’s friction coefficients.

- The codes and standards defining MOV surveillance requirements need review to include methods for detecting aging degradation. Current requirements do not address age related degradation of MOVs. Establishment of trending requirements for diagnostic testing and improved surveillance methods could correct this deficiency.
The Idaho National Engineering Laboratory (INEL) is performing motor-operated valve (MOV) research in support of the U.S. Nuclear Regulatory Commission's (NRC's) efforts regarding Generic Issue 87, "Failure of HPCI [High-Pressure Coolant Injection] Steam Line Without Isolation," and Generic Letter 89-10 (GL 89-10), "Safety-Related Motor-Operated Valve Testing and Surveillance." This paper presents the results of testing to assess valve and motor operator performance under varying pressures and fluid conditions. This effort included an examination of the methods used by the industry for predicting required valve thrusts and research to provide guidelines for the extrapolation of in situ test results to design basis conditions.

The research program included two full-scale flexible wedge gate valve qualification and flow interruption test programs, designated Phase I and Phase II. Six valves were tested, three 6-in. isolation valves typical of those used in reactor water cleanup (RWCU) applications and three 10-in. valves typical of those used in HPCI applications. In all, 31 flow interruption tests were performed, 10 of these at design basis conditions for the boiling water reactor (BWR) systems of interest. In addition, we evaluated the results from selected utility in situ differential pressure testing in response to GL 89-10.

This research has identified several inconsistencies in the existing industry stem thrust equation and challenged the overly simplistic assumptions inherent in the use of the equation. This paper discusses our development of a new equation whereby conservative stem thrust estimates can be calculated for some flexwedge gate valves, that is, those classes of valves whose operational characteristics have been shown to be predictable. We also present a method whereby the results of testing of such valves at low differential pressure can be used to estimate valve response at up to design basis conditions.

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a. Work supported by the U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Division of Engineering and Division of Safety Issues Resolution, under DOE Contract No. DE-AC07-76ID01570; G. H. Weidenhamer and O. O. Rothberg, Technical Monitors.
EXTENSIONS AND APPLICATIONS OF DEGRADATION MODELING

F. Hsu, W.E. Vesely*, M. Subudhi, and P.K. Samanta

Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973, USA

*Science Applications International Corporation, Ohio, USA

SUMMARY

Component degradation modeling being developed to understand the aging process can have many applications with potential advantages. Previous work has focused on developing the basic concepts and mathematical development of a simple degradation model. Using this simple model, times of degradations and failures occurrences were analyzed for standby components to detect indications of aging and to infer the effectiveness of maintenance in preventing age-related degradations from transforming to failures. Degradation modeling approaches can have broader applications in aging studies and in this paper, we discuss some of the extensions and applications of degradation modeling.

The extensions and applications of the degradation modeling approaches to be discussed are: a) theoretical developments to study reliability effects of different maintenance strategies and policies, b) relating aging-failure rate to degradation rate, and c) application to a continuously operating component.

The theoretical extensions of degradation modeling approaches being developed will explicitly show the reliability effects of different maintenance and test intervals, different maintenance and test efficiencies, and different repair times. The effect of different maintenance strategies beginning with minimal maintenance programs where only limited maintenance is performed, when failures are observed to comprehensive maintenance programs, and where both condition-directed preventive maintenances and major repairs at failure occurrences are performed, can be evaluated.

An important application of degradation modeling is to study the relationship between degradations and failures so that aging-failure rate can be predicted from degradation rates. We will present an analysis of this relationship where a linear model with a time-lag between these two parameters is studied. An example of an application using the data on RHR pumps shows a time-lag of 2 years for degradations to affect failure occurrences.

This work was performed under the auspices of the U.S. Nuclear Regulatory Commission.
Additionally, the application of degradation modeling approaches to study aging effects on air compressors and auxiliary feedwater pumps will be discussed. Extended degraded modeling approaches will be analyzed to study sensitivity effects of different maintenance programs and strategies.

Reference

RISK EVALUATIONS OF AGING: PROCEDURES GUIDE FOR AN AGE-DEPENDENT PSA WITH EMPHASIS ON PRIORITIZATION AND SENSITIVITY STUDIES

W.E. Vesely
Science Applications International Corporation

Based on the previous work which has been performed in the project, a procedures guide is being developed for carrying out an age-dependent Probabilistic Safety Assessment (PSA) for evaluating the core damage frequency with aging effects explicitly treated. A PSA is basically a Level 1 Probabilistic Risk Assessment (PRA). The emphasis of the guide is on prioritization and sensitivity studies. Focus is also on active components although consideration of aging effects in passive components is also treated. The guide is intended to become a NUREG/CR and is the first of three volumes which are being developed. The table of contents for the procedures guide is listed below with a short description of each topic. These topics with demonstrations and applications are described in the presentation.

Procedures Guide for an Age-Dependent PSA with Emphasis on Prioritization and Sensitivity Studies

Table of Contents

1. The Age Dependent PSA Versus the Standard PSA
   The differences in basic component reliability modeling, test and maintenance modeling, and results which are obtained are described.

2. Component Reliability Models Used in an Age Dependent PSA
   The definition of aging for reliability and risk application is presented. Specific models are presented for aging failure rates and for modeling the effects of testing, maintenance, inspection, and repair on aging. The models are detailed and comprehensive, and cover a wide spectrum of situations.

3. Approaches for Transforming a PSA into an Age-Dependent PSA
   Three approaches are described for transforming a PSA into an age-dependent PSA; the approximate step-wise approach by consecutively running a standard PSA, the direct approach by modifying the PSA quantification formulas, and the risk importance approach (the Taylor expansion approach) by using appropriate PSA importances. Features of each approach are described.

4. Application of an Age-Dependent PSA
   Data and Modeling requirements for different applications of a PSA are described.

5. Using a PSA to Evaluate the Risk Effects from Aging Passive Components
   The modeling of aging passive components is discussed, including the modeling of crack growth phenomena and corrosion phenomena.

Specific approaches for prioritizing the risk importances of passive components are given.

7. Prioritizations of Aging Contributors

Applications to prioritize aging contributors are presented.

8. Evaluations of Test and Maintenance Effectiveness

Applications to evaluate the risk effectiveness of an aging maintenance program are presented.

9. Sensitivity Studies and Uncertainty Analyses of Aging Effects

Applications are presented for aging sensitivity studies and aging uncertainty analyses.

Appendices
SBWR TECHNOLOGY AND DEVELOPMENT

A.S. Rao, R.J. McCandless, C.D. Sawyer
GE Nuclear Energy, San Jose, California, USA

ABSTRACT

The SBWR is based on utilizing to the maximum extent possible proven LWR technology developed through 30 years of operating plant experience plus the ABWR technology development program. For the unique features, developmental programs have been put in place to qualify the design.

Thus, the focus of technology development has been on the passive safety features - the gravity-driven ECCS (GDCS) and the containment heat removal (PCCS). For these features, a program with a combination of component tests, integral system tests and methods development has been put in place.

GE constructed a full-height, scaled, integral facility to demonstrate the GDCS concept and provide data for methods qualification. The testing was completed in 1989. In addition, a key component of the GDCS, the depressurization valve, underwent a design, development, and environmental qualification program leading to a fully qualified valve in 1990.

For the PCCS, a three-pronged program was implemented. Basic heat transfer data were obtained via testing at the Massachusetts Institute of Technology and the University of California at Berkeley. A full-height scaled integral facility to demonstrate the PCCS concept and provide data for methods qualification was constructed in Japan in 1989. Initial testing is now complete. Design of a full-scale heat exchanger unit is underway and testing is planned for completion in early 1993.
Initial Performance Assessment of the Westinghouse AP600 Containment Design and Related Safety Issues *

V. F. Nicolette and K. E. Washington,
Sandia National Laboratories
and
J. L. Tills,
Jack Tills and Associates

ABSTRACT

This work summarizes the Westinghouse AP600 advanced reactor design assessment calculations performed to date with the CONTAIN code. Correlations for modeling the important heat transfer phenomena are discussed as well.

A CONTAIN model of the AP600 was constructed for design basis accident (DBA) calculations. Insights gained from modeling of the smaller-scale Westinghouse Integral Test Facility were incorporated in the development of the AP600 model. The results of the DBA calculations are compared to the results of other researchers to serve as a point of reference for future severe accident calculations.

The CONTAIN calculations are reviewed to examine several parameters/phenomena of interest. For example, the air velocity in the annular gap between the containment building and shield building is discussed. The magnitude and distribution of this velocity are critical in determining whether forced convection heat transfer correlations are appropriate for the annular region. The ability of the CONTAIN code to model relatively long-time accidents (3 days or more) is also discussed.

The results of the calculations are also used to identify limitations of the CONTAIN code regarding application to advanced reactor containment designs. For example, one of these limitations is the inability of CONTAIN to model water films on the containment outer surface. As a result of this limitation, all of the CONTAIN calculations performed to date assume that the containment outer surface is dry. This assumption greatly reduces the ability of the AP600 passive containment cooling design to dissipate the core decay heat. However, the CONTAIN calculations provide a lower bound on the heat removal capabilities of the AP600 containment.

The most recent heat transfer correlations available in the literature are assessed for use in the flow regimes and geometries applicable to the AP600. Use of one of these correlations in CONTAIN may allow for a more accurate assessment of the AP600. A point of interest is the modeling of condensation phenomena inside containment. The ability of the condensation model in CONTAIN to accurately model condensation under accident conditions unique to advanced reactor design is discussed.

*This work performed at Sandia National Laboratories supported by the U.S. Department of Energy under Contract DE-AC04-76DP00789.
AN APPROACH FOR ASSESSING ALWR PASSIVE SAFETY SYSTEM RELIABILITY

Tania M. Hake

Reactor Systems Safety Analysis Division
Sandia National Laboratories
Albuquerque, New Mexico 87185

SUMMARY

Many of the Advanced Light Water Reactor (ALWR) concepts proposed for the next generation of nuclear power plants rely on passive rather than active systems to perform safety functions. Despite the reduced redundancy of the passive systems as compared to active systems in current plants, the assertion is that the overall safety of the plant is enhanced due to the much higher expected reliability of the passive systems. In order to investigate this assertion, a study is being conducted at Sandia National Laboratories to evaluate the reliability of ALWR passive safety features in the context of Probabilistic Risk Assessment (PRA). The purpose of this paper is to provide a brief overview of the approach to this study.

The quantification of passive system reliability is not as straightforward as for active systems, due to the lack of operating experience, and to the greater uncertainty in the governing physical phenomena. Thus, the adequacy of current methods for evaluating system reliability must be assessed, and alternatives proposed if necessary. For this study, the Westinghouse Advanced Passive 600 MWe reactor (AP600) was chosen as the advanced reactor for analysis, because of the availability of AP600 design information. This study will compare the reliability of AP600 emergency cooling systems with that of corresponding systems in a current generation reactor.

The term "passive" as used here refers to systems which rely heavily on natural processes such as natural circulation to perform their function, rather than on decidedly "active" components such as pumps. However, many passive safety systems do contain mechanical components, valves in particular, that must change state for the system to operate. Generally these valves will have to change state only once during their mission, and motive power is in the form of stored energy such as compressed air or battery power. Past reactor PRA methods are directly applicable for the component failure aspect of passive system modeling. On the other hand, accounting for uncertainty in the natural processes involved in passive system operation requires an alternate approach. This is essentially an uncertainty in the success criteria for the passive systems. Specifically, given proper component functioning (valves open or close as required), a measure is needed of the degree of certainty that the natural process (natural circulation, gravity-induced flow, evaporative cooling, etc.) provides the fluid

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driving force or heat removal required to avert core damage. This uncertainty derives to a great extent from uncertainty in parameters associated with the process of interest, such as heat transfer coefficients or friction factors.

This study's approach to assessing AP600 emergency cooling function reliability begins with examination of component failures. NUREG-1150 methods are used here. Event trees delineate the emergency cooling systems that can be used in response to an accident initiator, which for the AP600 includes credit for not only the passive safety systems but also the active non-safety systems. Fault tree models are constructed to represent component failures for the systems of interest. Failure data from past PRAs and existing databases are generally applicable due to similarities in components included in the AP600 design and in current operating units. There are a few exceptions which result from differences in areas such as component usage or system safety classification (i.e. safety versus non-safety grade).

In order to incorporate natural process uncertainties in the quantification of passive system reliability, selected accident sequences involving passive system operation are analyzed using thermal-hydraulic models. These sequences are selected based on quantitative results of the component failure analysis, and qualitative information on the expected importance of the natural process to passive system operation. The sequence-level approach to the analysis is necessary because evaluation on the system level, i.e. one passive system at a time, would not consider the influence of other passive and active systems which operate (or fail) in the sequence. Thus, each sequence potentially represents a unique set of conditions affecting the outcome of the thermal-hydraulic analyses, although some binning of similar sequences is possible.

First, thermal-hydraulic models of the AP600 are built, and sensitivity calculations are performed to determine the most important code input parameters, such as heat transfer coefficients. Next, expert elicitation is performed using the NUREG-1150 structured approach, to obtain distributions on values for the important input parameters. Latin Hypercube Samples are constructed from the distributions on the input parameters in order to build multiple input decks for the thermal-hydraulic code. The multiple code calculations are performed for a given sequence and the results analyzed to determine the contribution of the natural process uncertainties to overall sequence outcome. This result is combined with the quantitative results of the component failure analysis to provide an overall measure of the reliability of systems serving the emergency cooling function.

Initially, a demonstration of the approach to natural process assessment will be completed. This demonstration will include only a few selected sequences involving operation of passive systems associated with core cooling (as opposed to containment heat removal). For the AP600, the passive core and containment cooling systems are to a great extent coupled, and complete evaluation of the core cooling systems will require consideration of the containment systems. Such a complete analysis is to be performed in the implementation phase of the program, pending successful method demonstration. The full paper will provide more details concerning the tasks of this project, and will discuss current results and insights.
Westinghouse, the Department of Energy, and the Electrical Power Research Institute, have joined together to complete the design of an advanced PWR which incorporates passive safety features; the AP600. The design goals for the AP600 are to

1) provide enhanced safety by the use of passive systems and improved defense in depth;

2) reduce the capital cost of the plant by reducing the need for equipment and more optimal design;

3) reduce the cost of operation and maintenance by improved design which factors in the utility operating experience.

One key feature of the AP600 design is the use of safety grade passive safety systems which provide the ultimate core and containment cooling for postulated accidents. The key features are:

1) The Passive Residual Heat Removal (PRHR) - a heat exchanger is provided which can remove decay heat by natural circulation from the primary system.

2) Core makeup tanks -- large borated cold water tanks located on the cold legs which provide borated water inventory makeup to the reactor coolant system for small to large LOCA's.

3) Automatic Depressurization System (ADS) - this system is a series of 4-inch and 8-inch valves located on the pressurizer and hot leg of the reactor coolant system. As the core makeup tanks drain, providing inventory to the core, these valves will open in a programmed manner to depressurize the primary system to containment pressure and will provide an alternate hot side vent paths for the decay heat generated steam in the core.

4) Gas pressurized accumulators are used for large break LOCA protection and will inject once the reactor coolant system's pressure is below its setpoint (-700 psia).

5) Incontainment Water Refueling Storage Tank (IWRST) - this is a large 400,000 gallon tank of borated water which is elevated relative to the reactor vessel and is connected via check valves. When the ADS depressurizes the reactor coolant system, the IWRST can inject water into the reactor vessel downcomer to maintain the core cooling.

6) Passive Containment Cooling System (PCCS) - the PCCS uses the cylindrical containment steel shell to reject heat to the environment. The outside of the containment is cooled by evaporative film cooling with water supplied by gravity from a 350,000 gallon tank. The containment shield building and containment shell provide a path such that a natural draft of air will be
induced to flow over the falling liquid film promoting more cooling. Condensation will occur on the inside of the containment with the condensed steam recycling back to the IWRST.

The strategy in the AP600 test program is to concentrate on large-to-full scale component tests to develop the thermal-hydraulic models needed for the safety analysis codes.

1) There will be a 1/6-scale core makeup tank test to examine the condensation behavior in the tank, its draining performance and wall condensation.

2) There will be a full scale ADS test, with a full scale sparger to test valve performance and sparger loads.

3) There have been two series of PRHR tests conducted on three full length tubes to develop the heat transfer correlations needed for PRHR design.

4) A series of small scale PCCS experiments have been performed to examine water film behavior, evaporative film cooling, and condensation on inclined surfaces. Two series of integral containment tests were performed on a 3-foot by 25-foot vessel to study evaporative film cooling, air cooling, and condensation.

5. A 1/8-scale containment test is under construction which examines the containment behavior at a large scale and will be used for code verification. There is also a 1/8 full scale segment of the containment dome and portion of the side wall constructed to examine the water distribution on the top of the containment.

6) A check valve experimental program to verify the check valve pressure drop and flow behavior with small ΔP's characteristic of gravity driven systems is underway.

7) A 1/4-scale AP600 long-term cooling systems test is currently being designed. This test will examine the gravity feed core cooling behavior of the core makeup tanks, IWRST tank, and the reactor sump injection as well as the venting capabilities of the ADS.

The analysis program for the AP600 will use best estimate state-of-the-art computer codes to analyze and quantify the safety system performance and design margin. The WCOBRA/TRAC code will be used for the small and large break LOCA's. The containment analysis will be performed using the GOTHIC code which is an improved version of the COBRA/NC containment code and is capable of modeling the natural circulation and convective flow patterns, and condensation inside the containment as well as the evaporative film cooling on the outside of the containment shell.

Using the best analysis tools available, and by performing the above AP600 safety systems experiments, we believe that the AP600 passive safety systems can be accurately characterized such that the design will be acceptable to the Nuclear Regulatory Commission and the public.
SUMMARY

This paper presents results of preliminary analyses of the proposed Westinghouse Electric Corporation AP600 design. AP600 is a two loop, 600 MW (e) pressurized water reactor (PWR) arranged in a two hot leg, four cold leg nuclear steam supply system (NSSS) configuration [1]. In contrast to the present generation of PWRs it is equipped with passive emergency core coolant (ECC) systems [2,3]. Also, the containment and the safety systems of the AP600 interact with the reactor coolant system and each other in a more integral fashion than present day PWRs. The containment in this design is the ultimate heat sink for removal of decay heat to the environment [4].

Idaho National Engineering Laboratory (INEL) has studied applicability of the RELAP5 code to AP600 safety analysis [5] and has developed a model of the AP600 [6] for the U.S. Nuclear Regulatory Commission. The model incorporates integral modeling of the containment, NSSS and passive safety systems. Best available preliminary design data were used. Nodalization sensitivity studies were conducted to gain experience in modeling of systems and conditions which are beyond the applicability of previously established RELAP5 modeling guidelines or experience. Exploratory analyses were then undertaken to investigate AP600 system response during postulated accident conditions.

Four small break LOCA calculations and two large break LOCA calculations were conducted. All calculations were performed assuming normal operation of all safety systems. The small break calculations included simulation of a 3 in. communicative break in the cold leg Core makeup Tank (CMT) pressure balancing line, a 3 in. cold leg break, a 1 in. cold leg break, and a double-ended shear of the In-containment Refueling Water Storage Tank (IRWST) drain line at its tee connection to the safety injection line. The large break LOCA included a typical double-ended cold leg break and a double ended break of the pump inlet nozzle. The latter break is only typical for AP600 and was chosen to investigate the possible complexity of system behavior.

In the small break LOCA calculations the system response looks very similar to that of current generation PWRs until Automatic Depressurization System (ADS) actuation. This system is incorporated in the AP600 design to reduce the primary pressure so that long

* Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under DOE Contract No. DE-AC07-76ID01570
term cooling by gravity draining of the IRWST can be established. The calculations show that the first three stages of ADS are very effective in reducing the system pressure to approximately 1 MPa. The fourth and last stage of the ADS continues to reduce the system pressure however at much smaller rate due to the smaller energy content of the expelled coolant. The calculations indicated increased dependence on the fourth stage depressurization with decreasing break size as expected and concomitantly longer periods without any ECC flow during the period between CMT emptying and initiation of IRWST flow. A core heatup was observed prior to IRWST flow in the IRWST drain line break.

The large break calculations showed less severe AP600 core thermal response than for current generation PWRs. This is due to lower power density of the core and smaller break to coolant volume ratio. The passive safety systems were able to effectively quench the core. Competing flows between IRWST and CMTs are observed after the reflood. The calculations were conducted only to refill stage of the transient.

Because of preliminary, evolving nature of the AP600 design, conclusions concerning the adequacy of the design and efficacy of the passive safety systems should be avoided. The results of these calculations are indicators of potential system sensitivities, and support understanding of AP600 response. Furthermore, the calculations identify areas where modification or extension of RELAP5 may be required to better characterize AP600 transient response.

REFERENCES

Expanding the Modeling Capabilities of the Cognitive Environment Simulation

Emilie M. Roth
Westinghouse Science & Technology Center

Harry E. Pople, Jr.
Seer Systems

Randall J. Mumaw
Westinghouse Science & Technology Center

The U. S. Nuclear Regulatory Commission has been conducting a research program to develop more effective tools to model the cognitive activities that underlie intention formation during NPP emergencies. Under this program an artificial intelligence (AI) computer simulation called Cognitive Environment Simulation (CES) has been developed (cf. NUREG/CR-4862; NUREG/CR-5213). CES simulates the cognitive activities involved in responding to a NPP accident situation. It is intended to provide an analytic tool for predicting likely human responses, and the kinds of errors that can plausibly arise under different accident conditions to support human reliability analysis (HRA). Recently CES was extended to handle a class of interfacing loss of coolant accidents (ISLOCAs). This paper summarizes the results of these exercises and describes follow-on work currently underway. A full description of the results of the ISLOCA exercises are provided in NUREG/CR-5593.

As an AI simulation CES is composed of a knowledge base and an inference engine. The CES knowledge base contains information about the nuclear power plant, including plant parameters available to be monitored, and their normal operating limits; the inter-relationships among plant physical processes; goals for safe plant operation; abnormalities (e.g., power failures; breaks) and the effect they have on plant processes; and what actions can be taken to correct abnormalities. As such it provides a mechanism for modeling the kind of knowledge of NPP that an operator would be presumed to have based on training, procedures, and experience. The inference engine provides reasoning mechanisms that enable CES to monitor changing plant parameters, to formulate and revise situation assessments, and to generate intentions to act. While the particular reasoning mechanisms utilized by CES do not mimic in detail the cognitive processes of human operators (e.g., short term memory; detailed monitoring or diagnostic strategies), it performs the major cognitive activities that are required to successfully assess and respond to an NPP emergency event (i.e., the cognitive tasks that human operators would necessarily have to perform to successfully handle the event). By performing the major cognitive activities required to handle a NPP emergency event, it provides a tool for assessing the cognitive challenges imposed by different accident sequences (e.g., what evidence needs to be examined, what knowledge needs to be accessed, what alternative hypotheses arise that need to be discriminated, what safety goals need to be considered in managing the accident), and potential opportunities for error.

As part of the CES development process CES was recently exercised on two ISLOCA scenarios. The results of the exercises illustrate how cognitive simulations such as CES, coupled with small-scale data on human performance, can be used to provide input to HRA. They also point to a need to improve the ability of CES to simulate human processing limitations.
The ISLOCA scenarios involved a leak from the Reactor Coolant System into the Residual Heat Removal (RHR) System. These events were diagnostically challenging because they produced symptoms in multiple regions of the plant that are normally unconnected (i.e., abnormal radiation in containment and in the RHR system) suggesting (misleadingly) the possibility of multiple independent breaks. CES successfully diagnosed both ISLOCA scenarios. The line of reasoning employed by CES to diagnose the ISLOCA and subsequent break in the RHR was very similar to the line of reasoning employed by human crews run on the same events; However CES performed the diagnosis much earlier in the event than the human crews did.

The results of the ISLOCA exercises served to clarify the strengths and limitations of the current version of CES. CES was shown to be a powerful tool for analyzing the problem-solving demands of a situation. It can be used to uncover what knowledge an operator must possess, what plant parameters he must monitor, and what evidence integration he must perform to successfully handle the task. It can also provide a lower limit on how quickly a correct diagnosis can be made.

The ISLOCA exercises serve as a model of how a cognitive simulation such as CES can be used to provide practical insight on human reliability issues of concern. CES, as a computer simulation, provides a means of specifying objectively the NPP knowledge and cognitive activities required to diagnose an incident of concern. As such it provides an objective tool for establishing the generality of a human reliability analysis derived from limited observations of human crews.

At the same time, the ISLOCA exercises revealed limitations of the current version of CES. CES detected disturbances sooner, and followed implications of disturbances more thoroughly than the human crews did. This suggests a need to incorporate mechanisms in CES that more accurately simulate human processing limitations.

The NRC is currently sponsoring a follow-on research project. One goal of this project is to identify and implement additional features in CES to better model human processing limitations. A second goal is to convert CES to run on a more widely available computer to improve its accessibility to potential NRC users.
HUMAN PERFORMANCE INVESTIGATION PROCESS (HPIP)

By: Mark Paradies & Linda Unger, System Improvements, Inc.

INTRODUCTION

The Human Performance Investigation Process (HPIP) is a systematic method for use by U.S. Nuclear Regulatory Commission (NRC) personnel investigating incidents that include human error. The combination of techniques into an investigation procedure is designed to help investigators (for example, a typical resident inspector with little human factors training) find the root causes of human performance problems that contributed to the incident.

This summary briefly outlines the process and the techniques. The full paper will provide a more in-depth description of the process, the techniques, and their development and evaluation. The complete documentation of the process will be presented in NUREG/Cr-5455, Development of the NRC Human Performance Investigation Process (HPIP) to be published late in 1991.

BACKGROUND

The nuclear power industry has gone through a transition from a period of plant construction to a period of plant operations. As the industry went through this transition, the NRC has likewise changed its regulatory emphasis. Today, with nuclear construction and licensing complete, the NRC is focusing on safe operation and maintenance of nuclear plants. This change in focus has caused the NRC's regulatory efforts to be less oriented toward design basis studies and more oriented toward operating experience and the human contribution to plant performance. Therefore, plant performance indicators and operational events now receive special attention.

As increased attention is paid to human performance, its contribution to plant safety becomes more evident. In the NRC Regions, resident inspectors or other regional personnel frequently review human performance when investigating the causes of incidents and when assessing the adequacy of proposed corrective actions to prevent the recurrence of incidents. Therefore, HPIP is designed to:

- Help investigators more accurately pinpoint the root causes of human performance related incidents, thereby leading to a better understanding of human performance problems.
- Be easily understood by the user and require little initial training (perhaps no more than one day).
- Be easily understood by management.
- Be compatible with database applications for ease of trend analysis.
- Not require significantly more effort by investigators in the field.
A flow chart of the investigation process was used to integrate the techniques into the process. This flow chart, shown in Figure 1, outlines the basic process (to which there are many variations) that NRC personnel in the field would use to investigate a fairly complicated incident. The flow chart consists of three parts: (1) the NRC HPIP Flow (center column) displays the steps used to investigate an incident; (2) The Purpose (left column) lists the purpose of each major section of the process; (3) The Tools (right column) lists the techniques used to perform each major section of the process.

The six HPIP techniques ("Tools") help investigators decide where to spend their investigative effort and identify the incident's root cause(s). These techniques are:

- **Events and Causal Factors Charting** - a technique that lays out the flow of an incident in a graphic format so that potential causal factors can be identified for further investigation.

- **SORTM** - a simple paper-based expert system designed to lead the investigator to those human performance areas most likely to have contributed to human error.

- **HPIP Modules** - individual procedures for investigating common human performance problems and identifying their root causes. One of these procedures was developed for each of SORTM's six human performance areas: (1) procedures; (2) training; (3) human engineering; (4) supervision; (5) communications; and (6) organizational factors / management systems.

- **Barrier Analysis** - a formal method of identifying the events that, if avoided, would have prevented the incident from occurring or would have significantly mitigated its consequences.

- **Change Analysis** - a technique to help analyze an incident in a system that had previously been working properly, but that may have been negatively impacted by a change in the system or process.

- **Critical Human Action Profile (CHAP)** - a technique based on task analysis and used to identify the human actions most critical to the failure of interest and to document the requirements that were necessary if these actions were to be performed successfully.

Investigators will seldom, if ever, use all of these techniques during a single incident investigation. At a minimum, however, Events and Causal Factors Charting, SORTM, and one or more of the HPIP Modules will be used to identify the root causes of a human performance difficulty.

Once the root causes of an incident have been determined, the investigator needs to consider whether the cause is an isolated occurrence or the result of a programmatic weakness. Programmatic causes are often uncovered by the review of the plant's operating experience or by reviewing additional procedures for similar problems. If similar problems are detected, the investigator then needs to dig deeper to uncover the reasons for the programmatic problems.
## FIGURE 1: NRC HPIP Flow Chart

<table>
<thead>
<tr>
<th>Purpose</th>
<th>NRC HPIP Flow</th>
<th>Tools</th>
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<tbody>
<tr>
<td>Call to Resident or 50.72 Notification</td>
<td>Events &amp; Causal Factors Charting SORTM</td>
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<tr>
<td>Plan investigation</td>
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<td>Collect facts</td>
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<td>Collect Physical Evidence</td>
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<tr>
<td>Interviews</td>
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<tr>
<td>Understand event, ensure complete investigation, ensure accuracy of &quot;facts&quot;</td>
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<td>Identify human performance difficulties for root cause analysis</td>
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<td>Identify Barriers to Event and Potential Human Performance Difficulties</td>
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<td>Analyze Root Causes</td>
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<td>Analyze Programmatic Causes</td>
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<td>Evaluate Utility's Corrective Action &amp; Identify Violations</td>
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<td>Generate &amp; Present Inspection Report with Findings &amp; Violations</td>
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<td>Ensure corrective actions address root causes &amp; violations are identified</td>
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<td>Identify important trends or &quot;generic&quot; system weaknesses</td>
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<td>Accurately document event, Effectively present management with findings/violations</td>
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<td>Find correctable causes for the specific event</td>
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From the NRC’s regulatory perspective, the results of an HPIP analysis (the incidents root causes) can then be used to review the utility's corrective actions (Did they address all the root causes?) and identify potential violations of rules and regulations. If the incident has been identified as being similar to past incidents: why weren't the previous corrective actions effective; how are the corrective actions currently proposed different from those proposed in the past; and are additional barriers proposed?

EVALUATION PROCESS

As noted earlier, HPIP is currently being evaluated in the field. This evaluation includes:

- Review of the process by the NRC's Research Program Review Group to evaluate HPIP's readiness for testing and suggest testing criteria.
- Review of the process at a usability workshop during which NRC personnel suggested improvements to HPIP.
- Independent evaluations of two HPIP training courses with recommendations to improve the training's effectiveness.
- Field testing of HPIP by 34 trained investigators.
- Firsthand observation by the development team of HPIP's use in the field.
- Assessment of the improvements in reports generated when using HPIP.
- Independent human factors review of HPIP.

The feedback provided during the evaluation will be used to develop improvements for the process. These improvements will then be incorporated into HPIP prior to field implementation.
ORGANIZATIONAL FACTORS INFLUENCING IMPROVEMENTS IN SAFETY

Dr. Alfred Marcus  
Dr. Mary L. Nichols

Strategic Management Research Center  
University of Minnesota  
271 19th Avenue South  
Minneapolis, MN 55455

and

Dr. Jon Olson, Battelle, Human Affairs Research Centers  
Dr. Richard Osborn, Wayne State University  
Dr. James Thurber, American University

INTRODUCTION

Results of conceptual and empirical research conducted by this research team, and published in NUREG-CR 5437, suggested that processes of organizational problem solving and learning provide a promising area for understanding improvement in safety-related performance in nuclear power plants. In this paper, we describe the way in which we have built upon that work and gone much further in empirically examining a range of potentially important organizational factors related to safety.

The paper will describe (1) overall trends in plant performance over time on the NRC performance indicators, (2) the major elements in the conceptual framework guiding the current work, which seeks among other things to explain those trends, (3) the specific variables used as measures of the central concepts, (4) the results to date of the quantitative empirical work and qualitative work in progress, and (5) conclusions from the research.

1. TRENDS IN PERFORMANCE OVER TIME

Data compiled by the NRC shows performance on indicators including SCRAMS, Safety System Actuations, Significant Events, and Safety System Failures trending toward improved performance, on average, since 1985. For each of these indicators, plus critical hours and release of radiation, the entire population of publically-owned nuclear power plants in the U.S. were divided into three categories--the top 10 percent, the middle 80 percent, and the bottom 10 percent--based on their performance on a given indicator over the period 1985-1989. Results were graphed and show that, overall, the top 10% performs well at the beginning of the period and remains at approximately the same level over the period. That is, they remain good,
steady performers. The poorest 10% show fairly steady improvement over the time period. From these patterns, we conclude that the key questions which must be addressed are "how do plants improve their performance?"; "why do some not improve?"; and "how do plants that perform well sustain their performance?".

2. CONCEPTUAL FRAMEWORK

To address the above questions, a conceptual framework was developed which may be considered "A Model of Organizational Learning at Nuclear Power Plants." The criterion variables are plant performance on the NRC performance indicators. The six elements of the model and a brief rationale for their inclusion are as follows:

(1) **Past Performance.** In a model explaining improvement or decline, past performance must be taken into account because of its baseline effects. Furthermore, past performance is likely to have a very strong effect on subsequent performance because it if difficult for systems to overcome past inertia.

(2) **NRC Problem Identification.** In order for improvement to occur, problems must be recognized. One of the sources of problem recognition occurs via regulatory oversight processes conducted by the NRC.

(3) **Utility Financial Performance.** Nuclear power plants are part of a larger financial entity, the parent utility, which allocates resources for the plants and communicates performance expectations. Financial performance of the parent utility influences learning through its affect on the availability of resources to address problems once they have been recognized, and second through its ability to command the attention of utility executives, especially when it is below expectations.

(4) **Utility Application of Resources.** The decisions utilities make regarding how to allocate their resources influences plant performance in at least two major ways. Past decisions regarding plant and equipment reflect themselves as fixed costs, and create the setting in which the plant is operated. On-going decisions regarding how much to allocate to various categories of expenses, such as production expenses, operations supervision and engineering, and maintenance supervision and engineering, potentially reveal a great deal about actual activities in the plant, including relative emphasis on maintenance versus operations, etc. In reflecting actual activities, they may also therefore reflect operating philosophies.

(5) **Nuclear Power Plant Production Experience.** It has been demonstrated by economists and operations researchers that as operating experience accumulates, learning occurs which results in improved production efficiency. Therefore experience could also lead to increased understanding of what is required to achieve safe operations.

(6) **Utility Business Strategies.** The central purpose of a business strategy is to provide a focus which commands the attention of the firm, its managers, and its
resources. The extent to which attention is focussed on nuclear, or conversely, distracted from nuclear and attendant safety considerations, is expected to influence performance on the NRC performance indicators.

In all analyses based on this model of learning we control for type of reactor.

3. VARIABLES

The dependent variables in the quantitative empirical analyses are the six NRC performance indicators; the independent variables will be specified in the full paper. They are implied in the following summary of method and results.

4. METHOD AND RESULTS

Several sets of analyses will be reported in the full paper. They consist of data that were graphed, using categories described above (top 10% vs. bottom 10% of plants for a given performance indicator from 1985-89), where the categories were graphed against selected independent variables (e.g. return on assets). Analysis consisted of visual comparisons of the relationships among the plants in the top 10% and bottom 10% and their respective prior financial performance (return on assets) in 1980-85, prior problem identification (major violations) in 1980-85, and resource allocation (operations and maintenance expenditures) in 1985-89. Space does not allow a detailed description of results, but they can be summarized as showing that plants appear to fall into patterns of beneficent and vicious cycles which are difficult to break.

In the second set of analyses to be reported, Poisson regression models for scrams, safety system failures, significant events, and safety system actuations were tested. The models incorporated variables from each of the elements in the conceptual framework, and controlled for reactor type. Briefly summarized, results showed that plant performance and improvement or degradation in performance is influenced by the allocation of resources by the utility, the availability of resources, the regulatory process of problem identification and correction carried out by the NRC, the business and power generation strategies pursued by the utility, and the experience the utility has with nuclear power production. There are different sets of predictors--or "profiles"--for different performance indicators, but the past is the strongest predictor of future performance.

Finally results of qualitative studies will be described. These were undertaken to supplement the quantitative analyses and were aimed at examining organizational learning processes through review of detailed diagnostic evaluations and a series of case studies developed through site visits.

Conclusions from the research will be provided.
There is a basic syllogism that drives organizational research in the nuclear power plant environment. The syllogism is as follows:
Organizational factors may cause errors in nuclear plants.
Errors in nuclear plants increase risk to workers and the public.
Therefore, organizational factors can increase risk to nuclear workers and the public.

The syllogism is deceptively simple. For example, the major premise leads to the further questions of what organizational factors? Under what conditions? To what types of errors? Similarly, the minor premise also leads to additional questions. For example, which errors increase risk? What types of risk are increased? For whom is the risk increased? What are acceptable levels of risk? Finally, with respect to the conclusion, one might wonder how one can demonstrate the relationship between organizational factors and risk and what the boundary conditions for this proposed relationship might be.

The current research project directly addresses several of these questions. Most immediately, we are interested in identifying relevant organizational factors. We have chosen three methods for identifying these factors. Each method involves an interview as the data collection device. The first method consists of an elaborate theory based protocol that is administered by trained interviewers to key respondents similar to the cadre identified using the NOMAC scheme developed by Haber and her colleagues at Brookhaven. The protocol deals with several levels of the organization including upper level management (analogous to the Strategic Apex in NOMAC terms), interdepartmental organization and intradepartmental organization (work teams, shifts, subunits, etc.). In addition, the protocol deals with several different aspects of plant organization such as decision making procedures, interdepartmental coordination of activities, and communications. Particular attention is paid to organizational practices, staffing and human resource utilization, and employees' reactions to physical work environments. The protocol has been field tested using a "think aloud" pilot procedure. In addition, it has undergone field operational testing following revisions implied by the think aloud pilot.

The second method chosen is known as a Goals/Means/Measures protocol. The logic of this approach is to identify a priori safety related goals and determine: a) what particular safety goals are set by key players in the organization, b) what means do they use to achieve these goals, c) what evidence is there that these means are employed, and d) how would one know that these efforts have been successful. For purposes of this protocol, safety is defined as freedom from radiological release, freedom from contamination, and absence of core damage. As was the case with the first interview protocol, the goals/means/measures instrument will be administered to key individuals as identified in a NOMAC like structure of key respondents.

In addition to the "orthodox" operational definitions of safety, we will also be considering the seven commonly identified leading indicators as penultimate variables that would provide useful information about the ultimate measures. Part of the validation process for the interview formats will require us to show that variation in organizational processes is accompanied
by variation in outcome variables. These leading indicators may prove useful and surrogates for the ultimate measures, and, more importantly, surrogates with some variability across sites.

The third technique to be used for identifying key organizational variables and processes is known as Behaviorally Anchored Rating Scale (BARS) development. These scales are based on critical incidents presented by subject matter experts (incumbents and supervisors). This technique directly addresses the issue of which variables are central to the differences between safe and unsafe environments. At this point in the project, several sessions have been completed in the development of these scales and preliminary results will be presented.

It is hoped that these three methods will converge to yield most essential organizational factor in nuclear power plant safety. The importance of this convergence will be discussed in the paper.
INCLUDING TEST ERRORS IN EVALUATING SURVEILLANCE TEST INTERVALS

I.S. Kim, S. Martorell*, W.E. Vesely†, and P.K. Samanta

Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973, USA

*Universidad Politecnica of Valencia, Valencia, Spain
†Science Applications International Corporation, Ohio, USA

SUMMARY

To evaluate the risk impact of surveillance requirements defined in Technical Specifications, both the beneficial and negative effects of surveillance should be considered. The beneficial effect of a test results from the detection of failures by the test, and its risk impact has been analyzed elsewhere.1

On the other hand, the negative effect of surveillance testing can be caused by test errors, e.g., human errors of omission or commission including potential for common cause failures. As a consequence of the negative effect, the performance of periodic testing can have adverse impact on safety. This concern is aggravated by the "overwhelming" amount of testing presently required by Technical Specifications.

To address the problem with surveillance testing, i.e., the adverse safety impact exacerbated by the significant amount of testing, the U.S. Nuclear Regulatory Commission has performed a series of studies. NUREG-1024,2 made recommendations for enhancing the safety impact of surveillance requirements. NUREG-13663 implemented the recommendations by "qualitatively" examining all Technical Specifications surveillance requirements to identify those that should be improved. The following four different types of the negative effect were used in the NUREG-1366 study as the screening criteria: (a) leading to a plant transient, (b) unnecessary wear to equipment, (c) unnecessary radiation exposure to plant personnel, and (d) unnecessary burden on plant personnel.

This paper defines the various negative effects of surveillance testing from a risk perspective, and then presents a methodology to "quantify" the negative risk impact, i.e., the risk penalty or risk increase caused by the test. The quantitative methodology focuses on two important kinds of negative effects, i.e., test-caused transients and test-caused equipment degradations. These negative effects generate significant safety concerns due to: (1) plant abnormality which may challenge safety systems and plant operators and (2)

1This work was performed under the auspices of the U.S. Nuclear Regulatory Commission.
equipment wear-out which increases safety system or function unavailability and thereby reduces the plant's accident mitigating capability.

The paper will present (1) a PRA-based method to evaluate the negative risk impact due to test-caused plant transients and (2) a method based on a test-caused equipment degradation model and PRA to assess the negative risk impact associated with equipment wear-out. Also described in the paper are illustrative applications of the methods to specific surveillance tests conducted at boiling water reactors (BWRs) such as the tests of main steam isolation valves (MSIVs), turbine overspeed protection system, and diesel generators. Evaluation results of the risk effectiveness of the tests are presented along with the insights from the sensitivity analysis of the risk impact versus test interval.

These risk-oriented methods can be used in the regulatory decision making process to help establish optimal test intervals by explicitly considering the negative effect as well as the positive effect of testing.

REFERENCES


Aging Assessment of BWR Control Rod Drive Systems*

by Rebecca H. Greene
Oak Ridge National Laboratory

ABSTRACT

This Phase I NPAR study examines the aging phenomena associated with BWR control rod drive mechanisms (CRDMs) and assesses the merits of various methods of "managing" this aging. Information for this study was acquired from (1) the results of a special CRDM aging questionnaire distributed to each U.S. BWR utility, (2) a first-of-its-kind workshop held to discuss CRDM aging and maintenance concerns, (3) an analysis of NPRDS failure cases attributed to the CRD system, and (4) personal information exchange.

An eight-page questionnaire was prepared by ORNL and distributed to all domestic BWR plants. The survey solicited site-specific data on CRDM degradation and failure experience, maintenance and aging interactions, and current testing procedures. To obtain firsthand information on CRDM aging histories, a workshop was sponsored by ORNL to openly discuss CRDM performance and the overall questionnaire results with utility participants. The three-day meeting on CRDM aging and maintenance was attended by 26 utility personnel from 21 BWR plants and 14 vendor and commercial representatives. These attendees provided invaluable information needed for understanding degradation mechanisms and maintenance constraints associated with BWR CRDMs.

As part of this study, nearly 3500 NPRDS failure reports have been analyzed to examine the prevailing failure trends for CRD system components. A investigation has been conducted that summarizes the occurrence frequency of these component failures, discovery methods, reported failure causes, their respective symptoms, and actions taken by utilities to restore component and system service.

The results of this research have identified the predominant CRDM failure modes and causes. In addition, recommendations are presented regarding specific actions that utilities can implement to mitigate CRDM aging. An evaluation has also been made of certain practices and tooling which have enabled some utilities to reduce ALARA exposures received from routine CRDM replacement and rebuilding activities.

THE EFFECT OF AGING UPON CE AND B&W CONTROL ROD DRIVES

Edward Grove and William Gunther
Brookhaven National Laboratory
Upton, New York 11973

SUMMARY

Though mechanically different, the control rod drive (CRD) systems used at both CE and B&W plants position the control rod assemblies (CRA) in the core in response to automatic or manual reactivity control signals. Both systems are also designed to provide a rapid insertion of the CRAs upon a loss of AC power.

The CRD system consists of the actual drive mechanisms, power and control, rod position indication, and cooling system components. This aging evaluation included the individual absorber rods, and the fuel assembly and upper internal guide tubes, since failure of these components could preclude the insertion of the control assemblies.

Aging and environmental degradation have resulted in system and component failures. Many of these failures caused dropped or slipped rods which adversely affected plant operations by resulting in power reductions, scrams, and safety system actuation. No CRD system failure has ever resulted in the inability to shut down a reactor. However, unplanned, automatic trips challenge the operation of the plants safety systems. Consequently, their occurrence represents a potentially significant increase in plant risk. System and component failures have resulted in four Information Notices during the past decade.

CRD DESIGN AND OPERATION

The control rod drive mechanisms are flange mounted on top of the reactor vessel head and are designed to be operational at system temperature and pressure. Reed switches, which provide rod position indication signals, are located in a separate housing alongside the drive housing, and are subjected to the same operational and environmental stresses. All power and control system components are located in modularized electrical cabinets out of the primary containment.

The majority of CE plants use the magnetic jack control element drive mechanism (CEDM). The sequenced energization of the gripper coils actuate the mechanical latches and move the leadscrew and the mechanically coupled control element assembly (CEA). Fort Calhoun and Palisades use a rack and pinion CEDM. Similarly, the controlled rotation of the pinion gear produces rack movement resulting in CEA location.

B&W utilizes a roller nut control rod drive mechanism (CRDM), which in the presence of a rotating magnetic field, engages and spins around the leadscrew causing it to move vertically. Energization of two opposite stator poles causes the roller nuts to remain engaged with the leadscrew, which does not produce rotation, holding the CRAs stationary.

OPERATIONAL EXPERIENCE

Failure and degradation of power and control system components accounted for the majority of CRD system failure occurrences, primarily due to component aging or potentially aging effects. Due to the modularized, easily replaceable
system component design, root causes of failure were often not determined. For CE plants, power supply, current sensor, timer module, and power switch failures occurred frequently. Breaker failures were not common, however one molded case circuit breaker (MCCB) was reported to have tripped at a current 25% less than design. This necessitated the addition of all MCCBs to the preventive maintenance program to insure continued reliability for the 40 year design life. SCR gate drive actuator and system fuse failures were common problems within the B&W system. Several instances of component failures due to overheating were reported, demonstrating susceptibility to environmental stress and inadequate maintenance.

Primary coolant leakage due to seal aging degradation were reported for both CRD designs. The CE rack and pinion rotating seals, and the B&W flexitallic flange seals were the common failure sources. Primary coolant leakage in a high temperature area, will cause the boric acid to boil, increasing the acidity and corrosiveness. Uncorrected, it can accumulate and block stator cooling passages, resulting in failure.

Loss of rod position indication due to system aging were reported. Reed switch failure rate and the erratic operation due to the buildup of contact surface film, and the low closing force required from the decreased gap necessitated B&W to redesign both the switches and circuitry. The new, redundant four channel system incorporates high differential, glass enclosed switches and circuitry which remains operational in the event of reed switch failure.

Control Element Assembly Calculator (CEAC) failures in CE plants resulted in significant plant effects. The two safety related CEACs transmit penalty factors to the Core Protection Calculators based upon CEA position and misalignment data transmitted by the reed switches. Erroneous position signals has resulted in the generation of overly conservative penalty factors resulting in a power decrease or scram.

Instances documenting the inability to insert individual CRAs were reported. Debris from fractured internal CRDM leaf springs resulted in immovable mechanisms on two separate occasions. A complete circumferential, through wall crack on a CE control element rod, resulted in the lower portion of the rod to break off and fall, with the poison pellets, into the fuel assembly guide tube. This prevented the CEDM from fully inserting the CEA.

CONCLUSIONS AND RECOMMENDATIONS

The operating review for both the CE and B&W CRDs indicate that both have experienced age related component degradation and failure. Susceptibility to environmental stresses, maintenance and human error was also evident. Effects of these failures resulted in redundancy loss, power reduction, scrams, and ESF actuation.

Vendor recommended and utility maintenance, inspection, and surveillance practices have been assessed for usefulness in detecting and mitigating age degradation. Advanced inspection techniques such as infrared thermography, motor current signature analysis, and Electronic Characterization and Diagnostics (ECAD) may also be useful in detecting and trending aging, and should be evaluated further.
This study is being performed to examine the relationship between time dependent
degradation, and current industry practices in the areas of maintenance, surveillance, and
operation of steam turbine drives for safety related pumps. These pumps are located in the
Auxiliary Feedwater (AFW) system for pressurized water reactor (PWR) plants, and the Reactor
Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) systems for Boiling
Water Reactor (BWR) facilities. This research has been conducted by examining current
information in NPRDS, reviewing Licensee Event Reports, and thoroughly investigating contacts
with operating plant personnel, and by personal observation. The reported information was
reviewed to determine the cause of the event and the method of discovery. From this data
attempts have been made at determining the predictability of events and possible preventive
measures that may be implemented.

Findings in a recent study on the Auxiliary Feedwater System (NUREG/CR-5404)
indicate that the turbine drive is the single largest contributor to AFW system degradation. This
is evidenced in the loss of feedwater event at the Davis Besse Nuclear Plant in 1985. However,
examination of the data show that the turbine itself is a reliable piece of equipment with a good
service record. Most of the problems documented are the result of problems with the turbine
controls and the mechanical overspeed trip mechanism, which apparently stem from three major
causes:

1. Originally designed as a continuous drive mechanism, with a slow start-up
sequence, the turbines are now used in stand-by service with normal start-up in
approximately 30 seconds. The turbines are normally run once a month during
pump in-service testing. The data indicate that this lack of operation can actually
be a major contributor to turbine degradation. Moisture trapped inside the turbine
and controls can cause damaging corrosion to the governor and the governor
valve.

2. Maintenance enhancements have been poorly implemented, resulting in
corrective maintenance instead of preventive/predictive maintenance.

3. Design changes by the manufacturer have been implemented with varying
degrees of success because of the change in service from the original design.

Recent improvements in maintenance practices and procedures, combined with a stabilization
of the design seem to indicate that this equipment can be a reliable component in safety systems.
AGING, CONDITION MONITORING, AND LOSS-OF-COOLANT ACCIDENT (LOCA) TESTS OF CLASS 1E ELECTRICAL CABLES: SUMMARY OF RESULTS

Mark J. Jacobus
Sandia National Laboratories
Albuquerque, NM 87185

Summary

This paper will summarize the results of aging, condition monitoring, and accident testing of various nuclear power plant cable products. Four sets of cables were aged under simultaneous thermal (−95°C) and radiation (0.10 kGy/hr) conditions. One set of cables was aged for 3 months, a second set was aged for 6 months, a third set was aged for 9 months, and a fourth set was not aged. A sequential accident consisting of high dose rate irradiation (6 kGy/hr) and high temperature steam was then performed on each set of cables. The results of the tests indicate that the feasibility of life extension of some popular cable products is promising. Mechanical measurements, primarily elongation, modulus, and density, were more effective than electrical measurements for monitoring age-related degradation.

The broad objectives of this experimental program were twofold:

a. to determine the life extension potential of popular cable products used in nuclear power plants and
b. to determine the potential of condition monitoring (CM) for residual life assessment.

A number of condition monitoring measurements were performed on the cables during aging, including insulation resistance (IR), polarization index (PI, the ratio of IR at two different times), capacitance and dissipation factor, elongation at rupture, tensile strength, modulus profiling, modulus tests using Franklin Research Center's indenter developed under EPRI funding, hardness testing of insulation and jacket materials, and bulk density. Selected data from these measurements that has not previously been published will be presented in the paper.

Throughout the accident simulations, the cables were normally powered at a nominal voltage of 110 Vdc with no current. Insulation resistance (leakage current) was monitored throughout the accident simulations. This provides a basis to compare the accident performance of unaged cables with the accident performance of cables aged to three different lifetimes.

Some preliminary conclusions from the tests are as follows:

a. Of the condition monitoring parameters tested, elongation at break tends to show the most correlation with amount of aging for the most cable types.

b. For the jacket materials and some of the insulation materials, hardness and indenter modulus both increased with aging, but they did not change consistently...
for some other insulation materials. Indenter modulus was clearly the more sensitive of the two techniques.

c. Density normally increased with aging for most of the insulation and jacket materials.

d. Although there were some exceptions, neither tensile strength nor any of the electrical measurements had any significant, consistent trend with aging.

e. The maximum differences between the accident insulation resistance of unaged cables and cables aged to the three different lifetimes was about two orders of magnitude. In most cases, insulation resistance during the accident was lower for cables that had greater amounts of aging.

f. The accident insulation resistance of single conductor cables removed from multiconductor cables was typically higher than the insulation resistance of the multiconductor cable, indicating that testing of only single conductor cables may not adequately represent multiconductor cables.

g. Over the range from 50-250 V, insulation resistance was largely independent of test voltage during both aging and accident testing.

h. During accident testing, most cables tested behaved in a consistent inverse temperature fashion, i.e., as the temperature was reduced, the insulation resistance consistently increased.

i. During the initial team transients, some cables had insulation resistances that fell well below the steady state value and then recovered. Except for this overshoot phenomenon, periodic measurements of IR would have been sufficient to indicate cable performance.
Effects of Aging on Calibration
and Response Time of
Nuclear Plant Pressure Transmitters

H. M. Hashemian
Analysis and Measurement Services Corporation
AMS 9111 Cross Park Drive
Knoxville, Tennessee 37923
(615) 691-1756

Abstract

This paper presents the key results of an experimental research project conducted for the Nuclear Regulatory Commission to quantify the effects of normal aging on static and dynamic performance of nuclear grade pressure, level, and flow transmitters (hereafter referred to as pressure transmitters). The project involved laboratory testing of representative pressure transmitters manufactured by Barton, Foxboro, Rosemount, and Tobar (or Veritrak) companies. These manufacturers provide the four most commonly used pressure transmitters in the safety systems of U. S. nuclear power plants. The transmitters were tested under normal aging conditions as opposed to accelerated aging, even though accelerated aging will be used in the last few months of the project to determine the weak links and failure modes of the transmitters.

The project has been performed in two phases. The Phase I project which was a six month feasibility study has been completed and the results published in NUREG/CR-5383. The Phase II project is still underway with the final report due in the fall of 1991.

The project has focused on the following areas:

1. Effects of aging on calibration stability.
2. Effects of aging on response time.
3. Study of individual components of pressure transmitters that are sensitive to aging degradation.
4. Sensing line blockages due to solidification of boron, formation of sludge, freezing, and other effects.
5. Search of LER and NPRDS databases for failures of safety-related pressure transmitters.
6. Oil loss syndrome in Rosemount pressure transmitters.

To date, it has been determined that the overall performance of pressure transmitters is only moderately affected by normal aging with the experimental results providing an objective
basis which shows that periodic response time and calibration tests performed once every fuel cycle is adequate for management of normal aging of nuclear grade pressure transmitters.

The search of the LER and NPRDS databases has not shown any systematic problems in nuclear plant pressure transmitters, nor has it revealed any performance degradation trends that can be used to establish a degradation rate. However, a notable number of reports were found in both databases which indicate that excessive drift, response time degradation, and sensing line blockages do occur in nuclear plant pressure sensing systems.

The oil loss syndrome in Rosemount transmitters has been shown to affect both the calibration and response time of the transmitters. It has been determined that the oil loss problem is detectable on-line by a combination of drift monitoring and dynamic testing using the noise analysis technique. Although the drift monitoring for on-line detection of oil loss has been said to be more effective than dynamic testing, oil loss cases in Rosemount transmitters have recently surfaced where the drift monitoring alone did not reveal the problem as good as the combination of drift monitoring and dynamic testing.

The project has revealed that sensing line blockages can result in significant degradation of response time of a pressure sensing channel and that the severity of the problem depends on the volumetric displacement of the sensing element in the pressure transmitter. More specifically, some transmitters, such as Barton, are significantly affected by any obstruction in the sensing lines and some transmitters, such as Rosemount, are moderately affected by sensing line blockages.
APPLICATIONS OF RESEARCH RESULTS
FROM
NPAR SERVICE WATER SYSTEM STUDIES

D. B. Jarrell and R. C. Stratton
Pacific Northwest Laboratory (PNL)
Richland, Washington

SUMMARY

Under the Nuclear Plant Aging Research (NPAR) program, the U.S. Nuclear Regulatory Commission (NRC) has invested considerable resources in understanding the mechanisms, mitigating the damage, and managing the results of nuclear power plant aging degradation. Many direct benefits have resulted from this program in terms of improved plant safety through upgraded regulatory guidance. The "Service Water System (SWS) Aging Degradation Assessment" (NUREG/CR-537g), produced by the Pacific Northwest Laboratory (PNL), contributed to this program. In this publication, it was clearly demonstrated that corrosion is the primary degradation factor in SWS component failures and outlined chemistry control as the most effective mitigation strategy for these failures. Two other benefits have resulted from the NPAR SWS task: 1) a contribution to the content of Generic Letter 89-13 (nuclear plant SWS performance requirements) and 2) the development of a systematic and complete root-cause analysis (RCA) methodology for use in the solution of SWS as well as other system component failures.

Less recognized, but also of significance, are the spin-off initiatives from the NRC's NPAR program investigations. One such spin-off resulting specifically from the NRC-sponsored SWS research is the continuing development of the RCA methodology to facilitate the computerized integration of the following:
1. component condition monitoring
2. fault diagnostics
3. failure root cause analysis.

The combination of these three elements into an integrated system provides a means to automate both the identification and the mitigation strategy for most age-related component degradation mechanisms.

The Decision Support system for Operations and Maintenance (DSOM) project being developed at PNL, funded by the U.S. Marine Corps, is implementing the NPAR-developed RCA methodology utilizing an artificial intelligence (AI) approach. By using model-based reasoning methods and an object-oriented schematic representation, DSOM provides a real-time interactive means to systematically investigate, understand, and auto-document the mitigation of age-related component degradation.

The DSOM software system captures component state parameter, degradation sensor, and machinery history data as well as detecting the plant system configuration. The truly unique aspect of this system is that it integrates

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1 Operated for the U.S. Department of Energy by Battelle Memorial Institute under contract DE-AC06-76RLO 1830.
the component parametric data, basic process physics, a hierarchical fault
diagnostic logic, and failure history knowledge to determine a more complete
specification of the component's condition than was previously possible. This
condition specification is then compared to degradation models in an effort to
allow the active degradation mechanism(s) to be identified. Based on the
monitored performance trends and their correspondence to the active
degradation model, an accurate, technically based estimate of equipment
service life can be projected.

The PNL DSOM is designed to demonstrate whether or not the performance
characteristics of an in-service component are within the specification limits
stated in its design criteria. In addition, the system will interactively
assist in diagnosing a real-time malfunction. This diagnosis is based on
monitored component parameters and the integration of the diagnostic
conclusion with the component surveillance data and failure record (machinery
history database) information. Not only does the system track the current
performance level of the component considered but it also computes and tracks
its degradation characteristics by trending test and failure history data into
its active component knowledge base. In this way, these data are factored
into any appraisal of the component's ability to function at or above its
design requirements until further testing is scheduled and conducted.

The architecture of the DSOM system uses a hierarchy of reasoning agents
to reach diagnostic conclusions. This hierarchy is based on the physics of
the processes involved: it is constructed to facilitate the fault reasoner in
recognizing, localizing, and characterizing existing degradation and fault
behaviors. Any off-normal component behavior is systematically compared to
an exhaustive set of behavioral models of the component-specific degradation
mechanisms. The objective of this analysis is to produce a correspondence
between modeled degradation behavior and actual component data to provide an
identification of the causal degradation mechanism.

The objectives of the PNL project are to develop, install and test the
functionality of the DSOM system for a fossil-fueled plant. The project
includes developing the specification for the instrumentation required to
detect degradation, the associated data acquisition system, the hardware and
software for a model-based artificial intelligence reasoner, and a
display/interface for plant personnel interaction with the decision support
system. This integrated system will serve as a tool for assisting operations
and maintenance personnel in performing the following:

- component operation and condition monitoring - meeting design
  specification requirements
- reliability-centered preventive maintenance - equipment inspection
  and maintenance based on previous failure history and component
  condition monitoring
- predictive maintenance - surveillance and performance data trending
to optimize and effectively schedule component maintenance,
  refurbishment or replacement.

10-12
SNUBBER AGING ASSESSMENT: RESULTS OF NPAR PHASE II IN-PLANT RESEARCH

D. E. Blahnik and E. V. Werry
Pacific Northwest Laboratory
Richland, Washington

D. P. Brown
Lake Engineering Company
Greenville, Rhode Island

SUMMARY

Snubbers are safety-related devices used to restrain undesirable dynamic loads at various piping and large equipment locations in nuclear power plants. Each snubber must accommodate a plant's normal thermal movements and be capable of restraining the maximum off-normal dynamic loads postulated for its specific location. Snubbers are subject to the effects of aging, and the factors that degrade their safety performance needed to be better understood. This paper describes the in-plant aging research conducted to enhance the understanding of snubber aging and its consequences. The research methodology, results, conclusions, and recommendations are described in the paper. Recommendations for monitoring the service-life of snubbers are among the recommendations.

Phase II of the snubber aging study was conducted by the Pacific Northwest Laboratory and its subcontractors, Lake Engineering Company and Wyle Laboratories. The principal approach was an in-plant snubber aging investigation, in cooperation with several nuclear power utilities. Two methods were used to obtain data for the research. The first method included interviews with plant maintenance and engineering staff. The second method involved analysis of plant operating data, including maintenance records and in-service testing and examination records. Plant selection was based on several factors, including availability of staff, plant procedures, snubber types and length of service, and plant types (BWR and PWR).

Eight sites (13 plants) were visited during a three-month period. Snubbers used at five of the sites were primarily mechanical; snubbers at the remaining three sites were primarily hydraulic. In addition to the site visits, over 70 telephone interviews were conducted with knowledgeable staff at nuclear power plants throughout the U.S. Hands-on snubber research was also conducted at Lake Engineering facilities. This work involved the disassembly, examination, and measurement of mating parts associated with hydraulic snubber seals.

Recent performance histories of snubbers at the sites visited were evaluated to understand how snubbers age and degrade and to determine their failure characteristics. By distinguishing between snubber failures related to aging and failures related to non-aging causes, we concluded that approximately half of all snubber failures may be attributed to service-related (aging) influences. The influences can be categorized according to
the environment (temperature, humidity, etc.), routine transients, and vibration. These categories and their relationships to snubber failures will be described in the paper.

Influences, such as heat, vibration, and moisture, can degrade the performance of mechanical snubbers by increasing drag and breakaway forces and by changing the activation acceleration thresholds. Data in one plant indicated an increasing trend in mechanical snubber drag force with service time. For hydraulic snubbers, high temperatures in isolated operating areas can rapidly degrade seals. Radiation probably contributes less significantly to aging than originally hypothesized. The research indicates that fluid leakage in hydraulic snubbers is commonly associated with leaking hydraulic fittings, and that most seal leaks are not directly attributable to normal environmental conditions. However, the research indicated that, for at least one BWR plant, the incidence of seal leakage was higher at elevated temperatures in the dry well than in other areas of the plant. This supports the assumption that seal degradation can be accelerated by exposure to higher than normal temperatures.

The following recommendations for service-life monitoring guidelines are offered based on the results of the in-plant research:

- The root causes of snubber failures and degradation should be determined. Degradation or failure due to nonservice-related influences should be identified and separated from the data base used to verify service life.

- Because plant operating environments may differ from design specifications, local area snubber environmental conditions, e.g., temperature, should be monitored.

- Snubbers subjected to severe environmental influences should be identified and managed on a case-by-case basis.

- Service-life for the general snubber population should be established by trending relevant degradation parameters.

- Augmented evaluation methods, such as hand stroking, are useful to identify potential snubber degradation, particularly degradation caused by dynamic load transients.

- Evaluation of test parameter time traces obtained during routine functional tests is useful to identify performance anomalies that may be indicators of snubber degradation.

- Service-life projections based on data from snubbers exposed to the actual plant operating environment is preferable to analytical service-life projections.
OVERVIEW OF HUMAN FACTORS
ISSUES ASSOCIATED WITH
ADVANCED PASSIVE REACTORS

Julius J. Persensky, Ph.D.
Office of Nuclear Regulatory Research
Nuclear Regulatory Commission

Nuclear power plant operators in current plants do just that, they operate the plant. They monitor the state of the process, observe trends, react to system changes and take action to maintain the plant in a safe condition. In order to accomplish these activities, they (1) rely on information available to them from the various instruments and displays in the control room, (2) use hard copy procedures to direct their actions and (3) perform their control actions through the various switches, keyboards and knobs available. Proposals from vendors and our observations of activities in other countries indicate that the control room for advanced passive reactors is likely to be quite different from control rooms one would see in current power plants. These differences will go beyond the immediate appearances, e.g., smaller, more cockpit-like, to issues of automation, function-allocation, the changing role of the operators and their training and qualifications, and software verification and validation.

The papers which follow in this session describe some of the research which is now underway to address selected issues associated with advanced instrumentation and controls. These projects primarily address the development of technical bases for guidelines which would be used to review advanced designs. Mr. Beltracchi will discuss a workshop held this past Spring on methods to bring together the vast body of research results attained by Halden over the years into a form more amenable to review guideline development. Dr. Moray will report on research designed to test a method for comparing various displays to an operator's mental model of a system or process. A project in which we are cooperating with EPRI, the verification and validation of expert systems, will be described by our contractors from SAIC. And finally, Mr. Wachtel will discuss the development of human factors design review guidelines which will supplement existing guidance in NUREG-0700.

Traditionally, human factors research focuses on the human side of the human-system interface surface, i.e., displays, controls, procedures, the work environment, and the personnel subsystem as exemplified by the papers from Dr. Moray and Mr. Wachtel. However, because of the close interaction with digital instrumentation and control systems, our current human factors research program also encompasses research on the hardware and software aspects of the interface as is evident in Mr. Beltracchi's and the SAIC papers.

Over the last ten years the NRC's human factors regulatory and research focus has been on conventional control rooms and the systems which support current operators. While it is conceivable that future systems could be designed to operate safely without involvement by humans, it is more likely that, for the
foreseeable future, people will be needed to design, build, operate and maintain them. Therefore, one goal of the designers of these future plants should be to optimize the partnership among people, machines, and software. This would be accomplished by recognizing the inherent strengths and weaknesses of each and assigning the functions which each performs best to that element, in human factors parlance "function allocation." Since all U.S. nuclear power plants now operating were either operating or under construction ten years ago, when human factors concerns were first recognized by the NRC and industry, there was little opportunity to focus on the function allocation question.

With the potential for new plants and new designs, the importance of this allocation of functions must be considered from the beginning. We must recognize the importance of the role of humans from the initial design through fabrication and construction as well as testing, operation, and maintenance. This recognition must not be limited to consideration of the person-system interface but also must include the numbers, qualifications, and training of staff, use of job performance and decision aides made possible through digital systems, job and organizational structure and continuing performance feedback.

From a regulatory perspective new review guidance is needed for: evaluating advanced design interfaces, both inside and outside of the control room; establishing minimum staffing and qualifications; categorization of computer application based on safety significance; evaluating the effectiveness of alarm optimization techniques; evaluating verification and validation of artificial intelligence based systems; and establishing criteria for assessing technological change on human performance.

To meet these regulatory needs for review guidance for advanced designs, the Human Factors Branch in the Office of Nuclear Regulatory Research has recently completed three survey/workshop projects, has ten ongoing projects, and plans to initiate fifteen new efforts over the next year which will address safety concerns on both sides of the human-system interface surface. This research paper will discuss some of these planned efforts and how they relate to the human factors issues and regulatory user needs.
A Halden Reactor Project Workshop on "Understanding Advanced Instrumentation and Controls Issues" was held in Halden, Norway, during June 17-18, 1991. The objectives of the workshop were to (1) identify and prioritize the types of technical information that the Halden Project can produce to facilitate the development of man-machine interface guidelines and (2) to identify methods to effectively integrate and disseminate this information to signatory organizations. As a member of the Halden Reactor Project, the U.S. Nuclear Regulatory Commission (NRC) requested the workshop. This request resulted from the NRC's needs for human factors guidelines for the evaluation of advanced instrumentation and controls.

The Halden Reactor Project is a cooperative agreement among several countries belonging to the Organization for Economic Cooperation and Development (OECD). The United States began its association with the Halden Project in 1958 through the Atomic Energy Commission. The project's activities are centered at the Halden heavy-water reactor and its associated man-machine laboratory in Halden, Norway. The research program conducted at Halden consists of studies on fuel performance and computer-based man-machine interfaces.

The experience gained at the Halden Project in the design, implementation, and evaluation of computer-based systems can be very useful for the development of man-machine interface guidelines. However, it is difficult to convey this experience in technical reports and to anticipate the specific needs of different vendor, utility, and licensing organizations represented by the members of the Halden Project. There is a particular need for man-machine interface guidelines that systematically address the operator's cognitive activities such as problem solving, diagnosis, decision making, and planning. The workshop served as a forum to discuss these issues. The workshop of June 17-18, 1991, served as a first step in responding to these needs.

The workshop consisted of four parts. In the first part, signatory members of the Halden Project gave short talks identifying their needs for guidelines, stating specific requests of the Halden Project, and providing comments on the programs and experiments at Halden. The speakers in this session were from five different countries; 1) United Kingdom, 2) Sweden, 3) Spain, 4) United States, and 5) Japan.

The second part of the workshop consisted of a demonstration of the Integrated Surveillance and Control System (ISACS) at the Halden Man-Machine Laboratory. ISACS is computer-driven integrated operator interface and a limited prototype for an advanced control room and will be discussed.
The third part of the workshop consisted of working group sessions. In order to effectively meet the goals of the workshop, three working groups were formed to address the issues. The groups were:

- Working Group 1: Guidelines for the Design of Man-Machine Systems
- Working Group 2: Guidelines for the Evaluation of Man-Machine Systems
- Working Group 3: ISACS Evaluation Program

The fourth and last part of the workshop consisted of oral reports from the Chairman of each working group. Each working group Chairman identified the issues discussed and presented the recommendations made by the group. These recommendations consisted of a schema for classifying the degree of automation within a man-machine interface, the need for research on how to certify man-machine interfaces, and a human factors review of the ISACS interface and correction of discrepancies found prior to evaluation experiments. There was also a clear need expressed for the identification and classification of cost effective methods and measures for evaluating man-machine interfaces.
Testing the DeGroot Recall Paradigm to Evaluate Displays for Operating Personnel.

Neville Moray  
Department of Mechanical and Industrial Engineering  
University of Illinois at Urbana-Champaign

Barclay G. Jones  
Department of Nuclear Engineering  
University of Illinois at Urbana-Champaign

Jens Rasmussen  
Risø National Laboratory  
Denmark

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Contract Monitor L. Beltracchi

Abstract

In this paper we report the results of experiments based on deGroot's work to assess the value of memory tests for measuring the quality of displays and the level of expertise of operators.

Three kinds of display and people with three levels of expertise were included in the experiments. The displays were computer generated versions of traditional analog meters, traditional analog meters supplemented by a dynamic graphic representing the relation between temperature and pressure in some subsystems, and a dynamic graphic representing the underlying thermodynamics of power generation using the Rankine Cycle. The levels of expertise were represented by undergraduates with one semester of thermodynamics ("novices"), graduate students of thermodynamics and nuclear engineering, and professional nuclear power plant operators ("experts").

Each group watched a set of transients presented on the displays, using data generated by a high fidelity NPP training simulator, and were then asked three kinds of questions. The first measured their ability to recall the exact values of system state variables. The second measured
their ability to recall what qualitative states the system had entered during the transient. The third measured their ability to diagnose the nature of the transient.

The results of the experiments will be reported in relation to the possible use of memory tests to evaluate displays and the interaction of the quality of displays with the level of expertise of operators.
Development of Guidelines for the Validation and Verification of Expert Systems

Paper presented at 18th Water Reactor Safety Information Meeting

Lance Miller
Elizabeth Groundwater
Steve Mirsky

Science Applications International Corporation
1710 Goodridge Dr. McLean, VA 22102

ABSTRACT

Science Applications International has a joint contract with the NRC and the Electric Power Research Institute to develop guidelines for the Verification and Validation of Expert Systems for use in the nuclear power industry. This paper reports on the overall goals and tasks of this ongoing multi-year contract. We highlight the key findings of the two task surveys completed so far: survey of the applicability of conventional V&V techniques to expert systems, and survey of activities and techniques actually used for V&V of expert systems. We conclude with discussion of the likely directions and emphases for the remaining work.
Advanced control room (ACR) concepts are being developed and refined in the commercial nuclear industry as part of future reactor designs. These ACRs will utilize advanced human-system interface (HSI) technologies which may have significant implications for plant safety in that they may affect: (1) the operators' overall role (function) in the system; (2) the methods by which operators receive information about system status; (3) the ways in which the operators interact with the system; and (4) the requirements on operators to understand and supervise an increasingly complex system. The U.S. Nuclear Regulatory Commission (NRC) reviews control room designs to ensure that they incorporate good human factors engineering principles so as to support operator performance and reliability necessary to protect public health and safety. The principal guidance available to the NRC (NUREG-0700) was developed more than ten years ago and does not address new technologies. Accordingly, the guidance must be updated. This paper discusses the development of an NRC Advanced Control Room Design Review Guideline (Guideline).

Many issues impact the development of guidelines, including: (1) the "review environment," e.g. the diversity of plant types, recent efforts to standardize plant designs, and the range of active to passive safety system designs; (2) the diversity of technologies being developed for ACRs, as reflected by the differences in control room design approaches that might be characterized as "hybrid," "advanced" and "intelligent" CRs; (3) human factors issues associated with advanced technology, e.g. allocation of function and automation; and (4) the state-of-the-art of human factors guidelines for advanced HSIs. These four issues have led to the development of a top-down approach to the planned review of ACRs, which, when integrated, should permit the tracking of a CR design from initial conception through design implementation. To undertake such reviews, Guidelines are needed in two general categories - "Design Process Review" and "Design Implementation Review" - and each of these can be further divided into two subcategories.

"Design process review" guidelines can be used to evaluate designs prior to their final implementation, using commonly accepted human engineering practices. These guidelines can be divided into two modules - planning and analysis. Planning refers to elements such as: (1) the organization of the human factors team and its role in the design process; (2) the human factors program plan; and (3) the specification of high level design goals and objectives. Analysis refers to elements of the
plan such as the systems and task analysis, function allocation, specification of performance requirements, trade-off studies, and tests/evaluations at the system/function or part-task level.

"Design implementation" review guidelines can also be divided into two modules: human factors engineering (HFE) review and dynamic performance evaluation. The HFE review refers to the evaluation of control room interfaces according to accepted human factors design guidelines. Dynamic evaluation (a full-mission, real-time, person-in-the-loop process using a prototype or simulator) assures that the final, integrated design meets its design performance goals.

The guidelines development effort includes a "Guideline Development" task and three related support tasks: Electronic Document Development, Test and Evaluation, and New Guideline Development. The Guideline is being developed through an iterative process requiring several separate drafts. The first (Revision 0) has been completed. It contains the HFE guidelines. Revision 1 will contain the guidelines for design process and dynamic evaluation. Revision 2 will be prepared after initial usability testing; and Revision 3 will follow independent peer review. The objectives of these tests and reviews are to evaluate the scope and content of the Guideline (i.e. its adequacy for the review of advanced control room technology), and to ensure the usability of an electronic version of the guideline in terms of presentation, functionality, and user interface.

The guideline development effort began with an identification of existing human factors guidelines relevant to advanced HSIs. Through a review of the human factors literature and contact with organizations which sponsor such research, approximately 50 previous guideline efforts were identified. These were prioritized based upon their documented validity. Those guidelines with the highest ratings were codified as "primary sources" for the initial set of guidelines to be incorporated in the NRC document.

Revision 0 is organized according to a structure similar to that of documents that focus heavily on HCI, but modified to accommodate HSI trends in advanced control rooms. The seven major sections of the draft document include: Information Display, Operator Input and Control, Alarms, Operator Aids, Interpersonal Communication, Information Protection, and Workstation Design.

As indicated above, the Guideline is being developed in electronic as well as hardcopy form to facilitate guideline access and review, editing, extrapolation of relevant guidelines for a specific review, and incorporation of new guidelines as they become available.
MOST LIKELY VESSEl LOWER HEAD FAILURE LOCATION DURING SEVERE ACCIDENT CONDITIONS

J. L. Rempe, S. A. Chavez, S. D. Snow, G. L. Thinnes, and C. M. Allison
Idaho National Engineering Laboratory
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83415

ABSTRACT

The U.S. Nuclear Regulatory Commission is sponsoring a lower vessel head research program to investigate plausible modes of reactor vessel failure to determine: (a) which modes have the greatest likelihood of occurrence during a severe accident and (b) the range of core debris and accident conditions that lead to these failure. All major types of U.S. light water reactor vessels are being considered, and both high- and low-pressure conditions are being addressed for each reactor type. The research program includes analytical and finite element calculations. In addition, high temperature creep and tensile data for predicting vessel structural response were obtained. Calculational results used to predict which failure location is more likely in a particular reactor design during a severe accident are described within this paper.

Detailed analyses are being performed to investigate the relative likelihood of a BWR penetration and vessel to fail during a wide range of severe accident conditions. The analyses include applying a numerical model to obtain the penetration and vessel thermal response and applying an analytical model to investigate the relative likelihood of tube rupture and global vessel failure. Sensitivity studies consider the impact of assumptions related to debris composition, debris porosity, corium decay heat, vessel coolant mass, heat removal from the vessel, melt relocation time, and melt relocation distance on vessel and penetration response. In addition, analytically developed failure maps, which were developed in terms of dimensionless groups, are applied to extrapolate numerically-obtained results to geometries and materials occurring in PWR penetration/vessel configurations and a wider range of debris conditions.
COOLING OF CORE DEBRIS WITHIN THE REACTOR VESSEL LOWER HEAD

Robert E. Henry, James P. Burelbach, Robert J. Hammersley and Christopher E. Henry
Fauske & Associates, Inc.
16W070 West 83rd Street
Burr Ridge, Illinois 60521

and

George T. Klopp
Commonwealth Edison Company
1400 Opus Place, Suite 400
Downers Grove, Illinois 60515

SUMMARY

Under severe accident conditions, the most crucial action for recovery from the accident state is to cool the core debris and prevent or terminate attack on the remaining fission product barriers. One means of preventing attack on the containment structures is to retain the core debris within the reactor vessel. The TMI-2 accident demonstrated that this could be accomplished by water resident within the reactor vessel combined with injection on a continual basis to quench the debris and remove decay heat over the long term. Some accident situations could result in the transport of molten core debris to the lower plenum, as occurred in TMI-2, the boiloff of water in the lower plenum, and an inability to add water to the reactor coolant system (RCS). Even in this extreme set of circumstances, sufficient cooling may be available to prevent failure of the reactor pressure vessel (RPV) lower head and thereby retain the core debris within the vessel.

Containment configurations like Zion would result in substantial accumulation of water around the lower parts of the reactor vessel for most
accident sequences. For some PWR containments, there could be substantial water accumulation around the reactor vessel and the hot and cold legs. If this water could directly contact the carbon steel vessel surface and RCS piping, substantial energy could be removed from the primary system and in particular the RPV lower head.

Experiments discussed in this paper, which were performed in support of the Commonwealth Edison IPE and Accident Management Programs, demonstrate nucleate boiling heat removal rates from the outer surface of a simulated RPV lower head surrounded by typical reflective insulation used in nuclear power plants. Therefore, the heat flux is limited by thermal conduction through the carbon steel head. Experiments were performed in which the reactor vessel lower head was simulated with a 0.32 m (12.75") outer diameter pipe cap. Wall thicknesses of 1.75 cm (0.688") and 3.3 cm (1.312") were used to provide substantially different heat fluxes to the outer surface. The heat source was molten iron thermite, at a temperature of about 2400K, poured onto the dry inner surface of the lower head. Water provided cooling on the outer surface. Both uninsulated and insulated configurations were investigated. The measured heat fluxes were essentially the same for these two different cases. This clearly demonstrates that the water flow rate through the insulation is sufficient to supply cooling water to the RPV outer surface under such accident conditions. In addition, the measured heat fluxes are well in excess of those which can be attributed to film boiling. Hence, the vessel outer surface was cooled by nucleate boiling during the entire transient.
ON THE PREDICTION OF STEAM EXPLOSION ENERGETICS

T.G. Theofanous, S. Angelini, R. Buckles, X. Chen & W. Yuen
Center for Risk Studies and Safety
University of California, Santa Barbara

The primary purpose of this paper is to present recent results on certain key experimental aspects and mechanisms of steam explosions. In addition, sample results of integral calculations, incorporating the recent experimental results, are presented. The scope of this work is focused to the premixing and propagation phases of energetic explosions.

On premixing, we present experimental results in scaled (1/8-scale) geometries of the lower plenum of a PWR. The computer code PM-ALPHA (Amarasooriya & Theofanous, 1991) was used as the scaling tool to ensure the experiment is run under similar water depletion regimes as that predicted for the reactor. This water depletion (from the mixing zone) is the key physical mechanism that limits the energetics of such large scale explosions, and to a large extent these experiments are focused on it. Thus, instead of a molten material, hot solid particles are used in these experiments. Local, instantaneous liquid fraction measurements are made, within the three-phase mixing zone, by means of a new instrument developed specifically for this purpose.

For calculating escalation and propagation, the key ingredient is the fuel fragmentation kinetics. These kinetics are explored experimentally in this work, for the first time at conditions that simulate a propagating explosion. This is accomplished in a hydrodynamic shock tube where molten, superheated metallic drops are subjected to shock waves of magnitudes and shapes similar to an explosion front. Fragmentation is measured by radiography using high energy flash x-rays. Also, the degree of chemical reaction is deduced from the collected debris. The data are analyzed and thus incorporated in our three-fluid propagation code ESCPROSE (Medhekar et al., 1991).

Finally, illustrative integral calculations are presented for large scale pours in the lower plenum. The premixing transient is obtained from PM-ALPHA and various explosions are triggered, with ESCPROSE, at different times during the premixing transient. In addition, the maximum energetic potential transient is presented on the basis of the premixing transient and ideally efficient, locally, conversions. These results provide further support to the quantification of α-mode failure in Theofanous et al. (1987).

REFERENCES


RECENT DEVELOPMENT AND RESULTS FROM
SEVERE ACCIDENT RESEARCH IN JAPAN

K. Soda, J. Sugimoto, N. Yamano, K. Shiba
Department of Fuel Safety Research
Japan Atomic Energy Research Institute
Tokai-mura, Ibaraki-ken, Japan 319-11

SUMMARY

In recent years in Japan, severe accident research has been placed
among the top-prioritized reactor safety researches since experimental and
analytical investigations of a severe accident are needed to understand the
safety margin of a nuclear power plant and to further improve accident
management measures. It is thus indispensable to establish the knowledge base
in the course of performing severe accident research and to utilize it for
quantifying and reducing risks of a nuclear power plant.

The Government's Annual Plan on Reactor Safety Research describes
purposes of such severe accident research as;
* To identify phenomena associated with a severe accident,
* To develop analytical tools for source term analysis,
* To estimate a risk and safety margin of nuclear power plants, and
* To evaluate measures to prevent and mitigate severe accident by
design and/or accident management.

Severe accident research in Japan is pursued primarily at the Japan
Atomic Energy Research Institute (JAERI) experimentally and analytically in
accordance with the Annual Research Plan. In addition, demonstration tests at
the Nuclear Power Engineering Test Center (NUPEC) are conducted with an
emphasis on quantification of the safety margin of a nuclear power plant in
conditions beyond the design basis. Industries are making progress in
quantifying risks of nuclear power plants in Japan.

The summary of experimental and analytical investigations of severe
accidents in Japan is briefly given as follows.

Experimental Investigation

Experimental investigation of phenomena associated with a severe
accident is carried out to characterize accident progression from heatup of
fuels till containment failure. Fuel degradation is studied by using the Nuclear
Safety Research Reactor (NSRR), a unique research reactor which is capable of
simulating a transient of a reactivity initiated accident (RIA). Interaction of
component materials with fuel is examined in detail with emphasis on an
interaction between control rod materials with spacer grids. Debris from the
TMI-2 lower vessel will be examined in 1991.

Ex-vessel phenomena such as molten core and concrete interaction
(MCCI), molten core and coolant interaction, leakage through containment
penetrations, and an overall aerosol behavior in a containment are to be studied at the ALPHA (Assessment of Loads and Performance of a Containment in Hypothetical Accidents) facility which was fabricated at JAERI in 1990. The ALPHA facility is the major out-of-reactor experimental facility for severe accident research in Japan. Experiments to be performed in 1991 will be focused on coolability of molten core and occurrence of steam explosion during molten core and coolant interaction.

Behavior of fission products such as release from fuel at high temperature and in reduced environment and formation of organic iodide in radiation field are still recognized as a source of large uncertainties existing in source term evaluation. Experimental facility for fission product release from a fuel is now in designing stage and the facility will become operation in late 1993 at JAERI. Fission products retention in a water pool by pool scrubbing are investigated in which experiment conditions are at high pressure and temperature. Laboratory scale experiments and theoretical analyses of fission product chemistry are in progress at several universities funded by JAERI as a part of severe accident research program.

Analytical Investigation

Probabilistic Safety Assessments (PSAs) of selected reference plants have been performed by JAERI, the Japan Institute of Nuclear Safety (INS) of NUPEC and industries. Results showed much lower core damage frequency (CDF) than previously reported CDFs for similar plants. Comparison among the codes used in the PSAs identified the areas of uncertainties in severe accident analysis.

Code development activities continues at JAERI for THALES-2 code which is an integrated system code for source term analysis. Code comparison of THALES-2 with MELCOR has initiated in 1991. Application of detailed and mechanistic codes, SCDAP/RELAP5 and CONTAIN, to experiment analyses has made progress by participating international standard problem exercises of OECD. Containment aerosol analysis code, REMOVAL, has been further improved by implementing new models in. Accident management strategies have been investigated to supplement the USNRC's accident management analyses.

International Research Collaboration

Issues on severe accident are considered not to be unique problem to one country, but common to all countries utilizing nuclear energy. This inevitably necessitates to support international research collaboration. Development and achievement made in Japan of severe accident research should also be shared by nuclear communities through international research collaboration.

The intention of this paper is to provide achievements of severe accident research in Japan to the international communities for further enhancement of safety of nuclear power plants.
The work performed at IPSN concerning accident studies on nuclear installations is focused on the characterization of accidental sequences with three major aims:

- prevention;
- mitigation;
- organisation of counter-measures.

As criteria to optimize all efforts made to improve nuclear safety, the radioactive dispersal in the environment must be quantified as function of internal and external radioactive products transfers.

During the short-term phase of the accident, potential radioactive releases can be evaluated by the realistic code system ESCADRE; this system is validated by numerous analytical studies related to containment and fission product behaviour; it will be further qualified by the results of the global experiments performed in the PHEBUS FP facility at IPSN.

During an hypothetical crisis, IPSN, as technical support to the french safety authorities, would predict the evolution of the potential source term by the crisis system named SESAME, designed to quantify the source-term by use of available informations and measurements. From the utility side, operational means include design basis and ultimate measures to cope with the accident and application of the "Internal Emergency Plan".

Given the potential source-term from the installation, the radioactivity dispersal in the environment can then be predicted both by detailed codes and crisis tools. The CONRAD system is operational to predict the dispersal itself, while the CART project will produce a data base of relevant parameters for the countries surrounding nuclear sites.

These results are essential to determine the adequate counter-measures for the protection of the population, as they are planned in the so-called "Particular Intervention Plans".

The prediction of environmental consequences during a crisis would also be a guide line to elaborate the measurement strategy for the impact of releases in the environment, both for accidental and post-accidental phases. Concerning this latter phase, rehabilitation of contaminated environment is the
purpose of the "Post-Accidental action Plan". The technical actions to be undertaken are based on the results of predictions and measurements; their efficiency is studied by the experimental program named RESSAC, addressing soil and plant radionucleide transfers and contamination removal techniques.
The SCDAP/RELAP5/MOD3 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).

SCDAP/RELAP5/MOD3, created in January, 1991, is the result of merging RELAP5/MOD3 with SCDAP and TRAP-MELT models from SCDAP/RELAP5/MOD2.5. The RELAP5 models calculate the overall RCS thermal-hydraulics, control system interactions, reactor kinetics, and the transport of noncondensible gases, fission products, and aerosols. The SCDAP models calculate the damage progression in the core structures, the formation, heatup, and melting of debris, and the creep rupture failure of the lower head and other RCS structures. The TRAP-MELT models calculate the deposition of fission products upon aerosols or structural surfaces; the formation, growth, or deposition of aerosols; and the evaporation of species from surfaces.

The systematic assessment of modeling uncertainties in SCDAP/RELAP5 code is currently underway. This assessment includes (a) the evaluation of code-to-data comparisons using stand-alone SCDAP and SCDAP/RELAP5/MOD3, (b) the estimation of modeling and experimental uncertainties, and (c) the determination of the influence of those uncertainties on predicted severe accident behavior.

The evaluation of code-to-data comparisons using SCDAP and SCDAP/RELAP5 indicated that the calculations performed with SCDAP/RELAP5, and to a lesser extent SCDAP, described the important features of each experiment. However, the assessment identified several important modeling improvements, incorporated during successive releases of SCDAP and SCDAP/RELAP5, that considerably improved the agreement between calculation and experiment. These included the merger of SCDAP with RELAP5 and the addition of the new models for double sided oxidation, fuel dissolution, and axial heat transfer.

The overall code-to-data comparisons indicated that estimated variation between calculated and measured results was as follows. The thermal response, including variations in timing as well as magnitude, could typically be predicted within ±20 % with a few outliers in the ±40 % range. The ballooning and rupture could typically be predicted to a few percent. The hydrogen production had the worst overall agreement, particularly during bundle reflood, with a variation up to a factor of two.

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Sensitivity studies using SCDAP/RELAP5 indicated that the variations between SCDAP/RELAP5 calculations and experiments were due equally to uncertainties in (a) experimental conditions or results and (b) modeling important processes. Uncertainties in radial heat losses, power, flow conditions, hydrogen production measurements, and peak temperature estimates were dominant contributors for experimental conditions or results. Dominant modeling uncertainties were the initial relocation of liquefied fuel rod material, flow diversions due to changes in geometry, multi-dimensional flow patterns in the upper plenum region, and oxidation once the initial bundle geometry was lost. Variations in predicted and actual thermal-hydraulic response of the experimental systems - (a) liquid level and dryout times for temperatures below 1000 K, (b) local flow perturbations due to cross flows from outer assemblies or flow diversions from damaged bundles, and (c) radial heat losses, due to both experimental and modeling uncertainties had a dominant influence on the overall variation between experiment and calculation. User guidelines are being developed to minimize the influence of these uncertainties in modeling the thermal hydraulic features of these facilities.

Six specific damage progression model deficiencies were identified - (a) influence of ballooning upon flow and subsequent heatup, (b) oxidation of the inside of unpressurized fuel rod cladding, (c) the oxidation of relocating material or material that has formed a cohesive blockage, (d) additional hydrogen during reflood, (e) the porosity of frozen melt and the relocation of ceramic fuel rod material, and (f) the interaction occurring between bundle materials and complex flow of rivulets and droplets. A model development effort to resolve these deficiencies is currently underway.
One of the most important phenomenological issues in the progression of severe accidents after the reactor vessel has failed is whether or not the plant can be brought to a stable condition and the threat to containment integrity, whether by basement penetration or by containment pressurization, is avoided. The most commonly available mechanism for removing heat from discharged melt in LWR containments is water addition and the DOE, industry, and the NRC are all now working to develop and evaluate design criteria to address core debris coolability by water pools. One NRC element of this research is the WETCOR experimental program being performed at Sandia National Laboratories. These tests are intended to compliment and augment the ACE/MACE program sponsored by EPRI. Technically, the NRC approach will differ from the basic approach in the MACE tests by including heating of the experiment perimeter to reduce crust attachment and support. This is accomplished by inductively heating a 35cm diameter tungsten annulus which is filled with molten oxide mixtures of Al2O3, ZrO2, CaO, and SiO2 at temperatures of 2000-2400K and then flowing subcooled water onto the melt. The WETCOR tests are also designed to answer two additional questions: These are: (1) Is oxidic debris more or less coolable than metallic debris? (2) What are the limits of this coolability in terms of the debris depth, the debris power, and the debris composition? The WETCOR 1 test is being performed using a different oxidic debris type and under different boundary conditions than the MACE tests in order to focus on question number one. The remaining WETCOR tests will focus on conditions which will bound question number two.

The results from the initial tests performed under the WETCOR program will be reviewed and compared to the initial MACE program results. Data comparisons will include heat transfer to the overlying water, debris temperature histories, ablation rates, gas release rates, and the effects of water on aerosol production.
PARAMETER EFFECTS ON MOLTEN DEBRIS SPREADING AND COOLABILITY

F. J. Moody
K. M. Fruth

GE Nuclear Energy

The spreading, cooling, and freezing of molten core debris on a horizontal surface during a postulated severe accident are important considerations which influence the containment thermal response. Coolability of flowing debris is largely determined by its spreading configuration, as described by its time-dependent surface area and depth. If freezing occurs at the frontal edge of flowing debris, it may be sufficiently immobilized to prevent its contacting exposed containment boundaries. However, local mounding or piling up of debris could reach a depth at which decay and chemical heating cannot be removed by available cooling mechanisms, resulting in progressive, localized concrete degradation.

An important aspect of molten debris spreading is its pattern as it emerges from the pedestal doorway in a Mark I containment. Available programs for tracking the spread of flowing debris are based on an assumed spreading angle. This study employs a special formulation with the method of characteristics, which provides an unsteady spreading pattern prior to arrival at a containment boundary. It was found that the spreading configuration depends on the volume flow rate, doorway width, and debris surface tension. Additional analyses provide simplified formulations to estimate: how far molten debris can spread before immobilization by freezing and subsequent mounding; the effect of gaseous void fraction; and the effect of metal/oxide stratification on the vertical temperature profile.
Analyses of Corium Spreading in Mark I Containment Geometry

by

J. J. Sienicki and C. C. Chu
Argonne National Laboratory
Argonne, IL 60439

Under NRC sponsorship, analyses have been performed of melt spreading and containment shell heatup following postulated reactor vessel failure in the Mark I system. The overall objectives of this work are to lend support to confirmation of the conservative assumptions made in the probabilistic assessment of Theofanous et. al. in NUREG/CR-5423 (1989), extend the results from that study, and thereby assist in resolution of the Mark I shell vulnerability issue. The approach consists mainly of calculations using the MELTSPREAD-I computer code (Farmer, Sienicki, and Spencer, 1990) developed at Argonne under EPRI sponsorship as well as other supporting analyses. MELTSPREAD-I models fluid dynamic and heat transfer processes involved in the transient spreading of core and structural materials over concrete or steel and submerged beneath water as well as the heatup of a structural boundary such as the containment steel shell. The mechanistic MELTSPREAD-I calculations thus complement the simpler treatment of melt spreading incorporated in NUREG/CR-5423.

Consistent with the approach followed in NUREG/CR-5423, the MELTSPREAD-I analyses treat both Scenarios I and II from that document intended to encompass the two general modes of melt release behavior predicted by different in-vessel accident progression codes. In particular, melt release conditions for Scenarios I and II determined from NUREG/CR-5423 are also assumed as reference melt arrival conditions in the present investigation. As in NUREG/CR-5423, the cases of both wet and dry containments with water present on and absent from the drywell floor, respectively, are also considered. A number of calculations examine the sensitivities of the melt spreading and shell heatup behavior to current uncertainties and conservatisms inherent in Mark I melt spreading analyses including: variations in the conditions of melt released from the reactor vessel as predicted by various in-vessel analysis codes currently used in the U. S., the effects of melt splashing, freezing, and retention within the below-vessel principally CRD-related steel structure as well as melt breakup and quenching in water during relocation from the vessel to the pedestal floor; and uncertainties in the modeling of basic spreading related phenomena and melt-to-shell heat transfer.

To provide a basis for the variation of melt release conditions, a compilation was assembled of available relevant results from Oak Ridge model as well as MAAP, APRIL, and MELCOR code calculations. An analysis was subsequently performed of the interactions that melt undergoes with the below-vessel structure and water during the process of relocating from the reactor vessel to the pedestal floor. The analysis results show that if a melt jet released from the vessel is intercepted by a structural member, then melt splashing and freezing upon the structure will occur over a portion of the initial release phase until meltthrough of the structural member occurs. However, only a minor portion of the melt mass is calculated to be retained in the structure in Scenario I and a negligible part in Scenario II. In contrast, melt breakup and quenching in water residing on the pedestal floor is predicted to significantly modify the melt
arrival conditions on the pedestal floor from the original melt release conditions. In particular, a significant fraction of melt either dripping as drops from the structure or entering the water in a coherent jet mode is calculated to break up into droplets that freeze while settling through the water. Thus, a significant fraction of the melt released as liquid in Scenarios I and II is predicted to collect upon the floor as a liquid-solid slurry with a lower temperature. The portion of the melt that does not break up and thus impinges upon the sump covers as a molten jet is calculated to melt through the sump cover plates such that the sump volume fills up with melt during the initial release phase of both scenarios. The predicted melt arrival conditions were input to MELTSPREAD-I to examine the effects of the below-vessel melt-structure and melt-water interactions upon the subsequent spreading behavior. Results of the various MELTSPREAD-I calculations will be presented in the full-length paper.

References


ACE Program Phases C & D: 
Molten Corium Concrete Interaction (MCCI) Experiments 
and 
Corium Melt Coolability Experiments (MACE) 
by 
B.R. Sehgal 
Electric Power Research Institute 
and 
B.W. Spencer, D.H. Thompson, J.K. Fink and M.T. Farmer 
Argonne National Laboratory 

A series of experiments are being performed at Argonne National Laboratory (ANL) investigating the interaction of molten core material with concrete and its coolability with water. The work is supported by the ACE (Advanced Containment Experiments) International Consortium. This consortium consists of 19 countries; with the U.S. participation shared by NRC, DOE, EPRI and Westinghouse. The project is managed by EPRI.

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

The approach employed for the MCCI experiments is to use prototypic materials, i.e., UO₂, ZrO₂, Zr, stainless steel, inactive fission products and representative concretes in a furnace; with tungsten electrodes supplying sustained internal heat generation in the melt to reach temperatures of approximately 2500K. The physical size of the corium and concrete interaction zone is 50 cm x 50 cm and approximately 300 kg of molten corium material, is used.

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

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

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

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

A fundamental question is whether a thick stable crust is formed, which will substantially reduce the heat transfer rate from the melt to the water. The stability of the crust formed is scale-dependent, and the heat transfer rate and the long-term coolability depend on crust cracking allowing melt-water contact. A supporting project at University of Wisconsin is performing crust scaling analysis to determine and extend the applicability of the data obtained in the MACE experiments to reactor situations. Another supporting project at University of California, Santa Barbara, is investigating the conditions for formation of crust and its stability and morphology.

A scoping MACE test was performed in 1989. That test employed approximately 130 kg of corium melt interacting with a 30 cm x 30 cm block of concrete and cooled by water. A stable crust formed in that test and was supported by the electrodes, which had moved inwards from their original position. The heat transfer from melt to water was substantially reduced due to the crust. The MACE test design is being changed to preclude support by electrodes. The next two MACE tests will employ about 450 kg corium melt interacting with a 50 cm x 50 cm block of limestone-common sand concrete and siliceous concrete, respectively. These two tests are scheduled for completion in 1991. Further tests are planned in order to cover the other parameters which affect melt coolability. Pre-test analyses of the MACE tests are being performed by analysts at several member organizations of the ACE International Consortium.
The Surtsey Test Facility at Sandia National Laboratories (SNL) is used to perform scaled experiments for the Nuclear Regulatory Commission that simulate hypothetical 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 1:10 linear scale model of a reactor cavity. Debris is entrained by the steam blowdown into the Surtsey vessel, where specific phenomena, such as the effect of subcompartment structures, water in the cavity, and hydrogen generation, can be studied.

The most recent Surtsey DCH experiment was the first of a series of Integral Effects Tests (IET-I) using a small scale model of the Zion Nuclear Generating Station. The purpose of this test series is to investigate possible scale distortions between NPP scale and experiment scale. The Surtsey DCH experiments will be conducted at 1:10 linear scale whereas counterpart tests will be performed at Argonne National Laboratory (ANL) at 1:40 scale. Results of these experiments will allow assessment of scaling methodologies proposed by the SASM-TPG and by SNL.

For the IET test series one-tenth linear scale models of the Zion reactor pressure vessel (RPV), cavity, in-core instrument tunnel, and subcompartment structures were constructed. The RPV was modelled with a melt generator that consisted of a steel pressure barrier, a cast MgO liner, and a thin steel inner liner. The melt generator had a semi-hemispherical bottom head with a graphite limitor plate with a 3.5 cm exit hole to simulate the ablated hole in the RPV bottom head formed by tube ejection in a NPP severe accident.

In the IET-1 experiment, iron oxide/aluminum/chromium thermite was used as a corium melt simulant. Forty-three kg of molten thermite was ejected by slightly superheated steam at 6.2 MPa through the hole in the graphite limitor plate. Steam blowthrough entrained the majority of the molten debris into the Surtsey vessel, which had been pre-inerted with nitrogen (<0.1 mol.% O₂) to 0.20 MPa.

The cavity and in-core instrument tunnel were designed to withstand steam explosions of up to 7 MPa with a safety factor of four. In the IET-1 experiment, the cavity initially contained 3.48 kg of water, which
corresponds to condensate levels in the Zion plant. The inclined portion of
the instrument tunnel entered the bottom head of Surtsey at a 26° angle from
vertical, as it does in Zion. A false concrete floor was constructed in the
Surtsey vessel, similar to the floor of the Zion basement, so that the
inclined portion of the instrument tunnel was about 2.7 times the correct
scaled length of the Zion instrument tunnel exit. This floor was
constructed in Surtsey to match the configuration of the ANL facility.

The subcompartment structures included 1:10 linear scale models of the crane
wall, four steam generators, four reactor coolant pumps (RCP), the seal
table, the seal table room, the biological shield wall, the refueling canal,
the radial beams and the gratings at the RCP deck, and the operating deck.
The steam generators, reactor coolant pumps, and gratings were constructed
of steel while the other structures were made of reinforced concrete. All
of the subcompartment structures were coated with epoxy paint.

Instrumentation included 17 metal diaphragm, strain gauge-type pressure
transducers, 2 optical pyrometers, 9 aspirated thermocouples, and 12 gas
grab sample bottles. One optical pyrometer was located inside the
biological shield wall and the other was located outside the crane wall.
Both pyrometers were focused above the cavity exit through quartz windows.
These pyrometers gave accurate debris temperature measurements and also a
good estimate of the debris ejection interval.

The Surtsey vessel was inerted with nitrogen to virtually eliminate
metal/oxygen reactions in the vessel atmosphere. In addition, this
preserved the hydrogen produced by steam/metal reactions so that the
hydrogen concentration produced by the HPME could be measured. The gas
samples were analyzed using gas mass spectroscopy by Battelle, Pacific
Northwest Laboratories in Richland, WA.

This paper will give results of the IET-1 experiment. In particular,
results of the thermite discharge and steam blowdown will be reported. The
pressures and gas temperatures in the cavity, sub compartments, and upper
dome of Surtsey will be reported as a function of time. The temperature of
the molten thermite as it entered the subcompartment structures will be
given. Results of the measured amounts of hydrogen generated by steam/metal
reactions and a complete debris recovery summary will also be reported.
These results will be compared to the results of the ANL counterpart test
and possible scale effects will be discussed.
Postulated fission product releases (so-called "source terms") have played a major role in U. S. regulatory requirements both for reactor siting and in setting plant design requirements. The current source term is derived from report TID-14844, issued in 1962, which postulates the instantaneous release into containment of 100 percent of the noble gas inventory of the core, 50 percent of the Iodine fission products (half of which are assumed to deposit on interior surfaces very quickly), and 1 percent of the remaining fission products. Current regulatory guidance also assumes that Iodine is present primarily in the form of elemental Iodine.

At the present time, the NRC is pursuing several initiatives to incorporate research insights from severe accidents regarding source terms into regulatory guidance.

The reactor site criteria (10 CFR 100) are to be revised to remove source term and dose calculations for siting purposes and to add exclusion area size and population density requirements directly into Part 100. An interim revision of 10 CFR 50 will be carried out in parallel to retain a reference to a source term.

An updated report replacing TID-14844 is also in progress, and is expected to reflect changes in fission product timing, composition and magnitude, and iodine chemical form.

A final revision of 10 CFR 50 to incorporate updated source term and severe accident research insights will then be undertaken. Although this updated source term report is expected to be applied toward licensing of future light water reactor plants, insights arising from this study are also expected to be made available for voluntary use by existing reactor licensees. The present status of this effort and expected schedules will be described in more detail in the paper.
The analyses in this study were based on quantitative (calculated) results of seven severe accident sequences for light water reactor (LWR) nuclear power plants. These sequences represent a wide range of conditions that are significant risks. Both high- and low-pressure sequences were chosen for three principal plant types; a single sequence was considered for the PWR ice condenser. Each sequence was evaluated by the Source Term Code Package (STCP), and the thermohydraulics has been documented in previous NRC reports.$^1$ The issue that has been addressed is the chemical forms of iodine in the reactor coolant system (RCS) and in containment — not the ultimate disposition of these chemical forms.

In an LWR accident sequence, fission products released from the core will undergo changes in temperature and concentration as they pass through regions of the RCS. A chemical kinetic model used 20 reactions to determine the control volume where an equilibrium of the iodine, cesium, hydrogen, and steam species becomes "frozen." This means that the temperatures and concentrations of species in subsequent control volumes are not sufficient to reach an equilibrium in the mean residence time available. The "frozen" equilibrium is the species distribution entering containment. Separate equilibrium calculations were performed, using the FACT system,$^3$ to obtain the distribution of iodine species. The FACT system was chosen in this study because it can be used by anyone who wishes to examine the calculations and its data base contains only assessed data.

In six of the seven calculations, iodine entering the containment from the RCS was almost entirely in the form of CsI; the contributions of I or Hl were <0.1% of the overall percentage of iodine.

During the second half of the Surry AB sequence, there is a period during which temperatures in the core region are predicted to be in excess of 2000 K (3141°F) and subsequent volumes of the upper grid plates and guide tubes are at temperatures of only ~500 K (400°F). Under such conditions, the equilibrium compositions in the core region would be "frozen" by the rapid decrease in temperature. For this sequence, the overall iodine distribution was 2.8% as I, 0.4% as HI, with the remainder as CsI. Thus, a total of 3.2% as I plus HI was the largest fraction of iodine in a form other than CsI calculated to enter containment from the RCS in this study.

Once within the containment, CsI is expected to deposit onto interior surfaces and dissolve in water pools, forming iodide (I–) in solution. The dissolution of HI and HNO₃ (produced by irradiation of $\text{N}_2$ in the atmosphere) and the hydrolysis of $\text{I}_2$ tend to acidify films and pools of water.

Iodine behavior in containment was evaluated during the early stage of an accident sequence, up to ~1200 min. If pH is controlled in containment water pools so that it stays above 7, a reasonable value for the fraction of I– converted to $\text{I}_2$ is $3 \times 10^4$. This yields a small production of volatiles for PWRs, but virtually none for BWRs. Thus, if pH is maintained at 7 or above, only a small additional amount of $\text{I}_2$ is indicated to enter the gas phase in PWR systems.

If the pH drops below 7 (assumed uncontrolled pH), a larger fraction of aqueous I– will be converted to $\text{I}_2$. Evaporation of this volatile species so as to maintain equilibrium partitioning will result in greater atmospheric $\text{I}_2$, which, in turn, will yield higher organic iodide concentrations. As expected,
ESTIMATE OF RADIONUCLIDE RELEASE CHARACTERISTICS
INTO CONTAINMENT UNDER SEVERE ACCIDENT CONDITIONS

Hossein P. Nourbakhsh
Brookhaven National Laboratory
Upton, NY 11973

Estimation of Accident source term is important in nuclear safety regulation. Current regulations (10 CFR Part 100) requires that the suitability of the reactor site be judged based in part on a postulated fission product release associated with a substantial core-melt accident. The release into containment for this accident is derived from the 1962 Atomic Energy Commission (AEC) report TID-14844 ("Calculation of Distance Factors for Power and Test Reactor Sites"). The instantaneous release into containment of 100 percent of full power noble gas fission products in the core is postulated. Regulatory Guides 1.3 and 1.4 suggest that the bulk of the iodine (about 90 percent) should be assumed to be elemental (I\textsubscript{2}). This "maximum credible" accident, postulated for site analysis is a nonmechanistic event and no specific accident sequence leading to the postulated release is specified. Regulatory Guide 1.3 and 1.4 specify a large loss-of-coolant accident environment in conjunction with this accident.

The use of TID-14844 release assumptions has not been confined to a determination of site suitability alone. The regulatory applications of releases of this type cover a wide range, including the basis for (1) performance of important fission-product clean-up systems such as sprays and filters, (2) post accident habitability requirements for the control room, (3) the radiation environment for safety related equipment qualification, (4) post-accident sampling systems and (5) containment leak rates.

There has been significant research activity regarding severe accidents following the accident at Three Mile Island Unit 2 (TMI-2). A detailed review of the available source term information for light water reactors from this extensive research has been performed for the present study. This information is provided to support the generation of an updated estimate of source terms appearing in containment under severe accident conditions.

Estimates of radionuclide release and transport characteristics are specified for each unique combination of reactor coolant and containment system conditions. The characteristics of the radionuclide releases in this study are clearly different than the hypothetical source terms proposed in TID-14844.

A quantitative uncertainty analysis in the release into containment using NUREG-1150 methodology is also presented in this paper.
A detailed analysis has been performed to calculate the timing of the containment isolation signals and the first fuel pin failure for LOCA's of various sizes for a W-4-loop design (Seabrook) and a B&W design (Oconee). Sensitivity studies were performed to assess the impacts of fuel pin burnup, axial peaking factor, break size, ECCS availability, and main coolant pump trip on these timings. The analysis was performed using a four-code approach, comprised of FRAPCON-2, SCDAP/RELAP5/MOD3, TRAC-PFI, and FRAP-T6 calculations. In addition to the calculation of timing results, this analysis provides a comparison of the capabilities of SCDAP/RELAP5/MOD3 against FRAP-T6 and TRAC-PFI for fuel pin failure analysis and large-break LOCA analysis, respectively. This paper will discuss the methodology employed, the code development efforts required to implement the methodology, and the significant results obtained from this analysis.
On August 8, 1985, the NRC issued a policy statement on severe accidents which concluded that existing plants posed no undue risk to public health and safety. The policy statement, however, recognized the importance of past Probabilistic Risk Assessment (PRA) activities in understanding severe accidents at nuclear power plants, and in identifying previously unrecognized severe accident vulnerabilities. Based on this observation, the NRC issued Generic Letter 88-20 (and several supplements) which required licensees of commercial nuclear power plants to perform probabilistic safety assessments of their facilities. The assessments are to consider both internal initiated [termed Individual Plant Examination (IPE)], and external initiated [termed Individual Plant Examination for External Events (IPEEE)] events.

Over the next few years, seventy-eight IPE/IPEEE submittals covering 114 nuclear units will be reviewed by the NRC staff. The review will focus on the process used by the licensee in fulfillment of the IPE and IPEEE objectives prescribed in Generic Letter 88-20. The IPE/IPEEE review will be performed by a NRC staff review team consisting of four members with different technical expertise. The members will perform an initial check of the licensee's submittal for completeness and consistency with past PRA practices, meet with the licensee to discuss any technical review questions, and attempt to resolve any outstanding issues that stem from the examination. Based on the review team findings, a more detailed second step review may become necessary. In general, the second step review will employ contractors with different technical expertise, and require a site visit to audit supporting (or tier 2) information. The resulting contractor reports would then be considered by the NRC review team prior to a final assessment of the licensee's IPE/IPEEE process.

This paper will provide an overview of the IPE review process and summarize program status. In addition, a discussion of preliminary IPE/IPEEE findings and proposed licensee actions that result from their IPE effort will be presented. These findings will be stored in a computerized data system to be used at a later date to gain generic insight into plant behavior for different classes of plants. Generic IPE/IPEEE insights should prove useful to the NRC staff in evaluating potential strengths and weaknesses of the regulations.
SAFETY GOAL IMPLEMENTATION - SUMMARY AND STATUS

Brad Hardin
U.S. Nuclear Regulatory Commission

During the past year, a special NRC steering group has been preparing guidance for implementing the Commission's safety goal policy. The guidance is for interim use by the staff during a trial period to allow the staff to evaluate its effectiveness and to modify it as needed. At this time the guidance is limited to the consideration of safety goals in the regulation of operating plants only. The next step would be to extend its use to the licensing of the advanced reactor designs currently under review by the NRC.

In brief, the guidance involves a screening process to evaluate conformance of proposed regulatory actions for operating plants against two subsidiary safety goal objectives: (1) core damage frequency (CDF) of less than 1E-04/reactor year, and (2) frequency of a "large release" of radioactive materials from containment of less than 1E-06. An explicit definition of large release has not yet been finalized. The staff is currently evaluating alternative definitions and will make a recommendation to the Commission within the next year. When a decision has been reached by the Commission on the definition to be used for a "large release", the safety goal procedure will be modified to incorporate that definition.

The subject of safety goals has been under consideration for the past 10 years. In August of 1986, the Commission published it policy statement on Safety Goals for the Operation of Nuclear Power Plants. This policy statement contained two qualitative safety goals, two quantitative health objectives to be used in determining achievement of the qualitative safety goals and a general performance guideline for staff examination.

The qualitative safety goals are:

"Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health."

"Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks."

The quantitative health objectives to be used in determining achievement of the above safety goals are as follows:

"The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of the percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed."
"The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes."

In addition, a general performance guideline proposed by the Commission for further staff examination is as follows:

"Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall "mean" frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."

Other Commission guidance includes a Staff Requirements Memorandum in June of 1990, which provided guidance to the staff in the use of safety goals (including the possible use of the two subsidiary safety goal objectives described above) and requested the staff to develop a formal plan for implementation. In response to this request, the special steering group was formed and the interim plan for implementation developed.
Evidence for strong Holocene earthquake(s) in the Wabash Valley seismic zone

Steve Obermeier
U.S. Geological Survey

Many small and slightly damaging earthquakes have taken place in the region of the lower Wabash River Valley of Indiana and Illinois during the 200 years of historic record. Seismologists have long suspected the Wabash Valley seismic zone (see inset in fig.) to be capable of producing earthquakes much stronger than the largest of record (m_b 5.8). The seismic zone contains the poorly defined Wabash Valley fault zone and also appears to contain other vaguely defined faults (Hamburger and Rupp, 1988; Sexton, 1988) at depths from which the strongest earthquakes presently originate.

Faults near the surface are generally covered with thick alluvium in lowlands and a veneer of loess in uplands, which make direct observations of faults difficult. Partly because of this difficulty, a search for paleoliquefaction features was begun in 1990. The search was mainly in the Wabash River Valley, and to a lesser extent in the Ohio River and White River Valleys.

The Wabash River Valley traverses the central axis of the seismic zone. The valley contains extensive deposits of thick, late Wisconsinan (~14,000 years old), glaciofluvial braided stream sediments (almost entirely clean granular) and extensive deposits of younger point bar sediments (largely clean granular). The water table has probably been relatively shallow (< 10-20 feet deep) over large parts of the valley during almost all the Holocene (past 10,000 years) on the basis of depth of weathering profiles in the alluvium. This combination of a relatively shallow water table through time and the presence of widespread, thick, clean granular deposits extending up to shallow depths has provided an excellent opportunity for liquefaction to have formed in response to strong earthquake shaking.

The present status of the field search for paleoliquefaction features is shown in the figure. (Previous results of the study were reported in the 1 March 1991 issue of Science.) The figure shows those searched localities that have sediments susceptible to liquefaction. In almost all the field work, river banks were examined at places where erosion is severe during flooding each year. As the figure indicates, many miles of banks were examined.

All features interpreted as having a seismically induced liquefaction origin are expressed as planar sand-filled dikes that generally are vertical. At some places the sediment source stratum at depth was observed, and at some places venting to the surface took place. The dikes widen downward, are strongly aligned in many areas, and locally occur in elevated landforms. Because of these relations and properties just listed, I interpret an earthquake origin to be unequivocal.
Ages of the dikes are not well constrained. Radiocarbon and archeological data indicate that large gravelly sand dikes (at Sites GR, BR, and PB) formed between 1,500 and 7,500 years ago. Stratigraphic relations and similarity of weathering strongly suggest that all the large dikes are from a single event. This event probably exceeded $m_b$ 6.2 on the basis of the 1895 historic threshold earthquake near New Madrid ($m_b$-6.2) that produced a more limited amount of liquefaction. The concentration of the large gravelly sand dikes in a region of relatively uniform liquefaction susceptibility virtually proves that the epicentral region of the earthquake(s) that produced the Wabash Valley paleoliquefaction features was in or very near the Wabash Valley seismic zone. If both the most distant small dikes and the large dikes were formed by the same event, the earthquake almost certainly was much stronger than $m_b$ 6.2; its magnitude could easily have equalled the 1886 Charleston, SC, earthquake ($m_b$-6.7) on the basis of a comparison of distribution of liquefaction features.

It is possible (very likely, in my opinion) that the southernmost liquefaction features, along the Ohio River, were formed by the 1811-12 New Madrid earthquakes. Reasons for suspecting the 1811-12 earthquakes are that the dikes have no significant evidence of weathering (significant weathering would not be expected in 1811-12 liquefaction features), and the dikes lie between the 1811-12 epicentral region and the area of farthest 1811-12 historically observed liquefaction (see fig.). These sites offer the opportunity to do geotechnical engineering tests that can be used to back-calculate accelerations, far from the 1811-12 epicentral region.

Conclusions of the study are as follows: (1) an earthquake much stronger than any historic earthquake struck the lower Wabash Valley between 1,500 and 7,500 years ago; (2) the epicentral region of the prehistoric strong earthquake was the Wabash Valley seismic zone; (3) apparent sites have been located where 1811-12 earthquake accelerations can be bracketed.

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Searched localities having sediments susceptible to liquefaction; dots are sand pits, and heavy lines are stream banks. Sites having dikes are shown in capital letters. Maximum width of dikes shown as L (large, > 15 cm), M (medium, >6 cm), or S (small, <6 cm); question mark indicates uncertain earthquake origin or age. Blackened area shows the location of (small) vented sand volcanoes induced by liquefaction during the 1811-12 New Madrid earthquakes; these small sand blows are historically reported liquefaction features farthest from the 1811-12 earthquake epicentral region, which was near New Madrid.
PALEOSEISMIC INVESTIGATIONS IN THE NORTHEASTERN AND SOUTHEASTERN UNITED STATES: A STATUS REPORT

by

D. Amick, R. Gelinas, H. Kemppinen and K. Cato
Ebasco Services Incorporated
2211 W. Meadowview Rd., Greensboro, N.C. 27407

A systematic search for evidence of prehistoric earthquake activity along the Atlantic Seaboard found no conclusive paleoliquefaction evidence of large prehistoric earthquakes originating outside of South Carolina. In contrast, evidence of 6 large Holocene earthquakes has been found within the epicentral area of the 1886 Charleston earthquake. These findings support the hypothesis that during the past few thousand years earthquake induced ground motions have been strongest in the meizoseismal zone of the 1886 earthquake, and are in keeping with the concept of a unique seismotectonic setting near Charleston. The return period for these rare events was determined to be approximately 500 years. It is important to note that these past studies focused exclusively on coastal areas. As a logical extension the United States Nuclear Regulatory Commission is presently funding neotectonic/paleoliquefaction studies in non-coastal areas of the South Carolina - Georgia Seismic Zone (SCGSZ), the Southern Appalachian Seismic Zone (SASZ), and at several locales in New England.

Within the SCGSZ present paleoliquefaction investigations have targeted: (1) fluvial deposits near Union, SC; (2) lacustrine, fluvial and marine deposits near Bowman, SC; and (3) fluvial, lacustrine and marine deposits along the Coastal Plain segment of the Savannah River. The first two locales have experienced significant seismicity during historical times. The third has been relatively aseismic over the past several centuries and serves as a control area. Within the SASZ, the Giles County and eastern Tennessee seismicity clusters have been identified as high priority locales for future field investigations.

Several additional complementary lines of investigation are also underway SCGSZ. They include: (1) the potential use of Carolina Bays as barometers of past seismic activity; and (2) the evaluation of remote sensing data for evidence of surface faulting in the Charleston meizoseismal area. With respect to Carolina bays, our initial studies suggest that their age, internal stratigraphy, and hydrologic setting may make them a good place to look for evidence of Holocene strong ground motion. Saturated eolian derived sands and silts, which form the basal units in many of the bays are thought to be extremely susceptible to liquefaction under low to moderate levels of seismic loading. Further, the sandy rims of the bays would be especially susceptible to liquefaction and slumping during strong earthquakes. If prehistoric liquefaction and associated subaqueous venting or slumping of the rim materials has occurred in the past, then we would expect to see thin layers of these basal sands deposited within the overlying peat units. Each layer would represent a different, datable liquefaction episode. A limited "control" study of Carolina Bays is presently being
undertaken. This effort could aid in the development of recognition criteria that would be used to search for evidence of seismically-induced liquefaction in similar lacustrine deposits that are prevalent throughout much of the SCGSZ study area. With respect to surface faulting in the Charleston area, other investigators have identified a NNE trending zone on SPOT imagery that is generally coincident with the epicentral area of the 1886 Charleston earthquake. These investigators have speculated that it may represent surface faulting. An evaluation of multiple generations of aerial photographs from the 1940's through today is underway to test this hypothesis.

In New England investigations are exploring: (1) the potential use of pingos as barometers of past seismic activity; (2) the origin of wedge features previously reported as being the result of large prehistoric earthquakes; and (3) the potential use of deformed subaqueous lacustrine sediments as indicators of prehistoric earthquake activity. In addition, detailed paleoseismic studies are underway at three locales in New England where significant past earthquakes have occurred.

Initial studies of pingos have found that (like Carolina bays) their age, internal stratigraphy, and hydrologic setting may make them a promising place to look for evidence of Holocene strong ground motion. Internal stratigraphy consists of a basal unit of loose, saturated sand overlain by organics. The sands are thought to be extremely susceptible to liquefaction. If liquefaction and subaqueous venting of these sands has occurred we would expect to see thin sand layers deposited within the overlying organics. Each layer would represent a different datable liquefaction episode. With respect to wedge structures, previous investigators have attributed their genesis to: (1) seismically induced liquefaction features, (2) pseudo ice wedge casts, or (3) ice-wedge casts associated with permafrost conditions. Understanding the origin of these structures is important since they have some characteristics that are similar to paleoliquefaction features and may represent post glacial deformation which cuts the Holocene soil horizons. If they are related to prehistoric earthquakes, then their distribution could prove very useful in assessing the seismic history of New England. With respect to deformation of subaqueous lacustrine sediments, worldwide studies have shown that strong ground shaking can result in subaqueous slumps/slides, small scale subaqueous liquefaction features, and anomalous bedding. We feel this could prove to be a valuable line of study in New England. Control studies are presently underway in the Moodus and Ossipee areas to determine if lacustrine deposits could be used to aid in the search for evidence of past large New England earthquakes.

In parallel, detailed investigations in New England are focused on: (1) Holocene fluvial and lacustrine, and late Pleistocene glaciomarine, glaciofluvial, and glaciolacustrine deposits located near Moodus, Connecticut; (2) Holocene fluvial and lacustrine, and late Pleistocene glaciofluvial and glaciolacustrine deposits near Ossipee, New Hampshire; and (3) Holocene fluvial and late Pleistocene glaciomarine, glaciofluvial, and glaciolacustrine deposits near Newbury, Massachusetts. The three basic elements of the ongoing field program are: (1) the identification and evaluation of high priority locales followed by detailed site studies; (2) a general search of existing exposures for evidence of post glacial deformation features; and (3) the identification and evaluation of subaqueous lacustrine deposits for evidence of Holocene deformation.
GROUND AMPLIFICATION DETERMINED FROM BOREHOLE ACCELEROGRAMS

Dr. Ralph J. Archuleta  
Dr. Sandra H. Seale  
Institute for Crustal Studies  
University of California, Santa Barbara

The Garner Valley downhole array (GVDA) consists of one surface accelerometer and four downhole accelerometers at depths of 6 m, 15 m, 22 m, and 220 m. The five, three-component vertical array of dual-gain accelerometers are capable of measuring accelerations from $3 \times 10^{-6}$ g to 2.0 g over a frequency range from 0.0 Hz (0.025, high-gain) Hz to 100 Hz. The site (33° 41.60' N, 116° 40.20' W) is only seven kilometers off the trace of the San Jacinto fault, the most active strand of the San Andreas fault system in southern California and only about 35 km from the San Andreas fault itself. It is situated at the northern end of the Anza seismic gap. This section of the San Jacinto fault has accumulated at least 1.1 m of slip since the 1899 earthquake leaving a possibility of a M 6.5 - 7.0 earthquake. Small earthquakes occur regularly with at least 10 earthquakes with magnitude 1.5 or greater being recorded monthly at GVDA. Since being installed in July 1989 GVDA has recorded 232 earthquakes with magnitudes from 4.7 to 1.2.

The site is within an ancestral lakebed that is presently the site of extensive deposition of fine materials derived from both the crystalline rocks and from dissection of the older alluvial deposits. These deposits overlie granitic rocks that are dominantly hornblende tonalite. Drilling and coring by the Army Corps of Engineers shows that the soil is comprised of fine alluvial and granular silty sands, silts and some clay layers. Velocity logs taken by the USGS show an 18-m thick layer with a shear-wave speed about 220 m/s. Below this is a layer with a shear wave speed of 580 m/s extending to 60 m at which depth the velocity jumps to 1310 m/s. Having looked at the original records taken by the USGS, we would place the velocity step from 580 m/s to 1310 m/s around 40-45 m, a depth more consistent with the short-baseline refraction survey by the Army Corps of Engineers. The shear wave speed is 1310 m/s from this boundary to 100 m, the maximum depth that was logged. Approximately seven kilometers from GVDA the USGS operates two borehole stations that have velocity transducers installed. These holes drilled in similar igneous rocks, but without any sediment cover, show a shear wave velocity of 2650 m/s below 50 m. The upper 18 m of soil has been analyzed in detail using geotechnical methods by Pecker and Mohammadioun (1991). They find a nearly continuous increase in shear wave velocity from 90 m/s at 0 m to 280 m/s at 18 m. Their results are similar to that of the USGS if one assumes that 220 m/s represents an average velocity structure for the upper 18 m.

To examine the amplification as a function of frequency we selected 17 earthquakes $M \geq 2.0$ that were within 20 km of GVDA. For each of the 17 earthquakes we computed a whole record spectral ratio of surface to 220 m for each of the horizontal components. The earthquakes are at many different azimuths with respect to GVDA; we computed the average spectral ratio from 17 earthquakes. Ignoring peaks the average level of amplification is around 10 in the frequency band 1.5 - 30 Hz. Resonances around 1.7, 3.0 and 12 Hz are evident for both components. There is a low-frequency resonance around 0.4 Hz that shows up on only one component. The spectral ratio is rapidly decreasing above 35 Hz. This decrease is evident in the spectral amplitudes for the shallow accelerometers and also for the accelerograms at 220 m.

The effect of the weathered granite layer can be examined by computing spectral ratios of records at 22 m and 220 m. The average level of the spectral ratio between 2.0 and 30 Hz is around three. The overall shape of the spectral ratio observed for 22/220 is very similar to that for 0/220 which is simply a factor of three or so larger. Thus the weathered granite layer, which also
has low shear wave velocities, significantly affects the amplification of the waves. We observed a similar situation for other downhole strong motion records at McGee Creek, California, where an intermediate layer strongly affected the overall amplification of seismic waves.

To examine the effect of the different layers we consider the acceleration amplitude spectrum for S waves at each depth for two earthquakes that are close to each other. In this way we can minimize any effects due to distance or azimuth. Also we can compare two records of different magnitude. We consider the M4.2 event (epicentral distance 6.2 km, depth 14.5 km, azimuth 251.4° and a M2.5 earthquake (epicentral distance 6.6 km, depth 13.3, azimuth 249°). The data we examine are the combined horizontal S-wave spectra: the square root of the sum of squares. The spectral ratios of each depth are taken relative to the spectrum at 220 m. These data reveal several features. It is clear that the spectrum at 220 m has a significantly greater amount of high frequency energy than at any other depth. If the amplitude at low frequencies for 220 m is adjusted to be comparable to the low frequency amplitude at 22 m by multiplying by about 2.2, the high frequency spectral amplitudes at 220 m exceed those at 22 m by a factor of 20 for both events for frequencies greater than 40 Hz. The 22 m accelerometer is located in the weathered granite zone just below the soil. We infer that the weathered granite zone has amplified the low frequencies but the attenuation at high frequencies is much greater than the amplification. The overall spectrum at 220 m is again amplified by the soil as can be seen by comparing spectral amplitudes at 22 m with those at 0 m. The overall spectral S-wave amplification at low frequencies is around nine similar to what we determined by averaging the spectra of 17 different earthquakes.

Analysis of individual spectra and spectral ratios for the various depths shows that the zone of weathered granite has a pronounced effect on the spectral amplitudes for frequencies greater than 40 Hz. The soil layer impedance may amplify the high frequencies more than it attenuates. This result must be checked more thoroughly with special consideration of the spectra of the P-wave coda on the horizontal components. Analysis of the P-wave spectra and the spectral ratios shows an increased amplification in the same frequency range (60-90 Hz) where the S-wave spectral ratios imply a change in the attenuation. Comparison of acceleration spectra from two earthquakes, ML 4.2 and ML 2.5 that have nearly the same hypocenter, shows that the near surface amplification and attenuation is nearly the same for both earthquakes. However, the earthquakes themselves are different if we can assume that the recording at 220 m reflects the source spectra with a slight attenuation. The ML 2.5 earthquake has significantly greater high frequency content if the spectra are normalized at the low frequency, i.e., normalization by seismic moment.
TEMPORAL AND SPATIAL CLUSTERING OF PALEOSEISMIC EVENTS ALONG THE MEERS-DUNCAN-CRINER FAULT ZONE: IMPLICATIONS TO EARTHQUAKE HAZARDS IN THE CENTRAL AND EASTERN UNITED STATES

F. H. Swan
John R. Wesling
Kathryn L. Hanson
Geomatrix Consultants, Inc.
San Francisco, California

The northwest-striking Meers-Duncan-Criner fault zone lies along the complex northeastern border of the southern Oklahoma aulacogen, a fault-bounded trough that formed in the Middle Cambrian and extends into the craton from the southern margin of North America. The 200+ km-long fault zone separates a series of crustal uplifts on the southwest from very deep sedimentary basins to the northeast. Based primarily on differences in the geomorphic expression, the fault zone is interpreted to consist of at least five segments. Two of the fault segments, the Meers fault and the Criner fault, exhibit surface expression suggestive of Quaternary activity. Despite geologic evidence for recent activity on these fault segments, southern Oklahoma has had very little historical seismicity. There have been no macroseismic events associated with these faults, and there is no pattern of microseismicity that would suggest these faults are active.

Holocene surface faulting, which is characterized by prominent southwest-facing fault scarps, has been well documented along a 26 km-long section of the Meers fault and lineaments along the southeastward projection of this section suggest that the total length of recent faulting could be as much as 37 km. Detailed geologic mapping, trenching and morphometric analysis of the fault scarp indicate there has been pronounced temporal clustering of prehistoric surface faulting events on the Meers fault where there were two events within the past 3200 years preceded by a period of tectonic quiescence which lasted on the order of \(10^4\) to \(10^5\) years. Based on the net slip per event and the inferred rupture dimensions, these paleoseismic events were probably associated with earthquakes in the magnitude range of \(M_w 6\frac{3}{4}\) to \(M_w 7\frac{1}{4}\).

The Criner fault is approximately 120 km southeast of the Meers fault and has been mapped as the southeastern part of the mostly subsurface Meers-Duncan-Criner fault zone. Between the Meers and Criner faults, the fault zone is mapped as being overlain by Permian and Cretaceous strata and there is no surface expression of faulting. The 12 km-long mapped surface trace of the Criner fault trends approximately N45W along the southwestern margin of the Criner Hills anticline. En echelon folds within the Ardmore Basin to the northeast of the fault are oriented between 10 and 30 degrees more northerly than the Criner fault, suggesting left-lateral faulting along the Criner fault similar to the recent displacements on the Meers fault.

Geomorphic features suggestive of late Quaternary activity on the Criner fault include deflected drainages, springs, stream nickpoints in close proximity to the fault, and near-vertical, southwest-facing scarps in the limestone bedrock. The fault scarps range up to 1.0 m in height and are similar in morphology and orientation to bedrock scarps along the Meers fault. Where present across the mouths of stream gullies, the scarps coincide with prominent stream nickpoints.
Quaternary displacement on the Criner fault is indicated by an exposure of displaced fluvial gravels along Hickory Creek, which are estimated to be middle to late (?) Pleistocene in age. Two fluvial terraces along Hickory Creek, which are inset into the faulted fluvial deposits do not appear to be displaced by the fault. Mapping and dating of the Quaternary deposits along the Criner fault is still in progress. The available data suggest there have been no surface faulting events on the Criner fault since at least the middle Holocene. The differences in the Holocene histories of the Criner and Meers faults implies that the Meers-Duncan-Criner fault zone consists of several fault segments having markedly different rupture histories and that there has been spatial clustering of surface-faulting events along the fault zone.

The absence of any surficial evidence of faulting along the Duncan segment, which lies between the Meers and Criner segments, suggests that this segment of the fault has been inactive during the late Quaternary. However, the cumulative amount of Quaternary displacement on the Meers and Criner faults is very small (less than a few tens of meters) and the long term slip rates on these faults is very low. The rates of fault slip relative to denudation rates can have a significant influence on the preservation of evidence of surface faulting. It is possible that evidence of Quaternary surface faulting is not as well preserved along this segment of the fault which is characterized by less resistant sandstones and shales.

In summary, the following characteristics of the Meers-Duncan-Criner fault zone may have significant implications to the assessment of earthquake hazards in the central and eastern United States. Coincidence of this zone with the Oklahoma aulacogen suggests that the Quaternary faulting represents reactivation of a major zone of crustal weakness. During the late Quaternary the fault zone has been characterized by both temporal and spatial clustering of surface faulting events. This coupled with the absence of historical seismicity along the zone clearly demonstrates that the historical record is too short to identify some potentially hazardous seismic sources. Because of the potential for clustering of seismic events, low Quaternary slip rates do not a priori mean low seismic potential (i.e., small magnitude events and/or long recurrence intervals). Also, the small amount of cumulative displacement and the low slip rate relative to regional denudation rates can make it very difficult to detect Quaternary faulting along faults like the Meers-Duncan-Criner fault zone.
MODERN AND HISTORIC EARTHQUAKE-INDUCED LIQUEFACTION IN THE SAGUENAY-LAURENTIDE REGION OF QUEBEC

Martitia Tuttle and Leonardo Seeber
Lamont-Doherty Geological Observatory, Palisades, New York 10964, U.S.A.
Lorraine Wolf
National Research Council, Washington, D.C. 20418, U.S.A.
and
Tim Law
Carleton University, Ottawa, Ontario K15 5B6, Canada

The 1988 Saguenay event of moment magnitude $M_W$ 5.9 is the first modern earthquake in eastern North America for which liquefaction was documented and ground motion measured. It provides the opportunity to develop the usefulness of liquefaction features in paleoseismicity and to contribute to the assessment of the earthquake hazard in northeastern North America.

To date, six sites where liquefaction occurred during the Saguenay event have been documented in the Ferland-Boilleau valley of Quebec, between 25 and 30 km from the earthquake epicenter. Multiple sand boils, ranging in plan view from 10 cm to 15 m across, were observed at all of these sites. Below sand boils at three sites in Ferland, sand dikes, sills, vents and aprons (vented material) formed in fluvial sediments as the result of liquefaction of loose layers within the underlying glaciolacustrine deposit (Fig. 1). In addition, weathered liquefaction features, some of which were cross-cut by dikes that clearly formed during the 1988 earthquake, were observed at the same sites.

Important diagnostic characteristics of liquefaction features that formed in 1988 include upward-branching morphology of sand dikes, vertical and lateral grading of sediment filling the dikes, brecciation and faulting of host sediment above and adjacent to sand dikes and sand boils, and multiple fining-upward sequences of vented deposits above the soil $A$ horizon. These characteristics are indicative of rapid hydraulic emplacement of sediment from below. Observation and analysis of these liquefaction features made it possible to identify older, pre-1988 sand dikes, sills and vented deposits as earthquake-induced features.

Suites of organic samples associated with several older liquefaction features have been collected. To date, radiocarbon analysis of a few wood samples suggest that at least one of the older features formed between 1540 and 1680. Of the two large historic earthquakes, the 1638 and the 1663 events, that occurred in the region during this period, the 1663 earthquake is the more likely of the two earthquakes to have been responsible for the formation of the older liquefaction feature. The 1663 event which triggered landslides along the St. Lawrence River about 90 km to the
southeast of Ferland is thought to have occurred in the Charlevoix seismic zone. However, the well-constrained 1925 Charlevoix earthquake of MW 6.8 does not appear to have induced liquefaction in Ferland. Therefore, it would seem that the pre-1988 event, possibly the 1663 earthquake, that induced liquefaction in Ferland was either larger than the 1925 earthquake or located closer to Ferland. The relatively large size of the older liquefaction feature as well as the liquefaction of a relatively dense source layer as suggested by geotechnical testing support more intense ground shaking in Ferland during the earlier event than during the 1988 Saguenay earthquake.

Additional radiocarbon dating of pre-1988 features will help to determine the number and timing of previous earthquakes in the Saguenay-Laurentide region and subsurface investigations at broadly distributed sites will help to resolve the source areas and relative magnitudes of those events. A search for suitable sites for coordinated geological and geotechnical investigations is currently underway.

Figure 1. Trench log of modern and old liquefaction features that formed on the floodplain of Rivière des Ha! Ha! in Ferland, Quebec. Modern features including sand boil deposit and small (less than 1 cm in width) feeder dikes formed 30 km from the epicenter of the MW 5.9 Saguenay, Quebec earthquake. Older, somewhat weathered liquefaction features include a buried sand boil deposit and relatively large feeder dikes and sills. A glaciolacustrine deposit underlying the surficial fluval deposit is the probable source of the injected and vented sand within both generations of features.
HIGH-PRECISION ACCELERATOR-MASS-SPECTROMETER RADIOCARBON DATING OF SUBMERGED TIDAL-MARSH SOILS--AN APPROACH TO ESTIMATING THE FREQUENCY AND COASTAL EXTENT OF SUBDUCTION ZONE EARTHQUAKES IN OREGON AND WASHINGTON

Alan R. Nelson

U.S. Geological Survey, MS 966
PO Box 25046, Denver, CO 80225

SUMMARY

Has subduction of the Juan de Fuca plate beneath the North America plate in the Pacific Northwest produced great (magnitude, M>8) earthquakes during the late Holocene? Records of the past 200 years yield no evidence of great plate-boundary earthquakes in the Cascadia subduction zone. But along the coasts of Oregon and Washington peaty, tidal-wetland soils are interbedded with mud in estuarine stratigraphic sequences, and the submergence (relative rise of sea level) of some of these soils seems too widespread (>100 km), too large (>1 m), and too sudden (<10 years) to be attributed to any process except coseismic subsidence. How large were the Cascadia plate-boundary earthquakes that produced coastal subsidence and how often did they occur? Such questions are critical for earthquake hazard assessment in the Pacific Northwest; radiocarbon dating of buried tidal-wetland soils can help answer these questions.

Buried tidal-wetland soils of the Pacific Northwest may have been submerged by sudden coastal subsidence during any of three types of earthquakes. First, great earthquakes on one or more segments of the interface between the subducting and overriding plates may have produced a regional zone of coastal subsidence hundreds of kilometers long. Tidal-wetland soils similar to those in southern Washington and northern Oregon were submerged and buried along hundreds of kilometers of coast following historic great subduction earthquakes in Alaska and Chile. Second, deformation on shallow faults and folds in the overriding North America plate during great earthquakes on the plate-interface may have produced local areas of coseismic coastal subsidence. And third, some localized subsidence may have occurred during smaller earthquakes (M 6-73/4) in the overriding plate that were independent of plate-interface events.

Some tidal-wetland soils, however, may have been submerged and buried by non-tectonic processes. Peaty soils are commonly interbedded with mud in intertidal sequences of mid-latitude passive continental margins. Examples of non-tectonic processes that can produce such sequences include rapid changes in the rate of regional sea-level rise combined with changing sedimentation rates, or changes in the configuration of bars and channels in tidal inlets that led to local changes in tidal range. Tidal-wetland soils can be assumed to have been submerged by coseismic subsidence only where the abrupt upper contact of a widely-mapped, peaty soil gives strong evidence of a sudden, substantial change (>0.5 m) in water depth.
Precise radiocarbon dating can help distinguish soils buried following regional plate-interface earthquakes from soils buried during local upper-plate earthquakes or by non-tectonic processes. Synchronous ages for soils submerged along hundreds of kilometers of coast would be consistent with great plate-interface earthquakes, some perhaps as large as M 9. Non-synchronous ages would indicate lower magnitude plate-interface earthquakes with coseismic subsidence of much more limited extent, moderate-magnitude earthquakes on shallow structures in the upper plate, or soil submergence and burial by non-tectonic processes.

Each of the above processes may have submerged and buried tidal-marsh soils in the Coos Bay region of southern Oregon. This area includes the distal part of the active accretionary wedge of the North America plate. Analogies with deformation during great earthquakes in Chile, Japan, and Alaska indicate that regional uplift of as much as several meters and/or differential movements across folds and faults may occur in the accretionary wedge during great plate-interface earthquakes. In southern Oregon, late Holocene coseismic deformation has been inferred on shallow, upper-plate faults and folds in the South Slough arm of western Coos Bay. Surface deformation associated with these structures might be reflected as submergence or emergence events of 0.5-2 m in the tidal-marsh stratigraphic record. Many localized deformation events are probably coincident with plate-interface earthquakes, but shallow earthquakes of moderate to large magnitude (M 6-73/4) on smaller structures (<30 km long) in the overriding plate might also produce localized areas of coseismic subsidence or uplift.

The frequency of submergence events and the precision with which suddenly submerged soils can be radiocarbon dated determine if closely spaced submergence events can be distinguished from site to site. But in the Coos Bay region the precision of conventional 14C ages on peaty tidal-marsh soils is too low and the recurrence intervals between events are too short to distinguish local from regional submergence events.

The precision of age estimates for events in southern Oregon can be improved by averaging multiple accelerator-mass-spectrometer (AMS) 14C analyses of rigorously selected and pretreated plant macrofossils at the abrupt upper contacts of tidal-marsh soils. An initial test of this method in South Slough shows that standard deviations on AMS ages can be reduced to ±25-40 years. However, consideration of the total analytical errors in AMS analysis and age differences due to changes in the rate of 14C production in the atmosphere over time indicate that 95% confidence limits on calendar-corrected ages for submergence events range from 50 to 400 years. Events that occurred only 100 years apart may be distinguishable if their ages fall on favorable parts of the 14C calibration curve. In other cases, we may not be able to identify events that occurred as much as 400 years apart at different sites.
Soil Structure Interactions of Eastern U. S. Type Earthquakes

Chang Chen, Ph.D., P.E.
Assistant Chief Engineer
and
Samir Serhan, Ph.D.
Senior Structural Engineer
Gilbert/Commonwealth, Inc.
P.O. Box 1498
Reading, Pa. 19603

Two types of earthquakes have occurred in the eastern U. S. in the past. One of them was the infrequent major events such as the 1811-1812 New Madrid Earthquakes, or the 1886 Charleston Earthquake. The other type was the frequent shallow earthquakes with high frequency, short duration and high accelerations. Two eastern U. S. nuclear power plants, V. C. Summer, and Perry, went through extensive licensing effort to obtain fuel load licences after this type of earthquakes was recorded on sites and exceeded the design bases beyond 10 hertz region. This paper discusses the soil structure interactions of the latter type of earthquakes.

Current NRC staff interim position on this type of eastern earthquakes will prevent the operating plants from going through unnecessary licensing efforts if the future recorded events are less than a magnitude of $5 \frac{1}{4}$, within 25 km of the plants, and with acceleration spectrum exceedance beyond the 10 hertz region. EPRI also suggests the Cumulative Absolute Velocity (CAV) of 0.3 g·sec as the non-exceedance criterion. This kind of short duration, high frequency, and high acceleration earthquakes is non-damaging for passive systems, components, or rugged active mechanical components. The possible engineering significance of this type of earthquakes is the impact on active electric or control systems equipment. Under certain circumstances, operator's actions will be able to correct spurious actuations of safety systems due to relay chattering caused by this type of eastern earthquakes. Soil structure interaction of this type of earthquakes will assist to determine the need of operator's actions.

There were many soil structure interaction analyses performed in the past. However, there was a lack of recorded data to verify analysis results. For this reason, EPRI co-sponsored a seismic experiment with 1/4 and 1/12 scale models constructed in Lotung, Taiwan. The previously mentioned seismic data recorded at V. C. Summer and Perry Nuclear Power Plants were not adequate to verify soil structure interaction results. V. C. Summer data was recorded on an adjacent dam abutment not inside the plant. Perry data was recorded inside the plant not in the free field. However, Perry data was used to verify the structural model of dynamic analysis which successfully duplicated the high frequency responses on the steel containment as recorded.

On December 28, 1989, a magnitude 3.9 event occurred adjacent to Krsko Nuclear Power Plant, in the Republic of Slovenia, Yugoslavia. The plant is instrumented in accordance with USNRC Regulatory Guide 1.12. The strong motion instruments were activated. The records have the same characteristics of typical eastern U. S. earthquakes, namely, high frequency, short duration, and high accelerations. The recorded maximum horizontal component is 0.5 g in the free field at the surface. However, the recorded acceleration in the same direction in the Reactor Building base mat is only 0.05 g and that in the Diesel Generator Building is 0.2 g. Traditionally, an earthquake of such a small magnitude was not analyzed because of its lack of damaging potential. However, since this is one of the few cases that strong motion data was recorded both inside and outside of an operating nuclear power plant, it is useful for verifying soil structural interaction analysis results.
Furthermore this unusually high reduction factor from free surface to base mat needs investigation to determine whether the same kind of reduction can be expected for future earthquakes of larger magnitude.

Krsko Nuclear Power Plant is located in the Krsko block. In the geological past, regional tectonic conditions produced horizontal tension which caused the Krsko block to subside relative to boundary blocks. Thus, the plant is on top of about 700 m of soil deposit. At the current time the regional tectonics is under compression with maximum principal stress approximately in the NS direction. The Reactor Building has about 16 meters of embedment. The vectorized version of SASSI program on Cray computer was used to perform the soil structure interaction analysis. Parametric study was performed to obtain the optimum parameters and assumptions in order to match not only the acceleration but also the frequency content of the responses. This paper describes the parametric studies, their results and conclusions with regard to future applications.

The seismic wave environment is assumed to consist of vertically propagating shear waves due to the low reduction of the high frequency portion of the measured structural response compared with the free field response counterpart. SASSI soil structure interaction analysis of the December 1989 Earthquake resulted in encouraging qualitative and quantitative agreement with the recorded measurements for both horizontal East-West and North-South directions. Results of the analysis show clear evidence of strong soil structure interaction effects. The structural response motions displayed strong rocking mode participation due to irregular base mat shape in the vertical plane.
Abstract

A Proposal for Standardized Seismic Design (SSD) for Nuclear Power Plants

Thomas F. O'Hara, John P. Jacobson Yankee Atomic Electric Company
Walter J. Briggs Northeast Utilities

U.S. Nuclear Power Plants (NPP's) are designed, engineered and constructed to exceptionally stringent standards. Their seismic adequacy is determined by regulatory compliance and demonstrated by PRAs and seismic margin studies. However, the present seismic siting criteria requires improvement. The goal of proposed changes to the siting criteria is a predictable licensing process and a stable regulatory environment.

To satisfy this goal it is necessary to modify 10CFR100 Appendix A. At present NRC is scheduled to complete their redraft of Appendix A and supporting regulatory guides in 1991. Fundamental to the goal of stabilizing the licensing process is the removal of interpretive details from Appendix A and placing them in regulatory guides. The existing treatment of significant siting factors such as meteorology and hydrology in 10CFR100 provide the appropriate model for the treatment of seismic licensing issues. In the present 10CFR100, it is specified that "consideration" of the site meteorology and hydrology are required when evaluating the suitability of a new site. However, all of the details of how an applicant may characterize and protect against extreme meteorological and hydrological phenomena are provided in supporting Regulatory Guides and Standard Review Plans. This approach has been proven to provide adequate protection against these external hazards for our current generation of plants. Consistent with this treatment of the meteorological and hydrological concerns, it is proposed that seismic concerns in a new Part 100 be addressed by the following statement "Consistent with General Design Criteria 2 of Appendix A to Part 50 of this chapter, analyses sufficient to determine site suitability and to provide reasonable assurance that a nuclear power plant can be constructed and operated at a proposed site without undue risk to the health and safety of the public shall be developed."

We acknowledge that the proposed simplification of the existing regulations should be accompanied by new regulatory guides which endorse acceptable methodologies in detail for deriving acceptable seismic design criteria for any site in question. This paper discusses one approach which could be endorsed in the proposed new regulatory guides. This proposed methodology evaluates EUS NPP seismic designs relative to their respective site seismic hazards to determine a generic probabilistic transfer function. Using this transfer function we propose a new means to determine acceptable seismic design levels based on hazard, the Standardized-Seismic Design (SSD).

Two recent state-of-the-art studies evaluate the probability of exceeding seismic design for all eastern U.S. (EUS) NPP's: a Lawrence Livermore National Labs study (LLNL, 1989) funded by the NRC and similar research by the Electric Power Research Institute (EPRI, 1989) supported by the utilities. Both confirm that the present Appendix A has not provided consistent seismic design levels. In addition the present deterministic approach has lead to costly licensing delays and continual seismic re-evaluations.

SSD utilizes a probabilistic framework to accommodate alternative deterministic interpretations. SSD can provide a stable, predictable basis for licensing of future nuclear power plants. It uses seismic hazard input from EPRI and LLNL to produce consistent bases for seismic design. Combining deterministic and probabilistic insights provides a comprehensive approach for determining a site's Design Basis Earthquake (DBE). Deterministic methods are used both to provide the data input for the probabilistic model and to check the results.
A basic premise of SSD is the acceptance of seismic designs of present licensed plants as a conservative and allowable standard. Probabilistic Risk Analyses support this basis. A central tendency curve derived from existing site data (mean probability of exceeding seismic design) provides a transfer function for deriving future design spectra at new or existing sites.

In contrast with the use of these seismic hazard results in a solely quantative manner, SSD uses a "relative" approach to derive seismic design criteria. Absolute approaches fail to allow for limitations inherent in current EUS seismic hazard estimates. For example, it is not unusual to find EPRI and LLNL results for the same site that differ by two orders of magnitude. Results also show significant inconsistency from site-to-site. Past experience has shown that attempts to use these seismic hazard results in an absolute sense typically result in significant controversies.

SSD is a conservative and consistent technique to determine a seismic design level and spectral shape for any future NPP. The overall goal of this process is to link a consistent, conservative design spectrum with consistent and conservative design practice such that the end product satisfies all reasonable engineering standards.
Traditionally, probabilistic risk analyses of severe accidents in nuclear power plants have limited themselves to consideration of the set of initiating events occurring during full power operation. However, some analyses of accident initiators during low power, shutdown, and other modes of plant operation other than full power have been performed. These studies as well as the Chernobyl accident and recent operating experience at U.S. pressurized water reactors (PWRs) suggested that risks during low power and shutdown could be significant. As such, the analysis of the frequencies, consequences, and risks of these accidents was identified as one task in the Nuclear Regulatory Commission staff's study of the implications of the Chernobyl accident to U.S. commercial nuclear power plants.

This program is an ongoing high priority effort at Brookhaven National Laboratory (BNL) that is expected to continue for the next several years. The scope includes a Level 1 probabilistic risk assessment (PRA) with internal fire and flood for Surry Unit 1 (PWR). This program is also closely coupled to a parallel project for the Grand Gulf plant (BWR) being conducted by SNL. A catenating limited-scope project will combine the Level one information with accident progression, source term, and offsite consequence analyses to yield estimates of severe accident risks from these plant operational states which again will be available for direct comparison with the NUREG-1150 results.

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

The objectives of Phase 1 are:

- To provide initial insights as to any particularly vulnerable plant operational states (i.e. in addition to the well publicized state associated with mid-loop operations);

*This work was performed under the auspices of the U.S. Nuclear Regulatory Commission.
- To provide initial insights as to the set of initiating events applicable to each of the plant operational states especially initiators that do not apply to the full power analysis;

- To characterize on a high, medium or low basis the potential core damage frequency contribution associated with the leading sequences of each of the initiating events;

- To provide a preliminary characterization as to the potential risk associated with these leading sequences based upon 1) whether they are fast or slow developing, 2) whether they occur within the higher or lower core decay heat ranges, and 3) whether the containment might be open or closed; and

- To provide a foundation upon which the fully detailed Phase 2 analysis can focus its efforts.

Phase 1 of the program will be completed by September 1, 1991 and the results and insights will be discussed in the full paper.

REFERENCES

SUMMARY

The purpose of this paper is to describe the work being conducted at Sandia National Laboratories (SNL) with regard to the identification and quantification of accident sequences initiated in modes of operation other than full power for the Grand Gulf Nuclear Power Station.

As a result of the Chernobyl accident and other precursor events, the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research has undertaken a two-phase project to analyze the frequencies, consequences, and risk of accidents occurring during modes of operation other than full power.

This paper presents a brief overview of Phase 1, which encompasses a coarse screening of potential accidents that could occur at Grand Gulf while operating at other than full power. This coarse screening approach was adopted as a means of obtaining, in a relatively short period of time, some estimation of the potential for accidents during low power and shutdown conditions and some idea of the magnitude of the work necessary to perform a more detailed analysis of these operating states. A companion project for a pressurized water reactor (PWR) was conducted by Brookhaven National Laboratory. After the Phase 1 results have been examined, a Phase 2 analysis will be performed. This phase will incorporate a detailed analysis of the plant for these modes of operation.

The approach used is a modification of the standard Level I probabilistic risk assessment (PRA) approach used in many full power PRAs. The information contained in the NUREG-1150 analysis of the Grand Gulf plant was used as the starting point for this analysis. Event trees were constructed, top events were
modeled using fault trees of various sizes, and the top events were quantified using point estimates to produce the sequence frequencies. The IRRAS computer code was used in the construction of the event trees, modification of existing fault trees and construction of new trees, and the quantification of the accident sequences.

The Phase 1 analysis included all the normal tasks associated with a Level 1 analysis, with the exception that no significant recovery analysis task was performed. In addition to the normal set of tasks, a task to identify the reasons why a plant changes its mode of operation and to identify any equipment unavailabilities that resulted from such a change was necessary. While all the normal tasks, with the noted exception, were performed, some tasks made use of screening information only. The use of screening information and the lack of a significant recovery analysis task resulted in conservative estimates for the accident sequences. However, this reduced the possibility that, in this first comprehensive look at a BWR in these other modes of operation, potentially significant accidents would be eliminated before any detailed analysis of the accidents took place.

The approach, as described above, was applied to seven Plant Operational States (POSs) representing the plant's equipment status as the plant transitions from low power operation (less than or equal to 15 percent power) to refueling. In the beginning of this project, it was expected that the Level I approach would be applied to each of the Modes of Operation as defined in the plant's Technical Specifications with the analysis of Mode 1 (i.e. Power Operation) being limited to that time in which the plant operates at or below 15 percent power. As work progressed, it became apparent that another means of identifying plant operating conditions other than those included in the Modes of Operation would be necessary for the analysis to achieve its objectives. This resulted in the development of POSs. These POSs, defined in terms of a set of systems that are normally expected to provide the functions for maintaining the plant within a particular power, pressure, and temperature regime, provided a means of analyzing the plant as it transitions from low power operation to refueling.

The full paper will provide more details concerning the status of this project and will discuss the insights found thus far in the program.
An integrated Level 3 probabilistic risk assessment (PRA) was performed on the LaSalle County Station nuclear power plant using state-of-the-art PRA analysis techniques. The objective of this study was to provide an estimate of the risk to the offsite population during full power operation of the plant and to include a characterization of the uncertainties in the calculated risk values. Uncertainties were included in the accident frequency analysis, accident progression analysis, and the source term analysis. Only weather uncertainties were included in the consequence analysis. In this paper selected results from the accident frequency, accident progression, source term, consequence, and integrated risk analyses are discussed and the methods used to perform a fully integrated Level 3 PRA are examined.

LaSalle County Station is a two-unit nuclear power plant located 55 miles southwest of Chicago, Illinois. LaSalle County Station is operated by Commonwealth Edison Company and began commercial operation in 1984. Each unit utilizes a Mark II containment to house a General Electric 3323 MWt BWR-5 reactor. This PRA, which was performed on Unit 2, included internal as well as external events. External events that were propagated through the risk analysis included earthquakes, fires, and floods. The internal event accident scenarios included transients, transient-induced LOCAs (inadvertently stuck open relief valves), anticipated transients without scram (ATWS), and LOCAs.

The total core damage frequency at LaSalle from all events has a mean value of 1.01E-04/yr with a 5th percentile of 5.34E-06/yr, a median value of 2.92E-05/yr., and a 95th percentile of 2.93E-04/yr. The dominant accident sequence, 35.4% of the mean core damage frequency, involves a loss of all injection as a result of failures occurring after a loss of offsite power. The dominant cut sets of this sequence represent a short-term station blackout type accident. The second most likely sequence, 17.2% of the mean core damage frequency, is the result of a control room fire which is not suppressed and becomes large enough to require evacuation of the control room. Auto actuation of the systems fail as a result of the fire and the

* This work was supported by the U.S. Nuclear Regulatory Commission and performed at Sandia National Laboratories which is operated for the U.S. Department of Energy under contract No. DE-AC04-76DP00789.
operators do not operate the remote shutdown panel correctly due to the high stress. Loss of all injection occurs and short-term core damage results.

The cut sets generated in the Level 1 analysis were grouped into plant damage states (PDS) for propagation through the Level 2 and 3 analyses. The PDSs define the status of emergency systems and the condition of the containment at the time of core damage. This process resulted in thirty PDSs. For presentation purposes these thirty PDSs were combined into the following seven summary PDS groups: Seismic, Fire, Flood, ATWS, LOCA, Transients, and Transient-Induced LOCAs.

Given that core damage occurs, it is likely that the accident will proceed to vessel breach and the containment's integrity will be compromised. Although notable, the mean conditional probability of core damage arrest prior to vessel breach is still fairly small, approximately 0.15. For the short term station blackout accidents, the probability of core damage arrest is driven by the low probability that AC power is recovered early enough in the accident that restoration of coolant injection will arrest the core damage process. For the other accidents the lack of available or recoverable injection systems leads to the low values of core damage arrest. The mean conditional probability that the containment will remain intact throughout the accident is only 0.12. It is about equally likely that the containment will either be vented during the accident, mean probability of 0.46, or will structurally fail, mean probability of 0.42.

Although the conditional probability that the containment will fail during the accident is fairly high, the offsite risk to the public due to the operation of LaSalle County Station is still relatively low, especially with respect to the NRC safety goals. The mean individual early fatality risk within 1 mile is 1.1E-10/R-yr which is more than three orders of magnitude below the safety goal. Similarly, the mean individual latent cancer fatality risk is 8.5E-09/R-yr which is slightly more than two orders of magnitude below the safety goal. The mean values for early fatality risk and for latent cancer fatality risk are 1.2E-08/R-yr and 0.25/R-yr, respectively. The low values for risk can be attributed to the low core damage frequency, the fast evacuation of the public away from the plant, and plant features that reduce the source terms that result from a core damage accident. The risk at LaSalle is dominated by the Fire plant damage state (PDS) group and the Transient PDS group. The LOCA, and Transient-Induced LOCA PDS groups, on the other hand, are very minor contributors to the risk. The Flood, Anticipated Transients Without Scram (ATWS), and Seismic groups are in-between.
RISK-BASED PERFORMANCE INDICATORS*

M.A. Azarm and W.E. Vesely**
Engineering Technology Division
Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973

**Science Applications International Corporation

Summary

This research was sponsored by the U.S. Nuclear Regulatory Commission (NRC) to develop methods for monitoring an aspect of safety performance of licensed nuclear power plants; i.e., unavailability of safety systems.

Various options were evaluated for constructing system unavailability indicators using basic reliability principles and consistent with the content of data collected. The capabilities of each indicator option, in terms of detection probabilities and false alarm rates within a fixed response period for various types of anomalies, were evaluated through detailed simulation studies. As an example, the option of using system vs train downtimes to construct safety system unavailability indicators was evaluated. Comparison of response times indicated that the latter option has significantly shorter response periods (by factors of 10-30), i.e., more timely results.

Brookhaven National Laboratory performed two studies to evaluate the capability of NPRDS (Nuclear Plant Reliability Data System) as a data source for construction of system unavailability indicators. Specific indicators were developed consistent with the limited scope of NPRDS data. In order to generate system unavailability indicators in an efficient manner, the processes for NPRDS data acquisition, indicator construction, and analyses were automated using PC software. Limited pilot applications and verification were also performed.

An important aspect of risk-based indicators deals with the concept of dependent failures. The conventional dependent failure definition which consists of simultaneous failures of similar redundant components will generate a long response time for proper detection of anomalies, and it is not suitable for construction of an indicator. A recent study attempted to relax the definition of dependent failures by defining a time window within which the failure of similar redundant components can be accounted for. In this study, we have developed a model using basic reliability principles to treat the time between the failures of similar redundant components as a continuum rather than a truncated window. Using detailed reliability models we have developed a formulation for evaluating the risk implication of various failure patterns realized when the observation time is treated as a continuum. This methodology not only provides

*Work performed under the auspices of the U.S. Nuclear Regulatory Commission
the framework within which present PRA (Probabilistic Risk Assessment) methodologies can be enhanced but also provides the basic model upon which dependent failure indicators can be constructed. Specific indicators of dependent failures are presently being developed and evaluated.

Finally, limited pilot application of the indicator technology has revealed a large amount of information on short- and long-term trends in component failure rates, clustering of failures in similar components in and across systems, systematic effects and correlated trends, etc. Currently, there is no integrated methodology within existing PRA formulations that can incorporate such information and evaluate its risk implications. It is recommended that such methodologies be developed and that PRA models be quantified in a dynamic manner consistent with the information provided by the indicators. This concept is referred to as dynamic PRA.

REFERENCES


INTER-SYSTEM LOCA RISK ASSESSMENT

W. J. Galyean, D. L. Kelly and J. A. Schroeder
Idaho National Engineering Laboratory/EG&G Idaho, Inc.

Inter-Systems Loss-Of-Coolant Accidents (ISLOCAs) have been included in Probabilistic Risk Assessments (PRAs) since WASH-1400. While estimated as being relatively low contributors to core damage frequency, ISLOCAs have been identified as major contributors to risk at Nuclear Power Plants (NPPs). They have the potential to result in core melt and containment bypass, which may lead to the early release of large quantities of fission products. Recent events at several operating reactors have been identified as ISLOCA precursors. The occurrence of these events have raised concerns that the frequency of ISLOCA sequences might be underestimated in current state-of-the-art PRAs.

In order to expand the current state-of-the-art, an NRC sponsored research program is being conducted by EG&G Idaho, Inc. at the Idaho National Engineering Laboratory. The objective of the ISLOCA Research Program is to generate qualitative and quantitative information on the hardware, human factors, and accident consequence issues that dominate nuclear power plant risks for ISLOCA. To meet this objective, the approach being taken includes analysis of all interfaces between the primary reactor coolant system and other, lower pressure systems. The historical experience (primarily, licensee event reports) has provided the basis for determining the scope of the analysis with respect to potential failure mechanisms of the pressure isolation boundary. It is important to note that in the vast majority of these events, the dominant failure was a human error. Because of their significance, human errors are given particular attention in the present analysis.

The methodology comprises the following steps: 1) a screening of all primary system interfaces, including consideration of potential sequences initiated by hardware faults and/or human errors; 2) a probabilistic model of the interfacing system ruptures, both locations and occurrence; 3) consideration of the effects of ruptures in the auxiliary building, with respect to equipment operability and potential recovery actions; 4) consideration of recovery, post rupture; 5) normalization of calculated consequences to a nationwide-average nuclear power plant site.

The first step involved a review and screening of the plant’s reactor coolant system (RCS) interfaces using P&IDs and system descriptions. A review of plant operations was then performed to identify possible scenarios that could result in a breach of the primary pressure isolation boundary. This review relied upon the historical experience (i.e. applicable licensee event reports on ISLOCA "precursors"), plant procedures and practices, and discussions with plant personnel. The relatively high likelihood of human errors is well supported in the historical data. Therefore, in this portion of the analysis, extra attention needs to be focused on the possibility of human errors of commission that could lead to an ISLOCA type event.

An integral part of an ISLOCA sequence is the rupture of the interfacing system when exposed to pressures and temperatures much higher than their design...
basis. It is therefore important to assess the performance of these systems and components when they are subject to the high pressures associated with the RCS. Once the specific ISLOCA sequence has been postulated, and the relevant portions of the interfacing system have been identified, the basic approach for assessing the probability of a rupture in the interfacing system, consists of:

a) Estimating the realistic failure pressure for each piece of equipment in the low pressure rated system.

b) Thermal-hydraulic simulations of the interfacing systems are performed to estimate the pressure distribution in the system.

c) After the boundary conditions have been established, the actual rupture model is constructed utilizing an event tree format.

d) Each question in the event tree is answered by: (1) randomly selecting a failure pressure from the failure pressure distribution of the specified component being queried; (2) randomly selecting a local system pressure based on the results of the simple RELAP5 calculations, and; (3) comparing the selected component failure pressure with the selected system pressure.

e. Once the simulation is completed, the output is binned and estimates can be made about the relative frequency of equipment failures given system overpressurization.

Given the very rapid pressure rise predicted by the RELAP runs (approximately 30 seconds to equilibrium pressure for systems with flow, i.e. open ended or small relief valves), the potential for operator recovery before a rupture occurs in the interfacing system, is very limited. However, consideration is given to isolating the rupture and recovering from the sequence. Two considerations influence the potential for recovering from an ISLOCA sequence. One is the time to uncover the core, and the other is the effect of the rupture on equipment in the auxiliary building.

This application of the ISLOCA methodology to specific plants generated the following results. These result are preliminary, and extrapolating them to other plants would be speculative at this time.

1. Inter-System LOCA appears to be a relatively low contributor to the overall risk posed by NPP operation.

2. Human errors that could occur during startup and shutdown of the plant were found to be possible ISLOCA "precursor" events. However, in all likelihood these events are recovered without threatening core damage.

3. Good plant operating practices such as: clear procedures with cautions and warnings on the importance of maintaining pressure isolation boundaries, and training on potential ISLOCA events, are important in achieving relatively low ISLOCA risk.

4. Isolation of the break is a necessary part of preventing core damage from an ISLOCA. This is because the makeup capability to the BWST is insufficient to maintain an adequate reactor coolant inventory for breaks outside the containment that are larger than two inches in diameter.
U.S. ACCIDENT MANAGEMENT PROGRAM

Robert L. Palla, Jr.
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C.

Norman Lauben
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C.

The overall approach to accident management in the U.S. and the status of key regulatory, research and industry activities are presented. Five general areas are identified for utility assessment and enhancement as part of accident management implementation: (1) strategies for managing events beyond the design basis, (2) severe accident training, (3) guidance and computational aids for technical support personnel, (4) information needs during severe accidents, and (5) structure and process for decisionmaking. Anticipated improvements and important technical issues in each of these areas are highlighted. U.S. nuclear industry will play an important role in accident management by developing detailed guidance and information for use by utilities in implementing their accident management plans. The role of these products in the U.S. program is outlined. A process and schedule for implementation is presented. A research program in support of accident management is an essential part of the U.S. Nuclear Regulatory Commission's activities and is also described.
The ability of plant personnel to successfully manage severe accidents is strongly dependent on the availability of timely and accurate plant status information. The Nuclear Regulatory Commission (NRC) recognizes the importance of reliable plant information by making instrumentation an element of its accident management framework. This paper describes the results of research sponsored by the NRC to evaluate the availability of plant instrumentation for a range of possible severe accidents at a PWR with a large, dry containment using the approach developed in NUREG/CR-5513.

Availability of plant instrumentation for a range of severe accidents is performed by: 1) identifying severe accidents that influence risk for a PWR with a large, dry containment, 2) defining the expected thermal hydraulic, radiation and humidity conditions affecting the instrumentation, and 3) assessing instrument availability based on location and conditions.

Important severe accident sequences were identified using NUREG-1150 plant damage states and accident progression bins for Surry and Zion. The NUREG-1150 results were used because they are the most recent evaluation of all credible types of accidents that dominate core damage frequency and risk. The BMI-2104 and NUREG/CR-4624 results are used to determine the expected thermal hydraulic conditions for the plant damage states and accident progression bins for these plants. Natural circulation in the hot leg is factored into the evaluation. Radiation conditions were defined from the distribution of fission products in the plant given in BMI-2104 and NUREG/CR-4624. A humidity condition of 100 percent is assumed for this evaluation.

The instrument availability assessment is primarily based on temperature and pressure conditions that may occur during the severe accidents because the instrument hardware is more sensitive to these conditions. Radiation exposure may impact instrument availability during long term plant recovery. Relative humidity is not important because instruments are generally qualified to operate in an environment with 100 percent relative humidity.

Availability of an instrument is based on the assumption that instrument performance degrades significantly if instrument qualification limits are exceeded or if the system is operated outside of its range. Degraded instrument performance means that the instrument system output may be unreliable, that is,
the magnitude and/or trend of the parameter being monitored by the instrument is in error. The definition of degraded instrument performance includes the possibility of instrument failure.

Much of the plant data used to assess instrument availability are based on the Regulatory Guide 1.97 review for Calvert Cliffs. Important instrument data used are: the physical location; the range; and the qualification ranges for temperature, pressure, humidity, and radiation levels. The effect of station blackout is also considered.

The key results from the evaluation of instrument availability for a PWR with a large dry containment are summarized as follows:

- Instruments located in the reactor pressure vessel will experience temperature conditions beyond their qualification limits as the fuel approaches core melt temperatures. Exposure to these temperatures will degrade instrument performance and limit the availability of these instruments for further use in accident management.

- Instruments located outside the reactor vessel but within the reactor coolant system may experience temperature conditions beyond their qualification limits as a result of natural circulation during fuel heatup or fuel melting. Even if the qualification limit is not exceeded, some of the instruments that monitor temperature may be exposed to temperature conditions above their measurement range. Exposure to these temperature conditions will degrade instrument performance and limit the availability of these instruments for further use in accident management.

- Instruments located in the containment will be exposed to high temperatures in the event of multiple hydrogen burns or direct containment heating. Exposure to these temperature conditions may degrade instrument performance and limit the availability of these instruments.

Instrument failure resulting from high temperature conditions near the break location is possible for interfacing system LOCAs. Access to sampling and analysis equipment located in the auxiliary building away from the break location may not be possible, due to high radiation fields that may develop as early as when the cladding ruptures.

Because electrical power systems differ between plants, it is not possible to generally evaluate instrument availability for a station blackout. It is noted that many plants provide battery backup for all Regulatory Guide 1.97, Category 1 instrumentation, although this is not specifically called for in the document. If battery backup is provided, then most of the information required to monitor the status of the reactor coolant system and containment will be available until environmental conditions challenge instrument availability. Systems used to obtain and monitor samples of reactor coolant, containment atmosphere, and containment sump or cavity water may not be available in the event of a station blackout. As a result, information needs requiring sampling information may not be met.
MANAGING WATER ADDITION TO A DEGRADED CORE

P. Kuan and D. J. Hanson

Idaho National Engineering Laboratory
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, ID 83415

SUMMARY

In this paper we present an approach to the development of an accident management plan for adding water to a degraded core. Under certain degraded core conditions, adding water may lead to enhanced hydrogen production, changes in core geometry that would complicate recovery, steam explosions, or recriticality of the reactor core if unborated water is used. Therefore, a primary requisite for the development of an accident management plan for adding water to a degraded core is to ensure that undesirable consequences of water addition are understood so that: (1) their effects can be minimized and an accident can be terminated at the earliest possible stage, and (2) plant personnel can be better prepared to deal with plant responses that appear contrary to desired outcomes when water is added during a core degradation transient. The approach presented here addresses these concerns in the development of an accident management plan.

First, an event sequence of core damage is developed. Critical points on the event sequence where the core would have distinct responses to water addition include: (1) fuel rod ballooning and bursting, (2) rapid oxidation of zircaloy, (3) particulate debris bed formation, (4) cohesive debris bed formation, and (5) core relocation to the lower plenum. Based on the results of experimental and analytical work related to severe fuel damage and the knowledge gained from the TMI-2 accident, the potential consequences of water addition at these critical points are evaluated.

Second, evaluations are performed to determine the ability of plant instruments to diagnose core conditions before and after adding water to a degraded core. These include the characterization of the responses of reliable instruments and considerations of using the movable in-core self-powered neutron detectors and ex-core instruments during advanced stages of core degradation.

* Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research under DOE Contract No. DE-AC07-76ID01570.
Third, bounding calculations for core cooling by water addition are performed. These bounding calculations are preferred over best-estimate analysis in two respects: Uncertainty in accident conditions dictate simplified and conservative assumptions, and the results of the bounding calculations are not sensitive to the assumptions used. These calculations yield the minimum rate and amount of water addition to a degraded core that would not adversely affect subsequent evolution of an accident. In addition, the minimum rate and amount of water to quench or stabilize the core is also obtained in these calculations. The reference conditions for these calculations are the critical points on the event tree of core damage sequence.

Fourth, coolability boundaries are determined for expected geometries of core degradation. The geometries include those of cohesive as well as particulate debris beds. A cohesive debris bed is characterized by its size and the thermal conductivity of the materials comprising the bed. Given these two parameters, the coolability of the cohesive debris bed, defined by the potential of molten materials flowing out of the crust of the bed, depends on its power generating capacity. A particulate debris bed is characterized by its porosity and the size of the particles comprising the bed. Its coolability also depends on its power generating capacity. If the power generation inside the bed can support a heat flux at the surface of the bed exceeding the dryout heat flux of the bed, it is noncoolable. A noncoolable particulate debris bed will eventually melt. Quiescent responses to water addition to a core in advanced stages of degradation may indicate noncoolable core geometries. Accident management plans then must consider the possibility of subsequent relocation of molten core materials to the lower plenum of the vessel.

It is envisioned that an accident management plan for adding water to a degraded core would revolve around the core damage event tree. A time sequence should be developed for the event tree so that operators may anticipate the evolution of events. Identification of these critical points should be made with the aid of plant instruments; procedures should be established to monitor plant responses as water is added to the core, again, to anticipate the evolution of events. The minimum rate and amount of water to quench or stabilize the core, or to not adversely affect the damage progression could be used as engineering aids in making on-the-spot decisions by the technical staff. Finally, the management plan must include procedures to deal with core damage progression beyond the core itself after the core has become noncoolable.
U.S. NUCLEAR INDUSTRY APPROACH TO SEVERE ACCIDENT MANAGEMENT GUIDANCE
DEVELOPMENT AND IMPLEMENTATION

David Modeen
Nuclear Management & Resources Council

Larry Walsh
New Hampshire Yankee

Richard Oehlberg
Electric Power Research Institute

SUMMARY

The purpose of this paper is to discuss the United States nuclear industry activities occurring under the auspices of NUMARC, to define, develop and implement enhancements to utility accident management capabilities. This effort consists of three major parts:

1. Development of a practical framework for evaluation of plant-specific accident management capabilities and the subsequent implementation of selected enhancements.

2. Development of specific technical guidance that address arresting core damage if it begins, either in-vessel or ex-vessel, and maintaining containment integrity. Preventing inadequate core cooling or minimizing the consequences of offsite releases, while considered to be candidate areas for accident management enhancements, have been the subject of intense previous study and development.

3. Plant-specific implementation of accident management enhancements in three areas: (1) personnel resources (organization, training, communications); (2) systems and equipment (restoration and repair, instrumentation, use of alternatives); and (3) information resources (procedures and guidance, technical information, process information).

The integrated evaluation and application of insights from prior probabilistic risk assessments, plant-specific analysis, such as the Individual Plant Examination, and other industry and government programs, provide a means of improving a plant’s integrated capability to respond to rather unlikely, yet potentially severe, events.

At the same time, a very important question yet to be answered in the industry activities is what constitutes an appropriate allocation of utility resources to this effort relative to other plant priorities, and how one judges "success" in implementation of these enhancements.
A STRUCTURED APPROACH TO INDIVIDUAL PLANT EVALUATION AND ACCIDENT MANAGEMENT

G. T. Klopp Commonwealth Edison Company

Introduction:

The current requirements for the performance of individual plant evaluations (IPE's) include the derivation of accident management insights as and if they occur in the course of finalizing an IPE. The development of formal, structured accident management programs is, however, explicitly excluded from current IPE requirements. The NRC is following the NUMARC efforts to establish the framework(s) for accident management program development and plants to issue requirements on such development at a later date.

Commonwealth Edison Program:

The Edison program consists of comprehensive level 2 PRA's which address the requirements for IPE's and which go beyond those requirements. From the start of the IPE efforts, it was firmly held, within Edison, that the best way to fully and economically extract a viable accident management program from an IPE was to integrate the two efforts from the start and include the accident management program development as a required IPE product.

Given that recognition and stated intent, the entire character of the IPE takes on new focus. First of all, once one commits to using the IPE as the basic tool for accident management, one has committed to using the PRA in an operational mode. Moreover, that operational mode, severe accidents, is, arguably, the most critical in the plant envelope. As a result, one is forced to go to great lengths to insure that the IPE realistically represents the plant.

Secondly, there is an overwhelming need to be as comprehensive as possible in performing the IPE and in extracting insights from the IPE. This requirement also imposes a major requirement for a carefully structured approach to the IPE/AM effort. For Edison, with 12 units at six sites, that need is compounded by the need to insure a consistently high level of quality throughout the program.

Edison has put in place a highly structured IPE/AM program. The program planning and technical guideline documents fill seven volumes at last count. The techniques used to build the IPE's rest on realistic analyses and full acknowledgment of key uncertainties. This results in extensive use of transient
analyses to define plant responses, establish success criteria for systems, and help structure, along with EOP's, the plant logic diagrams. It has also resulted in the integration of the event trees for "front-end" and "back-end" into "plant response trees" which take into account the coupling effects throughout the plant. Lastly, the Edison effort has resulted in carefully formulated techniques for extracting and managing both IPE and AM insights from each key step in the process of performing an IPE.

Edison acknowledges that it would be difficult to establish the framework for an accident management program directly from the IPE insights alone. Therefore, two other, complementary techniques are employed to insure that a total perspective is maintained. The first of these is called a logical-intuitive approach and consists of a logical breakdown, by experienced personnel, of the NRC's five elements of accident management. The breakdown asks questions in an effort to "decompose" the five elements into the fine structure of a formulated program. This process, even if questions are asked and unanswered, serves as a check on the program derived directly from the IPE and offers a framework for assembling that program work. It also serves to identify the need for long-lead time items such as Edison's "SAMSON" neural network artificial intelligence AM tool.

The second technique involves an overview of Edison activities as conducted during emergency exercises. This overview is performed by a highly qualified behavioral scientist who performs a task analysis for selected positions in the emergency plan as it exists now and for those same positions as they would exist given expanded roles consistent with what has been concluded about the five elements of accident management. The task analysis is useful for a variety of purposes such as designing information displays, designing training programs, identifying work loads and organizational weaknesses, and evaluating communications. Also, this provides yet another check on the IPE derived program while adding a strong human factors element to the effort.

Conclusion:

The development of an IPE/AM program taking maximum advantage of the IPE effort has led Edison to a strongly integrated program. We believe that the benefits include realism, completeness, and significant long-term economies.
A FRAMEWORK FOR ASSESSING
SEVERE ACCIDENT MANAGEMENT STRATEGIES

W.E. Kastenberg, G. Apostolakis, V.K. Dhir
D. Okrent, M. Jae, H. Lim, T. Milici
H. Park, J. Swider, L. Xing, D. Yu

University of California, Los Angeles
Los Angeles, California

Accident management can be defined as the innovative use of existing and or alternative resources, systems and actions to prevent or mitigate a severe accident. Together with risk management (changes in plant operation and/or addition of equipment) and emergency planning (off-site actions), accident management provides an extension of the defense-in-depth safety philosophy for severe accidents.

A significant number of probabilistic safety assessments (PSA) have been completed which yield the principal plant vulnerabilities. For each sequence/threat and each combination of strategy there may be several options available to the operator. Each strategy/option involves phenomenological and operational considerations regarding uncertainty. These considerations include uncertainty in key phenomena, uncertainty in operator behavior, uncertainty in system availability and behavior, and uncertainty in available information (i.e., instrumentation).

The objective of this project is to develop a methodology for assessing severe accident management strategies given the key uncertainties mentioned above. Based on Decision Trees and Influence Diagrams, the methodology is currently being applied to two case studies: Cavity flooding in a PWR to prevent vessel penetration or failure, and drywell flooding in a BWR to prevent containment failure.

In general, the key uncertainties involve issues related to phenomena, operator actions, instrumentation and systems availability. The uncertainty in phenomena occur because operator actions change the progression of a severe accident, and introduce new regimes such as temperature or pressure, and new conditions such as the presence or absence of water. As a core melt accident progresses, the geometry change will also contribute to uncertainty. Uncertainties in phenomena exist with respect to the occurrence of steam explosions (both in-vessel and ex-vessel), hydrogen generation and combustion, and heat transfer in these new regimes and under these new conditions.
In addition to the traditional uncertainties in operator and system behavior regarding severe accidents, there is additional uncertainty in attempting to manage a severe accident using innovative means. This occurs because of the uncertain nature of the phenomena mentioned previously, a lack of knowledge regarding the state of the accident progression, and because the operators may not know whether or not their actions have been successful. Moreover, a lack of sufficient information due to damaged instrumentation may lead the operators to the wrong diagnosis and/or action.

When assessing a severe accident management strategy the following five criteria should be considered: the feasibility of the strategy, the effectiveness of the strategy, the possibility of adverse effects, information needs, and compatibility with existing procedures.

Consider flooding the cavity in a PWR to prevent vessel failure. In this case the **feasibility** is essentially a question of whether or not the operators will be able to fill the cavity up to the required level in sufficient time. The **effectiveness** has to do with whether or not the heat transfer is sufficient to keep the molten core in the vessel, given that the water is there on time. A possible **adverse effect** is a steam explosion, should the strategy be feasible but not effective, i.e., the core penetrates the vessel, and finds water in the cavity, which otherwise would not be there. Another **adverse effect** is Steam Generator Tube Rupture even though the strategy is feasible and effective in preventing vessel failure. **Information needs** refers to instrumentation availability, and **compatibility** considers the impact on existing rules and procedures.

**Influence Diagrams** are constructed to define both probabilistic and deterministic dependencies. The **Decision Tree** is then used to structure the decision taking into account feasibility, effectiveness, etc. The evaluation of such a tree would proceed as follows. The risks associated with each endpoint would be determined using **PRA methodology**. This risk might be in terms of early or latent fatalities, population dose, conditional probability of early containment failure, etc. The chance node probabilities would be evaluated using both deterministic and probabilistic methods. For example, the question of feasibility would require the use of Human Reliability Analysis (HRA) and a knowledge of system behavior (e.g., pump capacities, flow rates, etc.). The question of effectiveness would require mechanistic calculations regarding heat transfer, materials behavior etc. The same is true for questions regarding adverse effects.

At the present time, we have evaluated the Decision Trees for both cases described above. The most important considerations involve human actions, primary system failure such as steam generator tube rupture or surge line failure, steam explosions, reactor vessel pressure at breach, and timing of events.
ASSESSMENT OF TWO BWR ACCIDENT MANAGEMENT STRATEGIES

S. A. Hodge   M. Petek
Boiling Water Reactor In-Vessel Strategies (BWRIVS) Program*
Oak Ridge National Laboratory

Summary

A recently completed Oak Ridge effort proposes two management strategies for mitigation of the events that might occur in-vessel after the onset of significant core damage in a BWR severe accident. While the probability of such an accident is low, there may be effective yet inexpensive mitigation measures that could be implemented employing the existing plant equipment and requiring only additions to the plant emergency procedures. In this spirit, accident management strategies have been proposed for use of a borated solution for reactor vessel refill should control blade damage occur during a period of temporary core dryout and for containment flooding to maintain the core debris within the reactor vessel if injection systems cannot be restored.

The proposed strategy for poisoning of the water used for vessel reflood should injection systems be restored after control blade damage has occurred has great promise, using only the existing plant equipment but employing a different chemical form for the boron poison. The dominant BWR severe accident sequence is Station Blackout and without means for mechanical stirring or heating of the storage tank, the question of being able to form the poisoned solution under accident conditions becomes of supreme importance.

On the other hand, the proposed strategy for drywell flooding to cool the reactor vessel bottom head and prevent the core and structure debris from escaping to the drywell holds less promise. Although drywell flooding would preclude bottom head penetration failures and thereby greatly delay the release of debris, the bottom head would eventually fail by creep rupture. This is a consequence of not being able to completely surround the bottom head with water because of the gas pocket that would be trapped beneath the vessel support skirt. Since the drywell vents would have to remain open during and after the flooding process, the ultimate failure of the vessel wall would open a direct pathway for escape of fission products to the atmosphere. This strategy does, however, have potential for future plant designs in which passive methods might be employed to completely submerge the reactor vessel under severe accident conditions without the need for containment venting.

Recent plant experience has included many events occurring during outages at pressurized water reactors. A recent example is the loss of residual heat removal system event that occurred March 20, 1990 at the Vogtle-1 plant following refueling. Plant conditions during outages differ markedly from those prevailing at normal full-power operation on which most past research has concentrated. Specifically, during outages the core power is low, the coolant system may be in a drained state with air or nitrogen present, and various reactor coolant system closures may be unsecured. With the residual heat removal system operating, the core decay heat is readily removed. However, if the residual heat removal system capability is lost and alternate heat removal means cannot be established, heat up of the coolant could lead to core coolant boil-off, fuel rod heat up, and core damage.

A study was undertaken by the NRC and various contractors to identify what information was needed to understand pressurized water reactor response to an extended loss of residual heat removal event during refueling and maintenance outages. By identifying the possible plant conditions and cooling methods that might be used, the controlling thermal-hydraulic processes and phenomena were identified. Controlling processes and phenomena include: gravity drain into the reactor coolant system, core water boil-off, and reflux condensation cooling processes. Important subcategories of the reflux cooling processes include: the initiation of reflux cooling from various plant conditions, the effects of noncondensable gas on reflux cooling, core level depression effects, issues regarding the steam generator secondaries, and the special case of boiler-condenser cooling with once-through steam generators.

Recommendations for assisting staff evaluation of utility capability to effectively respond to a loss of residual heat removal system include: evaluating the capability for using gravity-drain processes and other means of feeding the reactor coolant system without AC power, determining the best options for using secondary system vents and backup sources of feedwater, and determining the capabilities of high point vents for removing air from the reactor coolant system. The primary area where research was needed regarded initiation and continuation of reflux cooling in a reactor coolant system containing air. Specific issues were the primary pressure increase needed to start and maintain the reflux process, and the effects of steam and air migration.

Two basic types of analyses were performed: feed-and-bleed cooling of an open reactor coolant system via the refueling water storage tank or accumulators; and reflux condensation cooling of a closed system using the steam generators as heat sinks (boiler-condenser mode in once-through steam generators). A total of five different reactor plants were evaluated, three for feed-and-bleed and two for condensation cooling. Major conclusions reached in the study are summarized below.

- For the three plants examined (one for each U. S. PWR vendor) all are theoretically capable of establishing a drain path
between the RWST and the RCS. However, the relative elevation difference between the RWST and the RCS which determines how much water is available, can vary significantly from plant to plant.

• Under ideal conditions, for the three plants studied, RWST feed-and-bleed of the RCS could maintain core cooling for as little as 0.4 hours (Waterford) to as much as 18 hours (Davis-Besse) assuming the loss of RHR occurred two days after shutdown.

• Feeding from the RWST and venting steam through a manway extends the time RWST water could be used to keep the core cool. But capability of controlling flow to achieve sufficient inventory needs to be established.

• The accumulators are not a practical source of makeup water as currently configured.

• Environmental heat loss from the RCS will only be a small fraction of even the lowest levels of decay heat.

• Closed RCS condensation cooling via one or more steam generators is a viable strategy to maintain core cooling after loss of RHR. However, there is a possibility that the "steady-state" pressure level in the RCS will threaten the integrity of temporary RCS closures.

• Analysis indicates that the RCS pressure level reached under quasi-stable conditions is dependent on a number of situational and phenomenological conditions. The model developed in this study predicted pressures ranging from 40 psia to 90 psia for H. B. Robinson and 50 psia to 70 psia for Oconee for cases where one or more steam generators were available for heat removal with their secondaries at atmospheric pressure.

• If the RCS pressures exceed the temporary thimble seal threshold pressure, the failure of ten seals could result in core uncovery in about seven hours and failure of all 58 could result in core uncovery in about three hours.

• If borated water addition from the RWST is used as an alternative cooling method and the RCS boils for an extended period of time, boric acid concentration will increase and precipitation could occur in RHR lines if RHR is recovered.

These results provide a better understanding of plant response to events occurring during outages. This understanding will be useful in achieving plant safety improvements in the following areas: operating procedures, training, instrumentation, equipment availability, and risk quantification.
A Model For Calculation of RCS Pressure During Reflux Boiling Under Reduced Inventory Conditions and Its Assessment Against PKL Data

D. E. Palmrose, INEL, EG&G Idaho Inc.,
R. Mandl, Siemens AG-UB KWU

Based on the occurrence of a number of plant incidents during low power and shutdown operating conditions, the USNRC has initiated several programs to better quantify risk during these periods. One specific issue of interest is the loss of residual heat removal (RHR) under reduced coolant inventory conditions. This issue is also of interest in the Federal Republic of Germany and an experiment was performed in the integral PKL-III experimental facility at Siemens-KWU to supply applicable data. Recently, an effort has been undertaken at the INEL to identify and analyze the important thermal-hydraulic phenomena in pressurized water reactors following loss of vital AC power and consequent loss of the RHR system during reduced inventory operation. The thermal-hydraulic response of a NSSS with a closed reactor coolant system (RCS) to loss of residual heat removal cooling capability is investigated in this report. The specific processes investigated include: boiling of the coolant in the core and reflux condensation in the steam generators, the corresponding pressure increase in the reactor coolant system, the heat transfer mechanisms on the primary and secondary sides of the steam generators, the effects of air or other noncondensable gas on the heat transfer processes, and void fraction distributions on the primary side of the system. Mathematical models of these physical processes were developed and validated against experimental data from the PKL IIIB 4.5 Experiment.

Sensitivity studies show which thermal-hydraulic parameters are the most important relative to the critical aspects of the plant response. In the case of boiling in the core and reflux condensation in the steam generators, sensitivity studies show that the secondary side pressure/temperature, the heat transfer mode in the primary side, the number of steam generators with nozzle dams, and the behavior of noncondensables in the pressurizer are the most important factors relative to RCS pressure increase. Conversely, the decay heat level and number of active steam generators has only a second order effect. In addition, the analysis shows that pure reflux condensation in the steam generators provides the lower limit on the pressure increase. That is, the other possible heat transfer regimes are expected to give lower heat fluxes and subsequently higher primary side pressure than pure reflux condensation.

The thermal-hydraulic models were first validated against experimental data from Semiscale Test NC-6. In this experiment, the facility is placed in a pure reflux condensation mode with reduced inventory for one loop with the pressurizer isolated. Four injections of nitrogen gas into the steam
generator inlet plenum were performed followed by a period of steady-state reflux condensation after each injection. The model calculations agree very well with the measured values with errors ranging from 6.5% down to 0.6%.

PKL experiment IIIB 4.5 began from mid-loop conditions for a shutdown maintenance period with a closed RCS. During the time without RHR cooling, the decay heat levels were 1.0 and 0.7 percent of full power. This results in a complex heat transfer process occurred in the steam generator U-tubes due to a countercurrent flow limitation at the U-tube entrance and a two-phase froth level swelling from the reactor vessel into the U-tubes. Unlike the Semiscale experiment, the PKL IIIB 4.5 experiment includes the pressurizer as an active volume in the RCS (it is not isolated and allows for free communication of gas and liquid with the rest of the RCS). Also, all of the coolant loops are available and unblocked. However, only one out of four steam generators had water on the secondary side making it available as a heat sink.

The system of thermocouples and differential pressure detectors throughout the PKL facility were used for: the determination of froth level in the RCS (including the pressurizer), the condensing length in the steam generator U-tubes, and the froth and water column heights in the U-tubes. A water column was present on top of the froth at each decay heat level and was approximately 0.6 to 0.9 m (2 to 3 ft) in height. The two-phase froth mixture entered the U-tubes at both power levels. The liquid in the pressurizer varied between one-fourth and one-half of its volume over the duration of the experiment.

The experimental data indicate a heat transfer coefficient in the primary side of the U-tubes of approximately 360 W/m²·K (63 Btu/hr·ft²·°F) (which is in the range for single-phase heat transfer). In addition to the two-phase effects, the mixing of nitrogen and steam, and the accumulation of liquid into the pressurizer were important effects.

The analytical model can bound the PKL experimental results. With a Nusselt film condensation correlation used in the model, the PKL system pressure is under-predicted by approximately 1.8 bar (26 psia) (model results of 4.8 bar (70 psia) vs PKL pressure of 6.6 bar (96 psia)). By changing to a single-phase heat transfer correlation (Dittus-Boelter), the model will over-predict PKL's pressure by 0.9 bar (model at 7.5 bar (109 psia) vs 6.6 bar (96 psia) for PKL). Clearly, the amount of pressure necessary to support the water column, the degree of frothing that reaches the U-tubes, and the pressurizer holding a fraction of the water inventory has a significant effect on the final RCS pressure. While neither of the correlations predict results within 10% of the experiment, the combination does set upper and lower limits which the RCS pressure must fall between. This means that one can determine bounds for possible RCS pressure that could result from a loss of RHR which are within reasonable limits.
Effects of Hydrogen Generation on Severe Accident Natural Circulation

J. E. O'Brien
Idaho National Engineering Laboratory
EG&G Idaho, Inc.
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SUMMARY

During certain pressurized water reactor (PWR) severe accident scenarios, natural circulation can play an important role in determining the failure location and time-to-failure of reactor coolant system (RCS) components. At the time in the transient when fuel cladding temperatures reach about 1800 K, hydrogen generation due to the steam/zirconium reaction becomes significant. Effects of this hydrogen production on natural circulation flow and heat transfer have been considered in this report. Possible flow disruption due to hydrogen stratification in the reactor vessel and/or hydrogen-induced steam generator blockage were both considered. Long-term hydrogen stratification in the reactor vessel appears to be unlikely due to turbulent mixing associated with high Rayleigh number in-vessel natural convection. Mixing times scales were found to be very short, even when a fully stratified starting condition was assumed. This finding was consistent with experimental results obtained by Westinghouse with a 1/7-scale PWR RCS model.

If mixing of steam and hydrogen is assumed to be rapid and no condensation occurs, analysis of the accident progression can be performed by simply accounting for the properties of the gas mixture as the composition changes. This is essentially what is done by RELAP5 since gas mixtures are treated as homogeneous. Therefore, recognizing the inherent limitations of a one-dimensional representation of a multi-dimensional phenomenon, RELAP5 should provide reasonable estimates of natural circulation effects, system temperatures, failure locations, and component time-to-failure with a steam/hydrogen mixture. If operator actions take place during an accident such that water is reintroduced on the secondary side of a steam generator, condensation would begin, possibly resulting in a sudden enrichment of the gas mixture hydrogen content in the steam generator tubing. This situation could potentially lead to a serious disruption in the natural circulation flow pattern due to U-tube flow blockage.
Primary System Thermal-Hydraulics During the Core Uncovery Phase of the TMI-2 Accident

David Bessette

There is a question of where the primary system of a PWR will fail in a high pressure severe accident scenario. The core heatup process involves a race between whether the hot leg or surge line will fail from PORV flows or buoyancy driven convection of heat or whether the lower head fails from melting and draining away of the core. The TMI-2 accident constitutes an important data point in reaching a determination. Any explanation or model of the process must not be inconsistent with what happened at TMI-2.

There is, however, the question, "What did happen at TMI?." Given that nearly half the core was molten, why were temperatures not experienced sufficient to melt upper plenum structures or fail the hot leg? The initial studies of TMI-2 concentrated on developing an overall description of the events that took place. Also, the studies were carried out prior to the identification of the issue of failure of the surge line or hot leg prior to vessel failure in a high pressure severe accident scenario. Subsequent analyses have been primarily aimed at characterizing the state of core damage and not the thermal hydraulics of the accident.

The TMI-2 accident sequence must, therefore, be examined in terms of energy distribution during the period of core uncovery. This lasted for approximately 1½ hours, from 111 minutes until shortly beyond 200 minutes. The evaluation that follows extends somewhat further, from 101 minutes to 230 minutes, to allow a more complete evaluation of coolant inventory and core behavior.

When the A loop reactor coolant pumps were tripped at 101 minutes into the TMI-2 accident, the primary system immediately transitioned from forced to boiler condenser natural circulation in the A loop. The B steam generator was not removing energy during this time. At the same time, mass and energy were being lost through the PORV, which opened near the beginning of the accident and remained open until 142 minutes.

Core uncovery began at 111 minutes, as evidenced by detection of superheat in the A hot leg. The core re-covered shortly after 200 minutes following HPI actuation. In between, at 174 minutes, operation of a B loop reactor coolant pump caused substantial core cooling.

Natural circulation to the A steam generator essentially ended at 128 minutes. The primary system then began repressurizing until 193 minutes. It appears that the accident was nearly averted at 128 minutes when primary system pressure reached a minimum of 610 psi. Had the A steam generator cooled an additional 3F, primary system pressure would have dropped below 600 psi, the accumulator pressure. Accumulator injection would have reflooded the core before the start of core damage.
Closure of the block valve at 142 minutes terminated the loss of primary system inventory. By this time, however, the core was substantially (i.e. about 1/2) uncovered. The first indication of cladding damage, a containment gas sampling radiation alarm, occurred at this time.

Judging from the amount of time the core was uncovered, the depth of core uncover, and the zircaloy oxidation process, the core had, aside from the subsequent pour at 224 minutes of material to the lower plenum, essentially achieved its final damage state by 174 minutes.

The primary system was repressurizing during the time when the flows exiting the core were high temperature steam and/or hydrogen. The core gases flowed to the voided regions of the primary system. Judging from the flow velocities, it does not appear there were return flows from natural circulation of cooler gas.

Once zircaloy oxidation is substantially underway, it dominates the progress of core degradation due to its highly exothermic nature. Progression of oxidation through the core would appear to be similar to a burn front. The process is an analogue to quench front progression, and modeling may require fine mesh (i.e. 1 mm) nodalization.

The only data obtained from above the core as part of the TMI-2 examination were temperature profiles of two control rod lead screws. These were shielded within flow shrouds. Although one of the lead screws was from over the center of the core, this was not the peak temperature location. Its maximum temperature was 900°C. Considering the heat capacity and total mass of the core exit gaseous flows, and considering the heat capacity of the upper plenum structures, this lead screw temperature is consistent with a simplified analysis.

There was no flow thorough the PORV during the period of substantial core damage, so there was no challenge to the surge line from high temperature burst failure. If the block valve had been opened, however, surge line failure appears likely. Considering the core exit gaseous flows, the heat capacity of the hot leg, and the heat transfer coefficient, hot leg failure was not expected at TMI. Had the accident progression continued, however, failure may have occurred later in time.
SCALING ISSUES FOR A THERMAL-HYDRAULIC INTEGRAL TEST FACILITY

L. Shotkin, NRC; T. Boucher, INEL; M. diMarzo, UMCP

As part of the design certification process for the new AP600 and SBWR reactors, the NRC is evaluating the vendor testing programs, both separate effects and integral tests, to determine if any additional data would be desirable. Particular attention is being focused on integral testing of the interactions of these new passive safety systems under accident conditions.

When considering an integral test facility for a new reactor, such as the AP600, a key question is the appropriate scaling rationale to use in its design. Scaling requirements are based on matching dimensionless parameters for both the model and the prototype. These parameters are derived from nondimensional forms of the thermal-hydraulic conservation equations (mass, energy and momentum). Therefore, typical scaling laws refer only to local phenomena and do not by themselves assure the correct global system behavior. To attempt to assure the latter, all relevant system components must be included in the test facility design and each must be properly scaled. For example, scaling condensation effects in a cold leg does not by itself account for the effect on cold-leg flow of condensation occurring in a steam generator.

Past integral test facilities, such as SEMISCALE, MIST and FIST, have been based on time preservation scaling. These were all full-height, full-pressure, water facilities. More recently, Ishii has developed a more general scaling principle which includes time preservation scaling as a sub-set. For natural circulation, these scaling principles can be illustrated by two simple relations:

\[ \tau = \sqrt{\lambda} \]  
\[ \sigma \tau = 1 \]

where \( \tau \) is the time scale ratio, \( \lambda \) is the length scale ratio, and \( \sigma \) is the power-to-volume scale ratio. The scale ratio is defined as test model divided by full-scale prototype. Equations (1) and (2) assume the same working fluid and pressure between model and prototype. Note that the test facility length and diameter, or volume, are left to the designer.

For time preservation scaling, \( \tau = 1 \). This leads to \( \lambda = 1 \) and \( \sigma = 1 \). That is, the test facility must be full-height and the power-to-volume ratio is preserved. For a test facility with reduced-height scaling, \( \lambda < 1 \), the time-scale is reduced (\( \tau < 1 \)) and the power-to-volume ratio is increased, relative to the full-scale prototype. A motivation for going to reduced-height is to allow larger diameter components for the same total volume, and approximately same cost. A larger diameter might be able to better capture multi-dimensional flow effects and, by nature of closer approximation of prototype length to diameter ratio, more closely simulate single-phase wall friction.
An additional scaling factor is the test facility pressure. The motivation for reduced test facility pressure is to minimize leakage, increase operational flexibility, allow for visualization and possibly to reduce cost. The NRC has sponsored one reduced-height, reduced-pressure test facility at the University of Maryland (UMCP) as part of the MIST program. UMCP concentrated on SBLOCA tests. Test data from UMCP and MIST were comparable when scaled based on system inventory. In addition, UMCP confirmed the major MIST test results and added to our detailed understanding of the Interruption-Resumption Mode (IRM) of cooling. UMCP also showed that a reduced pressure facility can model the full scale prototype if conservation of mass is the dominant conservation statement and energy and momentum account only for secondary effects. For example, for SBLOCA transients, UMCP results showed that the integral system can be considered "closed" and liquid inventory is the key independent variable. For integral systems where phenomena are controlled by the amount of energy available such as LBLOCA (or in general for "open" integral systems) pressure scaling may not be as straightforward.

In 1987, INEL evaluated several scaling concepts for operating PWRs and concluded that most local phenomena would be captured using a full-height full-pressure, same working fluid scaling concept. Recently, INEL performed a similar study for the AP600. Five concepts (all of which would use the same working fluid as the AP600) were considered:

- FHFP (Full Height, Full Pressure)
- FHLP (Full Height, Low Pressure)
- RHRP (Reduced Height, Reduced Pressure)
- RHLP (Reduced Height, Low Pressure)
- Small-Scale

A systematic process was followed in this evaluation and the results were independently reviewed. For the issues related to the AP600 design, the FHFP and FHLP options were the most cost effective. The specifics of full height scale may still be considered in light of the possible CCFL occurrences in the AP600 configuration. A reduced aspect ratio of the model may offer some advantages by better simulating the frictional versus gravitational force balance.
BWR STABILITY ANALYSES AT BNL*

W. Wulff, H. S. Cheng, U. S. Rohatgi and A. N. Mallen
Brookhaven National Laboratory
Upton, NY 11973

The March 9, 1988 instability at the LaSalle County-2 BWR power plant at Seneca, IL was simulated with BNL's Engineering Plant Analyzer (EPA) for the purpose of demonstrating that the EPA is suitable for simulating large-amplitude, limit-cycle power and flow oscillations. It was shown in Fall of 1988, by comparing all the available plant data from the STARTREC recording system of LaSalle-2 with EPA simulation results, that the EPA reproduces the LaSalle-2 oscillations without the use of stabilizing or destabilizing model or parameter modifications.

The power vs. flow map of the LaSalle-2 plant was also reproduced at five lines of constant control rod positions. The LaSalle-2 stability boundary was established with the EPA and confirmed within ±15% accuracy by comparing the EPA results with the results of the frequency domain code LAPUR of Oak Ridge National Laboratory. Comparisons of EPA simulation results with plant data from three Peach Bottom Stability tests show an agreement, based on mean and standard deviation, of -10±28%, -1±40% and +28±52% (low power) in the gain of the pressure to power transfer functions. This demonstrates that the time domain code HIPA in the EPA is capable of simulating instabilities.

The EPA simulation of the LaSalle-2 instability identified low core flow, caused by the dual recirculation pump trips, low feedwater temperature due to the inadvertent feedwater heater isolation, and power peaking as a result of fuel burn-up, to be the three causes for the instability; the absence of any one of the three causes would have prevented the instability. Moreover, of the fourteen questions posed by the NRC, the EPA served to answer all ten questions related to core-wide, in-phase power oscillations.

By simulating the LaSalle-2 instability with postulated scram failure, it was shown that the power oscillations peak at thirteen times rated power; the peaks can reach sixteen times rated power, if all feedwater preheating is lost after a turbine trip, while feedwater flow is maintained to maintain coolant inventory.

The EPA simulated ten ATWS scenarios under conditions of existing oscillations, or conditions inducive to instability, showing that the time it takes the suppression pool to reach its temperature limit can vary between 4.3 minutes (Turbine Trip with

*This work was performed under the auspices of the U.S. Nuclear Regulatory Commission.
out Bypass and no feedwater pump trips) and infinity (with Boron injection). Three additional transients have been simulated to show that restarting both recirculation pumps at the occurrence of an instability leads to scram, however in the case of scram failure, the power oscillations subside after peaking shortly at approximately five times rated power.

Modeling parameters were ranked in the order of their significance to stability. The influences of spatial increments and of time steps in the numerical solution techniques have been quantified, and the effects from using different integration routines have been assessed. It has been determined that the power amplitude is underpredicted by the factor of 3.5 if the dynamic simulation of the balance of plant is omitted. It was also shown that thermohydraulic instabilities cannot be simulated by imposing any boundary conditions, because the instabilities are self-induced and strongly impacted by closed loop feedback and resonance mechanisms.
SMALL BREAK LOCA RELAP5/MOD3 UNCERTAINTY QUANTIFICATION:
Bias and Uncertainty Evaluation for Important Phenomena
by
Marcos G. Ortiz, L. Scott Ghan, Judith Vogl
Idaho National Engineering Laboratory
EG&G Idaho Inc.
Idaho Falls, IDAHO 83415-2404

The Nuclear Regulatory Commission (USNRC) revised the Emergency Core Cooling System (ECCS) licensing rule to allow the use of Best Estimate (BE) computer codes, provided the uncertainty of the calculations are quantified and used in the licensing and regulation process. The NRC developed a generic methodology called Code Scaling, Applicability and Uncertainty (CSAU) to evaluate BE code uncertainties. The CSAU methodology was demonstrated with a specific application to a Pressurized Water Reactor (PWR), experiencing a postulated Large Break Loss-of-Coolant Accident (LBLOCA) [1,2]. The current work is part of an effort to adapt and demonstrate the CSAU methodology to a Small Break (SB) LOCA in a PWR of B&W design using RELAP5/MOD3 as the simulation tool1. The subject of this paper is the Assessment and Ranging of Parameters (Element 2 of the CSAU methodology), which determines the contribution to uncertainty of specific models in the code.

The specific models were chosen based on the Phenomena Identification and Ranking Table (PIRT, step 3 of the CSAU methodology) that was defined, earlier in the program, by two independent panels of experts [3]. The PIRT lists the phenomena believed to most strongly affect the transient scenario, as determined by their impact on a prescribed safety criterion. For the SBLOCA in a B&W plant, the selected criterion is the minimum liquid level in the reactor vessel.

Eight phenomena from this PIRT, regarded as highly important in determining the critical outcome (primary safety criterion) of the simulation, were chosen for performing sensitivity calculations. In this paper, we have selected the four phenomena of highest importance to demonstrate the evaluation of bias and uncertainty; these are the break flow, natural circulation, decay heat, and the temperature of the high pressure injection flow (HPI). We also present and discuss the results of the set of sensitivity calculations.

To determine the bias and uncertainty in the simulation of break flow, the critical flow model in RELAP5 required assessment using experimental data4. Time domain data were obtained and recast in the form of critical mass flow versus pressure, from which a comparison to the code model could be made. This produced both a bias and uncertainty in calculated breakflow as a function of system pressure.

1 Work supported by the U. S. Nuclear Regulatory Commission, Office of Research, Under DOE Contract No. DE-AC07-76ID01570.
For natural circulation the bias and uncertainty in the interfacial drag model used in the code was selected, as it plays the dominant role in calculating natural circulation. The empirical correlation used is based on a data base of over 1600 points. The bias and uncertainty were readily determined from this data base.

In the case of decay heat, the uncertainty was obtained from the ANS standard, supplemented by the work of Tran and Schrock. The decay heat is entered into the code input as a table of power versus time. The temperature of the HPI coolant is also a code input, and for the sensitivity study was set to 35°F for the low end (boron precipitation consideration) and 110°F for the high end (technical specification limit).

The sensitivity calculations have yielded interesting and useful results, including the conclusion that the observed bias in breakflow can lead to improved accuracy in SBLOCA calculations. Results from the key sensitivity calculations will be described and discussed in the context of the CSAU study.


Thermally-induced stresses in a reactor pressure vessel wall, as a result of high-pressure safety injection, are an essential component of integrated risk analyses of pressurized thermal shock transients. Limiting cooldowns arise when this injection occurs under stagnated loop conditions which, in turn, correspond to a rather narrow range (in size) of small-break loss-of-coolant accidents (Theofanous et al. 1989). Moreover, at these conditions, the flow is thermally stratified, and in addition to the global cooldown, one must be concerned about the additional cooling potential due to the downcomer plumes formed by the cold streams pouring out of the cold legs. In the NRC's Integrated Pressurized Thermal Shock (IPTS) study, this stratification was calculated with the codes REMIX/NEWMIX (Iyer & Theofanous, 1991a, 1991b). A comprehensive comparison with all available experimental data has recently been compiled (Theofanous & Yan, 1991). The stress analysis using this input was carried out at ORNL using a one-dimensional approximation with the intent to conservatively bound the magnitude of thermal stresses (Chevreton & Ball, 1984).

Recent publications (Neubrech et al., 1988; Geiß, 1987) compared the measured thermal stresses in PTS simulations in the HDR facility to predictions in the 1-D approximation and found the latter to be highly non-conservative (by almost a factor of x2). In particular, the stress field was found to be strongly asymmetric and in favor of the axial component. These comparisons are interesting because the experimental stress data are unique (in an integral experiment of PTS context) and the tests obviously, at first glance, prototypic.

This measured, strongly asymmetric, stress field is related to the elongated plume structure and should not have been unexpected. Consider, for example, the thermal stresses in a plane wall assigned a cold, uniform through the wall, elliptical spot. For a temperature difference of $\Delta T$, and major and minor ellipse semiaxis $b$ and $a$ aligned with the $y$ and $x$ coordinates, respectively, we have

\[
\sigma_x = -E \left\{ \frac{a}{a+b} \right\} \alpha \Delta T \tag{1}
\]

\[
\sigma_y = -E \frac{b}{a+b} \alpha \Delta T \tag{2}
\]

Thus

\[
\frac{\sigma_y}{\sigma_x} = \frac{b}{a} \tag{3}
\]

and the stress asymmetry increases with the elongation of the ellipse. On the other hand, the through-the-wall temperature gradient, in this example, would be zero, and so the calculated stress in the 1D approximation would also be zero.

The discrepancy in the above example is clearly an exaggeration; it serves to vividly illustrate, however, that the potential concern is legitimate. What remains to be done is to explore
quantitatively the impact of this concern on reactor predictions and to examine under what conditions, if any, the 1D treatment is adequate. The reason for this latter aspect is that in the scope of an IPTS study a very large number of wall stress calculations need to be carried out to properly sample the space of uncertainty in the key parameters, and transients, that have to be considered; as a consequence, a full 3D treatment becomes rather impractical. The purpose of this paper is to provide some results relevant to these goals.

We begin by discussing similarity of stratification and associated fluid thermal transients in the downcomer. We demonstrate that the particular HDR test (T32.18) chosen by Geiβ(1987) and Neubrech et al. (1988) suffers from gross dissimilarity to US reactor conditions. We find another HDR test, T32.34, to provide a very close simulation; unfortunately, no stress/strain data have been published for it. REMIX results are found to be in excellent agreement with both of these tests. Next, we develop finite element models for 1D and 3D treatments of the RPV and demonstrate that the 3D model accurately depicts the measured stress/strain fields. As expected, we find that the 1D treatment, based on the downcomer plume temperature at its origin as per IPTS study, is non-conservative for a significant fraction of the downcomer area (down to ~7 cold-leg diameters below the cold-leg centerline). By contrast for test T32.34, this area of non-conservative behavior is predicted to shrink down to an axial distance of only 4 cold-leg diameters. The same structural model is applied to the Calvert Cliffs geometry and thermal-hydraulic conditions similar to those of test T32.34. We find that the 1D treatment is even more appropriate than in test T32.34; only an area down to 2 cold-leg diameters is affected (i.e., the axial stress being greater than that computed in the 1D IPTS approximation); that is, the 1D treatment is entirely adequate. Finally, we generalize these structural analysis results to a simple, generalized similarity map that defines the regions of potential difficulty with the 1D IPTS prescription.

REFERENCES
This report contains summaries of papers on reactor safety research to be presented at the 19th Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel in Bethesda, Maryland, October 28-30, 1991. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, USNRC. Summaries of invited papers concerning nuclear safety issues from U.S. government laboratories, the electric utilities, the Electric Power Research Institute (EPRI), the nuclear industry, and from the governments and industry in Europe and Japan are also included. The summaries have been compiled in one report to provide a basis for meaningful discussion and information exchange during the course of the meeting, and are given in the order of their presentation in each session.

Reactor safety research
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