NUREG/CP-0047

Transactions of the Eleventh Water Reactor Safety Research Information Meeting

To Be Held at
National Bureau of Standards
Gaithersburg, Maryland
October 24-28, 1983

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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Date Published: September 1983

Compiled by: Stanley A. Szawlewicz, Consultant

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ABSTRACT

This report contains summaries of papers on reactor safety research work to be presented at the 11th Water Reactor Safety Research Information Meeting. The meeting will be held at the National Bureau of Standards in Gaithersburg, Maryland, October 24-28, 1983. The summary reports highlight the programs and results of nuclear safety research work sponsored by the Office of Nuclear Regulatory Research, USNRC. Summaries of invited papers are also included. The latter represent work on reactor safety research conducted by the electric utilities through the Electric Power Research Institute, the nuclear industry, and various government and industry organizations in Europe and Japan. The summaries have been compiled in one report to facilitate discourse and the open exchange of information during the course of the meeting.

TRANSACTIONS OF THE
ELEVENTH WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING

to be held at the

NATIONAL BUREAU OF STANDARDS
Gaithersburg, Maryland

October 24-28, 1983

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PREFACE

This report contains summaries of papers to be presented at the Eleventh Water Reactor Safety Research Information Meeting to be held October 24-28, 1983 at the National Bureau of Standards in Gaithersburg, Maryland. The summaries are indexed in the order of their presentation in each session. However, some of the speakers did not submit written material for inclusion in this report so that the table of contents does not represent the total agenda of papers to be presented at the meeting. A completed agenda will be distributed at the meeting.

Semiscale Loss-of-Power Test Results

John S. Martinell

EG&G Idaho, Inc.

The Semiscale Program and Test facility are located at the Idaho National Engineering Laboratory, and operated by EG&G Idaho, Incorporated for the U.S. Department of Energy. The system is a small-scale model of the primary coolant system of a pressurized water reactor (PWR) nuclear generating plant. It incorporates the major components of a PWR and is capable of attaining typical operating pressures and temperatures.

An experimental program designed to provide data from loss-of-power transients was conducted in Semiscale in FY-83 in support of programs sponsored by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. This Semiscale program provides information relative to resolution of the station blackout safety issue through validation of thermal-hydraulic computer codes used in accident sequence analyses. Description of the experimental objectives, scenarios, and results is presented below.

Research conducted in support of the series identified three major issues which were used to develop experiment objectives: 1) evaluation of transient signature for loss-of-offsite power only, and with coincident system failures; 2) evaluation of recovery procedures for loss-of-offsite power with coincident failures; and 3) assessment of thermal-hydraulic computer codes with respect to phenomena produced during loss-of-offsite power with coincident failures. A series of five experiments were conducted with variations in initial conditions and scenarios, including

1. Loss-of-offsite power with normal recovery (S-PL-1).

2. Loss-of-offsite and on-site ac power (station blackout) with concurrent failure of secondary heat removal, with no recovery (S-PL-2).

3. Station blackout with concurrent failure of secondary heat removal, delayed return of on-site ac power, with primary feed and bleed recovery (S-PL-3).

4. Loss-of-offsite power with concurrent 5% pump suction break, with ac-powered safety injection, accumulator, and secondary heat removal recovery (S-PL-4).

5. Station blackout with concurrent failure to SCRAM, with secondary heat removal recovery (S-PL-7).

Evaluation of transient signatures from Experiments S-PL-1, S-PL-2, and S-PL-3 indicate that three characteristic phases occur during loss-of-offsite and station blackout events. The first phase is characterized by single-phase natural circulation in the primary system at relatively constant pressure, with secondary boiloff heat rejection. Assuming loss of secondary heat removal, the second phase results in primary fluid heatup and expansion, with the primary pressure increasing to the safety relief setpoint. The third phase results in primary inventory depletion through the safety relief valves, leading to core uncovery. Provided that on-site power is restored early in the third phase and safety injection equipment operates, the extent of core uncovery can be mitigated through feed and bleed in the primary system.

The transient signature resulting from the S-PL-4 experiment was qualitatively similar to that resulting from previous Semiscale 5% cold leg breaks. In this case, break flow dominates the response with depressurization in the primary, and mass depletion resulting in slight core uncovery. The transient signature from the S-PL-7 experiment was characterized by initial primary fluid heatup, expansion, and pressurization, followed by single-phase primary natural circulation at slowly decreasing pressure. Core power was rejected via secondary boiloff to a stable inventory using auxiliary feedwater for makeup.

The recovery procedures simulated during S-PL-1 and S-PL-7 indicated that use of auxiliary feedwater was effective in removing core power and decay heat for stabilization and recovery of the plant. The use of delayed primary feed and bleed without access to secondary heat removal during S-PL-3 was effective in bringing the plant to a stable, two-phase condition in the primary system. Timing of the initiation of primary feed and bleed, and plant characteristics for power-operated relief valve (PORV) flow capacity and safety injection flow capacity influence the ability to achieve stable conditions without resulting in core uncovery. Access to ac-powered safety injection capability and passive accumulators resulted in effective recovery from the pump suction break following core uncovery during S-PL-4.

A summary of the phenomena produced from these experiments that are useful for the assessment of thermal-hydraulic computer codes includes:

1. Primary single-phase natural circulation with secondary auxiliary feedwater boiloff for rejection of core power and decay heat

2. Primary single-phase natural circulation with secondary inventory boiloff and depletion for rejection of decay heat.

3. Primary heatup, expansion, and pressurization due to imbalance in core power input and secondary heat rejection

4. PORV critical flow influence on primary mass inventory and distribution.

5. Primary two-phase natural circulation

6. Primary fluid condensation and flooding in the steam generator tubes

Use of this phenomenological data base for posttest assessment of the RELAP5 computer code is currently in progress. Results to date indicate that for the range of phenomena produced in this series, the RELAP5 code is capable of accurately predicting the response during loss-of-power transients. The data are available for similar assessment of other codes, which, along with RELAP5, can then be used to predict the outcome of various loss-of-power scenarios to support the resolution of the station blackout safety issue.

Semiscale Steam Generator Tube Rupture Test Results

Richard A. Dimenna

EG&G Idaho, Inc.

The Semiscale Program and Test facility are located at the Idaho National Engineering Laboratory, and operated by EG&G Idaho, Inc., for the U.S. Department of Energy. The system is a small-scale model of the primary coolant system of a pressurized water reactor (PWR) nuclear generating plant. It incorporates the major components of a PWR and is capable of attaining typical operating pressures and temperatures.

An experimental program designed to provide data from steam generator tube rupture is presently being conducted in the Semiscale Mod-2B System in support of programs sponsored by the U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research. The test series will provide computer code verification data for a spectrum of tube rupture events including various break sizes (modeling number of tubes ruptured) and various operator safety actions in response to the rupture. The test series will also be used to characterize accident signatures and plant response to typical emergency recovery procedures.

Each test in the Steam Generator Tube Rupture (SG) Test Series is designed in two parts. The first part will characterize the signature of the tube rupture event without operator intervention. In general, a 10-min period immediately following the rupture is used to represent accident identification time, during which only automatic safety functions (SCRAM, steam generator isolation, primary coolant pump trip, etc.) are allowed to occur. A spectrum of break sizes corresponding to the number of tubes ruptured will provide data on plant sensitivity during this period. The second part of the test will characterize particular operator response actions once the tube rupture signature has been identified. The first tests in the series employ single operator actions to test the plant response to individual recovery mechanisms. The principal recovery procedure employed throughout the test series is steam and feed of the unaffected steam generator. Additional recovery actions such as termination of high-pressure injection system (HPIS) flow or operation of the pressurizer power-operated relief valve (PORV) are tested in conjunction with steam and feed during the various tests in the series. Tests planned for the latter part of the series will include compounding events such as loss-of-offsite power or main steam line break.

Currently, the first three tests in the SG series have been completed. These included a one-tube rupture and two five-tube ruptures. Only the first two of these tests, S-SG-1 and S-SG-2, will be addressed here.

S-SG-1 represented a scaled, double-ended offset shear of a single steam generator tube located on the cold leg side of the tube sheet. The test included typical plant protection system response in the form of a low pressurizer pressure SCRAM, closure of main steam valves, and, upon safety injection system signal, initiation of HPIS/charging flow, termination of main feedwater flow to both steam generators, initiation of auxiliary feedwater flow, and primary coolant pump trip into a normal coastdown.

A 10-min accident identification period was simulated, during which no operator response was allowed. At the end of the 10-min period, a limited recovery operation commenced in the form of isolating the affected steam generator and beginning to steam and feed the unaffected steam generator. Following 3000 s of transient time, HPIS flow was terminated to study the effect of this action independently of other operator responses. The test was terminated at 5000 s when the primary system and affected steam generator secondary system pressures were nearly equal, and below the steam generator relief valve setpoint.

S-SG-2 represented a similar rupture of five steam generator tubes. The plant response during the first 10-min of the test following the same sequence of events as SG-1 (only automatic plant protection system actions were simulated). This 10-min period provided a comparison, therefore, between the one- and five-tube rupture events. Following the 10-min identification period, a plant recovery was again simulated by isolating the affected SG and commencing a steam and feed of the unaffected steam generator. In addition, pressurizer PORV operation was simulated in this test. PORV cycling was allowed only until the pressurizer had filled ~3/4 full with water. Following the termination of PORV operation, steam and feed continued as the only accident mitigating action. HPIS flow was terminated on the simultaneous satisfaction of pressurizer level and vessel upper head level requirements. The test was terminated at about 8000 s, when the primary system pressure had decreased to the accumulator injection setpoint.

Tests S-SG-1 and -2 both showed that the Semiscale plant could be recovered from a scaled steam generator tube rupture event with a minimal response from the operator. Recovery included a termination of break flow and reduction of primary system pressure to a specified value below the affected steam generator relief valve setpoint. A comparison of the two tests showed similar system responses in that the primary system pressure dropped rapidly causing a plant SCRAM, steam generator isolation, pump trip, and safety system injection. However, the two tests also demonstrated fundamentally different responses related to the difference in the magnitude of the break flow. S-SG-2, with a large break flow, resulted in emptying the pressurizer and reaching saturation conditions in much of the plant. Significant voiding was established in the vessel upper head and the primary side of the affected steam generator U-tubes. The steam generator U-tube voiding influenced the system pressure throughout the transient. S-SG-1, on the other hand, showed a smaller loss of primary system inventory because the HPIS more effectively compensated for the smaller break flow. As a result, the pressurizer never emptied completely and the pressurizer continued to control system pressure.

The results of the first two tests in the SG test series have indicated fundamental differences in plant response to a steam generator tube rupture as a function of break size. RELAP5 calculations reproduced the qualitative aspects of the differences as reflected in primary system pressure response. Future tests will explore other typical operator actions such as pressurizer auxiliary spray or restart of primary coolant pumps. In addition, the ramifications of break size will be investigated to identify the need for future testing of this parameter.

SALIENT RESULTS OF LOBI-MOD1 TESTS, AND PROGRAMME PLANS FOR THE LOBI* MOD2 FACILITY

W. L. Riebold
Commission of the European Communities
EURATOM-Joint Research Centre, Ispra Establishment

The LOBI research programme is being executed in the Ispra Establishment of the European Communities (E.C.) in the framework of an R&D contract between the Bundesminister Fuer Forschung Und Technologie of the Federal Republic of Germany and the Commission of the E.C. It represents an important commitment in the Water Reactor Safety Research programme of both contract partners and a multinational cooperation in this field.

The general strategy of the LOBI research programme is oriented towards the establishment of an extensive experimental data base for the development of analytical models and the verification of large system computer codes used for the safety analysis of LWRs.

The high pressure, integral system LOBI test facility is a 1:700 scale model of a 1300 MWE KWU PWR. The four reactor loops are represented by two active experimental loops designed for normal PWR operating conditions.

The design of the experimental facility was fundamentally aimed at achieving reactor-typical transient responses under a variety of accident conditions to be simulated. To this end, design compromises had to be accepted between the scaling criteria to be applied, technical feasibility and operational needs. Those compromises hold especially for components where diverse governing physical phenomena prevail during the evolution of the transient being reproduced. Examples are the downcomer gap width effect on the core flow behaviour on the one hand and the CCFL phenomenon, and hence the ECC bypass behavior, on the other. Also, the upper-plenum-to-upper-downcomer bypass effect on the loop seal clearing behaviour under intermediate and small break conditions.

Compromises also had to be made for components having design or operating characteristics different from those of their counterparts in the reference plant. Examples are the electrical heater rod bundle and the main coolant circulation pumps.

Any scaling-procedure-induced distortions limit the extent to which results from scaled-down experimental facilities can be directly extrapolated to real plants or directly used for their safety assessment.

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*LOBI = LPWR Off-Normal Behaviour Investigations

The present paper summarizes salient results of large break loss-of-coolant accident (LB-LOCA) tests performed in the LOBI-MOD1 facility. An assessment of some scale- and simulation-dependent phenomena observed in these tests has been carried out and has shown the overall validity of scaling and simulation criteria adopted in the facility design. Where appropriate, the possible impact of identified scaling distortions upon small break transients has been evaluated and the corrective measures applied are highlighted.

Short descriptions are given of the small break loss-of-coolant experimental programme to be performed in 1984/85, and of a proposed special transients experimental programme. The code assessment and the instrumentation development activities carried out within and in support of the LOBI activity are briefly outlined.

BWR Full Integral Simulation Test (FIST)

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General Electric Company

The Full Integral Simulation Test Program is a three pronged approach to the development of best-estimate analysis capability for BWR systems. An experimental program is underway in a new single bundle system test facility to extend the large break LOCA data base to small breaks and operational transients. An analytical method development program is underway to extend the BWR-TRAC computer code to model reactor kinetics and all major interfacing systems, to improve application modeling flexibility and to reduce computer running time. A method qualification program is underway to test TRACB02 against experiments in the FIST facility. The recently completed Phase 1 period included a series of LOCA and power transient tests and the successful analysis of a large and a small break LOCA test with TRACB02.

The FIST test facility is an integral system capable of full power steady state operation and real time simulation of transients. Combined with a full size electrically heated bundle, it is a full height simulation of the reactor vessel and internals, with scaled regional volumes, and includes all major interfacing systems and automatic trip signals. This provides full scale values for fluid conditions, velocities and static heads. Factors considered in the design of the facility are the BWR design features important to thermal-hydraulic performance. Since each fuel bundle in a BWR is individually channeled, the absence of cross flow in the core region means thermal-hydraulic conditions within the core are accurately simulated by a single bundle with boundary conditions imposed at the inlet and outlet plenums.

Large break, steam line break, and two small break LOCA tests have been completed. In addition, three power transient simulations and determination of the natural circulation flow map were completed. The large break test results, which are similar to a test from the Two-Loop Test Apparatus (TLTA) Program, show heat-up in the 700-800°F range and prompt quenching by the ECC Systems. The hydraulic response during the refill period following the quenching show, as expected, that the FIST design has overcome the geometric scaling compromises experienced in the TLTA and is somewhat affected by overscaled wall stored heat. The refilling response is slower than the corresponding BWR system, but does not adversely affect effective core cooling.

The small break LOCA shows heat-up in the 900°F range, similar to TLTA tests. Although, as expected, there is no bundle heat-up in the steam line break and the power transient tests, they provide useful thermal-hydraulic information for model comparisons.

The General Electric version of BWR-TRAC, TRACB02, which includes the BWR component models and phenomena models developed under the BWR Refill-Reflood Program, was used to analyze the FIST facility response in two LOCA tests. Particular attention in the system definition and application modeling was given to the lower plenum, two-phase level tracking, vessel stored heat, flow loss coefficients, and break geometry. The pre-test analysis of the large break LOCA tests is found to represent the observed controlling thermal-hydraulic phenomena very well. The system definition modeling correctly handles lower plenum flow split performance, and the resulting prediction of core flow and liquid inventory lead to representative thermal performance in the bundle. Additional detail in the vessel stored heat model and break geometry was added in the system definition modeling for the small break LOCA test. Again, the pre-test analysis is found to represent the observed thermal-hydraulic phenomena throughout the system, and the thermal response of the bundle, very well.

In summary, the method development program is providing improved system definition and modeling flexibility, as well as improved running time. The data base has been extended to a wider range of system response situations, and the testing of BWR-TRAC against LOCA tests has demonstrated its capability to handle these transients. Further work is planned for LOCA and operational transient tests, and additional qualification analysis with TRACB02.

FLECHT-SEASET Natural Circulation Systems Tests and Analysis

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E. R. Rosal, S. D. Rupprecht, B. R. Sinwell

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Since the accident at Three Mile Island, an increased interest has developed in single-phase, two-phase, and reflux condensation natural circulation cooling modes for a PWR which are expected to be the primary means of decay heat removal following a small break LOCA or an operational transient. A series of low pressure natural circulation experiments were performed to gain an improved understanding of these decay heat removal modes in the FLECHT-SEASET Systems Test Facility.

The experiments were performed on an integral system test facility which modeled a 4-loop PWR. The elevations associated with the facilaity were scaled 1:1 in order to preserve the prototypical gravity heads which drive the natural circulation flow, and the remaining system dimensions were power/volume scaled with a scale factor of 1:307 [1].

All tests were begun from a liquid solid forced circulation cooling mode at 2% decay power with the secondary side of the generators in the boiling mode with a primary pressure of 140 psia. Steady-state single-phase natural circulation was first established. T wo-phase and reflux condensation cooling modes were subsequently obtained by draining mass from the primary system. The ensuing system flow and pressure characteristics as a function of primary mass inventory were comparable to Semiscale [2] and PKL [3] natural circulation tests.

A series of non-condensible helium gas tests were also performed during each mode of natural circulation cooling. During two-phase natural circulation, the non-condensible gas blocked off selected U-tubes, and the system consequently pressurized and reverted back to a single-phase mode of natural circulation. The non-condensible gas initially collected on the downhill side of the U-tubes during reflux condensation, forming active passive heat transfer regions in the steam generator U-tubes. These active passive regions are similar to those observed by Hein [4].

In determining PWR natural circulation cooling modes, loop flows, and heat sink capability, the heat flux distribution in the steam generators is needed. Therefore, a method was developed for calculating the heat flux behavior in the full height U-tube steam generators. The basis of the method was an axial energy balance performed on the primary fluid in the steam generator U-tube during single-phase operation. Each generator was separated into three groupings of U-tubes, each with its own primary fluid temperature instrumentation. A least-squares curve fit was applied to these primary fluid temperatures providing temperature estimates at any point along the tubes.

The Dittus-Boelter correlation [5] was used with the calculated heat fluxes to compute the inside wall temperatures. The outside wall temperature was then calculated from the inside temperature and heat flux. At each wall thermocouple location, a thermocouple correction curve (calculated minus measured temperatures versus heat flux) and a "boiling curve" (calculated wall temperature minus Tsat versus heat flux) were developed. This data base formed the basis for calculating the steam generator heat flux distributions during two-phase primary conditions using only measured wall temperatures and secondary side saturation temperatures. Plots of primary and secondary fluid temperatures along with calculated and measured wall temperatures show that subcooled nucleate boiling and partial boiling regions exist on the secondary side and can be characterized by the local wall superheat. Reverse heat transfer (secondary to primary) was also seen to occur in the downhill side.

Heat flux distributions were calculated using the method described above for primary side single-phase, peak two-phase flow, and reflux condensation. The single-phase heat flux calculation typically exhibit the highest heat fluxes near the tubesheet on the uphill side and regions of reverse heat transfer above the tubesheet on the downhill side. In most two-phase primary flow cases, the calculated heat fluxes indicate a linear decrease along the tube length, with maximum value at the inlet side at the tube sheet. The reflux condensation plots strongly suggest the presence of a liquid film increasing in thickness near the bottom of each tube leg and thinnest at the top. It was seen that the heat flux was reduced at the tube sheets, contrary to the single-phase case, and highest at the U-bend. The nearly symmetrical reflux plots were supported by independent reflux meter readings indicating a near equal amount of condensate formed on the uphill and downhill sides of the generators.

Representative heat flux distributions before and after the injection of noncondensible helium gas were also calculated. Before injection, the tubes were in a reflux condensation mode with the highest heat transfer rate near the top of the bundle. After injection, the upper region of the bundle had a reduced heat flux, while the lower region had increased heat flux in order to reject a constant amount of energy. This shift in heat flux distribution was expected due to the buoyancy of the helium and indicates that the heat load in the generator will adjust to accommodate the noncondensible gas.

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REFLOOD CODE DEVELOPMENT WORK IN JAERI

Yoshio MURAO, Jun SUGIMOTO, Tsutomu OKUBO,
Tadashi IGUCHI, Hajime AKIMOTO, Kazuharu OKABE

Japan Atomic Energy Research Institute

1. Introduction

In the safety evaluation of the reflood phase of PWR, the empirical correlations on the reflood heat transfer and the carry-over rate have been used. For obtaining the flexibility of application to safety analysis, we are developing a computer code (called REFLA code) for the reflood phase based on the physical understanding of the phenomena. The schematic diagram of our study plan is shown in Fig.1. The REFLA code will be developed by the following steps:

(1) One-dimensional core code (REFLA-1D(core)),
(2) One-dimensional system code (REFLA-1D), and
(3) Integral system code (REFLA).

The latest version of the one-dimensional core code is REFLA-1D/MODE3. This version includes some new improvements in the core thermo-hydrodynamic models, which were developed with the JAERI small scale reflood test facility and the FLECHT facilities. One of improvements is in the correlation for the slip velocity in the two-phase flow.

In the dispersed flow region, two types of hydrodynamic models, i.e. Case 1 and Case 2, are introduced. In both cases, the liquid accumulation regions are assumed to exist above the quench front. The length of this region is now set to 0.3 m. Above the liquid accumulation region, there exists the normal dispersed flow region in Case 1, whereas the other liquid accumulation region in Case 2. In this region, the void fraction has a value between the values corresponding to the slug flow and the dispersed flow due to the integration of dispersed droplets.

The one-dimensional system code, REFLA-1D, calculates the overall system behavior in addition to the behavior in the core. In this code, the steam binding is assumed to be induced by single phase steam flow through a loop having a constant K factor.

The integral system code, REFLA, is an advanced system code in which the radial power distribution effect (local power effect) on the heat transfer is considered. The hydrodynamic behavior is assumed to be radially uniform, since the flow can easily communicate in horizontal direction. The basic models in the system components are based on the results of the CCTF test. In the code, four primary loops are considered and the variable equivalent K factor is used instead of a constant K factor of REFLA-1D. The pressurization of the pressure vessel due to the pressure loss in the broken cold leg nozzle is also considered.

2. Calculational results

In Figs. 2 to 4, indicated are the results of some simulation calculations with the REFLA-1D core code. The difference of the thermal responses of Case 1 and Case 2 seems to be small. The hydrodynamic responses of Case 1 and Case 2 ,however , are clearly different from each other . In Case 1, the local void fraction is almost unity until the quench front arrives near the region of concern, while in Case 2, the local void fractions at every location begin to decrease just after steam flow is developed and become noticeably smaller value than unity. The Case 1 appeared at lower flooding rate than 4 cm/s in FLECHT low flooding rate test and FLECHT SEASET test. In CCTF tests and JAERI small scale reflood tests, however, the Case 2 appeared even when the flooding rate in the low pressure injection period was about 2 cm/s. This difference can be attributed to the multi-dimensional effect. The water entrained by steam flow can be separated at the top of the core and reflux into the core if the core is wide enough or has cold structures which allow the separated water to form the falling liquid film.

Verification calculations with the CCTF test data are presented separately. These revealed that the one-dimensional core model is practically applicable to CCTF tests with the fairly wide core.

3. Summary

The status of the reflood code development is described. Although some improvements are neccesary, practically this one-dimensional reflood model can be used for prediction of reflood phenomena in reactors.

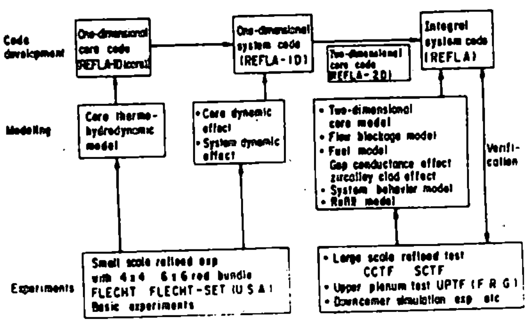

Fig. 1 Schematic diagram of research plan

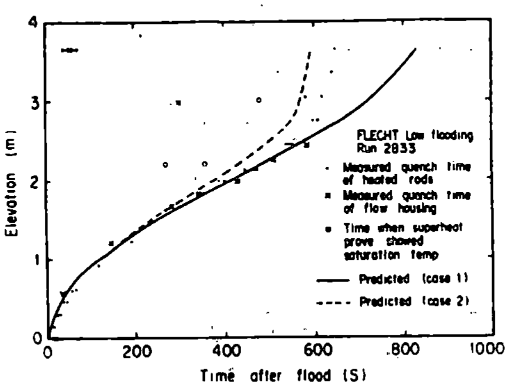

Fig. 3 Effect of water accumulation above quench front on quench envelope

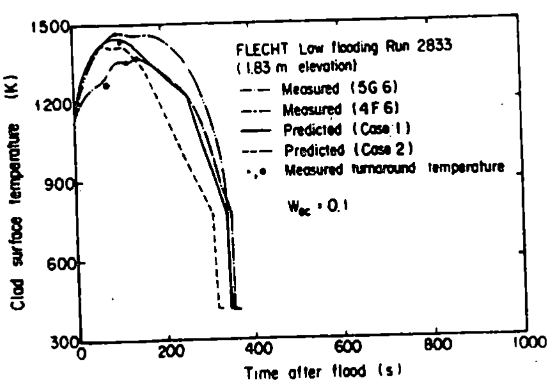

Fig. 2 Effect of water accumulation above quench front on history of clad surface temperature

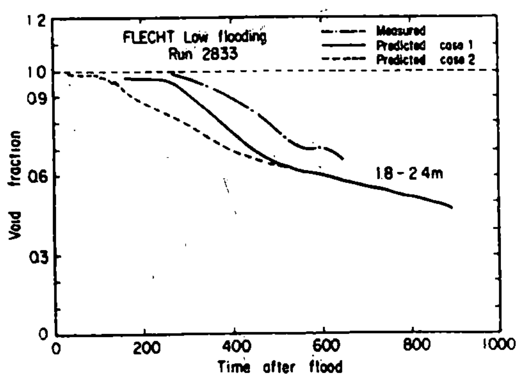

Fig. 4 Effect of water accumulation above quench front on void fraction

Steam Line Break, Jet Pump Drive Line Break and Natural Circulation Tests in ROSA-III Program for BWR LOCA/ECCS Integral Tests

K. Tasaka, M. Suzuki, Y. Anoda, H. Nakamura and M. Shiba

Japan Atomic Energy Research Institute

The ROSA-III facility is a volumetrically scaled (1/424) BWR system with an electrically heated core to study the response of BWR and the effectiveness of the ECCS during a LOCA. The core is consisted of four half-length simulated fuel assemblies. Each fuel assembly contains 62 fuel rods and two water rods which are spaced in a square 8x8 array. The coolant recirculation system consists of two loops, intact and broken loops, provided with a recirculation pump and two jet pumps in each loop. Jet pumps are installed outside the pressure vessel to simulate the volume and the relative height to the core satisfactorily.

Five tests have been conducted in the steam line break test series. The break location was assumed at the main steam line inside the containment except for one test T100R with a break outside the containment. The break area simulated 100%, 34% and 10% of the cross section of the BWR main steam line piping. The HPCS (High Pressure Core Spray) failure was assumed in the tests except for one 100% break test T100H. The ADS (Automatic Depressurization System) was actuated only for a 10% small break test T010S and a 100% break test T100R with a break outside the containment.

In a 100% steam line break test with a break outside the containment, the MSIV (Main Steam Isolation Valve) was closed at 3 s after break and the thermal-hydraulic phenomena were similar to those for a 5% recirculation line small break LOCA (Fig.1). However, the PCT (Peak Cladding Temperature) in the steam line break was lower than that in a recirculation line break because of less fluid loss in the steam line break. As for steam line breaks inside the containment, the void fraction and the mixture level in the downcomer were maintained high because of the continued depressurization. The mixture level inside core-shroud became lower than that outside the shroud before core level recovery by ECCS. The level swelling in the downcomer delayed the actuation of the LPCS (Low Pressure Core Spray) and LPCI (Low Pressure Coolant Injection) because they are actuated by the low L1 level signal in the downcomer with the failure assumption of the ECCS actuation signal by the high containment pressure. The PCT reached 1008 K (see Fig.2) in a 100% break T100S being higher than the PCT of 785 K in a 200% double-ended break at the recirculation pump suction but still well below the present safety criteria of 1473 K.

A Jet pump drive line single-ended break test T021D was conducted with the failure assumption of HPCS, LPCS and 1 LPCI out of three to study the fundamental thermal-hydraulic phenomena of a LOCA with a break at intermediate level of the reactor pressure vessel such as HPCS line piping. The break flow is limited at the jet pump drive nozzle with the total flow area of 21%. A steam line single-ended break test T015S with a break area of 15% was conducted for comparison. The ECCS conditions and initial conditions are similar to those of the test T021D. The test results of the collapsed liquid level inside shroud and the PCT are shown in Figs. 3 and 4 comparing with the results for the recirculation line split break tests with break areas of 15% (T015L) and 25% (T025L) with a HPCS single failure assumption.

In the test T021D the single-phase liquid water was discharged through the break until jet pump suction uncovery after break and the two-phase break flow continued thereafter for approximately 100 s. The level fall inside the shroud in a jet pump drive line break test T021D was more rapid than in a recirculation line break after uncovery of the exit to the recirculation line in the downcomer. Therefore, the cladding surface temperature started to rise earlier and the PCT in T021D reached 985 K being higher than the PCTs in the recirculation line break tests by approximately 130 K. In a steam line break test T015S the steam was discharged through the break, therefore, the residual mass in the system was larger than the other tests with breaks at two different locations limiting the PCT to a lower value of 649 K.

Natural circulation tests were conducted at the ROSA-III test facility changing the system pressure (7.35, 2.06MPa), core power (0-20%) and the downcomer liquid level (below L3 scram level) as test parameters. The following conclusions have been obtained through the analyses of the test data. (1) The ROSA-III test facility can simulate the ratio of the water level heights inside and outside the core-shroud in a BWR system during natural circulation in the low power or low flow rate region. (2) The decay heat power can be removed by the natural circulation if the downcomer liquid level is maintained above the jet pump suction line. (3) The Level swell inside the core-shroud becomes maximum at about 20% of the steady state power and the upper part of the core can be uncovered to the steam environment at low power region for the fixed liquid level above the jet pump suction in the downcomer.

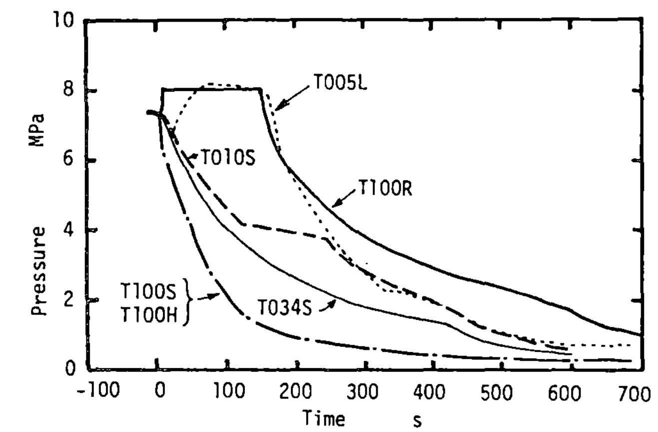

Fig. 1 Pressure Transients in Steam Line Break Tests

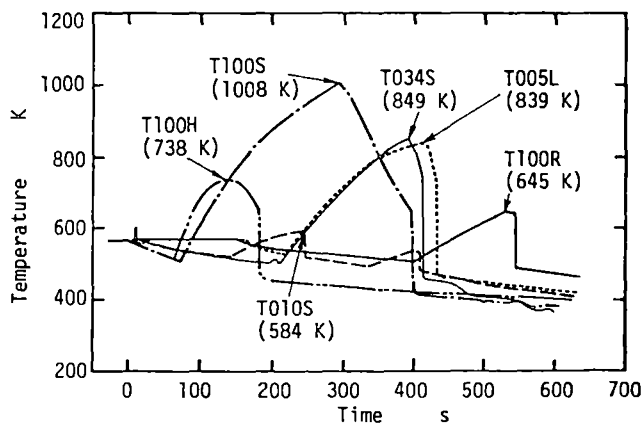

Fig.2 Peak Cladding Temperature

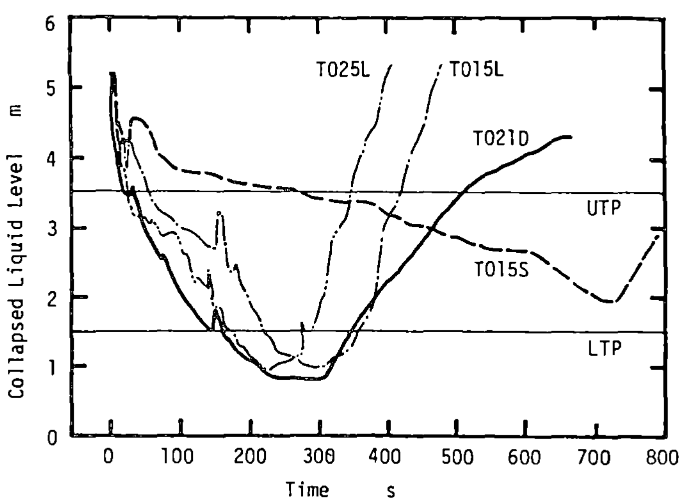

Fig. 3 Collapsed Liquid Level inside Shroud

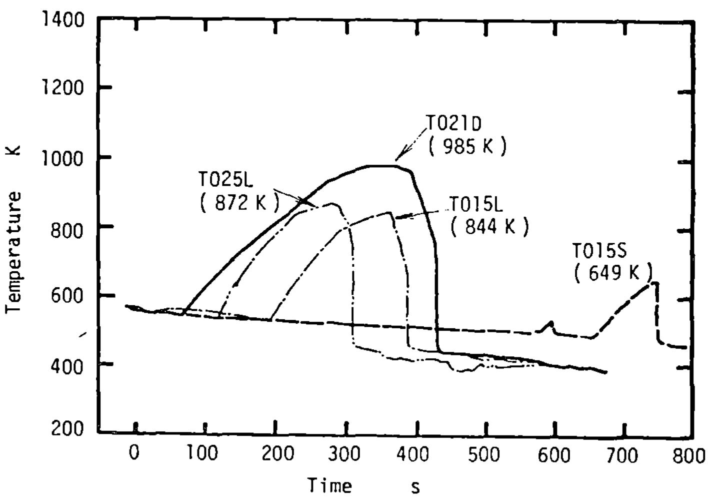

Fig. 4 Peak Cladding Temperature

COMPUTER STUDIES ON THE INFLUENCE OF SOME SCALING DISTORTIONS ON THE RESULTS OF AN INTERMEDIATE BREAK IN LOFT

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A number of important thermohydraulic reactor safety experiments has been done in the LOFT (Loss of Fluid Test) Facility in Idaho Falls, Id., USA. In applying the results on commercial pressurized water reactors the scaling problem must again be considered. It has two closely interrelated aspects:

- scaling of a test apparatus to a PWR, and
- scaling of the test results to understand the behaviour of a PWR because when doing the first scaling task a number of compromises was unavoidable.

This paper is solely concerned with the latter aspect. To our knowledge it has been adressed by a few authors only, all using the computer code RELAP 4.
In principle their approach was to use an input deck which gave reasonable results for a given test and modify it according to the parameters of the facility to be compared.
As a nuclear steam supply system is a very complex system, however, it is difficult with this method to sort out the different effects of different parameters.
Therefore this paper concentrates on understanding the influence of different parameters.

A RELAP 4/MOD 6 study was made based on the blowdown phase of the intermediate break experiment LOFT L5-1.

The method was to set up a base model and to vary parametrically some areas where it is suspected that LOFT differs from a commercial power plant.

The aim was not to simulate LOFT or a PWR exactly but to understand the influence of some parameters on the thermohydraulic behaviour of the system and the clad temperature.

With a RELAP 4/MOD 6 model of LOFT consisting of 17 volumes, 24 junctions and 8 heat slabs - which is less compared to the usual models with 4o volumes and more - we varied following parameters:

- stored heat in the downcomer (LOFT has rather large filler blocks in this part of the pressure vessel)
- bypass between the downcomer and the upper plenum
- core length.

The results of these calculations show that LOFT is prototypical for all calculated blowdowns.

As the clad temperatures decrease with decreasing stored energy in the downcomer, increased bypass and increased core length, LOFT results seem to be realistic as long as realistic bypass sizes are considered, they are conservative in the two other areas.

Integral Tests as a Tool for the Study of Physical Phenomena and Code Assessment

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Experience in the reactor safety research during the last fifteen years has shown the importance of a strong coupling of code development and experimental test programs. This will be discussed in connection with the analytical and experimental investigations for loss-of-coolant accidents (LOCA) in Light Water Reactors (LWR). The aspects discussed can, however, be applied to other research fields in reactor safety.

Three aspects have to be considered:
- Technical requirements
- Cost
- Use of integral tests as a tool to achieve a common understanding

Since it is not feasible to study a LOCA in a full sized nuclear plant, transient system behavior can only be predicted with codes. Therefore, reliable codes assessed by the analysis of separate effect tests, component tests, and integral system tests are required. Separate effect tests are aimed at a detailed study of single physical effects. Component tests are used to study component behavior while integral tests serve as a study of the behavior of the overall system.

The integral test facilities now available for LOCA and transient experiments are LOFT, LOBI and SEMISCALE, and with reduced system pressure PKL and CCTF. Each of these facilities has its own scaling criteria. Comparing LOFT, LOBI and SEMISCALE with a 1300 MWe PWR, the reduction in scale is obvious. Besides the reduction in size none of the facilities has the full number of loops.

More than 300 tests have already been performed using the available five integral test facilities to simulate LOCA and anticipated transients in PWR. Other test programs are underway e.g. test facilities like UPTF/SCTF, ROSA IV and BETHSY. The following table shows an approximate number of tests performed and planned:

PWR-Integral Tests

	Large Breaks Blowdown	Large Breaks Refill and reflood	Intermediate and small breaks	Anticipated transients
Performed Tests	~ 100	~ 140	~ 140	~ 30
Planned	~ 10-20	~ 130	~ 100-200	~ 50-100

However, only part of the tests performed have been compared with code calculations.

The initial integral tests performed in LOFT and SEMISCALE have been analysed using the 1st generation of system codes. These codes were based on the assumptions of homogeneous mixing under equilibrium conditions between the two phases. A comparison between experiment and calculation has shown, however, the need for more sophistication in the formulation of two phase phenomena. The new code generation like TRAC, RELAP5 and the German code system DRUFAN/FLUT are based on nonhomogeneous nonequilibrium formulation of two phase flow. Other codes like CATHARE and THYDE are under development.

Calculations to be compared with experimental results have to be performed with "best-estimate"-codes, because conservative assumptions with respect to the simulation of physical phenomena may lead to an unrealistic description of the component or system behavior.

Due to the high cost and time for code development and assessment it seems to be necessary to formulate a validation matrix for code assessment on an international basis. Thus, different codes could be compared. In the CSNI Task Group on Status and Assessment of Codes for Transients and ECCS different cross reference matrices are discussed for transients, small leaks and large breaks in PWRs.

Cost of separate effects tests is usually not very high and often each industrial nation has its own separate effect test facilities; the experimental - results are available through bilateral agreements to other countries.

Due to the high costs of integral test facilities, in some cases multilateral cooperation between the industrial nations has been achieved, e.g. with LOBI, 2D/3D and LOFT. In the LOFT consortium 9 OECD nations cooperate to carry out an integral test program of mutual interest.

Besides technical and cost aspects integral test facilities and their results will act as a tool to achieve a common understanding about the physical behavior, to exchange knowledge and to compare results of different code calculations.

Rough estimates have shown that more than one Billion Dollars was expended in LOCA research. Cost for experiments has been about 80 % and only 20 % has been available for test evaluation and code development. The future trend is to use relatively more money for test analyses and code assesment in order to increase confidence in the analytical results.

There is always the question if new test facilities or new tests are necessary. With respect to the study of physical phenomena, there seems to be no need for more integral tests after the already planned test programs have been completed. There is, however, a need to keep the integral test facilities available for operator training, operational transients and unforeseen events. For example after the TMI-2 event higher priorities were given to small leak experiments which were performed in SEMISCALE, LOFT and PKL.

AN OVERVIEW OF PIPE BREAKS FROM THE PERSPECTIVE OF OPERATING EXPERIENCE

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Review & Synthesis Associates

Currently the large pipe break, which is used as a design basis for nuclear plants, is being reviewed with the possibility of revision in the future. The classic case is the double-ended-guillotine break of the large primary pipe in BWR's and PWR's. While analytic studies will play a definite role in modifying existing pipe break criteria, it is my personal opinion that statistical analyses of available operational data on nuclear and non-nuclear piping systems will provide the ultimate bases for changes in such piping criteria. This approach was used to arrive at a decision concerning the acceptability or non-acceptability of nuclear reactor pressure vessels based on calculated probabilities of catastrophic failure.

Three collections of data exist for use in analyzing and predicting numbers and magnitudes of piping failures. These are a somewhat amorphous body of information on non-nuclear piping, available data on PWR piping and similar data on BWR piping. An assessment of failure mechanisms, crack growth rates, reliability of crack detection and sizing techniques, sensitivity of leak detection, and predicted modes of failure (when synthesized) should serve as a basis for modifying the existing criteria.

The preceding approach will be discussed in this paper. For example, seismic loads have been of concern so a brief discussion of piping response during earthquakes will be given. A similar approach will be used for other faulted (D) loads.

BWR's have suffered many cases of intergranular stress corrosion cracking in austenitic piping in various systems. The latest cases were potentially the most severe in that IGSCC has been reported in the main recirculating loops in the past year in several plants. The operational conditions contributing to the cracking will be reviewed as well as the problems inherent in volumetric examination of austenitic piping. Other classes of operating conditions contributing to BWR pipe cracking will be examined.

The PWR primary piping systems have been essentially free of cracking; however, there have been a variety of cracking mechanisms in other PWR piping systems related to mode of plant operation. Examples of thermal fatigue, IGSCC, erosion cavitation and vibrational fatigue, as well as failures due to dynamic loads such as water hammer will be reviewed.

Generally, the number of severe failures resulting from plant operation are quite low. A second category, those having the potential for severe failure, exists and requires further evaluation. The great majority of cracking incidents are relatively benign, posing maintenance and repair problems more than safety problems.

ELIMINATION OF THE DOUBLE-ENDED GUILLOTINE BREAK IN PRIMARY COOLANT LOOPS*

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Lawrence Livermore National Laboratory

The primary coolant loops (RCL) of a pressurized water reactor (PWR) are required to postulate pipe breaks. In a typical 4-loop PWR plant, about 28 break locations are postulated. The containment sizing, the ECCS systems, and the component supports adjacent to the RCL piping are designed to withstand a credible, yet unlikely, break load, i.e., the double-ended guillotine break (DEGB) load. In order to prevent pipe whip and to reduce the flow area upon the pipe rupture, pipe whip restraints are provided at each break location. These restraints, designed to withstand a pipe whip load of up to six million pounds, are huge and occupy a large portion of an already congested area sometimes blocking access for in-service inspections.

This paper reports on the effort by the Lawrence Livermore National Laboratory (LLNL) in developing a technical basis from which the Nuclear Regulatory Commission (NRC) can work to eliminate the DEGB design requirement in PWR primary coolant loops.

Our study considered both direct and indirect causes of pipe failure. Brittle or ductile rupture is considered a direct cause of failure of a piping system. Ductile failure can be introduced by such things as fatigue, corrosion, vibration, and residual stress which can lead to crack growth. Examples of indirect causes of pipe failure include crane failure which causes the rupture of the pipe, support failure resulting in shearing of the pipe, and missile impact.

We selected a probabilistic approach to assess the problem. A probabilistic approach is a mathematical formulation incorporating all available information generated through deterministic studies into a numerical tool. This tool enabled us to perform sensitivity studies whose results gave us a better understanding of the importance of each parameter while allowing us to incorporate the uncertainties associated with the data obtained from the deterministic studies into our assessment.

We developed two probabilistic models: the probabilistic fracture mechanics model for direct causes, and the support system reliability model for indirect causes. (Detailed descriptions for both models are documented in NUREG reports, CR-2189, 2301, and 2801.) We used these models to evaluate both Westinghouse and Combustion Engineering PWR primary systems. The results of our evaluations showed the probability for direct DEGB was in the range of

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*This work was supported by the United States Nuclear Regulatory Commission under a Memorandum of Understanding with the United States Department of Energy.

10^{-10} to 10^{-17} per year; for indirect DEGB, the probability ranged from 10^{-5} to 10^{-10} per year. The large variations in these ranges were the result of variations in the plants as well as in the uncertainties associated with the parameters considered in the evaluation.

We considered our results and concluded that the DEGB event is unlikely. Considering the unlikelihood of this event and the adverse effects pipe whip restraints introduce into the system, we suggest the DEGB design requirement be reevaluated. Before a well defined replacement criterion can be established, interim applications, such as the removal of the asymmetric blowdown load requirement, the decoupling of the DEGB and the SSE, and the elimination of pipe whip restraints in the RCL, can be justified.

The technical details and results of this study were presented to the ACRS who considered this an acceptable and proper approach to the handling of the pipe rupture problem. As an extension of our study, we are assessing the DEGB requirement for Babcock & Wilcox (B&W) PWR plants and General Electric (GE) BWR plants. The NRC plans to complete the results of both studies by the end of 1984. This approach is also being used to assess the reliability between rigid and flexible piping systems. Our ultimate goal is to use the results of these studies to develop revisions to the current piping design criteria which will allow piping design to be more balanced between the extreme dynamic load and daily thermal events. This in turn may lead to further changes in the piping design requirements, such as higher damping, less supports, and alternate methods to generate design spectrums.

THE VIEW OF INDUSTRY ON THE IMPACT OF PIPE BREAK CRITERIA

S. A. Bernsen - Bechtel
Representing the Atomic Industrial Forum

Historically, large pipe breaks in the types of materials used and under operating conditions similar to those in light water reactor service have not occurred. Nevertheless, the non-mechanistic assumption of a double ended pipe break of the early sixties, selected for loss of coolant accident analysis purposes, has become a mechanistic criterion for the design and arrangement of high pressure piping systems and their associated supports and enclosures in today's nuclear plants. It has been estimated that the cost for dealing with the pipe break effects on a typical large nuclear unit is somewhere in the 30 to 50 million dollar range. This large cost impact did not always exist; as late as the early 70's the effects considered and provisions made were quite nominal. However, as time has progressed, analytical techniques have become more and more sophisticated and experts continue to discover additional phenomena associated with the idealized instantaneous double ended pipe break-leading to more analysis, additional research and further cost increases. This trend is continuing.

While it seems reasonable and appropriate to continue to design the Emergency Core Cooling Systems for a range of loss of coolant accidents up to and including those that approximate the area of the largest pipe connected to the reactor vessel and to use this break in determining the loading and temperature rise rate for containment structures and equipment qualification, it no longer seems reasonable to provide precisely engineered break protection for a limited number of potential pipe break locations. This observation is gaining increasing support, particularly as engineering judgment and historical perspectives are being supplemented by both deterministic and probabilistic studies that indicate the potential for large instantaneous breaks in nuclear grade piping systems is virtually incredible. Fracture mechanics analyses support leak before break assumptions with wide margins and probabilistic studies indicate potentials for double ended pipe breaks in the range of less than one in a billion years.

Current design considerations for pipe break include:

- o The installation of two to three hundred pipe whip restraints designed to accommodate both the impact and the thrust forces from hypothetical pipe breaks. Elaborate analytical techniques have been developed. In most cases, designers resort to two phase dynamic blowdown modeling of the reaction force coupled with an inelastic dynamic analysis of the restraint structure. Restraint locations generally consider prevention of the formation of a plastic hinge in the pipe.

- o Careful tracking of the jet envelope resulting from the hypothetical pipe break (frequently ignoring the presence of intervening structure) including computation of drag forces on various shaped items and careful consideration of the effect of the jets on other safety-related systems, components and structures as well as the potential for formation of secondary missiles or debris. In some cases jet impingement barriers are provided.

- Where pipe breaks are allowed to occur without whip restraints, elaborate analysis of the effect of the whipping pipe on adjacent structure and piping has been performed.

- Detailed sub-compartment pressurization analyses are performed to develop transient loading conditions on structures - the most notable of these, of course, are the detailed analyses performed for pipe breaks within the primary shield cavity and the resulting transient asymmetric loadings on the reactor vessel. These are also coupled with the internal transient asymmetric loading analyses within the reactor vessel and the elaborate evaluation of the behavior of fuel and core internals and reactor vessel supports.

Provisions for pipe break create serious layout problems and significant design compromises which affect operator exposure and may to some extent reduce the reliability of piping systems. Some examples of these design compromises and their effects include:

- Opening up the spaces between the reactor primary shield and the vessel for ventilation purposes which may reduce shielding effectiveness and increase radiation levels during plant operation as well as during access for in-service inspection or during fuel handling operations.

- Provisions to reduce excessive loads on the reactor primary shield and the reactor vessel frequently interfere with the most effective arrangement for cooling air paths within the reactor cavity.

- Massive pipe whip restraints within the containment interfere with access for in-service inspection. In some cases portions of the restraints need to be removed for access, thus increasing the time that examination personnel need to spend in high radiation exposure areas.

- Frequently whip restraints are set with very tight tolerances on the piping system - increasing somewhat the probability that the restraint system might interfere with free movement of the piping and thereby increase stresses. Restraint lugs are occasionally attached directly to the piping, also increasing local stresses in order to accommodate hypothetical breaks.

- Concern over secondary missiles may prevent efficient use of easily removable platforms making access more difficult.

- Concern over the behavior of non-metallic insulation could cause projects to incorporate less effective insulation designs, increasing system heat losses and heat loads within the containment and other building enclosures.

Although industry has developed designs that fully accommodate current licensing criteria, it is believed that the level of effort, cost and complexity created by designing to accommodate these conditions is not safety effective. This view appears to be held equally by the NRC staff. ACRS has provided generally supportive comments on this issue also. With this broad consensus, it is anticipated that rapid progress will be made in establishing more realistic licensing requirements and substantially reducing the extent to which pipe break considerations will impact plant design.

PIPE-TO-PIPE IMPACT TESTS

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Pacific Northwest Laboratory

OBJECTIVE

The objective of the program is to generate experimental data and supporting analyses, to evaluate the Nuclear Regulatory Commission's criteria for anticipated damage due to pipe whip. These criteria are contained in the Standard Review Plan 3.6.2 and are stated as follows:

> "An unrestrained whipping pipe should be considered capable of causing circumferential and longitudinal breaks, individually, in impacted pipes of smaller nominal pipe size, and developing through-wall cracks in equal or larger nominal pipe sizes with thinner wall thickness, except where analytical or experimental, or both, data for the expected range of impact energies demonstrates the capability to withstand the impact without rupture."

RESULTS

The program test matrix has been completed with seventeen impacts between pipes of various sizes on simple supports and a 6" pipe being accelerated by the mechanism designed and constructed for the project. The target pipes varied in size from 12" Sch 60 to 3" Sch 160 and were sustained at PWR conditions. Vegetable oil was used as a fluid to avoid the potential energy release associated with water at these conditions. Impact velocity was varied for each of the target pipes, with new specimens being incorporated for every test.

Velocity and damage parameters were taken from each test, along with records of strains, pressure, temperature and impact loads. Many specimens experienced severe deformation and two specimens were ruptured.

It was decided to establish the degree to which the completed tests were representative of potential impact conditions in a representative PWR configuration. To achieve these data, a model of the SNUPPS plants was surveyed, using the assumption that pipe whip restraints did not exist. The distance through which a ruptured pipe could accelerate onto an adjacent high energy pipe was noted. It was concluded that the test data currently acquired lies within a low percentile of the data obtained from the survey.

In order to correlate the test data to make predictions for further testing and to establish potential pipe whip situations, a mathematical model has been developed. The model uses momentum relationships based upon crushing and plastic bending modes of deformation to arrive at energy distributions. The assumption is made that static crushing tests adequately provide the basis for dynamic crushing behavior. Good correlation with the test data has been achieved through this approach.

Two approaches have been studied to establish the potential for rupture in an impact situation. One approach considers the existence of a flaw and applies the impact strains to estimate the growth potential for that flaw. The second approach uses the impact strains to evaluate the likelihood of failure by ductile rupture. It is concluded that the contribution of flaws to rupture probability is much less than that attributed to ductile rupture.

RELATED WORK

The Atomic Energy Company Ltd., of Canada, has completed a series of impact tests on 2" Sch 80 pipes. There is a marked functional relationship between diameter change and impact energy that is satisfied by both the AECL data and the earlier test results achieved within this project.

FUTURE WORK

- The two analytical components of the rupture prediction capability will be completed.
- A study will be performed to determine the location and effect of plastic hinges formed at impact.
- The prediction capability will be applied to determine a suitable set of additional impact tests.
- These tests will be performed.
- A final report will be written.
- A value/impact evaluation will be performed in the event that licensing criteria changes appear appropriate.

Review of Current RES Programs in Structural Engineering

J. J. Burns, NRC

The research effort in structural engineering within the Nuclear Regulatory Commission is directed to the safety of nuclear power plant structures and components. The safety margin of structures remains the central issue. The principal events which may cause structures to be unsafe are severe accidents and extreme earthquakes and combinations of the two. The objective of current research is to establish the safety margins against severe accidents and earthquakes.

Because there are at present no new applications for construction permits of nuclear power plants in the United States (however, some still have not received operating licenses), research deals in large with the systematic evaluation of existing plants. The need exists primarily for determining the degree of safety of existing plants which were designed for less severe loading conditions than presently required by the NRC. Structures will degrade as plants grow older. More severe loading can result from upgrading seismicity based on new earthquake observations, or from new understanding of the behavior of structures to the effects of earthquakes. Additionally, service experience frequently reveals inadequacies in design practices or standards which require further research efforts.

Below are listed the current NRC projects in structural research. The objective, contractor, and expected date of completion of each project is given:

1. Engineering Characterization of Seismic Motion for Nuclear Plant Design
   Woodward-Clyde Consultants
   Program Conclusion - 1983
   Objective:

   Development of a basis for improving the way in which design earthquake motions are chosen for nuclear power plants. The improvements should account for the facts that instrumental peak acceleration is not a good measure of damage potential.

2. Probability Based, Load Combinations for the Design of Seismic Category I Structures
   Brookhaven National Laboratory
   Program Conclusion - 1986
   Objective:

   Provide a basis to improve the ways for combining loads for structural design. To develop a consistent design procedure, a systematic evaluation of event combination criteria for combining responses for different types of structures, and of limit state identification will be made.

3. Containment Safety Margins
   Sandia National Laboratory
   Program Conclusion - 1988
   Objective:

   Develop reliable methods for assessing the capabilities of containment structures at limit states significant to public health and safety by experiments on scaled steel and concrete models.

4. Integrity of Containment Penetrations Under Severe Accident Loads
   Sandia National Laboratory
   Program Conclusion - 1986
   Objective:

   Develop a methodology that can be used to provide reliable predictions of penetration leakage characteristics under severe accident conditions. The methodology will be primarily based on experimental evidence.

5. Category I Safety Margins
   Los Alamos National Laboratory
   Program Conclusion - 1988
   Objective:

   Develop experimental data for determining the sensitivity of the dynamic behavior of Category I Structures to variation in earthquake loadings beyond that used in the design of nuclear power plants. Structural fragility data will be developed from structural model tests.

6. Buckling of Steel Containments
   Los Alamos National Laboratory
   Program Conclusion - 1984
   Objective:

   Develop by experimental and analytical means acceptable methods of design for static and dynamic buckling for structures with openings and reinforcing, including the influence of shallow domes and hatches. Experimental evaluation of the area replacement rule for shell openings.

7. Standard Problems for the Evaluation of Structural Engineering Computer Codes
   Brookhaven National Laboratory
   Program Conclusion - 1988
   Objective:

   Determine the ranges of validity of analytical methods used to predict the behavior of nuclear-related structures under accidental and extreme environmental loadings, and determine the extent to which the predictive methods are supported by experimental data.

8. Containment Leak Test Study
   Oak Ridge National Laboratory
   Program Conclusion - 1987
   Objective;

   Evaluate the practicality of the containment leak testing program, and the compatibility of regulatory requirements and industry testing standards.

A SUMMARY OF "SCALE MODELING OF REINFORCED CONCRETE STRUCTURES SUBJECTED TO SEISMIC LOADING"

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Group Q-13
Los Alamos National Laboratory

Seismic experiments on scaled models are of considerable interest because of the very large size of the reinforced concrete structures used in nuclear power plants. Scale modeling techniques, including the modeling of reinforced concrete structures, are well developed; however, the use of scale models in seismic experiments in which the model structure is loaded into its nonlinear range involves some special problems. These problems include:

1. the necessity for, and difficulties in, appropriate time and amplitude scaling of the seismic excitation, and
2. the necessity for, and difficulties in, appropriately scaling the material properties in the nonlinear range, especially any time dependent properties.

In this paper the scaling, or modeling, laws are developed so that several methods of scaling can be investigated and compared. It is concluded that a "true" scale model can not be constructed and tested for the investigation of this dynamic, nonlinear problem. However, it is shown that the effects of distortions can be understood and minimized by the proper selection of scaling laws. The distortion of damping effects is given special consideration and the relative difficulties in scaling the effects of three different damping mechanisms (viscous, structural, and Coulomb) are considered.

EXPERIMENTAL ANALYSIS OF THE RESPONSE OF REINFORCED CONCRETE STRUCTURES SUBJECTED TO EARTHQUAKES

by

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Department of Civil Engineering
University of Illinois Urbana

A series of structural models of reinforced concrete frames and walls have been tested to study their response to strong base motion using the University of Illinois Earthquake Simulator. This work, which has been going on since 1967, has provided data indicating some of the strengths and weaknesses of small-scale models in understanding the response of full-scale structures in the nonlinear range of response.

The object of the presentation is to discuss the potential and limitations of dynamic tests of small-scale models to analyze earthquake response of reinforced concrete structures.

DESIGN AND ANALYSIS OF CONCRETE STRUCTURES

J. J. Ucciferro, United Engineers & Constructors Inc.

Category I concrete structures are designed to withstand extreme loads such as LOCA pressure and temperature, tornado, seismic, impulse and impact. Category I concrete structures can be broadly categorized into two groups, the Containment and other Category I structures. For the Containment the governing load combination varies from one region to another. Even within a particular region the design related to various stress components can be governed by different load combinations. For example, the design for radial shear reinforcement at the base might be governed by a load combination involving accident pressure alone but the meridional reinforcement in this region might be governed by a load combination containing a reduced pressure including seismic. This is not true for the other Category I structures, where seismic loading (OBE) usually governs the overall design. Results from seismic analyses illustrate that the OBE response accelerations, base shears and moments are approximately 65% of those corresponding to the SSE, for cases where the peak ground acceleration of the OBE is one half that of the SSE. This is due to the lower damping associated with the OBE. When the prescribed load factor (1.9 in this case) is introduced, the OBE controlled design can be as much as 25% more conservative than that required for the SSE. A smaller load factor for the OBE would be more appropriate.

The design process for nuclear structures originates when general arrangements are developed by the mechanical and nuclear disciplines with structural discipline input. Seismic analyses are then performed using lumped mass models which account for the distribution of mass and stiffness in the structure. Soil structure interaction analyses are performed if appropriate. Response spectrum analysis are performed to determine structural response unless the selected soil structure interaction method requires time history input.

For structures which are relatively simple such as less complex rectangular buildings, shears and moments are distributed by hand. For complex rectangular buildings three dimensional finite element analyses are performed to determine the distribution of lateral loads into the shear walls. These analyses typically use the acceleration profiles output from the seismic analyses as input to generate inertial forces. Effects such as openings in slabs and walls, transfer of horizontal forces from the slabs to walls, basement - wall interaction and floor offsets are considered. Once the structural response from seismic loading has been determined other loads are included in the prescribed load combinations as required. Out of plane effects due to lateral loads such as soil, water and seismic are considered separately and the resulting out of plane shears and moments are added to the other internal forces for design. For the case of the other Category I structures reinforcing steel is provided in accordance with the ACI-349 Code.

Containment design is somewhat different in that tangential, radial and peripheral shear forces exist in regions of high membrane tension. In addition, large discontinuity effects exist at the cylinder basement junction and in the vicinity of large openings where force concentrations are produced due to the interruption of membrane forces. Also large bending moments and shear forces are produced due to the presence of the thickened "boss" around the opening and the out of plane hatch loading.

Containments are analyzed to determine the distribution of internal forces in discontinuity regions including the basemat. The internal forces in the membrane region are statically determinate and are easily computed by hand calculation. Finite element analyses of discontinuity regions are performed using computer programs with capabilities to include the effects of concrete cracking, anisotropic material behavior and multiple layers of reinforcing steel. Stresses are computed for various combinations of tangential, radial and peripheral shear, bending moment and inplane tension. Results are compared to allowable values given in the ASME Section III, Division 2 Code. Due to the conservative shear provisions in the Code large quantities of diagonal reinforcing and transverse shear reinforcing are required. This has resulted in extreme congestion and associated difficulty in concrete placement. Additional research is required in this area in order to justify relaxation of the existing Code provisions.

THE PERFORMANCE OF ARTIFICIALLY DEFECTED LWR FUEL RODS IN AN UNLIMITED DRY AIR STORAGE ATMOSPHERE

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In 1982, a whole rod testing program using defected and intact LWR rods was initiated to provide the Nuclear Regulatory Commission (NRC) with information to allow evaluation of spent fuel dry storage licensing positions relative to (1) the long-term, low temperature (less than 250°C) behavior of spent fuel rods in dry storage, and (2) the radioactive contamination potential of crud from cladding in dry storage. The need for this information, philosophy behind whole rod testing, test matrix, apparatus description and rod characterization were reviewed at the NRC 10th Water Reactor Safety Research Information meeting last year. This paper describes results obtained to date.

The defective rod portion of the test used two H. B. Robinson PWR rods and two Peach Bottom-II BWR rods, each containing artificial defects in the form of 0.03 inch diameter holes. One defective rod of each type was placed in a sealed capsule containing one atmosphere of an Ar/1% He mixture, the second rod of each type was placed in a filtered capsule open to the hot cell air. To date the test has accumulated approximately 6000 hours at 229± 1°C.

During the first interim examination after 2235 hours the four defected rods were examined, the fuel rod strain measured, the capsule filters analyzed, the capsules swabbed and the swabs analyzed, and fuel rod particulate analyzed.

The general surface condition on the rods appeared unchanged. The defect located at 23-½ inches from the top of the BWR rod tested in air enlarged to approximately 0.09 inches diameter. It developed an asymmetrical axial crack extending 0.17 inches down the rod and 0.28 inches up the rod. The crack width at the widest point was 0.06 inches. Diameter measurements at both the defected and undefected axial positions indicated no strain occurred in the cladding at any PWR or BWR rod location except the one BWR upper defect mentioned above. At that location there was a strain of 9.8% with no apparent ovality. Although the cladding splitting only occurred for $\pm\sim\frac{1}{4}$ inch, the cladding strain did not drop to 5% until ~1 inch from the defect and preirradiation diameters were not reached until ~2 inches from the defect.

The filter analyses indicated no movement of fissile material out of the fuel rods. Co^{60} and Mn^{54} activity, components of crud, were found but the level was two orders of magnitude lower than measured on capsule smears. No U^{235} was found on the smears using gamma spectroscopy although delayed neutron counting indicated 250 ± 50 mg of total fissile material was present. Am^{241}, Eu^{154}, Eu^{156}, Ce^{144}, Sb^{125}, Cs^{134}, Cs^{137}, Mn^{54} and Co^{60} were all detected on the smears with Co^{60} being the predominant species. Not enough crud was obtained for chemical or structural analyses. The loose particulate in and around the split defect was collected and analyzed. Only 2.9 micrograms of fissile material was detected indicating the oxidized fuel was being retained within the fuel rod.

A second interim examination is in progress at the time of this writing. The cover gas in the six sealed capsules will be analyzed. The two defected rods in filtered air and two intact rods in air will be nondestructively examined, including capsule smears, visual examination and gamma scans. In addition, the defected rods will have a filter analysis, along with cladding strain and defect measurements. One or both of the defected rods in the filtered air will be destructively analyzed which will include metallography, ceramography, hydride analysis, cladding hardness, x-ray diffraction for fuel structure, and SEM examination of the crack tip.

Using oxidation incubation data obtained from unirradiated unclad pellets, the first interim examination indicates an oxide front velocity of $\sim 5.3 \times 10^{-5}$ cm/min which agrees well with Boase and Vandergraaf. Only speculation can be offered at this time why only one of the four defects exhibited cladding strain and split; hopefully the second interim examination will shed some light on this question.

In summary, based on the first interim examination of the dry storage at 229°C of defected BWR and PWR for 2235 hours, the major observations are: (1) significant oxidation and some cladding splitting occurred at only one of four defects that were tested in air; and (2) little crud spalled from the rods during this initial part of the test. Examination of the rods after 6000 hours at temperature is now underway.

In summary, based on the first interim examination of the dry storage at 229°C of defected BWR and PWR rods for 2235 hours, the major observations are: (1) significant oxidation and some cladding splitting occurred at only one of four defects that were tested in air; and (2) little crud spalled from the rods during this initial part of the test. Examination of the rods after 6000 hours at temperature is now underway.

EVALUATION OF NUCLEAR FACILITY DECOMMISSIONING PROJECTS PROGRAM

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Major studies have been undertaken in recent years by the U. S. Nuclear Regulatory Commission (NRC) and others on the technology, safety, and costs associated with decommissioning nuclear facilities. The program described in this presentation is being undertaken by the NRC to compile and evaluate the activities of ongoing decommissioning projects. Assessment and evaluation of the methods, impacts, and costs will provide bases for evaluating licensee's decommissioning proposals and for future decommissioning direction and regulation.

The Evaluation of Nuclear Facility Decommissioning Projects Program is being undertaken by the NRC to compile and evaluate the activities of ongoing decommissioning projects. Assessment and evaluation of the methods, impacts and costs will provide bases for evaluating licensees decommissioning proposals and for future decommissioning direction and regulation.

The primary objective of this program is to provide the NRC licensing staff data which will allow an assessment of radiation exposure during decommissioning and the implementation of ALARA techniques. The data will also aid in determining the funding necessary to ensure timely and safe decommissioning operations.

The tasks required to achieve the objective include the following:

- Identify candidate projects for evaluation
- Establish working agreements with licensees
- Develop a site specific reporting format
- Prepare progress reports of activities
- Prepare summary reports of activities
- Compare data with existing decommissioning studies
- Evaluate the facility's decommissioning techniques
- Compare similar projects

The selection of facilities to provide meaningful data for the Evaluation of Nuclear Facility Decommissioning Projects Program is based upon the following criteria:

- Mode of decommissioning, i.e. SAFSTOR, DECON, or ENTOMB
- Availability of decommissioning data
- Willingness of facility owner or managing agency to cooperate in program
- Usefulness of data for future decommissioning project analysis

The collection of data during the field evaluations of decommissioning projects provides as complete as practicable chronology of the facility's decommissioning. Data accumulation starts at the earliest feasible time to include engineering, ALARA review and other efforts preliminary to the actual facility decommissioning.

Three principal end results govern the scope of data collection during actual decommissioning. These are: 1) how actual decommissioning costs, methods, and radiation exposure usage compare to those assumed in the Battelle-PNL and ORNL NUREGs on decommissioning, 2) how responsive the decommissioning activities are to ALARA objectives, and 3) the lessons learned from actual decommissioning experiences.

A computerized data collection system, has been developed for data accumulation and analysis. This computer program incorporates information related to the following:

- Facility categorization, based on design, configuration, constructor, operator, and operating history
- Predecommissioning engineering
- Predecommissioning radionuclide inventory and dose assessments
- ALARA efforts
- Waste disposition, for both contaminated and non-contaminated materials, including packaging, transportation, and disposal site
- Decommissioning Costs
- Radiation exposure
- Special techniques and tooling utilized
- Termination survey data
- Lessons learned

To date reports have been prepared for the following reactor decommissioning projects: Elk River (NUREG/CR-2985), Fermi-1 (NUREG/CR-3116), Ames Laboratory Research Reactor (NUREG/CR-3336), and North Carolina State University R-3 Reactor (NUREG/CR-3370).

Residual Radionuclide Contamination Within and Around Nuclear Power Plants: Origin, Distribution, Inventory, and Decommissioning Assessment

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The U.S. Nuclear Regulatory Commission (NRC) has been charged with the responsibility of developing a general decommissioning policy for commercial nuclear facilities in the United States, including nuclear power plants. Since the nuclear industry has matured to the point where some of the early nuclear power plants have reached or will be nearing retirement status, it is imperative that the NRC develop detailed information to provide guidance for the decommissioning of these plants.

Several studies have conceptually assessed the technology, safety and costs associated with various alternatives for decommissioning nuclear power plants. One of the key elements of such assessments is a characterization of the radionuclide inventory remaining within a nuclear power plant. This information is essential for understanding the radiological problems which will be encountered during decommissioning. However, empirical data relating to the composition, distribution, and quantity of residual radionuclides attached to piping, hardware, equipment and concrete surfaces within nuclear power plants have been extremely meager, and calculated or estimated quantities have mainly been used in previous assessments.

To provide the NRC with an actual data base of residual radionuclide measurements within nuclear power plants, Pacific Northwest Laboratory (PNL) has conducted a comprehensive sampling and analyses program at seven nuclear power plants. The objectives of these studies are: 1) to provide information on the range of composition, quantities, and locations of radionuclide residues likely to be encountered in retired reactor power stations (exclusive of the reactor pressure vessel) and in the immediate station environs, 2) identify the origin of the radionuclide contamination, and correlate observed radionuclide inventories with reactor operating histories and procedures, 3) develop, to the degree possible, predictive capabilities to permit generic assessments of residual radionuclide contamination in nuclear power plants, and 4) provide a data base for use in formulating policies and strategies for decommissioning retired nuclear power plants.

The nuclear power stations which have been studied include 4 BWR's and 3 PWR's. The BWR stations include Pathfinder Generating Plant, Humboldt Bay Power Plant, Dresden Nuclear Power Station-Unit 1, and Monticello Nuclear Generating Plant. The PWR stations include Indian Point Station-Unit 1, Turkey Point Plant-Units 3 and 4, and Rancho Seco Nuclear Generating Station.

On-site sampling and measurement programs were conducted at each of the above nuclear power plants to acquire samples of contaminated piping, hardware, equipment, concrete, and soils for comprehensive radionuclide analyses for determining residual radionuclide concentrations and inventories at each of the plants. The program has focused on the measurement of radionuclides which have been transported from the pressure vessel and deposited in the various piping systems and components of the plants, and does not include the radionuclide inventories in the neutron activated pressure vessel and internals. Emphasis has been placed on measuring those radionuclides which will be most abundant at the time of shutdown (e.g. ^{51}Cr, ^{54}Mn, ^{55}Fe, ^{60}Co, ^{65}Zn, ^{90}Sr, ^{134}Cs, ^{137}Cs, ^{141}Ce, ^{144}Ce), plus those radionuclides having very long half-lives which will become dominant after tens to thousands of years following reactor shutdown (e.g. ^{59}Ni, ^{63}Ni, ^{94}Nb, ^{99}Tc, ^{129}I, and the transuranic radioisotopes of Pu, Am and Cm).

This paper presents the results of this study and discusses their significance in relation to: 1) the radiological conditions existing in and around nuclear power plants at the time of shutdown, 2) the processes responsible for the observed residual radionuclide distributions, 3) disposal of radioactively contaminated materials in compliance with low-level waste disposal rules specified in 10CFR61, and 4) various decommissioning alternatives, including mothballing, entombment, and dismantlement.

AN OVERVIEW OF DECONTAMINATION AS A PRECURSOR TO DECOMMISSIONING

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The Pacific Northwest Laboratory (PNL), operated by Battelle Memorial Institute, has performed a program for the U.S. Nuclear Regulatory Commission (NRC) to examine decontamination as a precursor to decommissioning. Specific objectives of the program were to examine the effect of the decontamination step on reduction of occupational and population exposure and on the reduction of waste volumes during the decommissioning of LWR plants.

The study was directed toward LWR systems and facilities now in operation. Decontamination, decommissioning and dismantling of separations plants, mixed oxide fuel facilities and UO_2 production, preparation and fabrication were not included nor were designs of advanced light water reactors examined.

The paper describes results of the study in the areas of: 1) a review of the literature on decontamination and planning, 2) the experimental evaluation of decontamination processes, 3) the evaluation of decontamination waste solidification and disposal, 4) an examination of safety considerations and radiation exposure anticipated in such operations, and 5) the cost factors involved.

BETA PARTICLE MEASUREMENT AND DOSIMETRY REQUIREMENTS AT NRC-LICENSED FACILITIES

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There are a number of indications that current beta measurement techniques are often not sufficiently accurate to assure compliance with limits of skin dose required by 10 CFR 20.101. Most survey meters used to measure beta radiation are adaptations of gamma survey meters. Thus, the user should understand that: 1) the area and depth of the beta source must be considered, 2) the distance and angle between source and survey meter are important, and 3) an estimate of the range of beta energies present in the measured radiation field must be obtained. This information can be used to determine a correction factor to be applied to survey meter results after subtracting gamma radiation measurements. In NRC-licensed facilities, many of those using beta survey instruments do not have enough information to determine an appropriate correction factor.

Information, similar to that required for interpretation of survey meter measurements, is required when personnel dosimeters are used to measure beta radiation doses. However, the individual processing the dosimeter usually has no first-hand knowledge of the exposure conditions. Thus, he must either assume an average correction factor or design the dosimeter to provide some of the information and rely on the wearer to supply the remainder.

Researchers from Pacific Northwest Laboratory (PNL) have conducted beta radiation measurements under laboratory and field conditions to assess the degree of the measurement problem and offer suggestions for possible remedies. PNL used an active spectrometer based on a thick silicon surface barrier detector and a passive spectrometer consisting of thermoluminescent dosimeters covered by varying thicknesses of aluminum.

Although the surface barrier detector was useful in laboratory studies of beta emitters, it was difficult to use in the field and suffered from some of the problems encountered with survey meters. The passive spectrometers provided a large part of the useful data gathered at NRC-licensed facilities.

Commercial survey meters and dosimeters, typical of those used by licensees, were used to measure beta fields in locations where data was obtained with active and passive spectrometers. In many cases, the commercial survey meter and dosimeter measurements results were widely divergent from the spectrometer measurement results. However, in some instances this was due to a very weak beta field.

(a) Operated for the U.S. Department of Energy by Battelle Memorial Institute.

Based on the results of PNL's studies, there appear to be two types of radiation fields where beta dose rates may be limiting. The first case is obvious and involves sources which are essentially "pure" beta emitters. All the measurements made at the fuel fabrication facility fall into this category. The second case, involving the commercial nuclear power plants, occurs where sources are very thin and relatively small. Sources of concern at power plants are nearly always fission products. Workers most likely to receive improperly measured beta radiation doses include maintenance personnel who disassemble cleanup pumps and pipes, remove components near the reactor core or handle equipment associated with spent fuel storage.

Although some of the difficulties encountered is measuring beta dose and dose rate can be corrected by a better design of survey meters and dosimeters, regulatory guidance is needed in the application of these devices and the interpretation of results. Additional training of survey meter and dosimeter users is also recommended.

ANALYSIS OF MEASUREMENTS WITH PERSONNEL DOSIMETERS AND PORTABLE INSTRUMENTS FOR DETERMINING NEUTRON DOSE EQUIVALENT AT NUCLEAR POWER PLANTS

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In a recent document (NUREG/CR-3400), we reported on an analysis of two series of measurements of neutron spectra and responses of neutron personnel monitoring instruments in containment at nuclear power plants. This paper consists of excerpts from that document.

One series of measurements was made by PNL at six nuclear power plants; the other was made by EML and RPI at eight nuclear plants. We chose measurements at 18 locations within the plants studied by PNL and at 37 locations within the plants studied by EML and RPI. At all these locations measurements were made with multispheres as well as with several types of survey instruments and personnel dosimeters.

In reviewing these measurements we find that there is no single clearly superior technique for estimating dose equivalent received by workers. Even the best techniques gave errors in the dose equivalent of more than 40%, about one third of the time.

We have divided the possible techniques into four categories:

(1) use of calibrated TLD albedo dosimeters

(2) use of remmeters such as the Eberline 9" spherical remmeter, the Andersson-Braun (Snoopy) remmeter, and use of the 10" sphere of the multisphere set.

(3) multisphere measurements

(4) techniques that are not recommended because results are too unreliable (use of gamma-ray exposure rate to predict neutron dose equivalent rate), or because instruments are not ready for routine use (^{3}He spectrometer or tissue equivalent proportional counter).

We now consider each of these categories.

The TLD albedo dosimeter has the advantage that it can be worn by a worker and responds as he moves through fields of varying intensity. If the dosimeter is calibrated with a moderated Cf source, and some spectral index such as the response ratio of 9" to 3" spheres has been measured at locations where the worker is likely to be, the dosimeter can be expected to predict dose equivalent with a standard deviation of about 40%. The disadvantage is that the calibration factor may vary in an unknown way as the worker moves through areas with different spectral quality.

The remmeter has the advantage that it can be calibrated to record the approximate dose equivalent in any area of interest. The Andersson-Braun type of remmeter is a stable device that generally gives good reproducibility. However, it would have to be used with an instrument such as a 3" sphere in order to give a measure of spectral quality.

The Eberline 9" spherical remmeter is used most extensively and together with the Eberline 3" sphere gives spectral information. However, there is a wide variability in calibration factors. The 10" sphere gives a smaller spread in calibration factors. However, the device would have to be engineered for routine use as a remmeter.

Measurements with a set of seven or eight multispheres, together with an unfolding computer code, yield detailed information about the neutron spectrum. A weighted sum of responses from four of these spheres can be used to calculate dose equivalent without the help of an unfolding code. However, predictions by this technique agreed only to within a factor of about 2 with dose equivalents derived from a full set of multisphere measurements.

Although the ratio of neutron dose equivalent rate to gamma-ray exposure rate was near unity for many of the locations examined, there were variations of a factor of 2 to 4 and, in some cases, a factor of 100 in this ratio. Both tissue equivalent proportional counters and ^{3}He spectrometers show promise for estimating dose equivalent. At present, however, they are in the class of research instruments and require sophisticated analyses and interpretation.

In view of the fact that no one technique seems clearly superior at this time, we recommend that both TLD albedo dosimeters and remmeters be retained for estimating dose equivalent to workers. In particular we recommend:

Continued Use of TLD Albedo Dosimeters: calibrated with a D_2O-moderated ^{252}Cf source to obtain a reference calibration factor; corrected for spectrum hardness, based on R_9/R_3 or R_{10}/R_3 measurements or on separate average values for PWR's and BWR's.

Continued Use of Remmeters: calibrated with a D_2O-moderated ^{252}Cf source or calibrated with a bare ^{252}Cf source together with a factor of about 2 to adjust the calibration to moderated ^{252}Cf; accompanied by a 3" Eberline sphere (if a 9" spherical remmeter is used) in order to establish correction factors for spectral hardness, with electronics improved to produce greater stability and precision; accompanied by a 3" sphere of the multisphere set (if a 10" sphere is used) even though the data indicate a minimal dependence of the response of the 10" sphere on the spectral quality, and with efforts to develop the 10"sphere for use as a routine remmeter.

One-time, or Occasional Measurements with a Set of Multispheres: made either in-house or by a contractor. If a power plant has an in-house capability, a set of 4 spheres may be used to estimate dose equivalent more frequently and at more locations than might be feasible for a contractor.

Techniques Not to be Used: the use of gamma-ray exposure rates, with an average n/γ ratio, is not recommended at this time; the development of ^{3}He spectrometers and tissue equivalent proportional counters for determining dose equivalent is being actively pursued at several laboratories. While showing much promise for the future, these systems are still under development and are not available for routine measurements.

PRELIMINARY RESULTS OF TESTING BIOASSAY ANALYTICAL PERFORMANCE STANDARDS*

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Bioassay is required (10 CFR, Part 20) for determining a worker's exposure to radioactive material. Bioassay involves whole-body counting for internally deposited radionuclides and excreta analysis. The correct assessment of internal dose is dependent upon accurate bioassay results. However, a major problem for licensees is that bioassay laboratories may not be providing accurate results.

A Health Physics Society standards committee was formed to draft ANSI performance criteria for radiobioassay, and the NRC issued advance notice of intent to require licensees to obtain bioassay services from "accredited" in-house or commercial laboratories. Most bioassay laboratories welcome the concept of accreditation.

This study was established to evaluate the appropriateness of the draft ANSI Standard by conducting a two-round, nation-wide bioassay intercomparison test. The analytical performance of *in-vivo* and *in-vitro* bioassay laboratories is now being studied in light of minimum criteria for accuracy and precision specified in the draft Standard.

The *in-vitro* testing involved the preparation and distribution of 560 samples of artificial urine containing carefully controlled quantities of either tritium, ^{90}Sr, ^{241}Am+^{238}Pu, ^{137}Cs, or uranium. The radionuclides were provided by the National Bureau of Standards. Laboratories volunteered to participate in the following categories: liquid scintillation counting, gross-beta measurements, alpha spectrometry, gamma spectrometry and fluorometry.

The *in-vivo* measurements testing involved preparation of a realistic torso phantom with interchangeable sets of lungs tagged with ^{60}Co, ^{235}U, or ^{241}Am, and a whole-body bottle phantom tagged with mixed fission products (^{60}Co, ^{137}Cs, and ^{144}Ce). These phantoms are currently being shipped to whole-body counting facilities around the country, including several nuclear power plants.

To date, 74 results from 14 different *in-vitro* laboratories have been received. Preliminary results from the first round of testing are shown in the following table:

*Work performed jointly for the U. S. Nuclear Regulatory Commission and the U. S. Department of Energy under Contract No. DE-AC06-76RLO-1830.

Nuclide	Measurement Category	% Failure[a]
H-3	Liquid Scintillation	17
Sr-90	Beta Counting	10
Am-241	Alpha Spectrometry	50
Pu-238	Alpha Spectrometry	50
Cs-137	Gamma Spectrometry	43
U-nat	Mass Determination	33

[a] Data indicates MDA > AMDA, $B_r < -0.25$, $B_r > +0.50$, or $S_b > 0.40$.

These results show that many laboratories have difficulty meeting the current draft ANSI Standard criteria for accuracy (B_r), precision (S_b), and minimum detectable amount (MDA). Alpha spectrometry failures usually related to difficulties in precision, whereas fluorometry failures were largely attributable to unacceptable MDA's. These results show that considerable improvement is necessary before accurate internal exposure assessments can be generated from bioassay measurement data.

Results of the *in-vivo* testing will not be available until each of the participating sites has had the opportunity to make measurements on the calibration phantoms.

CONSIDERATIONS IN FACTORING OCCUPATIONAL DOSE INTO VALUE-IMPACT AND COST-BENEFIT ANALYSES

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An area of growing concern in recent years has been the apparent increase in levels of collective radiation dose to workers at nuclear power plants in the USA. NRC decisions and rulings related to inservice inspection, retrofits, and plant upgrades have been primarily intended to reduce the risk of public radiation exposure resulting from either routine release of radioactivity or potential accident situations. However, implementation of required control measures and procedures can often result in increased levels of occupational exposure. Recognizing the need to incorporate occupational dose into probabilistic risk assessments (PRA), value-impact, and cost-benefit analyses, the NRC has sponsored this study with the objective of developing an appropriate methodology to factor potential worker exposures into overall safety analyses.

ALARA guidance for optimization of radiation exposures to the general public requires a consideration of all relevant social and economic factors. Clearly, any resultant increase in occupational dose should be included as a cost in such assessments. However, a review of several previous PRAs and cost-benefit analyses indicates this factor has seldom been considered. In some cases the implementation of decisions intended to reduce public dose can actually result in collective occupational dose levels exceeding the potential public collective dose averted. To correct such situations, guidance is required on methodology for factoring risk to workers into decision processes associated with public safety.

Factoring occupational dose and its consequences into a risk assessment on public safety is not a straightforward exercise. Several inputs to such assessments necessarily involve value judgements. Among the questions requiring resolution are:

- What is a reasonable measure of the equivalency of collective occupational dose to that of public dose? Should similar levels of effort and resources be devoted toward avoidance of a worker man-rem as is given to a public man-rem? If not, what equivalency factor is justified?

- How can determination of the relationship between marginal cost and collective dose versus increasing stringency of control best be accomplished as required for incorporation in value-impact and cost-benefit analyses?

- In considering methods for reduction of either public or occupational risk, what weight should be applied to the reduction of failure probability versus consequence mitigation (e.g. Should equivalently stringent controls be applied to routine radiation release situations as opposed to potential accident situations of equal risk?)

- What monetary value can most reasonably be ascribed to the avoidance of occupational and/or public collective dose? Should the value of \$1000/man-rem (Appendix I, 10 CFR 50) be applied across the board?

Since current policies do not provide definitive answers to the above questions, a review of plant operations, procedures, and associated technical literature has been undertaken with the objective of assembling the best current information to assist in decision making in areas where the goals of public and worker safety may be in conflict.

As a part of this study, general formulas for use in optimization of radiation exposure are being developed. These formulations will be used to evaluate specific operations and controls (i.e., inservice inspection and test, retrofits, and upgrades). Additionally, sub-models for specific of portions the problem are being developed for cases where generalized methods are determined to be insufficient. For example, Markov models for treatment of redundant trains of a system will provide the analyst with a tool for addressing the changes in failure probabilities as a function of inservice inspection frequency. The methodology will be applied to the extent possible within the constraints of data availability to several examples to related current NRC issues.

Dose Reduction at Nuclear Power Plants

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Introduction

Collective dose equivalent at nuclear power plants increased from approximately 1,250 rem in 1973 to nearly 54,000 rem in 1980 (1). This rise is attributable primarily to an increase over the same time period in nuclear generated power from 1,289 MW-yr to 29,155 MW-yr; and secondly, to increased average plant age. However, considerable variation in exposure occurs from plant to plant depending on plant type (BWR or PWR), refueling schedule, maintenance problems, etc. In order to understand the factors influencing these differences, the Nuclear Regulatory Commission has recently contracted with Brookhaven National Laboratory (BNL) to study dose-reduction techniques and effectiveness of as low as reasonably achievable (ALARA) planning at light water plants.

Objective of Proposed Work

The project objectives are to 1) identify and evaluate high-dose maintenance tasks in LWR s to enable the NRC and industry to focus on major dose-reduction targets; 2) investigate the extent to which high-reliability, low-maintenance equipment is used in the nuclear industry in order to develop a data base for use by NRC in providing guidance and evaluating licensee performance in reducing corrective maintenance doses; 3) review present procedures and equipment used in handling LWR radwaste packages on site in order to enable NRC to provide guidance on methods involving lower doses; 4) identify, evaluate and describe incentives for reducing the collective occupational dose in the nuclear industry in order to provide new, cost-effective and positive steps that NRC can take to provide such incentives; and 5) collect and organize for ready use by license reviewers the occupational dose reduction information that becomes available. These objectives will be coordinated with industry efforts ongoing at the Institute for Nuclear Power Operations (INPO), and the Electric Power Research Institute (EPRI) and with the Nuclear Regulatory Commission. In general, a data base containing background data for evaluation of the effectiveness of ALARA programs will be compiled and recommendations for new, positive steps that NRC or industrial organizations could take to reduce occupational doses will be formulated.

Significant Findings to Date

Maintenance work contributes about 79% to the annual collective dose at BWRs and about 70% at PWRs. These differences are attributable primarily to differences in plant design and layout. BWRs tend to have a greater number of contaminated components since their turbines are exposed to activity generated in the primary system. However, the steam generators which separate primary and secondary systems in PWRs get contaminated and, due to many recent steam generator tube failures, this system proves to be the largest source of exposure in PWR systems.

Numerous engineering modifications, which will reduce workers' exposure, are being considered in both new plant designs and as retrofits to existing plants. Examples of some of the more promising techniques will be discussed in terms of dose reduction potential and optimization (ALARA). For example, in new PWR plant designs, a collective dose reduction by about a factor of two appears to be possible by using low cobalt steel (Inconel 600 with a maximum cobalt content of 0.015%) in the steam generator tubes. This can be obtained at comparatively little extra cost.

Some foreign plants have recently been designed with predicted collective dose considerably below current U.S. experience. For example, the Sizewell 'B' PWR plant to be built in England has a design target of 0.2 rem/MW(e)-y collective dose. This may be compared to 0.5 rem/MW(e)-y for similar four-loop Westinghouse plants built in the U.S. since 1974. In addition, a design target of 1.0 rem maximum individual effective dose equivalent has been imposed. Several design features are included to meet the <1 rem/y criterion, even though these features were not considered cost-effective (based on a sliding scale of $60 to $1,500/rem over the dose range 0 to 5 rem/y).

An important parameter, which is being treated differently in various reports, is the influence of interest rates in optimization calculations. For example, an effective interest rate of about 20% was used in the AIF report (2) on engineering techniques and modifications whereas real interest rate (usually 3 to 7%) may be more appropriate. The real rate is corrected for inflation which influences both the cost of borrowing money and the cost of future health effects by about the same factor. The difference is highly significant since it results in cost estimates that differ by a factor of 3.94 for 3% vs 20% interest rates.

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DECONTAMINATION IMPACTS ON WASTE MANAGEMENT AND DISPOSAL*

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The nuclear industry is actively considering the potential advantages of primary system decontamination to reduce occupational exposure and to ensure the safe operation of light water reactors (LWR). The Nuclear Regulatory Commission (NRC) is responsible for insuring the public health and safety, and will therefore require a careful evaluation of different decontamination processes and the unique wastes they produce. The major areas of concern for the NRC in evaluating the effectiveness and safety of chemical decontamination processes are: the compatibility of the chemical system with the primary system materials, the long term stability of the primary system following one or more decontaminations, the rate of radioactive contamination buildup after decontamination, and the type, volume and toxicity level (radiotoxicity as well as chemical toxicity) of the radwaste streams generated by the decontamination as well as their subseqent management at the plant and at the disposal site.

In addition to the question of materials compatibility of a decontamination method and subsequent radiation reduction during normal operations, the NRC will require assurance that a proposed decontamination method is acceptable in terms of the additional occupational exposure incurred during a decontamination and during subsequent radwaste handling at the plant and at the burial site.

The decontamination processes considered for potential use all employ large quantities of chelating or complexing agents. The wastes containing these reagents will be disposed of in a shallow land burial site. Studies of existing sites have indicated that disposal of these types of wastes may impact on the ability of a site to retain radionuclides. The basic issue surrounding the disposal of wastes containing complexing or chelating agents is the potential of these wastes to enhance the migration of radionuclides. Each site, each waste type, and the way it is packaged for disposal will impact on the hazard posed by disposal of these wastes. It is desirable then to examine all the technologically feasible methods for safe management of these wastes, and to examine the interplay of site and waste characteristics. The complexity of the interactions that occur between radionuclides, a complexing agent and the soil, raise questions about the ability to model or predict the impact of the decontamination wastes on the performance of a

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

shallow land burial site. This issue alone emphasizes the need to investigate alternative methods for managing these wastes, as well as the need to determine whether there are packaging and solidification procedures that will lessen the potential hazards associated with disposing of large quantities of organic complexing agents. Work, in this program, is focused on determining if the decontamination wastes can be treated at the plant prior to disposal in a manner that will mitigate the potential hazards associated with disposal of organic complexing agents. This includes an assessment of the occupational hazards. Any method used for processing decontamination wastes should be safe with respect to the potential for increased exposure.

This paper will summarize the work done to evaluate the effectiveness of existing process technologies for managing decontamination resin wastes including the ability to solidify these wastes directly. A preliminary assessment of those areas that may lead to increased occupational exposure during waste management will be summarized.

PRESSURIZED WATER REACTOR STATION BLACKOUT

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The purpose of the Severe Accident Sequence Analysis (SASA) Program was to investigate accident scenarios beyond the design basis. The primary objective of SASA was to analyze nuclear plant transients that could lead to partial or total core melt and evaluate potential mitigating actions. The following summarizes the pressurized water reactor (PWR) SASA effort at the Idaho National Engineering Laboratory (INEL). The INEL is presently evaluating Unresolved Safety Issue A-44 - Station Blackout from initiation of the transient to core uncovery. The balance of the analysis from core uncovery until fission product release is being performed at Sandia National Laboratory (SNL). The current analyses involves the Bellefonte Nuclear Steam Supply System (NSSS), a Babcock and Wilcox (B&W) 205 Fuel Assembly (205-FA) raised loop design to be operated by the Tennessee Valley Authority (TVA).

The station blackout transient was selected from dominant core melt sequences identified by the Accident Sequence Evaluation Program (ASEP). The transient is initiated by a loss of offsite ac power followed by a failure to provide onsite ac power, and failure to provide steam generator cooling by way of the auxiliary feedwater system. The transient is characterized by a rapid boildown of the steam generator secondary liquid followed by primary system saturation, voiding, and resultant core uncovery. The thermal-hydraulic analysis for the Bellefonte station blackout transient was performed with the RELAP5/MOD1.6 computer code. The calculated system response indicates steam generator secondary dryout at 310 s, primary system saturation at 950 s, and core uncovery to begin at 1600 s. Core fuel cladding heatup to the zirconium-steam reaction initiation temperature of 1273 K occurred at 2550 s, terminating the RELAP5 portion of the analysis.

In the near future an event tree of specific actions required of the hardware and the operators to recover the Bellefonte plant from the station blackout transient will be developed. Potential mitigating operator actions derived from this probabilistic analysis will be evaluated deterministically using the RELAP5 computer code to confirm successful plant recovery and determine timing of the sequence. Additional station blackout calculations are also planned for a Westinghouse 4-loop PWR to determine the effect on the transient of differences in design features (internal reactor vent valves for B&W vs none for Westinghouse, for instance).

The SASA program is presently involved in extending the thermal-hydraulic analysis beyond core uncovery to examine system response following core uncovery. The primary safety concerns to be evaluated are primary piping heatup beyond yield stress limits, fission product transport and plateout on metal surfaces, and effects of core degradation on system thermal-hydraulic behavior. The Severe Core Damage Analysis Package (SCDAP) will be used in conjunction with RELAP5 to provide core performance and loop thermal hydraulic analytical capabilities. In addition to evaluating SASA concerns, the results of the extended analyses will be used to support Power Burst Facility (PBF) Phase II Test Planning at INEL.

RELAP5 BROWNS FERRY STUDY BWR ATWS

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The subject of this study is a postulated Anticipated Transient Without Scram (ATWS) at unit one of the Browns Ferry nuclear plant, a boiling water reactor (BWR). The development work is being conducted at the Idaho National Engineering Laboratory (INEL) for the U.S. Nuclear Regulatory Commission (NRC). It has long been recognized that the dominant ATWS transient in BWRs is the main steamline isolation valve (MSIV) closure pressurization type of event. This is because the steam supply system is isolated from its condenser or thermal sink, and because the resultant pressurization causes a large power excursion, or because other ATWS transients ultimately degrade into this type. Resolution of this ongoing safety concern can only be effected through plant transient simulation.

The analytic tool used in this study is RELAP5/MOD1.6. This version of RELAP5 has the capability to simulate BWR plants in that several special process models, such as a jet pump momentum mixer model, have been installed. The code has the capability to transport dissolved boron throughout the 1-D hydrodynamic solution using a strict donoring formulation, and to calculate the effect of this poison on the reactor power.

Extensive checkout and verification of the Browns Ferry BWR model was performed. We will present some results of the comparison of the model to plant transient data during both a two pump trip test, and a generator load rejection. In both cases, data comparison is quite favorable.

Subsequent to model verification, a nominal MSIV closure ATWS simulation was performed. This benchmarking effort assumed that the operator of the plant adhered to the General Electric's Emergency Procedure Guidelines (EPG), and that all of his actions tend to maximize the mitigation of the transient. The results of this study compare favorable to those of General Electric. The nuclear core is effectively shutdown by the standby liquid control system and the transient is relatively benign.

Future ATWS studies are planned. The probability and risk assessment group at the INEL is analyzing deviations from the EPG nominal case to ascertain the most probable route to core damage. This sequence will then be simulated using the benchmarked RELAP5 model. The results of the analysis will be fed to both the containment and core damage groups of the INEL and Oak Ridge National Laboratory.

CONTEMPT BROWNS FERRY STUDY, BWR ATWS

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Under the auspices of the U.S. Nuclear Regulatory Commission (NRC) extensive safety analyses by the Accident Sequence Evaluation Program and the Interim Reliability Evaluation Program have identified plant specific boiling water reactor accident sequences that are the most likely occurrences at the Tennessee Valley Authority's Browns Ferry Nuclear Plant (BFNP). Of these sequences, anticipated transients without scram (ATWS) are scenarios of great interest.

Part of the Idaho National Engineering Laboratory charter is to perform thermal-hydraulic analyses and predict front-end containment behavior during an accident. CONTEMPT/LT-028 was selected for this purpose from several codes because it was judged to have the best potential for immediate success in deterministic containment behavior studies. However, CONTEMPT/LT-028 was originally developed as a tool for short term LOCA analyses, and the applicability of the code for long term accidents and operational plant transients remained an open question. Therefore, under the Severe Accident Sequence Analysis Program, efforts were undertaken to model an ATWS scenario to demonstrate CONTEMPT code feasibility and limitations. The objective was to model BFNP containment behavior driven by primary system failure to scram. The CONTEMPT code was supplied with boundary conditions from RELAP5/MOD1.6, and the transient was structured in the spirit of the General Electric Emergency Procedure Guidelines.

Preliminary calculations show that CONTEMPT/LT-028 is indeed a viable tool for accident sequence analysis. The ATWS analysis is subject to the significant limitation that the code remains unqualified by actual BFNP transient data. However, qualitative results indicate that a scram failure, when handled according to the General Electric Emergency Procedure Guidelines, is a benign transient. Assuming that the pressure suppression pool is well mixed and that steam condensation in the pool is complete, CONTEMPT/LT-028 calculates that the Mark I pressure suppression design performs its intended function.

SCDAP SEVERE CORE DAMAGE STUDIES:

BWR ATWS AND PWR STATION BLACKOUT

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The Severe Accident Sequence Analysis (SASA) Program, sponsored by the U.S. Nuclear Regulatory Commission (NRC), is addressing a number of accident scenarios that potentially pose a health hazard to the public. Two of the scenarios being analyzed in detail at the Idaho National Engineering Laboratory (INEL) are the station blackout at the Bellefonte nuclear plant and the anticipated transient without scram (ATWS) at the Browns Ferry-1 plant. These scenarios were identified from probability risk assessment (PRA) studies as being the highest probability events that could lead to a degraded nuclear reactor core.

The INEL analyses of the station blackout and ATWS have been divided into four parts, which represent the sequence being followed in this study. First, the evaluation of long term irradiation effects prior to the station blackout or ATWS was conducted using the FRAPCON-2 fuel rod behavior code; second, the reactor primary and secondary coolant system behavior is being analyzed with the RELAP5 code; third, the degradation of the core is being analyzed with the SCDAP code; and finally, the containment building response is being analyzed with the CONTEMPT code. This paper addresses only the SCDAP/MODO degraded core analyses for both the station blackout and ATWS scenarios. Other papers in this session address the other aspects of the INEL effort.

Since SCDAP models the behavior of only the reactor core region, the boundary conditions to the core must be provided by a coolant system code such as RELAP5, or must be estimated using engineering judgement. For these analyses, both RELAP5 calculated values and engineering judgement were used to establish core boundary conditions.

For the station blackout case, the RELAP5 calculation indicated that core uncovery would occur and the hot rod in the core would attain 1478 K (2200 °F) by about 2250 s. At this time, the coolant system pressure was essentially constant at 15.9 MPa, the inlet coolant mass flow rate was 8 kg/s, and reactor power was obtained solely from fission product decay heat. These conditions were used as core boundary conditions to initialize the SCDAP calculations prior to 2250 s after event initiation when the RELAP5 calculation was terminated, as well as beyond 2250 s when boundary conditions are believed to remain essentially constant.

Good agreement was noted between RELAP5 and SCDAP calculated temperatures prior to 2250 s, when SCDAP was being initialized with the RELAP5 boundary conditions.

The first signs of core damage were noted at 2600 s, when cladding ballooning and rupture started to occur. At 3200 s, a cladding surface temperature of 1850 K was attained at the hot spot. Thereafter, enhanced cladding oxidation and hydrogen generation started to occur, as modeled in SCDAP/MODO by the Urbanic-Hiedrich model. The first signs of fuel rod melting were noted at 3300 s, when part of the hot zircaloy cladding melted and flowed downward to cooler portions of the core. The full extent of core melt was not known at the time of this writing, but will be reported in the full proceedings.

For the ATWS case, the RELAP5 calculation that used proposed Emergency Procedure Guidelines (EPGs) as scenario decision logic, did not calculate elevated core temperatures to occur, even though the core was partially uncovered. As a result, SCDAP calculations for the ATWS case have not been performed to date. Alternate pathways using the operator action event trees are currently being investigated to determined if core damage can be attained. Future SCDAP analyses of the ATWS are pending the results of the investigation.

Future severe core damage studies will support many SASA-related issues. Work will continue in the PRA-dominant scenarios area. The station blackout, ATWS, and their derivatives will be evaluated for not only hydrogen and fission product source term data, but also for accident management insights.

PWR SHUTDOWN DECAY HEAT REMOVAL ANALYSES IN SUPPORT OF TAP A-45

by

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INTRODUCTION

The primary method for removal of decay heat from pressurized water reactors (PWRs) is via the steam generators to the secondary system. This energy is transferred on the secondary side to either the main feedwater or auxiliary feedwater systems. The probablistic risk assessment reported in WASH 1400, later reliability studies, and related experience from the Three Mile Island Unit 2 accident have reaffirmed that the loss of capability to remove heat through the steam generators is a significant contributor to the possibility of core damage.

The US Nuclear Regulatory Commission (NRC) currently considers the adequacy of shutdown decay heat removal to be an unresolved safety issue (USI A-45). The purpose of Task Action Plan (TAP) A-45 is to "evaluate the adequacy of current licensing design requirements, to ensure that nuclear power plants do not pose an unacceptable risk because of failure to remove shutdown decay heat." A major part of TAP A-45 is concerned with the transition from reactor trip to hot shutdown. Also of interest is the transition from hot shutdown to cold shutdown and maintaining cold shutdown conditions. Although a limited number of alternative means for removal of shutdown decay heat from PWRs are being examined by the NRC, this paper will focus on activities at the Los Alamos National Laboratory to investigate the application of the "feed and bleed" concept as a diverse alternative method of removing decay heat which does not rely on the use of the steam generators.

ANALYSIS ACTIVITIES

An extensive program of decay heat removal analysis using the TRAC-PF1 code is in progress at Los Alamos National Laboratory. Thermal-hydraulic analyses of accidents involving loss of all secondary cooling are being performed. These studies are evaluating the capability of Babcock and Wilcox (B&W), Combustion Engineering (CE), and Westinghouse (W) plants to remove decay heat using a "feed-and-bleed" operation. Audited plant models for specific plants have been developed and are in use. The specific plants are Oconee-1 (B&W), Calvert Cliffs-1 (CE), and Zion-1 (W). A common set of transient analyses has been identified and are being performed for each plant. The set of transients consists of 1) a loss-of-offsite power (LOSP) induced loss-of-feedwater (LOFW) event, 2) a LOFW event, 3) a combined main steamline break (MSLB) and LOFW event, 4) a combined single tube steam generator tube rupture (SGTR) and LOFW event, and 5) a combined main feedwater line break (MFLB) and LOFW. For each of the five events in the common set, a minimum of three transients are calculated. First a base-line transient is calculated for which there is no actuation of the safety injection (SI) system and no operator intervention. This transient which leads to core dryout, establishes the timing of critical events such as steam-generator dryout, primary system saturation, containment overpressure, and the start of core heatup. The second transient evaluates plant thermal-hydraulic performance considering "feed" only operation after the safety-injection (SI) system signal, usually containment overpressure. The "feed" of the emergency core coolant (ECC) is at a primary system pressure determined by the power-operated relief valve (PORV) setpoint. The third transient evaluates the effectiveness of "feed and bleed" procedure conducted according to the appropriate operator emergency guidelines. An additional transient is calculated using a LOFW initiator to determine the effectiveness of a "feed and bleed" procedure in cooling and depressurizing the plant to the design operating conditions of the residual heat removal (RHR) system.

STATUS AND RESULTS TO DATE

The status and results obtained to date are presented below for each plant type. It should be noted that the results are for the specific plant models indicated and do not apply to all reactors manufactured by the vendor. However, efforts are in progress to extend the conclusions to similar plants of the same vendor.

Oconee-1 (B&W)

All calculations in the common transient set have been completed with the exception of the combined SGTR/LOFW transient. Compared to CE and W reactors, steam-generator-secondary dryout occurs early in the transient because the once-through steam generators have less secondary liquid inventory. However, the Oconee-1 ECC flow capacity at the PORV setpoint is large and the plant can successfully maintain a stable operation in the "feed" mode until the water supply is exhausted. If two high pressure injection (HPI) pumps are actuated before 1600 s for the LOSP transient and 900 s for the LOFW transient subcooling can be maintained. Operating in the "feed and bleed" mode aids cooling by lowering the primary pressure and increasing HPI output. Although the early (≤1000 s) character of the combined MSLB/LOFW transient differs from the LOFW transient, after 1000 s the transients are similar and the conclusions for the LOFW transient apply. The combined MFLB/LOFW transient is similar to a LOFW transient with a compressed time scale to steam-generator-secondary dryout and a similar timescale thereafter. "Feed and bleed" cooling in a once-through mode can be maintained for approximately 9 h. Cooldown and depressurization to RHR system design conditions was not successful. Either additional relief capacity or additional long-term water supply is required.

Zion-1 (Westinghouse)

The LOSP, LOFW and combined MSLB/LOFW calculations have been completed. The Zion-1 plant is characterized by a large PORV relief capacity and an ECC system incorporating centrifugal charging pumps that provide moderate flows at the PORV setpoint and high head SI pumps that begin delivering at an intermediate pressure (< 1500 psig). The plant can successfully maintain a stable operation in the "feed" mode until the water supply is exhausted. The latest time at which injection can be started and still maintain subcooling has not been determined. Operating in the "feed and bleed" mode aids cooling by lowering the primary pressure and increasing ECC output. Cooldown and depressurization to RHR system design conditions using a "feed and bleed" procedure was successfully accomplished. As with Oconee-1, the early character (< 3000 s) of the combined MSLB/LOFW transient differs from the LOFW transient. After dryout of the steam generators with intact steam lines at ~3100 s, the transients are similar and the conclusions for the LOFW transient apply.

Calvert Cliffs-1 (CE)

Only the LOSP and LOFW transients have been completed. The Calvert Cliffs-1 plant is characterized by an intermediate PORV relief capacity (larger than Oconee-1 and smaller than Zion-1), and a low SI capacity at the PORV setpoint. For the LOSP transient, the plant can maintain a stable operation when "feed" is initiated on containment overpressure. The plant can be maintained in this condition until the water supply is exhausted. For the LOFW transient, the plant cannot maintain a stable operation and cool the core when "feed" is initiated on containment overpressure. The limited SI capacity at the PORV setpoint is insufficient when the reactor coolant pumps (RCPs) are operating and adding ~17.5 MW(t) to the primary system. A "feed and bleed" operation will be required for this plant and is expected to be successful.

THE EFFECT OF SMALL-CAPACITY HIGH-PRESSURE INJECTION SYSTEMS ON BWR TRANSIENT INITIATED LOSS OF INJECTION ACCIDENT SEQUENCES

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The Severe Accident Sequence Analysis (SASA) Program* at Oak Ridge National Laboratory (ORNL) performs detailed BWR accident analysis for dominant sequences previously identified by the application of probabilistic risk assessment (PRA) methodology. One of the dominant accident scenarios identified by almost every BWR PRA is the transient loss of vessel water injection sequence (designated as the TQUV sequence). Due to the multiplicity of both high- and low-pressure injection systems, the loss of vessel water injection after a transient is an improbable event. However, typical PRA results assign a substantial fraction of the total core melt probability to this event. The percentage of total risk to the general public is even more significant since failure of all vessel water injection after scram is normally assumed to lead to an early core melt, followed by a large offsite release in less time than would be required for effective evacuation of the populace around the site.

In a TQUV sequence, there is a transient followed by reactor scram, followed by failure of all systems that would normally be relied upon to inject cooling water into the reactor vessel. If the reactor scram occurs from 100% power and no vessel makeup water is provided, the uncovering of active fuel begins after about 0.5 h, and is followed by destruction of the core if sufficient makeup water flow is not initiated. This paper, based on a recently completed SASA study,† provides an analysis of the ability of the Control Rod Drive Hydraulic System (CRDHS) and the Standby Liquid Control (SLC) System to adequately cool the reactor core during TQUV accident sequences at Browns Ferry. These small-capacity, high-pressure injection systems have traditionally been ignored in BWR risk assessment.

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*Research sponsored by the U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research under Interagency Agreements DOE 40-551-75 and 40-552-75 with the U.S. Department of Energy under contract W-7405-eng-26 with the Union Carbide Corporation.

†R. M. Harrington and L. J. Ott, "The Effect of Small-Capacity, High-Pressure Injection Systems on TQUV Sequences at Browns Ferry Unit One," NUREG/CR-3179, ORNL/TM-8635 (August 1983).

It is possible for the small capacity systems to have a significant effect on TQUV sequences because only moderate flows are required for shutdown cooling. The CRDHS can provide enough flow to replace coolant lost due to decay heat steam production within 0.5 h after a reactor scram from 100% power.

The CRDHS operates continuously during normal power operation, injecting a flow of 3.79 L/s (60 gpm) of water into the reactor vessel. After a reactor scram, the CRDHS flow increases automatically to ∿7.06 L/s (∿112 gpm), high-pressure injection, and with operator action this flow can be increased to as much as 18.9 L/s (300 gpm). The SLC system is on standby for operator actuation in the event of an ATWS but, with some realignment, a single, positive displacement, SLC pump can inject ∿3.53 L/s (∿56 gpm) of unborated high-pressure water into the reactor vessel. The CRDHS pumps are centrifugal pumps and; therefore, additional CRDH injection is possible at lower vessel pressures.

The beneficial effect of these low capacity injection systems on the TQUV scenario depends on sequence-specific events: initial conditions (i.e., pre-scram power level, time after scram of injection failure, etc.), CRDH and SLC systems availability, and operator action. This study has considered various sequence-specific scenarios. The study was performed on a representative U.S. BWR plant of the GE Mark I design, specifically, Unit 1 of the Tennessee Valley Authority's Browns Ferry Nuclear Plant located near Athens, Alabama. Pre-core uncovery analysis utilized the ORNL-developed BWR-LACP computer code and post-core uncovery analysis was performed with an extensively modified ORNL version of MARCH 1.1.

BWR SEVERE ACCIDENT SEQUENCE ANALYSES AT ORNL
— SOME LESSONS LEARNED — *

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The Severe Accident Sequence Analysis (SASA) Program was established in October 1980 by the Division of Accident Evaluation of the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission (NRC), to study the possible effects of potential nuclear power plant accidents. Under the auspices of the program, boiling water reactor (BWR) studies are being conducted at the Oak Ridge National Laboratory (ORNL) using Browns Ferry Unit 1 as the model plant. Assistance and complete cooperation is provided by the plant owners and operators, the Tennessee Valley Authority (TVA).

The function of the SASA program at ORNL is to conduct detailed analyses of the dominant (most probable) BWR accident sequences, which have been identified by probabilistic risk assessments (PRAs). The objectives include determination of the sequence of events and the magnitude and timing of the associated fission product releases.

All studies to date have concerned Unit 1 of the Browns Ferry Nuclear Plant, a BWR-4 with MK-I containment design, which is typical of most BWR plants in operation today. Four accident studies have been completed, resulting in recommendations for improvements in system design, emergency procedures, and operator training. Necessary severe accident analysis computer code improvements and modifications are an important by-product. The fission product transport work has indicated areas where further basic research is needed. The published reports concern Station Blackout (NUREG/CR-2182), Scram Discharge Volume Break (NUREG/CR-2672), Loss of Decay Heat Removal (NUREG/CR-2973), and Loss of Injection (NUREG/CR-3179).

The ORNL SASA studies have shown that several BWR modeling needs for Severe Accident analyses are unique and very important. The channel boxes and control blades in the core and the operation of the safety/relief valves must be represented if adequate analyses of the period of core uncovery and fuel heatup are to be performed. After core degradation, the presence of the control rod drive guide tubes and mechanism assemblies in the reactor vessel lower plenum and the numerous penetrations in the reactor vessel bottom head must be considered in the analyses.

*Research sponsored by the U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research under Interagency Agreements DOE 40-551-75 and 40-552-75 with the U.S. Department of Energy under contract W-7405-eng-26 with the Union Carbide Corporation.

In the course of performing the Browns Ferry accident analyses, many lessons have been learned regarding the response of Browns Ferry Unit 1 to the conditions of a Severe Accident. It is the purpose of the paper to discuss some of the more important points, which are of general applicability to BWR plants of this design. The points discussed in the paper are listed below:

1) The automatic and irreversible shift of the HPCI pump suction from the condensate storage tank to the pressure suppression pool upon high sensed pressure suppression pool level threatens the viability of the system in most Severe Accident sequences.

2) The ability to inject water into the reactor vessel from the condensate storage tank is important to the prevention and mitigation of Severe Accidents; the condensate storage tank capacity should be large.

3) The BWR scram system automatically provides increased reactor vessel injection when a scram is in effect.

4) A manual scram follow-up to an ineffective scram cannot be accomplished unless the initial scram can be reset.

5) The action of the reactor building fire protection sprays in a BWR is analogous to that of the Primary Containment sprays in a PWR. The effect is important in BWR accident sequences in which the pressure suppression pool is bypassed.

6) A large-capacity standby gas treatment system can provide filtered exhaust of all releases into the BWR secondary containment even if the reactor building and refueling floor blowout panels are lifted early in the accident sequence.

7) The magnitude of the fission product releases to the atmosphere during a Severe Accident would depend heavily on the timing of the releases. Releases subsequent to reactor vessel failure have a greater potential for reaching the environment.

A study of Anticipated Transient Without Scram (ATWS) accident sequences for Browns Ferry Unit 1 is currently underway at ORNL. This study involves a cooperative effort between the SASA and Human Factors programs at ORNL and the SASA program at Idaho National Engineering Laboratory as well as pre-publication review of the results by both TVA and General Electric Nuclear Division representatives. A draft report is scheduled for March 1984.

FISSION PRODUCT TRANSPORT ANALYSIS FOR THE LOSS OF DECAY HEAT REMOVAL ACCIDENT SEQUENCE AT BROWNS FERRY*

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As part of the Severe Accident Sequence Analysis (SASA) Program at Oak Ridge National Laboratory (ORNL), radionuclide transport and deposition within the nuclear plant are estimated for individual BWR core-melt accident sequences. Previous SASA efforts[1,2] have estimated radioactive source terms for the station blackout accident sequence and the scram discharge volume line break accident sequence, both at Browns Ferry Unit 1 (a Mark I BWR). The overall emphasis in all the SASA fission product transport analyses is on increasing the level of realism in source term estimation by considering previously undertreated factors, for example, by considering the effects of physicochemical interactions, the detailed impacts of potentially important plant systems (including systems other than engineered safety features), and the effects of coupling thermal-hydraulic and radionuclide transport considerations. In line with the primary SASA fission product transport goal of improving the understanding of the factors which may significantly affect radioactive accident source terms, both model sensitivities and overall sources of uncertainties are explored.

Because the emphasis is on considering heretofore undertreated factors, currently available approaches are not generally adequate to perform the indicated fission product transport analyses. Thus the assumptions and procedures used in SASA to estimate the accident source terms are a composite of methods previously developed by others and of ones developed at ORNL specifically for the SASA transport analyses. Some of these methods have been described in prior reports.[1,2] Thermal-hydraulic input for the transport analyses is based on calculations obtained using MARCH as modified at ORNL for problems specific to BWRs,[3-5] as well as on additional calculations performed to describe the environment in the secondary containment.[6]

Factors considered in the current analysis for the loss of decay heat removal accident sequence at Browns Ferry include initial releases from the core materials (both in-vessel and ex-vessel), plateout and reactions in the coolant system and the drywell, scrubbing by the suppression pool, both

*Research sponsored by the Division of Accident Evaluation of the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission under Interagency Agreements DOE 40-551-75 and 40-552-75 with the U.S. Department of Energy under contract W-7405-eng-26 with Union Carbide Corporation.

natural and spray-induced removal processes in the secondary containment, and processing by the standby gas treatment system. Four Reactor Safety Study radionuclide groups (noble gases, halogens, alkali metals, and "chalcogens"), as well as structural and control rod materials, are studied in the current analysis. In addition to "major" radionuclide pathways to the environment (such as escape from the reactor building or refueling bay), leakage pathways (such as escape via the main steam isolation valves to the turbine building,...) are also considered.

Among the topics discussed in the paper are the following:

1. the impacts of chemical interactions on the release and transport of iodine, cesium, tellurium, and antimony;

2. the effects of reactions of control rod materials (boron carbide) on radionuclide transport and deposition;

3. the possible modes of operation of the reactor building fire protection system sprays during severe accidents and their effects on source terms;

4. the relative importance of certain "secondary" leakage pathways; and

5. some major sources of uncertainty in the estimated source terms.

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ANALYSIS OF BWR DEGRADED CORE PHENOMENA AT RENSSELAER POLYTECHNIC INSTITUTE (RPI)

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The purpose of the research performed in the Department of Nuclear Engineering at Rensselaer Polytechnic Institute (RPI) on the BWR severe accident analysis* was to provide ORNL with new, improved models for the in-core phenomena leading to core degradation and meltdown. These new models should be used to enhance the applicability of the MARCH code for BWR accident analysis.

As a result of the work accomplished so far, a new BWR core model has been developed and implemented into a computer code called MELRPI. This code can be used as a subroutine to augment the existing BOIL subroutine in MARCH, or can be run as an independent computer program.

MELRPI addresses a wide range of problems, specific for BWR accident phenomena and important for a realistic assessment of the consequences of reactor severe accidents. Among others, a two-dimensional, multi-material core nodalization scheme is used to account for such specific components of a BWR core as, cannisters, and cruciform control rods. The nodal grid is fixed for the intact geometry, but may vary with time for the rubblized geometry, allowing for evaluating the core deformation process due to the debris bed formation in various nodes. This nodalization scheme is used in deriving a two-dimensional model for core heat transfer. The modes for heat transfer in the transverse direction are: heat conduction between fuel and cladding, heat convection/radiation between the fuel rods/channel boxes/control blades and the coolant, and the radiation heat transfer between the fuel rods, channel boxes, control blades, and the core shroud. The axial heat transfer model accounts for node-to-node heat conduction, and heat radiation to the upper and lower core structures. In the model for coolant thermal-hydraulics, two axial core regions are considered: the single-phase/two-phase pool, and the dry stream-hydrogen mixture region. The coolant flow in each node is divided into two parts: one in the coolant channels and the other in the interstitial region. In order to evaluate the hydrogen production rate, the zirc oxidation model developed for the ZRWATR subroutine of MARCH is used for both cladding and canister walls. In addition, a separate model is used for stainless steel oxidation.

Three different fuel failure modes are considered in MELRPI. Two of them deal with clad breach (from inside and outside), the third one is the criterion for rod slumping and rubble bed formation. The relocation of molten materials, such as zircaloy and stainless steel, is evaluated based on the model of a liquefied slug falling down along a solid surface. If the slug freezes, the thickness of the resulting crust is used to calculate the reduction in the area for coolant flow.

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* This work was sponsored by the Severe Accident Sequence Analysis (SASA) Program at Oak Ridge National Laboratory (ORNL) under the Containment Systems Research Branch of the Division of Accident Evaluation, Office of Nuclear Regulatory Research.

If the conditions for rubble bed formation are met in an arbitrary node, the intact geometry model for this node is replaced by a rubble bed model. The new model is characterized by such parameters as: the node porosity and average temperature, and mass of component materials. The volume (height) of a rubblized node is a function of the node porosity and the mass of solid components (which can decrease due to melting). When the only component left in a node is UO_2, and it starts melting, too, it is assumed that the molten fuel fills up the pores in a solid debris. The melting process can propagate both upwards and downwards, with the bottom of the fuel region located at the bottom of the lowest node to reach the melting temperature of UO_2. When molten fuel appears in the core bottom nodes, it starts leaking to the lower plenum.

The numerical method, applied in MELRPI for integrating the system of differential equations describing the core model, is based on the IMSL library routine DGEAR, and uses such specific techniques as dynamic resetting of the vector of state variables, for instance.

MELRPI was tested for various BWR accident scenarios following core dry-out. Some results of the test runs are presented in this paper.

In addition, to the models mentioned earlier, the new version of MELRPI, which is being finished now, includes a model for emergency core cooling systems (ECCS) used in both BWR/4 and BWR/5-6.

Although the MELRPI code accounts for various phenomena important for BWR severe accident analysis, some of these phenomena are modeled by using a simplistic approach, and require further development. Among others, the following improvements are planned for the near future: a space dependent model of the simultaneous mass and heat transfer between the liquefied slug and solid structures, a mechanistic model for molten fuel propagation, and a new thermal-hydraulic model for the BWR recirculation system, to replace a simplistic approach used in MARCH.

SAND83-1829A

Ice-Condenser Containment Loadings During Severe Accidents*

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Sandia National Laboratories is participating in several NRC-sponsored programs to study severe accident phenomenology. Part of that effort involves the combined use of the computer codes MARCH and HECTR to examine hydrogen behavior during severe accidents. MARCH has been used to model the primary system and provide hydrogen and steam source terms for HECTR. HECTR has been used to model the containment pressure-temperature response. The application of these codes to an ice-condenser containment is described in this paper. Sequoyah was used as the reference plant for this analysis; however, comparable results would be expected for similar ice-condenser plants such as Watts Bar.

MARCH 1.1, with certain improvements in the modeling of combustion and in-vessel steam generation, was used to generate hydrogen and steam source terms. The scenarios analyzed were initiated by either small or intermediate diameter pipe breaks or total-loss-of-feedwater transients. Three different paths of accident progression were then assumed. One assumption was that emergency core cooling was reestablished such that core melt was arrested and 75% of the Zr was reacted. The other two paths assumed progression through complete core melt and vessel failure, the difference arising in the total fraction of Zr reaction. In one case, assumptions were made with MARCH input parameters such that Zr oxidation was minimized, while in the other case 100% Zr oxidation was assumed.

HECTR is a lumped-volume containment analysis code developed at Sandia for calculating the containment response to hydrogen burns. A nine compartment containment model was used in this analysis. The compartments included the dome, the ice-condenser upper and lower plenums, the lower compartment, a dead-ended region, and four ice regions.

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

A large number of runs were made as part of this analysis to investigate the sensitivity of the results to combustion parameters, engineered safety feature availabilities, phenomenological parameters governing the steam-hydrogen source term, and the surface heat transfer correlation. In the interest of code comparison, runs were also made with HECTR using input assumptions similar to those used in previously reported analyses performed with CLASIX and COMPARE. In the core-melt cases, the potential benefits of a containment vent were also investigated.

Summarizing, we found the results to be very sensitive to parameters describing the combustion processes, the steam-hydrogen source and availability of engineered safety features. Both the characteristics of individual burns and the burn sequences were altered when these parameters were changed. In general, unavailability of the containment air recirculation fans or containment sprays, and higher values for ignition limits, combustion completeness and flame speed, produced higher burn pressures. Failure of the fans usually resulted in steam inerting of the lower compartment and high hydrogen concentrations in the ice-condenser. The results were relatively less sensitive to changes in the ice-condenser modeling assumptions. The potential benefits of a containment vent were strongly dependent upon the fraction of Zr reacted and the assumed failure pressure of the containment.

In our judgement, there is not yet enough information available to determine which set of assumptions are the most "realistic". Future analytic and experimental research, including large scale combustion tests and the development of more phenomenologically based core degradation codes, may permit more definitive statements to be made from future HECTR analyses.

Structural Analyses of PWR Containments
Subjected to Internal Pressurization*

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Introduction

Structural analyses of the Watts Bar, Maine Yankee and Bellefonte containment structures were performed as part of the Severe Accident Sequence Analysis (SASA) program. The objective of these analyses was to obtain realistic estimates of the ultimate pressure capabilities of these containments.

The three containments considered are representative of the different containment types. The Watts Bar containment is a hybrid steel type, the Maine Yankee containment is reinforced concrete and the Bellefonte containment is prestressed concrete.

Watts Bar

The Watts Bar containment structure is a ring-stiffened steel structure consisting of a cylindrical wall, hemispherical dome and a bottom liner plate encased in concrete. Analyses of the containment shell (without penetrations), equipment hatch, containment anchorage system, and personnel lock were performed.

Since the objective of the analyses was to obtain realistic estimates of the ultimate internal pressure capacity of the containment, it was necessary to use analysis techniques which are valid for loadings beyond the initial yielding of the material. The finite element analyses conducted were performed with either the MARC or ABAQUS finite element computer codes. Both codes have large deformation and finite strain capabilities. Average actual material properties were used for these analyses so that realistic results would be obtained.

An axisymmetric shell analysis of the containment shell subjected to static internal pressurization was performed using the MARC finite element code. The results indicate that onset of general yielding of the cylindrical wall will occur at 120 psig but after significant yielding of the shell, failure (based on a maximum von Mises stress criteria) would occur in the dome region at 175 psig. However, the maximum radial deflection of the cylindrical wall at this pressure is of the order of 40 inches suggesting that failures related to excessive deformation, e.g., piping failures or shield building interactions are possible. Thus a reasonable failure pressure range for the containment shell is between 120 and 175 psig.

* Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under Memorandum of Understanding DOE-40-550-75 with the U.S. Department of Energy.

An axisymmetric analysis of the Watts Bar equipment hatch subjected to static pressurization was performed to determine if buckling could cause a loss of seal and, hence, containment capability. From the elastic-plastic analysis results (using the ABAQUS code) it was concluded that buckling would occur at 140 psig. This is significantly lower than the 230 psig value produced from an eigenvalue estimate for the lowest buckling mode. This indicates the potentially non-conservative nature of using elastic eigenvalue estimates for buckling loads.

A finite element analysis of the door and bulkhead of the personnel lock subjected to pressurization up to 150 psig was performed. Although large deformation and significant yielding occurred, no strong statements about failure could be made.

Yielding of tie down bolts is expected to occur at 172 psig.

From the above analyses it was concluded that a realistic pressure range for the Watts Bar containment is between 120 and 140 psig. These pressure levels correspond to initial containment yielding and equipment hatch buckling.

Maine Yankee

Axisymmetric analyses of the reinforced concrete Maine Yankee containment subjected to static internal pressurization was performed using the ADINA finite element code. This program was selected because of its concrete material model which allows for concrete cracking and crushing. Two analyses were performed. The first did not allow for basemat uplift while the second did. The results from the first analysis indicated that extensive hoop concrete cracking would occur at 31 psig, liner yielding would begin at 73 psig and that hoop reinforcing bar yielding would occur at 118 psig. The second analysis which allowed for basemat uplift showed almost identical behavior through first liner yielding, but more extensive damage to the concrete in the cylinder wall basemat region was detected. In fact, this analysis was terminated at 96 psig due to a numerical instability associated with points in the cylinder wall-basemat region. Whether the instability corresponds to a true structural failure is questionable and, in fact, may be a shortcoming in the current state-of-the-art in concrete analysis.

Bellefonte

The Bellefonte containment is a prestressed concrete containment. Finite element analyses of the containment shell and equipment hatch using the ABAQUS code were conducted. Yielding of the dome tendons was calculated to occur at 130 psig and general yielding of the cylinder wall tendons is expected to occur at 139 psig. The equipment hatch capability is expected to be above 150 psig.

BNL SEVERE ACCIDENT SEQUENCE EXPERIMENTS AND ANALYSIS PROGRAM*

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In the analysis of degraded core accidents, the two major sources of pressure loading on light water reactor containments are:

(i) steam generation from core debris-water thermal interactions, and

(ii) molten core-concrete interactions.

Experiments are in progress at BNL in support of analytical model development related to aspects of the above containment loading mechanisms. The work supports development and evaluation of the CORCON (Muir, 1981) and MARCH (Wooton, 1980) computer codes. Progress in the two programs is described below.

Core Debris Thermal Hydraulic Phenomenology

The thermal hydraulic quench heat transfer characteristics of superheated ex-vessel debris beds are discussed. Experimental results and analytical models are presented for quenching of superheated debris beds cooled by overlying pools of coolant. The experimental program is directed towards prediction of the steam generation rate and the quench behavior characteristics of ex-vessel debris beds. Results are presented for packed beds of 0.89 mm - 12.7 mm spheres heated to initial temperatures of up to 977 K. The beds were contained in a preheated 100-mm diameter pipe, and bed heights up to 400 mm were studied. Water was poured over the bed and the steam generation rate and bed temperature transients were recorded. Selected data are presented. The results for beds of particles with diameter less than 6.35 mm indicate that they cooled in a bi-frontal quench process. Following arrival of the water, a cooling front moved down the column. The portion of the bed ahead of the front showed no evidence of cooling until arrival of the front. Upon arrival of the front to the base of the bed, approximately 60% of the internal energy remained stored in the particle. A second front then proceeded up the column removing the remaining stored energy. The quench behavior of the 12.7 mm spheres was significantly different. In this case, water penetrated immediately to the base of the bed and began an upward-directed frontal refill process. An analytical model is described which characterizes the quench behavior observed experimentally. This model has been modified to include the effect of decay heating. Predictions based on the model are compared with the experimental data. Calculations are presented which describe the effect of decay heating on the debris bed quench characteristics.

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

To aid in the development and assessment of codes (e.g., SCDAP) dealing with the transient quenching of in-vessel debris beds, an experimental and model development program has been initiated. This program is directed towards the case when a superheated debris bed interacts with coolant that is injected from below. Experiments are being performed with stainless steel particle beds heated up to 810 K. Saturated water serves as the coolant, and is injected at a constant rate by means of a positive displacement pump. Measurements include pressure drop, steam flow rate at bed exit, and temperature traces at various points within the bed. Data from the initial series of experiments are being processed. Early results suggest that a "quench front" propagates up the particle bed. For small water injection rates (superficial water velocity at entrance, $J_\ell \leq 1$ mm/s), the quench front seems to be reasonably smooth and horizontal. At higher water injection rates ($J_\ell \geq 4$ mm/s), however, the quench front is highly irregular.

Heat Transfer in Core-Concrete Interactions

To analyze the interaction between molten core debris and concrete, the CORCON code was developed at Sandia National Laboratories. Separate effects experiments and related analyses are performed at BNL to develop or improve specific heat transfer models of phenomena that may occur during core-concrete interactions. One particular phenomenon which has recently been considered is the existence of a water coolant layer overlying the molten core debris as it attacks the concrete. The water layer would be in film boiling over another more dense, immiscible liquid layer with transverse non-condensable gas flux emanating from the concrete below and passing through the liquid-liquid boiling interface. At present, experiments are in progress with various coolants boiling on liquid metal pools without the non-condensable gas flux.

In the experiments performed to date, two distinctly different modes of boiling heat transfer have been observed. For the case of saturated R-11 boiling on a pool of liquid metal, a very stable, quiescent interface was photographically observed between the two liquids, similar to film boiling from a polished solid surface. Quantitative data for the boiling heat flux as a function of surface superheat indicates that the liquid-liquid film boiling data for this fluid are in good agreement with the Berenson film boiling model predictions.

However, for the case of water boiling on the surface of similar liquid metals, quite different behavior was observed. Photographic investigation revealed considerable interfacial mixing between the liquid metal and the water with no stable liquid-liquid interface. Quantitative experiments demonstrated the potential for steam explosions between liquid metal and water in layered pool geometry, with bismuth, lead, and Wood's metal. Work is continuing in this area with improved instrumentation, metallic as well as non-metallic melts, and non-condensable gas flux through the film boiling interface to develop a firmer understanding of liquid-liquid film boiling and pool-geometry vapor explosions and their impact upon core-concrete interactions.

COMPARATIVE ANALYSIS OF AEROSOL SOURCE TERMS DERIVED FROM THE ORNL CORE-MELT PROGRAM AND THE KfK SASCHA PROGRAM

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An evaluation of the variances between the ORNL core-melt release data and the KfK data continues to be of concern to both laboratories. Both projects make use of surrogate chemical additivies to simulate fission product buildup in UO_2. The difference in scale of the two melting experiments is about one order of magnitude, and the total aerosol mass release difference (structural material sources) also approaches this amount. Radiologically, the differences are small since the volatile components (iodine and cesium) are dominant; however, a more significant difference is apparent in the behavior of tellurium which is also of potential radiological importance. A high tellurium release is observed in the KfK experiments while at ORNL, the tellurium appears to remain with the residual metallic phase containing stainless steel, zirconium, silver, and other fission product metals.

Several contributing factors inherent in the different experimental approaches toward demonstrating core-melt release seem to account for the variances. However, there is no obvious means of resolving the differences, because they appear to be locked in with the unique approach of each of the experimental routines. These factors may be briefly outlined as follows:

1. Mass and surface area. The smaller scale SASCHA experiments seem to produce extra surface area by redistribution of the melt, and thereby form very thin, molten layers that are more readily subject to vaporization.

2. Hot-wall vs cold-wall environment. Because of the outside high-temperature heat source, the SASCHA melts are more likely to be overheated on contact with the crucible walls. In the ORNL core-melt configuration, the relatively cold walls (which cannot be overheated) maintain a frozen boundary layer on both the sides and bottom of the melt, through which vaporization is not possible.

3. Complete steam oxidation vs limited oxidation. Several indications tend to support the conclusion that steam oxidation of the metal constituents in the SASCHA tests is complete (or more nearly complete) than in the ORNL tests. The most striking evidence for this is in the fission-product simulant releases in which the effect of zirconium oxidation is most evident. These effects include both the extra high SASCHA releases of materials which could be dissolved in a residual zirconium phase if there were one (e.g., Sn, Te, and Ag), also the smaller releases of Sr and Ba which have been shown to depend upon reduction by hot metallic zirconium.

4. Core-melt accident sequence environment vs experimental configuration environment. A unique part of the ORNL approach to the core-melt release experiment is the extra effort to make the test environment simulate as closely as possible the actual meltdown conditions and the thermal and hydraulic conditions actually conceived for a specific accident sequence. In choosing the AB or AD PWR sequences (which are low-pressure, fast-melt processes) the rate of metal-water reaction and the rate of incorporation of the fuel bundle into a coherent melt phase is intended to match the stepwise predictions derived from the relevant MARCH code analysis.

A summary is given in Table 1 for both the latest ORNL and SASCHA fractional release values in order to compare the resulting release and vaporization data derived from these two experimental programs and also, a set of values from the NUREG-0772 follow-on study are given and they are obviously nearly the same as the SASCHA data. A further comparison of the total aerosol mass release estimate from a fast-melt sequence (before reactor vessel melt-through) is given in Table 2. The same extent of similarity between the Nuclear Regulatory Commission (NRC) estimates and the SASCHA data is evident.

Table 1. Comparison of fission product release data

Midlife reactor inventory[a] (kg)		Aerosol released (%)		
Element or compound	PWR	Measured in ORNL 1-kg ART experiments	Measured in small-scale SASCHA experiments	Values proposed by NUREG-0956[b]
Sr	54.3	7.8	0.2[c]	10.4
Mo	157.3	0.14	0.2	6.8[c]
Ru	111.8	0.0	0.2	0.8
Te	22.8	0.0[c]	81.0	76.5
I	11.8	100.0	100.0	88.3
Cs	139.1	100.0	100.0	88.5
Ba	69.6	6.6	0.2[c]	19.6
Ce (La)	155.9	0.0	0.2	
Sn	380.7	1.2[c]	20.0	54.7
Mn	40.3	12.0	18.0	
Ag	2,159.0	6.1[c]	75.0	76.5
In	342.3	5.4	20.0[c]	
Cd	265.9	53.0	100.0	
Fe	1,410.0	0.08[c]	2.0	1.5
UO_2	100,000.0	0.00[c]	0.4	0.23

[a]Structural materials, control rods, and fission products.
[b]NUREG-0772 follow-on study.
[c]Major differences.

Table 2. Comparison of whole-core aerosol release estimates

Constituents of aerosol	Mass (kg)		
	KfK[a] (1981)	ORNL[b] (1982)	NRC[c] (1983)
Ag, Cd, In	1800	290	1836
UO_2	450	10	260
Fe, FeO, Mn, Cr	630	7	470
Sn (from Zircaloy)	126	5	350
Fission products	465 (175)	160	164
Total	3471 (3181)	472	3080

[a]H. Albrecht and H. Wild, paper presented at the ANS topical Meeting on Reactor Safety, Sun Valley, Idaho, pp. 458-68, CONF-81-0803 (1981).
[b]G. W. Parker et al., paper presented at International Meeting on Thermal Nuclear Reactor Safety, p. 1078, NUREG/CP-0027 (1982).
[c]U.S. Nuclear Regulatory Commission, *Radiation Release Under Specific LWR Accident Conditions* (Draft), NUREG-0956 (1983).

The Ex-Vessel Fission Product Source Term*

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An uninterrupted core meltdown accident eventually leads to core materials being expelled from the reactor vessel and interacting in the reactor containment. The expulsion of the core debris, possible steam explosions as a result of debris interactions with water, and core debris/concrete interactions can be important sources of both radioactive and nonradioactive aerosol during a severe reactor accident. These sources of aerosol generation and fission product release are being characterized in research programs sponsored at Sandia National Laboratories by the Containment Systems Research Branch of the U.S. Nuclear Regulatory Commission.

Expulsion of debris from the reactor vessel was not considered to be a major aerosol source term in the Reactor Safety Study. Experiments have shown that when high temperature melts are ejected from a pressurized vessel, as would be expected in about 3/4 of all core meltdown accidents, intense aerosol generation occurs. The aerosols are size distributed with modes at nominally 0.5, 5, and 50 µm. Vaporization, gas effervescence, and hydrodynamic atomization are believed responsible for these modes in the size distribution. The composition of aerosols produced by condensation of vapor is dependent on the vapor pressures of the melt constituents. The other two processes are mechanical in nature and yield aerosols having the bulk melt composition. Because these aerosols would carry highly reactive metals into the containment atmosphere, it is possible that aerosol generation could affect containment integrity.

The Reactor Safety Study recognized steam explosions caused by molten material interactions with water to be a potential fission product release mechanism. Ruthenium was expected to be

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

the most important radionuclide released by this mechanism. Analyses are now indicating that kinetic factors may substantially inhibit the release of ruthenium from fragmented core debris produced by a steam explosion. These analyses suggest experimental verification of the steam explosion source term is needed.

Aerosols produced by core debris interactions with concrete were the dominant source of ex-vessel fission product release considered in the Reactor Safety Study. Aerosol production by this process has been studied in both large-scale and small-scale tests. Results of these tests have been used to formulate the VANESA model of aerosol production during melt/concrete interactions. Results obtained with this model agree well with experimental data. Applications of the model to reactor accident analyses show that aerosol production continues far longer than estimated in the Reactor Safety Study. Releases of radionuclides predicted by the model can be either higher or lower than estimated in the Reactor Safety Study depending on accident-specific features of the situation.

The revised ex-vessel source terms that come from this research can have dramatic effects on the overall estimate of the radiological consequences of severe accidents. Quantitative and qualitative changes in the inventory of radioactive material suspended in the containment atmosphere can be brought on by ex-vessel sources of aerosols.

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BEHAVIOR OF U_3O_8, Fe_2O_3, AND CONCRETE AEROSOLS IN A CONDENSING STEAM ENVIRONMENT

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A project is being conducted in the Nuclear Safety Pilot Plant (NSPP) to study the behavior of aerosols assumed to be generated during light water reactor (LWR) accident sequences and released into containment. The program plan provides for individual tests using U_3O_8, Fe_2O_3, or concrete aerosol, as well as for tests involving mixtures of these aerosols. This project is sponsored by the Division of Accident Evaluation, NRC, and the purpose is to provide experimental qualification for LWR aerosol behavioral codes being developed at other laboratories.

The NSPP facility includes a test containment vessel (38.3 m^3), plasma torch aerosol generators, aerosol sampling equipment, and steam system parameter measuring equipment. Aerosol measurements include mass concentration, fallout and plateout values, size distribution, and physical appearance. System parameters include moisture content of atmosphere, steam condensation rates on wall, atmosphere temperature and pressure, and wall temperature gradients.

The purpose of this paper is to relate observations on the behavior of U_3O_8 aerosol, Fe_2O_3 aerosol, and concrete aerosol in a condensing steam-air environment within the NSPP vessel. Tests have been conducted on each aerosol under both dry air conditions and steam-air conditions; the primary experimental variable has been aerosol mass concentration.

Under dry conditions the three aerosols behave in a somewhat similar manner in regard to rate of disappearance from the vessel atmosphere. The aerosols are agglomerated in the form of branched-chains; the aerodynamic mass median diameter (AMMD) of the U_3O_8 and Fe_2O_3 aerosols ranged between 1.5 and 3 μm while that of the concrete aerosol was about 1 μm.

A number of tests have been conducted in a condensing steam environment: five involved U_3O_8 aerosol, five involved Fe_2O_3 aerosol, and one involved a concrete aerosol. Each test was conducted under quasi-steady-state steam conditions. The test atmosphere was prepared by using steam to bring the vessel atmosphere (air) to the desired temperature and resultant pressure; upon achieving this condition the rate of steam injection was reduced and the accumulated steam condensate removed from the vessel. Low-level steam injection was maintained for six hours to balance steam losses by wall condensation.

Comparison of the behavior of each of these three aerosols reveals that a condensing steam environment, as would be present in LWR containment during and following an accident, causes the U_3O_8 and the Fe_2O_3 aerosols to behave in a manner different from that observed in a dry atmosphere; the primary effect is an enhanced rate of removal of the aerosol from the vessel atmosphere. The same observation cannot be made for the behavior of concrete aerosol; based upon the results of the two tests, the presence of steam does not seem to affect the removal rate to any great extent. Electron microscopy showed the agglomerated U_3O_8 and Fe_2O_3 aerosols to be in the form of spherical clumps of particles and not as intermingled branched chains as observed in the dry tests; the AMMD appeared to be in the range of 1 to 2 μm. In the case of concrete aerosol, the presence of steam seemed to have a lesser influence on the physical shape of the aerosol.

With respect to the U_3O_8 and Fe_2O_3 aerosols the enhanced rate of removal could be the result of several factors: the change in aerosol shape enhancing gravitational settling; condensation of water on particle surfaces increasing mass and enhancing settling; and the influence of steam flux (diffusiophoresis) and thermal gradients near the wall enhancing plateout. The lack of enhancement of the removal rate of concrete aerosol by steam may be the result of differences in surface characteristics of these aerosol particles.

Computer modeling of aerosol behavior in a steam environment is under way, both in this country and abroad, but as yet no code is fully qualified that can account for the experimentally observed influences of steam on the behavior of these aerosols in containment.

COMPARISON OF AEROSOL CODE PREDICTIONS WITH EXPERIMENTAL OBSERVATIONS ON THE BEHAVIOR OF AEROSOLS IN STEAM

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Several computer codes have been developed to predict the behavior of aerosols in steam, a situation which is expected to occur in a light-water reactor accident. Among the codes with capabilities in this respect are MAEROS, devised at Sandia National Laboratories, AEROMECH, under current development at the University of Missouri at Columbia, and NAUA-Mod4, which was originated at Kernforschungszentrum Karlsruhe.

NAUA is of particular interest because it was specifically developed to model steam-aerosol behavior in hypothetical accidents in pressurized water reactors. It is unique in being based upon an experimental program conducted in connection with the design of the code. In that series of experiments, aerosols of uranium oxide, platinum oxide, and sodium nitrate were generated in a steam atmosphere. A particular goal of the experiments was to verify the use of Mason's equation to model steam condensation on particle surfaces. It was concluded that aerosol shape factors could be set to unity and all agglomerated aerosol particles could be treated as waterlogged spheres of 50% theoretical density.

The input to NAUA permits arbitrary time variation of the aerosol source, the steam flow rate, the vessel temperature, and leakage. The code can accept assumed initial aerosol size distributions that are either bimodal log normal functions or tabulations. The fundamental equations allow for gravitational settling and diffusional plateout. Brownian and gravitational coagulation are included. Steam condensation on particles is calculated from Mason's equation which uses the steam saturation ratio continually determined by the NAUA code on the basis of the steam input rate and temperature conditions specified by the user. Finally, the code is of the "multi-channel" type, in which the size distribution is a calculated function of time.

Several series of experiments have been conducted in the NSPP vessel (38.3 m^3) using aerosols generated by plasma torch in a steam-air atmosphere. From these experiments we have selected those performed with iron oxide or uranium oxide for comparison with various computer codes, principally NAUA-Mod4. The experiments were performed at maximum concentrations ranging from 1 to 7.5 g/m^3 in the iron oxide experiments and 5.5 to 26 g/m^3 in those performed with uranium oxide. We discuss the effectiveness of the various assumptions made to model the experiments, using the aerosol concentrations calculated as a function of time as the main basis of comparison. Comparisons of particle size, while more uncertain, are displayed also. The results of parameter studies to determine the sensitivity of the calculated results to the steam input level, initial particle size, deposition parameters, and assumed particle density are presented.

SAND83-1828A

HYDROGEN PHENOMENOLOGY IN LWR ACCIDENTS*

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This paper describes recent results produced at Sandia National Laboratories in a program to study hydrogen behavior in light water reactors. During the past year, we have produced significant new experimental results, completed the construction of new experimental facilities, and developed additional analytical capability.

A wide variety of experiments have been completed during the past year dealing with deflagrations, accelerated flames, detonations, and diffusion flames. We have primarily used the FITS and VGES facilities to examine deflagrations, although the results produced by other researchers have also been examined. We examined mixtures of hydrogen:air and ternary mixtures of hydrogen:air:steam and have provided more insight into flammability limits, flame speeds, effects of igniter location and initial conditions, heat fluxes, etc. It has become apparent to us that many different parameters are important in understanding deflagrations, and we are now beginning to understand these diverse effects.

Small-scale experiments at McGill University have focused on the phenomena of flame acceleration and detonations. It has been shown experimentally that flames in the presence of certain types of obstacles can accelerate to speeds that are many times the laminar flame speeds. If the flames are highly accelerated, the burns become more adiabatic, and dynamic loads may result. This work is of particular importance because of the very complex geometries found in reactor containments. While current research has shown that flame acceleration clearly can occur under certain conditions, it also appears that transverse venting or, possibly, the addition of diluents can significantly reduce the flame speeds. We have recently completed the construction of the intermediate-scale (6' x 8' x 100') FLAME facility. This facility will be used to examine the effects of scale and will allow us to mock-up specific parts of reactor containments (such as an ice-condenser) that may be susceptible to flame acceleration.

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

Detonations are becoming increasingly well understood. We have now measured detonation cell size versus hydrogen concentration for a variety of mixtures. Experiments to examine various geometries have provided information regarding the number of cell widths necessary for a detonation to propagate. Combining these experiments, we can now predict with a high degree of confidence what hydrogen concentrations are required to propagate detonations in various simple geometries. What is not yet well understood is the initial transition to detonation. This phenomenon is being addressed as part of the effort to study flame acceleration.

We have examined diffusion flames in a small-scale steam:hydrogen jet facility. A significant finding was that stable flames are possible for mixtures that are very steam rich (steam-to-hydrogen ratios greater than 0.8). This work and the research of the Hydrogen Control Owners Group have indicated that diffusion flames may be important (and sometimes prevalent) in a wide range of reactor accidents.

Analytical efforts have focused on the development of the HECTR code. HECTR can now model all of the important containment systems in ice-condenser plants and Mark III BWRs. An analysis of ice-condenser plants is presented in another paper at this meeting. Analysis of the Grand Gulf Mark III BWR containment is currently underway. The combustion models in HECTR are based on FITS and VGES experimental data. The combustion models and other models in the code are being continuously updated as additional experimental data become available. The hydrogen transport models in the code have been recently checked against experimental data and have performed well. Accident analyses to date have emphasized the sensitivity of the results to a wide variety of input and modelling assumptions and the need for caution when interpreting the results of this and other containment analysis codes.

Additional analytical capability is being provided by the development of the Vortex-Dynamics code and the modification of the CONCHAS-SPRAY code. These codes are being used to model the details of flame propagation and acceleration in the presence of simple obstacles. These codes will be used to model future flame acceleration experiments, and, where appropriate, specific regions of containment where flame acceleration may be expected to occur.

An Overview of the CONTAIN Code for Severe Accident Analysis*

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The CONTAIN computer code is intended to be the Nuclear Regulatory Commission's principal calculational tool for the analysis of the physical and radiological conditions inside the containment building following a severe nuclear reactor accident. The code has been under development for a number of years, and an interim version was released to a small number of laboratories in the U.S. and abroad in March, 1981 to obtain review and feedback. Since then, substantial improvements and extensions have been made to the models and numerical analysis, and the first version of the code suitable for release to the general reactor safety analysis community is expected to be completed early in CY1984. This version will be designated CONTAIN 1.0.

A large number of phenomena may significantly affect containment conditions following a core-melt accident. For practical reasons, some compromises must be made in the level of analysis. For example, CONTAIN neglects any possible feedback effects which might occur between primary system conditions or ex-plant conditions and containment response. Thus, the code accepts source terms for steam, gas, aerosols, fission products and core debris as input in the form of tables, presumably generated by a primary system code. Similarly, CONTAIN provides sources of fission products, gases and aerosols to the outside environment, which can serve as input to a radiological consequence code. A feedback loop which is not ignored is the coupling between aerosol and fission product behavior and thermal-hydraulic conditions (this coupling makes CONTAIN unique among containment analysis codes.) That there can be important two-way interactions between aerosol behavior and thermal-hydraulic conditions has been shown in a number of

*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 Number DE-AC04-76DP00789.
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accident sequence analyses using CONTAIN. (For example, see the paper by Tills, et al. on Surry source term calculations at the end of this session.)

The level of analysis used in CONTAIN is as physically realistic as practical, but like most system level codes, the treatment is 0-dimensional (i.e., control volume approach) or at most 1-dimensional. However, the configuration is not restricted in the number of cells or type of interconnections; i.e., parallel flow topologies are allowed.

The code is relatively large, with approximately 40,000 lines of Fortran, but it has a modular structure which allows the incorporation of new or alternate models relatively easily. CONTAIN 1.0 will be written in standard Fortran 77, and will be designed to run on a wide variety of machines (CDC, Cray, IBM, Vax).

Models to be operational in CONTAIN 1.0 can be grouped into four categories, and are listed below:

1. Atmosphere Phenomena.

Aerosol behavior; Two-phase thermodynamics; Intercell flow; Condensation/evaporation from structures and aerosols; Heat conduction in structures; Hydrogen burns.

2. Pool Phenomena.

Coolant boiling and evaporation/condensation; Chemical reaction framework; Concrete ablation by molten debris; Heat transfer to structures and atmosphere.

3. Engineered Safety Features.

Containment spray (including fission product and aerosol scrubbing); Fan coolers; Ice condenser; Liquid transport network.

4. Radioisotope Inventory.

General fission product decay scheme; Transport via host materials; Input-controlled transfer between hosts; Decay heat due to individual isotopes and balance of ANSI standard decay heat.

The next two papers in this session will discuss the extensive validation efforts which have been undertaken in the past year and present results of calculations for source term reevaluations.

EXPERIMENTAL VALIDATION OF THE CONTAIN CODE*

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A substantial effort is presently being devoted to CONTAIN code validation; this effort includes participation in code validation exercises involving blind predictions of experiments, as well as the analyses of published experiments. The features of the code exercised in the blind predictions and experiment analyses include atmosphere thermodynamics, intercell flow, structure heat transfer, aerosol behavior, and spray removal of fission products.

CONTAIN has recently made highly successful blind predictions of the outcomes of the V44 steam blowdown test in the German HDR facility and the ABCOVE AB-5 aerosol test at HEDL. In addition analyses of the NSPP 504 aerosol experiment in a steam environment and of the A-9 spray experiment in the Containment Systems Experiment (CSE) series have been completed, with good agreement between the calculated and experimental results.

HDR Experiment V44

The V44 experiment (German Standard Problem No. 6) was one of a series of full-scale water and steam blowdown experiments conducted by KfK to simulate design basis loss-of-coolant accidents. A total of five computer codes, including CONTAIN, were used to produce blind predictions in this code validation exercise. With the exception of CONTAIN, these other codes were designed to analyze design basis accidents.

Recently, KfK published a Quick-Look report on this experiment (as well as on V42 and V43, in which CONTAIN did not participate). Comparisons between predicted and measured results over several different time scales (0-2 sec., 0-200 sec., and 0-300 min.) were presented in graphical form. An independent

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evaluation of all of the graphs presented for absolute pressure, differential pressure, and atmosphere temperature indicated that CONTAIN could be judged second in overall performance. The predictions were particularly good in the pressure and temperature relaxation region following the blowdown. This is the regime of most interest for a severe accident code like CONTAIN. The overall performance of CONTAIN is particularly impressive considering its orientation toward severe accident analysis.

The ABCOVE Program

The Aerosol Behavior Code Validation and Evaluation (ABCOVE) program is a cooperative effort between the USDOE and the USNRC to validate aerosol codes. The first two tests in the program, AB-5 and AB-6, have been conducted. CONTAIN blind predictions were submitted for both tests.

The AB-5 test was a high source rate, single species test. Six different laboratories participated, and six different codes were used to generate eleven sets of predictions. The MAEROS stand-alone aerosol code, which is the CONTAIN aerosol module, was run independently by HEDL. A HEDL draft report comparing all the code predictions with the experimental results is now available. In this report, the CONTAIN and MAEROS predictions for the suspended mass and the aerosol median size are ranked first and second in quality among the eleven sets of predictions.

The CONTAIN prediction for the suspended mass is in excellent agreement with experiment up to the point where the aerosol concentration is reduced by three decades from the peak value. It tracks the concentration for three more decades with agreement to within a half decade.

The AB-6 test is designed to test the treatment of co-agglomeration in the case of multiple aerosol species. The CONTAIN prediction for AB-6 will be discussed if the experimental results become available.

Experiment Analyses

The NSPP aerosol experiments in steam environments provide an opportunity to evaluate the modeling of aerosol processes not present in a dry environment. The analysis of NSPP 504 provides a check, in particular, on the modeling of diffusiophoresis. In addition, the CSE A-9 experiment provides an opportunity to evaluate the modeling of the spray removal of gaseous and aerosol fission products and the modeling of the natural depletion of the aerosols under condensing conditions in the atmosphere. These analyses have produced good agreement between the calculated and experimental results.

CONTAIN Calculations of Severe Accident Sequences at the Surry Nuclear Power Plant*

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The CONTAIN computer code has recently been used to analyze phenomena occurring in the Surry reactor containment during a postulated severe accident sequence. The main purpose of the calculation was to demonstrate CONTAIN as an integrated thermohydraulic and aerosol transport code, capable of handling complex application problems. The choice of the Surry plant provided an opportunity to utilize input data previously generated and reported in Reference 1. Analysis of Surry also meant that CONTAIN results could be compared to MARCH/NAUA calculations presented in that reference.

The accident sequence selected for analysis was the AB sequence which is a large loss of coolant accident (LOCA) followed by a failure of the electric power to engineered safety features (ESF). Tabulated data obtained from Reference 1 were reduced to provide source tables for ex-vessel fission product and aerosol release rates as required by CONTAIN. Sources of steam, hydrogen, carbon monoxide, and carbon dioxide to containment were obtained from MARCH 1.1 runs made at Sandia using input data supplied by Battelle. Structure geometry for the CONTAIN calculation was identical to MARCH input with the exception of a floor heat sink which was not present in the MARCH deck. Both the CONTAIN and MARCH calculations used a single cell for the containment volume.

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The CONTAIN calculation was run for two cases, AB-γ and AB-δ, representing early and late time containment failure, respectively. For the AB-γ calculation, failure of containment occurred at 72 minutes due to a pressure spike produced by a hydrogen burn. Since most of the reactor coolant system (RCS) aerosols were still airborne in containment at this time, the fraction released was high; approximately 60% of the core inventory of Cs and I was released, and 85% of the noble gases. (The percentages are totals released, not release _rates_, though the rates were also high.)

For the AB-δ sequence, the late time failure provided greater opportunity for natural deposition to occur within containment. It was observed in the CONTAIN calculation that aerosols generated in the reactor cavity subsequent to vessel failure can cause a rapid depletion of the RCS aerosols as a result of co-agglomeration of the aerosols with resulting enhanced fallout of the RCS aerosols.

The releases from the CONTAIN calculations for both AB-γ and AB-δ sequences are a factor of two to three higher than those from the MARCH/NAUA calculations. This difference arises because the MARCH/NAUA calculations predict that significant condensation of steam on the RCS aerosols and enhanced aerosol settling occur shortly before vessel failure. In the CONTAIN calculations, the containment atmosphere is superheated during this period, and no condensation on aerosols can occur. Comparison of the CONTAIN and MARCH/NAUA calculations indicates that the effect of condensation on aerosols in the MARCH/NAUA calculations is to deplete the aerosols released from the RCS prior to vessel failure. This extra depletion occurs in both the AB-γ and AB-δ MARCH/NAUA calculations.

Efforts are now continuing with CONTAIN to quantify effects of RCS and cavity aerosol mixing on the rate of aerosol deposition. Multi-compartment calculations are being run to determine the extent of mixing and the rate of aerosol deposition within containment.

Reference 1: "Radionuclide Release under Specific Accident Conditions," BMI2104, Volume 1, Battelle Columbus Laboratories, draft.

FLECHT-SEASET Blocked Bundle Test Results and Analysis

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Testing of the 163-rod bundle test results has been completed and a data report describing the 163-rod bundle data will be issued by October 1983. The data evaluation and analysis portion of this task has been revised and Westinghouse will be working with Battelle Northwest Laboratories to utilize the COBRA-TF code to analyze the 21-rod bundle blockage data as well as the 163 rod blockage data.

The 163-rod bundle utilized two 21-rod blockage islands which had stainless steel sleeves to simulate 44% peak strain and blockage. The sleeves are arranged in a non coplanar array which is represntative of multirod bundle burst tests like NRU or REBECA. Flooding rates below 2.54 cm/sec were investigated as well as other typical reflood parameters.

The 163-rod bundle utilized two 21-rod blockage islands which had the same blockage as a previously conducted 21-rod bundle test. The 21-rod bundle tests, however, had no flow bypass, whereas the 163 rod bundle test had ample flow bypass. In order to provide a comprehensive, yet simple comparison of the flow blockage results to qualify the effects of flow bypass, the temperature rise difference between the blocked and unblocked bundles was calculated as a function of elevation and flooding rate. The temperature rise reflects the integrated heat transfer effect of the flow blockage and bypass.

In order to provide the most appropriate comparisons between the 21-rod bundles and the 163-rod/161-rod bundles, only the heater rods in the two 21-rod blockage islands of the 163-rod bundle were utilized. (No bypass rod data was utilized).

The results of these comparisons indicate the following effects:

1. The temperature rise difference between a blocked and unblocked bundle is greater for the 21-rod bundle tests than for the large 163-rod bundle tests. This is attributed to the flow bypass effect in the large 163-rod bundle which decreases the flow through the blockage region. However, even with the flow bypass effect, the blocked bundle maximum temperature rise is less than that for the unblocked bundle indicating that the blockage heat transfer effects offset the flow bypass effects.

2. As the flooding rate decreases, the temperature rise difference between the unblocked and blocked bundles increases indicating that the maximum temperature in the blocked bundle decreases. The maximum temperature in the blocked bundle decreases due to the improved heat transfer downstream of the blockage. The amount of the heat transfer improvement is affected by the absolute level of the heat transfer. As the flooding rate decreases, the overall heat transfer level decreases; therefore, any improvement in the heat transfer significantly affects the measured rod temperature. Also, with reduced flooding rate, the period of two-phase dispersed flow is

increased with respect to time which means that the droplet break-up effect is increased.

3. As the distance downstream of the blockage increases, the temperature rise difference between the unblocked and blocked bundles decreases indicating that the maximum temperature in the blocked bundle increases with distance from the blockage. However, the blocked bundle maximum temperature is still less than that for the unblocked bundle.

Currently, the Appendix K requirements state that cooling can only be by steam at flooding rates below 1"/sec and that if flow blockage is calculated, the bypass effects of the steam should be considered. These requirements result in a calculated peak clad temperature penalty for several PWRs. The FLECHT-SEASET 21-and 163-rod blocked data, as well as other data, all indicate a heat transfer enhancement, not penalty, if flow blockage occurs.

The 21-rod bundle data, the Karlsruhe FEBA 25 rod bundle data, and the 163-rod blocked bundle data will be analyzed and used to develop and assess hot channel flow blockage and grid thermal-hydraulic models for the COBRA-TF code. The heat transfer mechanisms which are believed to be important for spacer grids include: single phase convection enhancement, grid rewetting and quenching, and droplet breakup caused by the grid. Models are being developed with the NRC, EPRI, Battelle and Westinghouse for these effects. Similarly, for flow blockage the important heat transfer effects are believed to be boundary layer separation and reattachment, droplet shattering, and droplet impact. Again models are being developed for each of these effects which will be put into COBRA-TF. The resulting version of COBRA-TF will be compared to the 163 blocked bundle data.

REVIEW OF FEBA BLOCKAGE DATA

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A major concern relative to the safety of pressurized water reactors (PWRs) during a large break loss-of-coolant accident (LOCA) is the coolability of rod bundles with cooling channel blockages. The influence of size, shape, and distribution of flow channel blockages on emergency core cooling heat transfer is currently the subject of considerable research. The Full Length Emergency Core Cooling Heat Transfer-Separate Effects Tests and System Effects Tests (FLECHT-SEASET) Program, jointly sponsored by the U.S. Nuclear Regulatory Commission (NRC), Electric Power Research Institute (EPRI), and Westinghouse Electric Corporation (W) is examining blockage heat transfer for various blockage shapes, sizes, and configurations. Tests were performed for electrically heated bundles with 21 and 163 rods.

The Slab Core Test Facility (SCTF) of the Japan Atomic Energy Research Institute (JAERI) has examined blockage heat transfer for a single shape and configuration in an eight bundle electrically heated facility with two of the eight bundles blocked. Pacific Northwest Laboratory conducted in-reactor experiments in the National Research Universal (NRU) reactor, with a 32 rod nuclear fuel rod bundle to examine the heat transfer effects of rod ballooning and rupture. The Flooding Experiments with Blocked Arrays (FEBA) Program, sponsored by the Federal Republic of Germany at the Institut für Reaktorbauelemente, conducted experiments with 25 rod electrically heated bundles to examine heat transfer effects of coolant channel blockages and mid-plane spacer grids. All of these experimental programs are contributing significantly to the reflood flow blockage heat transfer data base which should effectively resolve the questions of coolability of blocked fuel arrays during large break LOCAs.

The FLECHT-SEASET and FEBA Programs have had some degree of coordination in an attempt to strengthen the overall data base. In particular, the FEBA Program has produced data for blockage sizes not examined by the FLECHT-SEASET Program. These data will be used as part of the FLECHT-SEASET reflood blockage heat transfer model development effort. The Idaho National Engineering Laboratory (INEL) in technical support of the NRC in the surveillance of NRC/Industry Cooperative Programs has reviewed selected data from the FEBA data base. The intent of the review was to evaluate the data base relative to the needs of the FLECHT-SEASET model development effort.

The FEBA Program conducted tests on eight separate bundles. The INEL review examined tests from the following four bundles:

1. Bundle 1 - unblocked with seven spacer grids.
2. Bundle 2 - unblocked with six spacer grids.
3. Bundle 3 - nine rods blocked with 90% blockage.
4. Bundle 4 - nine rods blocked with 62% blockage.

The test boundary and initial conditions were examined to determine the usefulness of comparing Bundles 1, 3 and 4 tests with comparable Bundle 2 tests to isolate the effects of spacer grid and 62 and 90% blockages. It was determined that the pressure and flooding rate parameters were controlled well and proved to be adequate for test comparisons. The initial rod temperatures at the initiation of reflood was not controlled as well but for most tests was considered adequate for test comparisons.

An evaluation of the data base was performed by comparing blockage (Bundles 3 and 4) or grid (Bundle 1) tests with comparable unblocked tests (Bundle 2). The grid effects tests clearly demonstrated that the heat transfer downstream of a spacer grid is significantly enhanced for a range of flooding rates and system pressures. Data from Bundles 3 and 4 tests indicate that the heat transfer downstream of blockages as high as 90% is enhanced relative to a comparable unblocked configuration. The INEL evaluation of the FEBA data suggests that these data are a significant contribution to the blockage heat transfer data base and are useful for mechanistic model development and verification.

COBRA-TF: FLOW BLOCKAGE HEAT TRANSFER PROGRAM

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The COBRA-TF (Coolant Boiling in Rod Arrays-Two Fluid) computer program is being developed at the Pacific Northwest Laboratory by the U.S. NRC to provide a best-estimate LWR hot bundle analysis capability. In particular, this program has two main objectives:

- develop a hot bundle/hot channel code to predict coupled thermal-hydraulic/rod deformation behavior
- incorporate and assess flow blockage heat transfer models for the reflood phase of a LOCA (loss of coolant accident).

A two-step approach is employed in the hot bundle analysis concept. First, a simulation of the primary system response to a designated transient would be performed with a best-estimate two-fluid system's code. From this calculation, which employs a somewhat coarse noding of the core region, the transient boundary conditions (upper and lower plenums) would be defined. Using these boundary conditions, several hot bundle transients employing a much finer mesh in the core region may be run, varying such parameters as local peaking factors. Local subchannel fluid conditions and rod temperatures will be calculated by COBRA-TF and input to a fuel rod deformation model. An active link will permit the feedback of the current fuel rod state (material properties, gap conductance, and clad ballooning) from the rod deformation model to the rod temperature and hydrodynamic solution algorithms of COBRA-TF, and vice versa. This procedure will provide a hot bundle analysis capability for coupled thermal-hydraulic/rod deformation calculations.

Concurrent with this effort, a joint project with the FLECHT-SEASET program has been initiated. This cooperative effort between Westinghouse and Battelle will produce a version of COBRA-TF capable of addressing Appendix K concerns about reflood heat transfer with flow blockages. This program has three distinct phases:

- assess the ability of COBRA-TF to predict reflood heat transfer in an unblocked rod bundle (FLECHT-SEASET 161 and 21 rod unblocked, FEBA Series I and II)
- incorporate flow blockage models and assess reflood heat transfer in blocked bundles (FLECHT-SEASET 21 rod blocked, FEBA Series III and IV)
- perform quantitative assessment of flow blockage heat transfer capability (with a fixed version of COBRA-TF) for the FLECHT-SEASET 163 rod bundle (blockage islands with bypass).

During the past year, the ability of COBRA-TF to predict low flooding rate-high temperature reflood tests was greatly enhanced. This improvement was realized both by modification of constitutive relations (e.g., entrainment, interfacial heat transfer, and froth front heat transfer) and by the inclusion of the following:

- two-phase enhancement of convective heat transfer in dispersed flow
- subchannel thermal radiation model
- grid spacer models:
  - convective enhancement
  - grid rewet and interfacial heat transfer
  - droplet breakup and micro-drop evaporation.

A description of the above models and data comparisons with FLECHT-SEASET 161-rod unblocked bundle and FEBA grid effect tests will be presented.

Both the flow blockage heat transfer and rod deformation models will be implemented in COBRA-TF and assessed against FLECHT-SEASET, FEBA, and NRU reflood tests this year.

MEASUREMENTS OF GRID SPACER'S ENHANCED DROPLET COOLING UNDER REFLOOD CONDITION IN A PWR

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The prediction of the peak cladding temperature during the transient of the reflood phase of a loss-of-coolant accident (LOCA) in a pressurized water reactor (PWR) is a matter of great importance [1]. Comparison of selected experimental cladding temperatures and heat transfer coefficients with the analytical data post-calculated by a typical analysis code produced satisfactory matching only by the use of artificially adjusted values of the empirical code input parameters for the water entrainment model and the liquid weighting factor for the dispersed flow regime after the experimental results had already been known [2]. This apparent lack of a rational justification in such a procedure points to the pressing needs of improved understanding of the very physics which provides the foundation for the crucial mist flow regime of LOCA reflood in the development of the analytical prediction codes.

The enhanced mist cooling downstream of grid spacers has recently been identified as one of the most important heat transfer mechanisms for the development of the cladding temperature transient from results of a transient thermal-hydraulic experiment over realistic ranges of system parameters in a simulated fuel rod bundle [3,4,5]. A suggested explanation is the effective cooling of smaller droplets, due to their large surface to volume ratios, generated from the thermally relatively inactive large droplets which are intercepted by the grid spacer. To check on the validity of this suggestion, a direct measurement of droplet dynamics across the grid spacer is needed. Such an endeavor would have been considered nearly unrealizable until the recent development by Lee and Srinivasan [6,7] of a special laser-Doppler anemometry technique for the *in situ* simultaneous measurement of velocity and size of particles or droplets in *a dilute* two-phase suspension flow. Using this optical scheme, Lee et al. [8,9,10] conducted a series of systematic studies of the influence of a simulation grid spacer plate at room temperature on the droplet size population and velocity distributions in the mist flow downstream for several preselected initial mean droplet sizes in the millimeter range in the mist flow upstream. Their results reveal that regardless of the initial mean droplet size in the mist flow upstream of the grid spacer plate, the mean droplet size in the mist flow downstream of the grid spacer plate has been found to assume a stabilized value on the order of 200μm. The measured order-of-magnitude increase in the population of the smaller croplets in the mist flow downstream of the grid spacer plate is indeed mostly due to the re-entrainment of droplets from the accumulated liquid from the deposition on the plate of some of the larger droplets in the initial flow upstream of the plate.

However, the aforementioned experiment corresponds only to the situation of a totally quenched grid spacer and similar experiment for a heated rod bundle is clearly desirable [11]. In the past few months, a major effort has been put into the design and fabrication of a test system in which similar optical droplet dynamics measurement can be made at elevated temperatures in a steam environment. The flow channel consists of a simulated 1.60m -long pressurized water reactor (PWR) fuel rod bundle of 2 x 2 electically heated rods with heating elements embedded in MgO filling inside the 1 mm thick NiCr cladding of the rods which are 10.75 mm in outside diameter and 14.30 mm in pitch. Embedded thermocouples are used to measure the rod cladding temperature at various axial levels and an unshielded Chromel-Alumel thermocouple sheathed by an Inconel tube of 0.25 mm outside diameter is traversed in the center of the subchannel to measure the temperatures of the water and steam coolant phases at various axial levels. Some preliminary results are presented.

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Measurement of Axially Varying Nonequilibrium
in Post-Critical-Heat-Flux Boiling in a Vertical Tube

by

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In convective film boiling, under both blowdown and reflood conditions of nuclear safety concern, one often finds that wall heat flux, wall temperature, fluid equilibrium quality, and the local heat transfer coefficient all vary with axial position. Since the two-phase flow may exist in a nonequilibrium state, one needs to also know the axial variations of fluid actual quality and vapor superheat.

To supplement the very limited steady-state data available, experiments have been run with quench fronts propagating up into a tubular test section. During this slow "reflood" process, it was possible to obtain measurements of wall heat flux, wall temperature, and nonequilibrium vapor temperature as functions of distance beyond the quench front. Since the time required for the quench front to propagate a few millimeters corresponded to many fluid residence-times, transient convective heat transfer theory indicates that the thermal data thus obtained are quasi-steady state. Comparisons of wall superheats and vapor superheats at axial distances greater that one meter beyond the critical heat flux (CHF) point with data obtained from previously reported fixed quench front experiemnts at similar inlet conditions confirm this conclusion.

An unexpected finding was a "transition" region immediately downstream from CHF where the two-phase fluid remained close to the equilibrium thermodynamic state. It was hypothesized that the liquid requires a finite axial distance to engage with the faster flowing vapor, and in this engagement region the volumetric presence of liquid is greater than heretofore understood. The greater liquid volume fraction would decrease the tendency for vapor superheating in this near-CHF region. At greater distances, the vapor superheat increases rapidly with axial distance indicating a surprisingly ineffective vapor to liquid heat transfer process.

Thus, the significant experimental findings indicate a zone near the CHF location where the vaporization source intensity (Γ) is relatively high, followed by a far zone where the source intensity drops off to a relatively low magnitude (approaching zero). If these findings are confirmed by future experiments, phenomenological modeling of the nonequilibrium heat transfer process in convective film boiling must account for these two regions of different behavior.

TRAC-BWR HEAT TRANSFER

Rex W. Shumway

The Idaho National Engineering Laboratory has been developing a boiling water reactor (BWR) version of the TRAC code as part of the ongoing safety concern sponsored by the U.S. Nuclear Regulatory Commission. The purpose of the heat transfer part of the code development is to develop a best estimate package and assure that heat transfer processes important for predicting off-normal BWR transients are adequately characterized by the heat transfer logic in the code and to identify where experimental data is lacking.

The TRAC-PD2 heat transfer correlation package was used as the starting point for developing the TRAC-BWR heat transfer package. Modifications to the TRAC-PD2 heat transfer package to make it applicable for BWR analysis were the addition of a radiation heat transfer model, inclusion of a critical quality-boiling length correlation for the departure from nucleate boiling, addition of a subcooled boiling model, and the reintroduction of the modified Zuber critical heat flux (CHF) correlation for low flow conditions. In addition, the film boiling and critical heat flux correlations were simplified and smoothed.

Comparisons of TRAC-BD1 with Bennett hot tube data indicated that the interfacial heat transfer was too small. Changing to either the Saha-Shiralker-Dix or Webb-Chen-Sundaram model resulted in good agreement. The later model was adopted because of its broader data base. Many other steady state tests ranging from subcooled boiling to steam only cooling were calculated with good results. However, some test results for low-mass-flux, low-quality, and low-pressure did not compare well, because the code predicted intermittent oscillations not evident in the experiment. On high inlet quality tests no oscillations were predicted and the predicted steam temperature was low but within the data uncertainty. The need for void measurements was identified.

Comparison with transient tests relied heavily on ORNL THTF data. Time to departure from nucleate boiling and the following temperature excursion were predicted well by TRAC-BD1 and the present correlations are adequate for high-mass-flux, high-pressure predictions. Here again the need for low-pressure low-flow rate data with measurements of steam temperature and void fraction was identified.

The BWR heat transfer package will next be compared against Lehigh University and INEL post-CHF from boiling data. The test conditions are applicable to some BWR transients and steam temperatures were accurately measured so that calculated interfacial heat transfer errors can be separated from wall heat transfer errors. The ability to separate interfacial drag errors from the other errors does not exist during high void fraction film boiling because no accurate void data exists.

Phenomenological Modeling of Two-phase Flow in Water Reactors
(Inverted Annular Two-phase Flow Experiments and Modeling)

by

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The objective of this NRC sponsored research program is to develop rigorous two-phase flow models and correlations which are the foundation of reliable and accurate LWR safety analyses. With the present highly advanced capability in numerical analyses, the essential limitations of various safety analysis codes are imposed by not-well understood two-phase thermal-hydraulics under various accident conditions. Therefore, accurate two-phase flow models and properly scaled correlations will significantly improve the reliability and accuracy of the advanced codes and safety analyses.

In FY 1982, (1) inverted annular flow hydraulics, (2) two-phase flow loop scaling criteria, and (3) pool entrainment phenomena have been studied in detail. However in the present paper only the results from the first task related to the inverted annular flow experiments and modeling are presented.

The inverted flow regimes appear after the occurrence of the critical heat flux (CHF) or during the reflooding phase of an overheated core. In the post CHF region, the continuous contact between the hot wall and liquid is not possible due to various CHF mechanisms. This implies that the continuous vapor phase is in contact with the wall in the post CHF region until the wall is rewetted by quenching. The structures of flow and interfacial transfer mechanisms are very different in these inverted flow regimes (inverted annular, slug and bubbly flow). Modeling of this inverted flow is particularly important for accurate predictions of reflooding phenomena, post CHF heat transfer and maximum cladding temperature during severe accidents.

In view of the above, the inverted flow has been studied in detail both analytically and experimentally. As a first step, simple simulation experiments of co-axial adiabatic jets of liquid and gas have been performed. From these, preliminary models for flow regime transitions and droplet size have been developed. It has been found that both the classical jet instability and the roll-wave entrainment play a very important role in the jet disintegration and droplet generation. The jet break-up leads to slug-like large drops whereas, the entrainment process produces small droplets having diameters less than about 1 mm.

In view of the several shortcomings of the above adiabatic experiments such as the wetting problems and downward flow direction, a new experimental apparatus has been constructed, in which film boiling heat transfer can be established in a transparent test section. This test section consists of two coaxial quartz tubes. The annular gap between these two tubes is filled with a hot, clear fluid (such as syltherm 800, so as to maintain film boiling temperatures and heat transfer rates at the inner quartz tube wall. Temperatures of up to 250°C and heat transfer of up to 4.5 KW can be established in this quartz annulus. Inverted annular flow film boiling can be established by introducing saturated or subcooled test fluids (such as R-113 or liquid N_2) into the inner quartz tube directly. In addition, simplified inverted annular flow geometry (liquid core, gas annulus) can be simulated, by introducing test liquids into the test section core through thin-walled tubular nozzles coaxially centered within the inner quartz tubing, while vapor or gas is introduced into the annular gap between the liquid nozzle and the inner quartz tube.

Since the test section is transparent, direct visual and photographic observation can be made of inverted annular flow along with observation of downstream inverted slug or dispersed droplet flow. Data on liquid core stability, core break-up mechanisms, and dispersed-core liquid slug and droplet sizes can be obtained. Similar data was previously obtained in an adiabatic simulation of inverted annular flow (see NUREG/CR-3339, ANL-83-44). However, the absence of film boiling conditions in this previous study (unlike the present experimental program) may be significant, since inverted annular flow liquid core stability could be influenced by vapor generation at the liquid-vapor interface, and since wall wetting in the adiabatic test section limited the observation of inverted slug or dispersed flow downstream of the simulated inverted annular flow hydrodynamic destabilization point.

In light of these limitations, initial experiments involving the heated, transparent test section will be used to investigate the applicability of the results obtained from the previous adiabatic simulation study. As in the adiabatic study, inverted annular flow geometry will be simulated using liquid nozzles coaxially centered within the test section, with various gases introduced as a simulated vapor annulus. The test liquid will be saturated and subcooled Freon 113, with superficial velocities of 0.12 to 1.1 m/s, and the gas annuli will be nitrogen, helium, and Freon vapor (to give a range of gas densities of 0.16-5.2 kg/m^3) with superficial velocities ranging near zero to 40, 100, and 20 m/s, respectively. The inner quartz tube will be 1.36 cm ID, and various liquid nozzle diameters will be used, to establish initial void fractions from 0.16 to near 1.0. The heated test section length will be 100 cm, and flow will be upward. The data from the above experiments are used to test the validity of the inverted flow model and droplet size correlations developed under the present study.

Heat Transfer and Fluid Dynamics Under Simulated Degraded Core Conditions

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An understanding of the physical processes governing two phase flow and heat transfer characteristics of a porous layer is essential to assess the coolability of a degraded core. For example, for a fixed pressure drop across the core, the flow rate of coolant undergoing phase change in the core will depend on the particle size characterizing the core and relative permeabilities of liquid and vapor phases. In the present work basic experimental and analytical investigations of two phase flow through porous layers composed of nonheated and heated particles and simulating a degraded core have been made. For both co- and counter current flows of gas and liquid, two phase pressure drop and void fraction have been measured in porous layers composed of nonheated particles. In the experiments, the particle size, particle shape, the particle size distribution, bed porosity and superficial velocities of gas and liquid are varied parametrically. A model based on drift flux approach has been developed for the void fraction. Using the two phase friction pressure drop data, bounding values of the relative permeability multipliers for the gas and liquid phase in homogeneous beds have been determined. The void fraction and two phase friction pressure gradient in beds composed of mixtures of spherical particles as well as sharps of different nominal sizes have also been examined. It is found that the models for single size particles are also applicable to mixtures of particles if a mean diameter is defined. However, it is true only if the size range does not deviate too much from the mean. The conditions for the onset of fluidization of porous layers composed of single size and mixtures of several size particles have been obtained by using the expressions for relative permeability multipliers and void fraction.

Several processes involving two phase heat transfer during steady state or transient cooling of particulate layers simulating a degraded core have been studied. These processes include pool and forced flow cooling of a layer of volumetrically heated particles and quenching of a hot particulate bed by bottom flooding. The above models for two phase drop and the energy equations for the fluid and the particles are used to determine the state of the fluid and the temperature of the particles in different regions of the porous layer. Observations of the particle temperature and quench front velocity are found to compare well with the predictions. Under certain flooding conditions, self-fluidization of the particulate beds has been observed. The particle fluid heat transfer coefficients in the fluidized state are found to be considerably high. Conditions leading to self-fluidization of the hot particulate bed are delineated.

The above discussed one-dimensional models are extended to treat several two-dimensional situations arising in a degraded core. For example in place coolability of a degraded core containing regions of different permeabilities and heat generation rates is assessed.

MODELLING OF SIMULATED CLAD BALLOONING BLOCKAGES IN THE THETIS RIG AT AEE WINFRITH

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1. OBJECTIVES OF THE TEST PROGRAMME

The objective of this programme was to investigate the thermal conditions in a severe coplanar clad ballooning configuration under conditions of forced and gravity reflood, to produce relevant information on the timescale of the Sizewell B Public Inquiry in the UK and to provide high quality data for computer code validation. The tests carried out in the THETIS Rig covered pressures from 1.3 to 4 bar and reflood rates from 1 to 6 cm/s. The test configuration consisted of a 7 x 7 array of full length (3.58 m) heater pins in which an off centre array of 4 x 4 pins was provided with simulated co-axial balloons. There was thus a substantial by-pass flow through the undistorted heater pins and no direct radial conduction of heat from the blockage to the shroud. In order to meet the timescale on which the information was needed, an existing set of Watlow fuel pin simulators was used. These consisted of a 12.2 mm Inconel tube with a central spiral nichrome heating tape and an insulant of 89% BN, 11% MgO. The pitch of the sprial was varied to produce a chopped cosine axial form factor of 1.25. Each rod had 12 x 1 mm thermocouples located in the insulant just under the tube wall.

The clad ballooning configuration was simulated by fitting a metal shell of appropriate dimensions and thickness over the centre section of the heater pin. The shape of the ballooned section was based on results from REBEKA experiments and consists of a 200 mm long parallel section with upstream and downstream conical tapers of 200 mm and 50 mm respectively. In the parallel section the simulated clad strain is 50%, giving a blockage factor of 90%. This figure of 90% blockage was a major design parameter for the experiment. It was intended to produce a blockage more severe than the worst blockage, following clad burst that was assessed for the reactor and to demonstrate that such a blockage did not achieve unacceptable clad temperatures at realistic reflood rates. Great attention was paid to ensuring that the ballooning shells were of an appropriate thickness and the space between the balloons and heater pins was filled with helium or nitrogen and pressure balanced to limit any further distortion during testing. These precautions are essential to ensure that radial heat conduction effects in the blockage are correctly simulated.

Particular attention has been paid to instrumentation, with the intention of producing high quality data for code validation. The computerised data aquisition system records data from 800 inputs. There are 588 thermocouples which effectively record heater pin near surface temperatures and 48 thermocouples attached directly to the inside of the shells simulating clad ballooning. An inverse conduction calculation was also successfully used to infer temperatures at the surface of a clad balloon from pin internal thermocouple measurements. Grid and shroud temperatures were also measured. Twelve pressure tappings along the axial length of the shroud also provide some data on voidage profiles. Measurements were also provided of inlet flow velocities, liquid levels and outlet flow quality.

Tests were carried out with single phase cooling (nitrogen) and under forced and gravity reflood conditions. The main value of the single phase runs, which were repeated at intervals through the test programme, is to provide evidence that the ballooning configuration did not distort or deteriorate during the test sequence. In practice they also provided valuable information on the enhancement of heat transfer in the neighbourhood and downstream of the grids.

The main purpose of the gravity reflood tests was to demonstrate that important U-tube oscillation effects, which would limit the validity of data from the forced flood experiments, did not occur. This appeared to be confirmed but in practice the design of the rig meant that excessive time was taken in filling the downcomer simulation and this limited the effectiveness of this phase of the experiments.

A matrix of test results was provided covering a range of constant reflood rates and pressures at steady rod powers. Four additional tests were carried out at 2 bars and at fixed reflood rates from 2 to 6 cm/s with a power variation to simulate the change of decay heat over the period (taken as beginning 40 seconds from LOCA start). These results were intended to provide the best available simulation of the reactor situation. However it is recognised that reactor calculations do not result in a constant flooding rate and that the heater pins have the wrong diameter, thermal characteristics and axial flux profile. It is therefore accepted that these tests have limitations and are not an accurate simulation of reactor transient conditions.

The important conclusions of the test series were:

1. Rewetting fronts slowed as the power profile peaked and locally travelled quickly through each grid. The blockage rewet at almost the same time as the bypass - at most only 10 s later.

2. The present experiments confirm other evidence that the spacer grids have an important effect on reflood heat transfer. The grids enhance heat transfer in both single phase and two phase flow but the enhancement is significantly increased through steam de-superheating if the grids are wet.

3. Heat transfer in the blockage was comparable to that in the bypass in the early stages of reflood and at this stage the blockage may be cooler but conditions then deteriorate until the rewetting front arrives. The hottest part of the blockage was at the top end of the 90% blocked region in the centre blocked subchannel and the blockage peak temperature occurred much later than in the bypass.

4. Despite the severity of the blockage it was in general well cooled and heat transfer coefficients in the blockage are not too different from those in the bypass. A range of different phenomena are involved and the overall phenomena are very complex. The key issue is however whether sufficient water, as water droplets, passes through the blockage. If this occurs, steam superheat is held under control and the blockage remains well-cooled. Conversely temperature conditions deteriorate very rapidly if most of the water droplets are diverted from the blockage. In the tests which model a decaying reactor power but with constant reflood rate, blockage heat transfer began to deteriorate significantly as the reflood rate fell from 3 cm/s to 2 cm/s.

The key to an understanding of blockage temperatures is therefore steam superheat controlled by the presence of droplets. In the tapered section entrance to the blockage there is partial flow stagnation and hence low steam velocities. Water droplets can only penetrate this stagnation region if they have sufficient upward momentum. Droplet velocities in this stagnation region become lower as the wetting front approaches the blockage though the droplet effectiveness as desuperheaters increases as they travel more slowly. This is a model which is consistent with a heat transfer regime which remains similar to that in the bypass until the rewetting front is at a certain distance from the blockage. The heat transfer then diminishes sharply until the rewetting front arrives at the blockage. Effects from rewetting of grids then complicate this picture.

These ideas have been incorported into the computer code BERTHA and will then be used in further updates of the MABEL code. Work using both these codes to analyse the 90% blockage results is proceeding.

ITALIAN RESEARCH ON STEAM GENERATOR PERFORMANCE UNDER ACCIDENT CONDITIONS

by G. Palazzi, ENEA, Italy

The purpose of this paper is to provide a brief overview of the principal Italian activities concerning the behavior of a PWR U-tube steam generator under accident conditions.

Two projects are managing the experiments in progress: the first, "LWR Safety Research Project", is in charge of the activities ordered by DISP (Italian Inspectorate of Safety & Health); the second, "Nuclear Component Project", is in charge of the activities ordered by Ansaldo DBGV (Italian company which supplies nuclear components).

The activities are the following:

1) GEN 3x3 test section, in collaboration with CISE. Facility placed at the SIET Laboratories, Piacenza. This test section includes 9 straight tubes having pitch and diameter typical of the steam generator and length equal to half an average length of the U-tube. Support plates are inserted in the same positions as in the steam generator. Both hot and cold side of the steam generator can be simulated by feeding the primary fluid to the bottom or to the top of the test section, respectively. The experimental loop supplies the test section with both primary and secondary fluids at the nominal conditions of the steam generators: 155 bar, 325°C for the primary side, saturation at 67 bar for the secondary side. The loop can also simulate all the abnormal conditions required by the experimental program. The experimental tests, at present in progress, have the principal aim to validate the Italian computer code TRAGEN, which describes the steam generator behavior in transient and accident conditions.

2) FREGENE test section, in collaboration with Ansaldo DBGV. Facility placed at CRE Casaccia ENEA, Rome. This test section represents a scaled down model of an actual steam generator. For simulating the secondary flow, freon 12 is used, in order to achieve experimental conditions which are less severe than in a water-steam mixture. The test section consists of a 15 U-tube bundle reproducing the tube array of a prototypical Steam Generator with exactly the same pitch, diameter and tube wall thickness. The tube bundle length has been reduced by a factor of 0.4 approximately. The S.G. performance will be investigated by decreasing the power down to the instability threshold.

3) GEST-GEN test section, in collaboration with Ansaldo DBGV and CISE. Facility placed at the SIET Laboratories, Piacenza. The GEST-GEN test section allows testing in actual thermalhydraulic conditions a significant full length subassembly of a natural circulation U-tube Steam Generator. This test section reproduces the main features of a prototypical Steam Generator: downcomer, tube bundle having 98 full length U-tubes (with the same tube diameter, wall thickness, pitch, array and material), tube support plates, flow distribution baffle, J-nozzle spargers, steam/water separator and dryers. The thermal power is 18 MW at nominal conditions. The start of the first tests is scheduled in the year 1985.

STEAM SEPARATORS DEVELOPMENT: EXPERIMENTAL PROGRAMS FROM SCREENING TO ACTUAL ENVIRONMENT TESTS

D. Pitimada

ENEA - ANSALDO DBGV

- The Italian program for developing and testing steam separators for PWR is presented. This program is carried out in the frame of research agreement between ENEA and ANSALDO DBGV S.p.A.

  The experimental activity includes:

  - small scale experiments intended to obtain first screening of steam separator geometry: these experiments are carried out in air - water facility (L.AR.A.). Results are merely qualitative;
  - full geometric scale experiments are carried out in air water facility (ARAMIS).
    These experiments are intended to obtain quantitative results concerning carryover, carryunder, efficiency, pressure drops.
  - full scale tests intended to qualify both new designed and Licence models steam separators. The tests will be carried out in Gest-sep loop operating at the actual reactor conditions.

The presented paper contains:

- LARA loop description
- ARAMIS loop description
- Gest-sep description
- Some scaling criteria utilized to choose the air water operating parameters
- Some results obtained in air water small scale experiments

The configurations of new designed steam separators aren't presented here due to waiting for patent.

REFLOODING OF A PWR BUNDLE

EFFECT OF SPACER GRIDS

by

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Spacers located all along the cladding of fuel rods in a reactor influence the cooling efficiency in comparison with results obtained on smooth heated walls like tubes :

- heat transfer in the dry region is increased at the level and just ahead of spacer grids, while peaks of cladding temperature are observed between grids.
- quench front propagation is expected to be accelerated at least at the level grids which have a low thermal inertia and no internal heat source.

Experimental information is, however, rather scarce in this area and moreover, a reference without any grids seems not to be reasonably possible due to superimposed deformation effects if the chosen reference case is a bundle without grids.

The selected approach in a joint program between CEA* and EDF** consisted here in performing reflood experiments with different kinds of spacer grids : grids used in French PWR, with (MG) and without (G) turbulence promoters, and very slight grids (SG) especially designed and manufactured in order to reduce as much as possible two phase flow disturbances and however able to hold the heater rods in the right position during the transient. These tests were carried out in the full length 6 x 6 rod bundle on ERSEC loop, the flooding rate was varied from 2.5 to 10 g/cm^2s and two values of pressure and inlet water subcooling were investigated : 1 and 6 bar, 20 and 80°C respectively. All these parameters were kept constant all along a test.

This work contributes to clarify the effect of grids and provides a data base which is thought to be very useful to help modeling. It was shown

* CEA : French Atomic Energy Commission

** EDF : French Utility

for example that the consequences of the phenomena involved are rather different according as the flooding rate is high or low. For high flow rate tests, the local effect of spacer grids upon the clad temperature rise (ΔT) was found to be very significant, especially when steam generation is large. In this case, the largest values of ΔT occur between the grids and are, as expected, such that $\Delta T_{SG} > \Delta T_G > \Delta T_{MG}$. Quench front velocity (U) is also influenced by the design of the grids and especially by the presence of turbulence promoters : $U_{SG} \lesssim U < U_{MG}$, the acceleration of the quench front being due both to local effects (very low temperature rise and sometimes early quenching at the level of grids) and improved precursory cooling between grids owing to droplets breakup and increase of vapour turbulence.

At low flow rate the liquid fraction in the dry region is smaller ; so the local effects on ΔT are considerably reduced, and are only still significant in the upper third of the test section. In this region we get $\Delta T_{SG} > \Delta T_G > \Delta T_{MG}$ and $U_{SG} < U_G < U_{MG}$, the mechanisms being the same as for high flooding rates. In the lower half part of the test section, heat transfer in the dry region is decreased for spacer grids equipped with turbulent promoters ($\Delta T_G < \Delta T_{MG}$). The postulated mechanism is the collection of water on the surface of grids and possible fall back of large drops from the grids, these drops being not entrained by steam which velocity is rather small in this region. This explanation seems to be supported by the fact that at the same time the quench front moves faster ($U_{MG} > U_G$) and the entrainment of liquid which start is delayed, is smaller with mixing grids.

As it appears from this work that grids strongly influence two-phase flow and then heat transfer in the dry region, that this influence is dependent upon the design of the grids (in particular whether the grids are equipped with turbulent promoters or not), it is concluded that :

- the presence of grids has to be taken into account in reflood models.
- reflooding tests in heater rod bundles must be performed with the grids of the reference reactor.

REFLOODING OF A PWR BUNDLE

PERICLES PROGRAMME AND FIRST RESULTS

by

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Service des Transferts Thermiques

The Pericles program, which is a joint program between CEA*-EDF** and FRAMATOME, partially sponsored by European Communities, aims at studying the thermalhydraulics of a PWR core during Emergency Core Cooling, either for large or small breaks.

The first part of the program is underway and makes use of a rectangular test section which has been designed to allow the investigation of 2D phenomena during level swell experiments as well as forced and gravity reflood tests. In an actual core, the power may be indeed significantly different from one fuel assembly to another, and this non-uniform radial power distribution may be responsible for two-dimensional phenomena such as cross-flow of water and steam as well as possible fall-back of water in cold assemblies after partial or total steam-water separation in the upper plenum. These 2D effects are expected to significantly influence the cooling efficiency in comparison with predictions from models based on phenomena observed on test sections like tubes or small bundles which behavior is especially one-dimensional. The rectangular test section is divided into three 7 x 17 electrical simulator assemblies, the lateral (cold) and central (hot) ones being supplied with two independent power sources. The upper part is designed to be as representative as possible with respect to fall-back phenomena. In each assembly, inlet mass flow rate, wall temperature (24 levels), fluid temperature (6 levels), outlet steam mass flow rate, pressure drop (6 levels),are measured. Each upper plenum is equipped with a separator from which water can either be drained and then liquid carry-over measured, or fall back on the upper core plate.

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* CEA: French Atomic Energy Commission

** EDF: French utility

Level swell experiments were performed with different water heads in the following conditions :

- pressure 3 bar
- inlet subcooling 60°C
- imposed mean heat flux on the center assembly : 5 and 3.5 W/cm^2
- radial peaking factor 1.43 and 1.85

In addition tests at uniform power have been performed for references. Results show, first, that in each assembly the swollen levels can be considered as identical and no significant differences occur in axial void fraction profiles as they are deduced from ΔP measurements. Comparison of experimental data (outlet steam flow rate, wall and steam temperature in the dry region, void fraction), with calculations performed under two extreme assumptions regarding radial mixing (perfect or no radial mixing), clearly indicates that perfect mixing prevails for the range of parameters investigated. This conclusion holds for all the power levels and radial peaking factors investigated.

Regarding forced reflood tests with non-uniform radial power distribution, work started with the investigation of the effect of the upper tie plate (UTP) which might influence heat transfer in the dry region through partial phase separation at the UTP inlet and water fall back in the assemblies. A test has been repeated with and without UTP associated or not with guide tubes which might play a role with respect to these particular phenomena. The experimental conditions were chosen so as to maximise this eventual effect : largest inlet flow rate, time decaying power, no fall back of water from the upper plenum. Recent work dealt with the study of the radial peaking factor influence ($1 \leq fxy \leq 1.85$) as well as the partition of inlet flowrate between the assemblies. The results are presented which give a first insight about the thermalhydraulic behaviour of the assemblies when 2D phenomena occur.

OMEGA TESTS

BLOWDOWN OF A 36 ROD BUNDLE

by

C. CHAULIAC

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Service des Transferts Thermiques

In the framework of LOCA studies sponsored by CEA(*) and EDF(**), the Omega program is an experimental separate-effect study about heat transfer during blowdown. Experimental results about a 36 rod bundle test section (6 x 6) are presented here. Rods are scale one in length (3.657 m) and external diameter (9.5 mm). They are electrically directly heated and provide axial cosine flux and radial uniform flux. They are assembled with PWR mixing grids.

Each end of the test section is connected to one tank on which one break can be opened.

Parameters of one test are :

- Pressure : 13.5 or 15 MPa
- Inlet temperature : 285°C
- Mean heat flux : 0, 0.6 10^6 W/m^2, or 10^6 W/m^2
- number (one or two) and size of the breaks, and (in the case of two breaks) the breaks section ratio.

(*) CEA = French Atomic Energy Commission

(**) EDF = French Utility

During blowdown, 120 channels are measured every 50 or 100 milliseconds. Main measurements are :

- wall temperature, measured on the dry side of the rods with 180 thermocouples (distributed 5 per rod on 14 different levels).
- Mass flow rate measured with two spool pieces located between the test section and the tanks, in vertical position. Each spool piece consists of one turbine flowmeter, one symmetrical venturi and two gamma ray densitometers.
- Pressure, fluid temperature, water level in tanks.

Variation of breaks size ratio (R = hot break section/cold break section), breaks section (S = hot break section + cold break section) and heat flux is studied.

Varying breaks size ratio changes the position of stagnation point during the transient and has an effect on DNB time and wall temperature axial distribution. But different trends are observed according to the value of S.

Effects of mixing grids are shown in particular when flow feeds the test section downwards.

PATRICIA STEAM GENERATOR TESTS

by

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The Patricia GV program which is a joint program between CEA* - EDF** and FRAMATOME aims at a better knowledge of the steam generator behavior of a PWR under small break LOCA conditions.

The first part of the program, Patricia GV1, deals with the behavior of the primary side of a single full-size U-tube of steam generator. The temperature outside the tube was forced by a flow of organic liquide. The tests were conducted in steady state, forced flow (from 0.5 % to 18 % of nominal rate), high pressure (7 and 4 MPa), in two phase flow (quality ranging at the entry from 0 to superheated steam). Non-condensable gases can be injected. The flow pattern can be visually observed through windows. A device at the entry of the U-tube allows to measure the flow rate of the falling film.

We have shown that the limit of stratification (observed at the top of the U-tube) can be predicted with a certain precision by means of the Cathare's criterion. The limit of flooding determined by Wallis can be used for studying the heat pipe behavior. Non-condensable gases have no evident effects on the heat transfer coefficient,but important ones on the pressure profile.The pressure profile has been measured in function of mass flow rate and inlet quality, and we have been able to determine an instability area inaccessible when a great number of parallel mounted tubes are available.

* CEA : French Atomic Energy Commission

** EDF : French Utility

The second part of the program, Patricia GV2, going on at the present time, deals with the secondary side, especially the dry out, in steady or transient mode. The test section is made of the complete straight lengths of 4 full section tubes, 8 half section tubes and 4 quarter of section tubes, fitted on support plates. The temperature inside the tubes is imposed by a flow of organic liquid. The wall and fluid temperatures will enable us to determine both amount of heat transfer and the heat transfer coefficient. The pressure profile will be used to assess the void fraction profile and the level. The test facilities allow us to simulate up to the nominal condition. Transient tests will be performed to check the results (especially the level) obtained in steady mode.

HYDRODYNAMIC LOAD MEASUREMENTS DURING SAFETY/RELIEF VALVE ACTUATION AT KUOSHENG PLANT

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Over-pressure protection of BWR plants is provided by releasing reactor steam through safety/relief valves (SRV's) into a suppression pool wherein the steam is condensed. The discharge of steam into the suppression pool applies hydrodynamic loads to the containment structure, piping and equipment. This phenomenon has been studied in various small scale and large scale model tests to provide an experimental hydrodynamic load data base for design and generic licensing applications.

In 1981, the Kuosheng Nuclear Power Plant became the first GE BWR-6/Mark III containment plant in the world to begin start-up testing. Therefore, a comprehensive in-plant SRV discharge test program was conducted to demonstrate safe plant operation during SRV actuations and support the issuance of an operating license. The Taiwan Power Company also work with the USNRC in the development of this test program. The test results are expected to be utilized as a benchmark of the Mark III containment plants in addition to demonstrating plant safety at Kuosheng.

This paper describes the test program, test instrumentation and the test results from the Kuosheng SRV discharge test. The test results show that the dynamic loads are within design limits and predicted values, and support the conclusion that the Kuosheng Nuclear Power Plant operates safely during SRV discharge.

THE SEISMIC SAFETY MARGINS RESEARCH PROGRAM - AN OVERVIEW

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J. J. Johnson, SMA

The Seismic Safety Margins Research Program (SSMRP) is an U.S. NRC-funded multiyear program conducted by Lawrence Livermore National Laboratory (LLNL). Its goal is to develop a complete fully coupled analysis procedure (including methods and computer codes) for estimating the risk of an earthquake-induced radioactive release from a commercial nuclear power plant. The analysis procedure is based upon a state-of-the-art evaluation of the current seismic analysis and design process and explicitly accounts for uncertainties inherent in such a process.

The SSMRP is the first effort to trace seismically induced failure modes in a reactor system down to the individual component level, and to take into account common cause earthquake-induced failures at the component level. The presentation presents the results of our seismic risk analysis of the Zion nuclear power plant using the SSMRP methodology.

The risk analysis included a detailed seismological evaluation of the region around Zion, Illinois which provided the earthquake hazard function and an appropriately randomized set of 180 time histories (having pga values up to 1.8 g). These time histories were used as input to dynamic structural response calculations for four separate Zion buildings. Detailed finite element models of the buildings were used. Calculated time histories at piping support points were then used to determine moments throughout critical piping systems. Twenty-one separate piping systems were analyzed. Finally, the responses of piping and safety system components within the buildings were combined with probabilistic failure criteria and event tree/fault tree models of the plant safety systems to produce an estimate of the probability of core melt and radioactive release due to the occurrence of earthquakes.

The base case is our best estimate of the configuration of the Zion plant, its current normal and emergency operating procedures. Some important assumptions as to the consequences of certain localized structural failures were made . For this base case, the median probability of core melt was computed to be 1E-5 per year. The upper (90%) bound on the core melt probability was computed to be 2E-3 per year, and the lower (10%) bound was computed to be 1E-7 per year.

Three additional cases were analyzed to test the effects of fundamental assumptions made for the base case. The base case probability of core melt and radioactive release was due primarily to (1) failure of pipes between the reactor building and the auxiliary building caused by relative motion between the two buildings and soil failure and uplift of the containment basemat, and (2) loss of on-site emergency AC power caused by failure of the service water pump enclosure roof slab, and the assumption that all six service water pumps would be damaged by the falling slab. If the assumptions as to the consequences of the basemat uplift and roof slab are removed, then it is found that the probability of core melt and release are reduced by a factor of 2.

In addition, the base case analyzed assumed that the operator could perform a "feed and bleed" operation to provide core cooling in the event the auxiliary feedwater system had failed. If this assumption is not made, the core melt probability increases by a factor of 3. The radioactive release (expressed in terms of man-REM/year) increases by only 13%, however, because the additional accident scenarios lead to release via basemat melt-through rather than overpressure failure of the containment.

Other cases were analyzed which showed that including the local site soil profile under the plant was an important effect, while the effects of structure-to-structure interaction and the assumption of rigid foundations were not significant. Finally, several cases were analyzed showing the effects of correlation between responses in the plant due to the common ground shaking, and correlation between fragilities. It was found that neglect of seismically-induced correlation between equipment failures could result in an order of magnitude under-estimate of risk.

Potential Overdesign for the Extreme Load Condition

Current PVRC "Technical Committee on Piping Systems" Activities

By

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1.0 Introduction

A major concern of some members of the industry has been the increased rigidity of piping systems resulting from seismic and other dynamic load requirements and the response of the designer to those requirements. Note that the concern is two-fold: (1) requirements and (2) industry response.

After years of concern being voiced and the presentation of papers and reports in the public forum the industry formed a special committee under the auspices of the Pressure Vessel Research Committee. This is not to say that substantial activities were not underway or completed by other investigators and organizations because they were. What was needed was an organization to review all the completed and ongoing activities as well as determine those areas which were most susceptible to modification. "The Steering Committee on Piping Systems" was organized and proceeded to appoint a "Technical Committee on Piping Systems" (the Committee) to perform the technical review of current data requirements, Codes, Regulations, etc., and make recommendations for modifications. The Steering Committee is an oversight group as well as having significant potential for making changes occur in the appropriate Codes or Regulations.

The Committee is currently looking at four major areas: (1) Damping Values, (2) Spectra, (3) Dynamic Stress Criteria, and (4) Industry Practice.

2.0 Progress To Date

Based on the background to the problem it was quite obvious that the most beneficial areas to address were related to damping and dynamic stress criteria.

2.1 Damping Values

The Committee has recommended that damping values for seismic events be modified significantly. This recommendation is based on all of the experimental work available to the Committee at this time. The majority feeling is that damping is not frequency related, however little data at high frequency-high amplitude is available at this time and the recommendations must be substantiated by more than feeling.

2.2 Spectra

The Committee has recommended that broadened seismic floor spectra be used to determine which piping frequencies are to be excited and then independently excite each pipe frequency within the broadened band. The loading to be used is an envelope of the resulting cases considered. This approach is currently allowed as an alternative by Appendix N of Section III. The Committee has recommended that it be the primary approach to seismic analysis of piping and expanded on in Appendix N.

In addition, designs in the United States are governed by the Operating Basis Earthquake (OBE) rather than the more severe Safe Shutdown Earthquake (SSE). Many individuals feel that it is unreasonable to control piping and support design by an event not directly related to safety (i.e., OBE).

2.3 OBE Vs. SSE

According to 10CFR100, Appendix A, when an earthquake exceeding OBE occurs, shutdown of the nuclear power plant is required. Licensees must then demonstrate that no functional damage has occurred to those features necessary for continued operation without undue risk to public health and safety. Because lower allowable stresses and damping values are permitted with the OBE, many piping designs at nuclear power plants are controlled by the OBE rather than the more severe Safe Shutdown Earthquake (SSE). Loads for both the OBE and the SSE are estimated using elastic dynamic analysis procedures; however, the allowable for the SSE implies local nonlinear behavior. It has been argued that it is unreasonable to allow an earthquake which must be resisted without sustaining damage to control the sizing and proportioning of pipe supports and other elements. While the two-earthquake approach to design is commonly adopted in many countries for nuclear reactor design, there is growing sentiment that the OBE must be set to a lower value or that allowable stresses must be increased such that piping design is not controlled by an event not directly related to safety. This specific issue, once adopted for nuclear reactor piping, is likely to have important implications for other structural features at nuclear power plants.

2.4 Dynamic Stress Criteria

The Committee is currently evaluating a number of approaches to modifying dynamic stress criteria for seismic design. Basically these fall into four different categories: (1) Dynamic Margin, (2) Inelastic Spectra, (3) Fatigue, and (4) Increasing Allowable Stresses

2.5 Industry Practice

Application of the current Codes, Regulations and Standards by the designer has also been a problem and the Committee is currently preparing a report which will address a number of issues that should assist the owner and his design agent in making decisions that will reduce the number and magnitude of supports on piping systems. Current plans call for addressing the following: (1) Limiting use of snubbers, (2) Small bore piping, and (3) Analysis-design interface.

3.0 Conclusions

Progress is being made to change the approach to seismic design of piping systems. It is anticipated that the current recommendations have the potential for eliminating a substantial percentage of supports on piping systems and reducing the size of those that remain. More work is needed. The Committee must address dynamic loadings other than seismic, extend current damping recommendations so that they are not frequency dependent (or prove that they, in fact, are), select one or more approaches to modifying dynamic stress criteria and address the OBE-SSE magnitude comparisons.

ENGINEERING CHARACTERIZATION OF EARTHQUAKE GROUND MOTION FOR NUCLEAR POWER PLANT DESIGN

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INTRODUCTION

The objective of the study is to develop recommendations for methods to be used in selecting design response spectra or time histories to be used to characterize motion at the foundation level of nuclear power plants. The problem of selecting design motions can be divided into two broad parts. The first part consists of choosing translational design criteria based on free field motions with consideration of the response and performance of nuclear plant structures. The second part consists of choosing translational and rotational design criteria at structure foundation levels with additional considerations given to spatial variations of ground motions and soil-structure interaction. This paper covers the results of Phase I addressing the first part of the problem. Phase II addressing the second part of the problem is currently being performed.

In Phase I, analyses were made to examine ground motion characteristics that provide a good description of damage for degrading stiffness ductile structures such as those found in nuclear power plants. These analyses include elastic and inelastic analyses of single-degree-of-freedom shear wall type structure models at several selected frequencies subjected to 12 different earthquake ground motion time histories. From the analyses performed, the characteristics of earthquake ground motion which are significant parameters in terms of potential seismic structural damage are identified and an approach has been developed by which inelastic response spectra may be accurately predicted.

ENGINEERING CHARACTERIZATION OF GROUND MOTION

Both the elastic and inelastic response of stiff structures (i.e., 1.8 to 10 Hz) can be adequately approximated by the U.S. NRC Regulatory Guide 1.60 response spectra anchored to an "effective" peak acceleration for earthquake ground motion of relatively long duration. In the case of inelastic response, the Regulatory Guide spectrum must be converted to an inelastic spectrum. For earthquake ground motion of relatively short duration, response of structures cannot be adequately approximated by any broad frequency content spectrum and it is necessary to develop a narrowbanded design spectrum representative of the central frequencies and frequency bands of the design earthquakes. Uncertainties in the frequency content should be accounted for through the use of narrowbanded spectra in which the central frequency is shifted throughout the range of uncertainty and not by use of a single broad frequency content design spectrum.

The definition of "effective" peak acceleration which resulted in the closest agreement with actual earthquake response was:

$$A_{DE} = \left(\sqrt{2\ln(2.8T_D'\Omega')}\right) a_{rms}$$

where a_{rms} is the rms acceleration. The best correlation was achieved by defining strong motion duration T_D', as the time associated with the first zero crossing of the accelerogram following the maximum acceleration or the time associated with 75% of the total cumulative energy, whichever is greater, minus the time associated with 5% of the total cumulative energy. The central frequency, Ω', is defined in terms of moments of the power spectral density function. The breadth of the frequency content is defined by the frequency range from f_{10} to f_{90} where 10% and 90% of the cumulative power lies at frequencies below f_{10} and f_{90}, respectively.

For earthquake records examined which had a local magnitude, M_L of 6.4 or greater, strong duration, T_D' of 3.4 seconds or greater and frequency content breadth, f_{10} to f_{90} of at least 1.2 to 5.5 Hz, the Regulatory Guide 1.60 spectrum provided an adequate engineering characterization when anchored to A_{DE}. For earthquakes with M_L of 5.7 or less and T_D' of 3.0 seconds or less and f_{10} to f_{90} less than 1.2 to 5.5 Hz, the broad frequency content spectra did not adequately represent the actual elastic or inelastic structural response. Based on a limited number of records, it appears that earthquakes with M_L less than 6.0 do not have sufficient energy content to be capable of producing high accelerations, long duration and broad frequency content spectra. For small earthquakes, a narrowbanded design response spectrum obtained by averaging only records with similar central frequencies and frequency bands seems more realistic.

PREDICTION OF INELASTIC RESPONSE SPECTRA

Inelastic analyses of single-degree-of-freedom shear wall type models of several elastic frequencies were performed for the 12 ground motions records considered. For this analysis, the model was designed to be at the onset of yielding for the actual ground motion input and this input was scaled by a factor F such that various ductility levels were achieved. In this manner, the required factors F to reach ductility levels, μ, of 1.9 and 4.3 were determined for each earthquake ground motion record. The input scale factor, F is equal to the inelastic spectral deamplification factor by which elastic spectra must be divided to obtain inelastic spectral accelerations.

It was found that the inelastic response spectra could be accurately predicted by either of two methods from the elastic response spectrum and an approximate knowledge of T_D'. By the point estimate approach, the inelastic spectral deamplification factor, F_μ is given by:

$$F_\mu = \mu(f_e'/f)^2 \, S_a(f,\beta)/S_a(f_e',\beta_e')$$

where f and β are elastic frequency and damping and f_e' and β_e' are effective linear frequency and damping which account for frequency lowering and damping increases during inelastic response. By the spectral averaging approach, Eq. 2 is utilized with average effective frequency, damping and spectral acceleration values. Relations have been developed to evaluate the effective frequency and damping as functions of ductility, μ and strong duration T_D'.

The recommended approach has been compared to the Sozen and Iwan methods for predicting effective frequency and damping and to Newmark and Riddell methods for estimating F_μ. It is concluded that either the point estimate or spectral averaging approach provide significantly more accurate estimates for F_μ than do other commonly used approaches for shear wall types resistance functions.

RELIABILITY ANALYSIS FOR STIFF VERSUS FLEXIBLE PIPING*

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Conservative design procedures adopted for nuclear piping systems usually result in stiff piping designs with excessive use of support devices such as rigid supports, snubbers, and pipe whip restraints. Use of piping support devices has created certain safety concerns. For instance, addition of rigid supports usually introduce higher thermal stresses during daily plant operation, which may then reduce the overall reliability of piping systems. On the other hand, use of snubbers tends to introduce unexpected seismic stresses because snubbers may lose their function during a seismic event. Actual test data on snubbers has clearly indicated that reliability for these devices is low enough to warrant safety concern. As for pipe whip restraints, pipe may be in contact with a restraint device due to imperfect installation creating stresses not originally computed in the design analysis.

The overall objective of this research project is to develop modified design requirements and criteria which will improve piping reliability and minimize the use of pipe supports, snubbers, pipe whip restraints, etc. In Phase 1 of this project, the necessary ground work will be established. Our specific goals are (1) to confirm that flexible piping design as a result of reducing piping supports and restraints to an extent is more reliable than stiff piping design under certain circumstances; (2) to search for a piping redesign with optimum flexibility and assess its reliability subject to given changes of design requirements, criteria, and practice; and (3) to develop a better understanding of how each design element contributes to current piping problems and identify feasible changes of piping design requirements, criteria, and practice.

Confirmatory piping reliability assessment for stiff versus flexible piping systems indicated that removing rigid supports in general tends to reduce thermal stress but increases seismic stress in the pipe. As a result, piping redesign can be made more reliable by reducing rigid supports to an extent. We also observed that piping design using snubbers among support devices may not exhibit the intended reliability because snubbers often fail to perform the desired function. It was demonstrated that certain piping systems with snubbers removed actually exhibit higher reliability than the original design.

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* This work was supported by the United States Nuclear Regulatory Commission under a Memorandum of Understanding with the United States Department of Energy.

The Steering Committee on Piping Systems established by the Pressure Vessel Research Council (PVRC) has investigated changes to be implemented in Regulatory Guide (RG) 1.61 and RG 1.122 aiming at more flexible piping design. RG 1.61 specifies damping values whereas RG 1.122 defines acceptable floor response spectra for seismic design of nuclear power plants. An independent impact assessment conducted by Lawrence Livermore National Laboratory concluded:

1. PVRC proposed changes substantially reduce calculated piping responses;

2. calculated responses exhibit sufficient safety margin when compared with statistical time-history results;

3. proposed changes allow piping redesigns with significant reduction in number of supports and snubbers without violating ASME code requirement;

4. the more flexible piping redesigns are capable of exhibiting reliability level equal to or higher than the original more stiff design.

To accomplish the objectives of this research project, on-going and future work will concentrate on the following areas:

1. change in piping reliability due to elimination of pipe whip restraints;

2. effect of reducing piping support devices on on-line components, i.e., pumps, valves, etc.;

3. feasibility of other possible changes resulting in piping design with optimum flexibility;

4. development of improved piping design requirements and criteria for design of future nuclear power plants as well as modification of existing plants.

DAMPING STUDIES

Arthur G. Ware

EG&G Idaho, Inc.

A series of vibration tests on 3- and 8-inch diameter carbon steel pipes with different support configurations was conducted to determine the damping characteristics of each. The purposes of the tests were to achieve a better understanding of the physical nature of piping system damping, to determine typical damping values for the various supports, and to supply data for use by the U.S. Nuclear Regulatory Commission (USNRC) and Pressure Vessel Research Committee of the Welding Research Council in their pipe damping programs. The final goal of these programs is to ascertain whether the current USNRC guidelines for piping system damping values can be increased, thereby requiring less seismic supports and making the systems less susceptible to thermal stress failure, less costly, and more reliable.

The test fixture consisted of a large rigid structure which supported the pipes to be investigated. The piping system was a 32-foot straight section supported at the ends (both pinned and nearly fixed conditions were used), with additional piping supports at various positions along the length. The system was excited using hammer, hydraulic shaker, and snapback methods. Data acquisition was recorded from accelerometer, strain gage, and LVDT displacement probe measurements.

The parameters which were investigated in this program include the effect of the following on damping:

1. Pipe size
2. Excitation magnitude
3. Response frequency
4. Individual piping supports
5. Support installation
6. Damping calculation methods.

A significant result of the study is that damping is related to the position of the supports to each mode. When a mode exercises an energy dissipating support, higher damping is induced; whereas a support located at a nodal point for a particular mode will not affect damping. Gaps and other nonlinearities introduce considerable amounts of damping, when more than minimal excitation occurs. This is particularly true of snubbers and rod hangers with gaps between the eye of the rod and the connecting bolt. While damping varied at low load levels, increasing the strains toward yield values induced higher damping. For linear systems, all methods of measurement and calculation gave similar results. For nonlinear systems, snapback tests with the logarithmic decrement calculation method proved most reliable.

VIBRATION TESTS OF A THREE-DIMENSIONAL PIPING SYSTEM

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The U.S. Nuclear Regulatory Commission and the Electric Power Research Institute have jointly sponsored a piping research project. The intent of this research effort, which involves design, analysis, fabrication, erection, and dynamic testing of a 6-inch outside diameter (O.D.) piping system, is to provide the following benefits:

1. increase the currently limited data base regarding damping in piping systems at response levels at and above OBE stress levels (ASME Code limits);

2. obtain a data base for benchmarking computer methods for analysis of pressurized piping systems with representative (nonlinear) supports and for response levels below and above pipe yielding and including pipe support failure; and

3. stimulate recognition of safety margins implicit in ASME Code rules for Class 2/3 piping by demonstrating the existence of large design margins for seismic response above SSE stress limits, and by providing safety margins data that could impact licensing issues involving existing nuclear power plants.

Two test configurations have been planned to achieve the project objective. One is a three-dimensional 6-inch diameter pipe without branch lines,and the second has a main 6-inch run with branch lines of smaller diameters. All lines are pressurized at room temperature with simultaneous simulated earthquake time history input at supports.

Test of the first line has been completed. A total of 72 tests were conducted covering a wide range of support, loading magnitude, and loading direction variations. The first natural frequency of the system is around 4.4 Hz. Preliminary evaluation of data indicates that the piping system damping is in the neighborhood of 2 to 3% of critical for below yield conditions and the system survived the earthquake input ten times that necessary to achieve the level D stress limit without any damage.

In-depth study of the test data is being pursued. The findings will be incorporated into the design and testing of the second three-dimensional system with branch lines.

Multiple Independent Pipe Support Motions

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Efforts under the Mechanical Piping Benchmarks Program have continued through FY 83.

Major work areas during this period were the investigation of multiple independent support motion (ISM) analysis methods and the confirmation of analysis methods by the consideration of and comparison to piping test results (physical benchmarks). In the study of ISM analysis methods, response predictions based on ISM response spectrum methods and alternate seismic anchor movement (SAM) analysis methods are compared to ISM time history results for various nuclear power plant piping configurations. In all, 14 variants of inertial response prediction and 5 variants of SAM analysis are being considered. To increase the data base, 33 seismic excitations are being used for two of the piping systems under study. The time history response data for these problems has been provided by LLNL. The levels of conservatism associated with each of the computational procedures is being assessed.

Two physical benchmark confirmatory evaluations, the Indian Point Rigid Strut configuration and the Extended Z Bend ANCO test T6R1R, were completed. The former involved the analytical simulation of a snap back test of a feedwater line in the Indian Pont Nuclear Power Plant Unit 1. The latter involved the blind prediction of the response of a simple pipe configuration seismically excited by three pipe mounted actuators. The correspondence of results achieved for the snap back test were only fair. The correspondence of results for the Z bend, as assessed by an alternate NRC contractor, were good. The physical benchmarking is being extended to the evaluation of larger and branched systems subjected to multiple independent excitations in a cooperative program between NRC, EPRI and ANCO Engineers Inc.

SAFETY IMPLICATIONS OF CONTROL SYSTEMS

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The Safety Implications of Control Systems Program is responsive to USI A-47 and has three interrelated goals: 1) to assess the safety implications of control systems and attendant plant characteristics, 2) to formulate a methodology for assessing the failure modes and effects of failures of control systems on the basis of common cause, common mode, and other multiple failures such as cascade failures, and 3) to develop criteria for establishing the relative importance of control systems important to safety and recommend design and operational criteria for these systems based on their relative importance to safety. The program is divided into three major tasks, 1) an augmented failure modes and effects analysis (FMEA), 2) a plant electrical system study (PES), and 3) a hybrid-computer model to be used as a tool in further analyzing significant scenarios identified in the FMEA and PES.

Failure modes and effects analyses are being conducted on control systems to determine whether their normal operation or malfunction may stimulate transients which impact safety system function or which exceed the expected consequences of an abnormal operating occurrence. Systems under close investigation include the ICS/NNI, main steam, main feedwater, auxiliary feedwater, reactor control, makeup/letdown, pressurizer, reactor coolant pumps, boron, and turbine. More remote systems whose influence may be felt in numerous proximate systems are also being reviewed, including the plant electrical, pneumatic, and service water systems.

Initial FMEA emphasis has been on events which lead to steam generator overfill in Oconee Unit 1 (the first plant studied). Twenty three specific scenarios were identified. A preliminary report finds that: overfill may be caused by the low level indicator failing low; failure of the selected 394 inch pressure tap may cause generator dryout; most other overfill scenarios studied may be terminated by the high level main pump trip (not safety grade). Current FMEA emphasis is on events which may lead to overcooling or to inadequate cooling of the primary system.

The plant electrical system study is assessing the effects on control systems and their supports when electrical systems are exposed to unusual environmental conditions, component failures, and abnormal energy sources and sinks. To determine the effects of bus failure within the highly interconnected electrical system of Oconee 1, buses have been coded into a computer program in which the effects of bus failure can be analyzed by an automated data management system. The analysis fails one bus at a time and computer output lists other buses that consequently fail and/or that transfer to an alternate source of power. Specific plant components that are lost with a bus are to be added to the program. Tracking will include the power sources to each of the modules of the integrated control system (ICS). Output may then be studied to determine which failures involve control elements with safety implications.

The hybrid computer model will be used to further analyze failed-component configurations found to be significant in the above analyses. Credible power failures and component losses will be initial conditions for simulations which track the dynamic consequences in detail. The integrated control system, including analog and digital logic, is modeled in depth to accurately reproduce characteristic response under normal and off-normal transients. The ICS is patched on the analog portion of the hybrid machine to exploit its interactive capabilities. Operator actions can be simulated during computer runs and the consequences of acts of omission and commission studied. The balance of the simulation, including neutronics, thermohydraulics and component submodels, is developed in sufficient detail on the digital portion of the computer, to provide a suitable support for the control system. All principal plant components between the heat source in the fuel pins and the ultimate heat sink are explicitly represented: e.g., control rods, core flood tanks, coolant pumps, pressurizer, pipes, and steam generators for both loops; and the turbine-generator, high and low pressure feedwater heaters, condensate-booster pumps, main feedwater pumps and various control valves in the secondary side. The overall modeling approach uses existing advanced state-of-the-art procedures available in production codes or in the literature, streamlined and tailored to the specific plant of interest for computing efficiency. The simulation is designed primarily to address mild to moderate transients that can occur at least partially under action of the non-safety control system. The tool will be used to screen a large number of cases involving potential system malfunctions. Whenever transients exceed the model's limitations, further analysis will be made with broader spectrum codes sush as RETRAN.

THE USE OF PRESSURE NOISE IN PWR DIAGNOSTICS

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Pressure fluctuations normally present in the primary coolant system of a pressurized water reactor (PWR) carry information about the state of the system.[1,2] This paper describes ongoing work at Oak Ridge National Laboratory which is aimed at achieving a more complete understanding of these pressure fluctuations, with the goal of using them to detect and diagnose problems at an early stage.

Simple models of PWR pressure fluctuations are being constructed and tested against a bench-top flow loop as well as the Loss-of-Fluid Test (LOFT) reactor. A generalized modeling strategy has been developed, and a computer program capable of modeling a wide range of hydraulic systems has been written. The computer program also compares the model to measurements and thereby deduces primary coolant system parameters such as the amount of pressure fluctuation caused by the reactor coolant pumps.

The modeling work is concentrated on pinpointing the underlying causes of pressure fluctuations. The reactor coolant pumps and pressure control system are obvious sources; others include heat transfer fluctuations in the core and steam generator, local flow turbulence at the pressure sensors and (at LOFT) coolant pump speed fluctuations.

Actual problems have been diagnosed on the bench-top loop and hypothetical problems are also being studied for the LOFT reactor. These include (1) loss of the pressurizer steam bubble, (2) creation of a steam bubble in the reactor vessel upper plenum, and (3) pressure sensor degradation. The diagnostic analyses are performed using noise analysis techniques which are very similar to the neutron noise analysis techniques that have been used successfully in the past to diagnose such in-vessel problems as excessive core barrel motion[3] and instrument tube vibrations.[4]

Relatively simple models are being used in order that physical relationships remain clear and the resulting building blocks can form part of an easily manipulated description of a PWR. Such a pliable model could be implemented on a computer for performing on-line, automated diagnosis of PWR problems. Some strategies for performing such automated diagnoses have been formulated and are being tested on the bench-top flow loop.

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EVALUATION GUIDELINES FOR MICROPROCESSOR-BASED SYSTEMS IMPORTANT TO SAFETY

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There is significant interest in using digital computer technology in the nuclear power industry as a means of improving protection and control systems and enhancing the man-machine interface. Several papers, reports, studies, and programs have been initiated to support or research this application of digital technology. The IEEE[a] has recently developed a standard to provide guidance in this area. The Nuclear Regulatory Commission has asked for public comment on the standard and may endorse it as a regulatory guide in the future.

In a parallel sponsored effort, EG&G Idaho has developed a set of 19 design issues related to the use of computers in safety and control systems. These issues integrate hardware and software requirements with a design methodology. Our purpose is to (1) provide a guide for the design of computer systems, (2) require that computer systems be designed to minimize the adverse effects of digital technology, (3) advocate computer designs that exploit the capability of this technology, (4) avoid placing undue restrictions on these designs, and (5) minimize cost to the government and the nuclear community. To meet these goals we also structured the hardware and software requirements and design methodology into three classes of systems. These three classes provide a graded approach to safety system requirements so that more stringent requirements are applied to systems "related to safety" and less stringent requirements or imposed on other systems "important to safety." These classifications are established with respect to the function performed and potential impact on safety equipment.

In establishing these three classifications, EG&G analyzed the instrumentation and control system design standards as governed by the Code of Federal Regulations (primarily 10 CFR 50 including Appendix A and B) along with other regulatory guides and standards associated with instrumentation and control. Although many of these standards were written prior to the maturation of computer technology, a careful review establishes that they do not prohibit or inhibit the use of digital computers. The concepts and engineering development methods expressed in these standards are fundamentally sound and for the most part independent of the technology used to formulate the concept.

Thus Class I systems are those systems "related to safety," defined by 10 CFR 50 and IEEE Standard 323-1974, that are required for the safe shutdown of a nuclear reactor. Our requirements are the most stringent for Class I.

Class II systems are defined as nonsafety-related (but important to safety) electrical equipment whose failure under postulated environmental

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a. "Application Criteria for Programmable Digital Computer Systems in Safety Systems in Nuclear Power Generating Stations," ANSI/IEEE-ANS-7-4.3.2--1982.

conditions could prevent the satisfactory accomplishment of required safety functions by safety-related equipment, or equipment that could give rise to a situation (state of the reactor) that challenges a Class I system. Typically this class of equipment will include process control and man-machine interface equipment.

Class III systems include those components that (1) were used in the development or testing of either Class I or Class II systems, (2) may impact or inhibit the satisfactory operation of a Class II (or a Class I) system, or (3) applications that use computers but are not Class II systems.

The 19 design issues are grouped in three categories (1) Defense-in-Depth, (2) Susceptibility, and (3) Reliability. The evaluation guidelines then require the resolution of each of these design issues, depending on the classification of the system (Class I, II, or III), within a design methodology. For example, a requirements specification (as part of a design method) should include requirements for electromagnetic compatability (as one of the 19 design issues). The other steps of the design method (design alternatives, system specification, development, qualification, and installation) should include provisions for electromagnetic compatability design requirements, testing methods, qualification, and verification procedures. The other design issues are examined in a similar manner. The evaluation guidelines also make recommendations for the solution of these design issues but do not specify the design.

Work will be continuing on these guidelines in the future as many of our recommendations are preliminary. Additional work is needed in several areas including diversity, fault-tolerant architectures, computer security, and the use of military specifications. We believe, however, that these guidelines constitute a framework and basis for both design and review of digital computers in nuclear power application.

We plan in fiscal 1984 to develop a triply redundant, fault-tolerant computer system and install the unit in the Experimental Breeder Reactor-II at the Idaho National Engineering Laboratory as a demonstration and/or research project. This triply redundant, fault-tolerant system is representative of highly reliable computers that are being used in critical control applications for commercial and military aircraft. The results of this effort will provide additional insight for further development of the evaluation guidelines.

AN ULTRASONIC LEVEL AND TEMPERATURE SENSOR FOR POWER REACTOR APPLICATIONS

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INTRODUCTION

A liquid-level sensor based on the ultrasonic technique is a promising long-term solution for monitoring core cooling. It can display temperature and density profiles of high resolution along a chosen path through the reactor. The density data can be used to indicate voids (boiling), froth, and actual level as well as collapsed level. Correlation of these parameters with the outputs of other plant sensors provide event independence.

The continuous nature of the temperature, level, and density information is another advantage of the ultrasonic method over those indicating at a limited number of discrete points. Compatibility with current reactor designs and the ability to perform under both normal and accident conditions is realized by the simple, all metal construction of the sensor, and by isolation of the transducer from the reactor core area.

EVALUATION PROGRAM

Prior work done at ORNL identified a number of specific questions pertaining to the instrumentation needs, the physical structure, and the environment of a nuclear power reactor which required evaluation before a sensor complete with instrumentation could be constructed for field testing. Satisfactory answers to these questions were arrived at after a thorough study of the physical parameters comprising a proto-typical ultrasonic wave-guide sensor.

Acoustic Attenuation.

The total sensor length, including any necessary acoustic lead-in from the transducer section, will be comparable to the top-to-bottom dimension of a reactor vessel. Detailed measurements were made on a 13 m section showing that acoustic attenuation over reactor-scale distances is not a problem.

Material Fatigue.

Two tests were designed to study any possible effects of self-induced aging--one wherein pulses at a rate of 30 Hz excited a test probe for three months; the other wherein 3.3 G pulses were obtained in a week (equivalent to five years of continuous operation). Neither test indicated any change in the elastic moduli of the sensor materials to within 0.5%.

Radiation Damage.

To estimate damage, a sensor was placed next to the core of the Oak Ridge Research Reactor. The thermal flux was around 500 G n/cm/cm/sec at that location. The data obtained over one refueling cycle show that radiation damage increases the elastic moduli but is small enough to be accounted for by a slowly-varying change in the probe calibration constants.

Flowing Water and Voids.

A test stand for subjecting a probe to rapidly-flowing water in both the axial direction and across the sensor was used with a pump providing a 5 m/s water velocity. For axial flow, the only noticeable effects were fluctuations in the signal due to observed changes in the water level. When air was injected to test the effect of two-phase flow, the mechanical vibrations of the system increased in amplitude and intensity, resulting in the torsional transit times varying up to 10% due to expanding and collapsing voids.

Similar results were found for the cross-flow tests. In addition, lateral forces due to the flowing water, and impulse forces when air was added, pinched the sensor between the impinging flow and the housing, blocking the torsional wave in a manner similar to a mechanical restriction.

Several ways to overcome this are: (1) divert the flow from that part of the sensor sensitive to lateral forces; (2) design the probe housing to break up the lateral forces while still allowing voids formed outside the housing to contact the sensor; (3) design the probe itself so that it would be deflected under a lateral force but not constricted.

Temperature Effects.

The temperature behavior of a sensor requires calibration due to the combined effects of temperature on sensor length, density, and elastic moduli. Stainless steel (300 series) behaves linearly with respect to the square of the transit times with a temperature coefficient of +0.0004 to within 10% for both the extensional and torsional waves over a range from 20 deg C to 400 deg C.

Restoration of Torsional Signal.

Loss of torsional signal observed during a pressurizer test at high temperatures could be recovered and the signal augmented by applying a dc current along the length of the rod, passing under both excitation and pick-up coils. Subsequent tests wherein the entire transducer section was placed in an oven, showed that the torsional signal should be maintained over the entire PWR temperature range.

Curved Waveguides.

An ultrasonic level sensor installed in an existing plant must conform to the restrictions imposed by structures already in place. Thus knowledge of the acoustic transmission around waveguide bends is necessary. In practice, it was found that as long as the radius of curvature of any bend was kept larger than 30 to 100 times the wavelength, the losses were small (on the order of a few percent) even for a 360 degree bend. For the proposed sensor, a radius of curvature of 30 cm, is permissible. This should impose no undue restrictions on retrofitting a sensor to existing power plants.

Multisectioned Sensors.

One of the strengths of the ultrasonic method is its ability to provide multi-variate outputs: a profile of density and temperature along the sensor length. This is accomplished by dividing the sensor into several sections. Successful probes with up to 5 zones have undergone long-term testing under a variety of conditions. Due to the losses inherent in a practical device, a maximum of perhaps 10 zones would be practical.

CONCLUSIONS AND RECOMMENDATIONS

We have undertaken a detailed study of an ultrasonic waveguide employed as a level, density, and temperature sensor. The purpose of this study was to show how such a device might be used in the nuclear power industry to provide reliable level information with a multifunction sensor, thus overcoming several of the errors that led to the accident at Three Mile Island.

Some additional work is needed to answer the questions raised by the current study--most noticeably the damping effects of flowing water. However, the problems encountered are not of a fundamental nature and would be resolved by a modest effort. The next phase should concentrate on nuclear qualification in a joint effort with a vendor and a utility group.

Non-Invasive Water Level Measurements at LOFT Using a Neutron Detection System

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The Three Mile Island accident demonstrated the need for pressure vessel level instrumentation. This need was formalized in NUREG 0737.[1] As result of this requirement, a variety of instrumentation systems have been proposed. These systems include both intrusive and non-instrusive devices.

This paper describes a non-invasive water level system developed by the authors at The Pennsylvania State University Nuclear Engineering Department. The system uses the variation in fast neutron leakage with water level and density to measure the hydraulic conditions in the reactor pressure vessel. The concept is based on the anomolous source range detector response observed during the TMI-2 accident.

The system consists of a string of neutron detectors placed vertically along the biological shield external to the reactor pressure vessel. Cadmium and polyethylene are used to shield the detectors from thermal neutrons. The shielding enables the detectors to see predominantly epicadmium neutrons which come directly from the core region adjacent to the detector. The feasibility of this system was demonstrated through both experiments and calculations. The results of this study were reported in NUREG/CR-3290.[2]

In January 1983, a system based on this preliminary work was installed in the Loss of Fluid Test facility (LOFT) at the Idaho National Engineering Laboratory (INEL). This system consists of four Reuter Stokes fission chambers shielded by cadmium and polyethylene. Fission chambers were chosen for the detectors because they can be operated in high gamma fields and they can be operated in either a pulse or current mode. The detectors are installed along with their shielding in one of the LOFT primary shield tank instrument wells. One detector is located near the bottom of the core, another slightly below core midplane, another near the top of the core, and one slightly above the core.

Data from the detectors is monitored using two HP-85 computers. One computer collects information on chamber current by measurement of the voltage drop across resistors in series with each of the detectors. The other computer collects pulse information from the detectors using computer controlled scalers.

This system was used to monitor water level and density during the February 1983 LOFT loss of feedwater experiment, LP-FW-1. This test involved complete loss of feedwater with cooldown and decay heat removal accomplished by bleed and feed of the primary system. No significant boiling or level change was observed by either this system or other systems installed at LOFT. By comparing readings, however, from the Penn State system with readings from LOFT instrumentation during this and previous tests, it was determined that this system would have been sensitive to level changes, had there been any.

Data obtained from the LP-FW-1 test provides base line information for subsequent tests. The data obtained was essentially what would be expected of a normal shutdown. Thus, comparison of data from subsequent tests with this data allows identification of abnormal condition in the reactor.

In June 1983, the system was used to monitor hydraulic conditions in the LOFT reactor vessel during the LP-SB-1 test. This test simulated a small diameter (3-inch) break in the hot leg followed by early primary coolant pump trip. Pre-test predictions indicated that while there would be no core uncovery, there would, however, be large amounts of boiling in the upper core region.

Comparison of the result of the LP-FW-1 test with those obtained in the LP-SB-1 test showed that the two lowest and the uppermost detectors followed the typical shutdown curve observed in the LP-FW-1 test. The detector at the top of the core did not. Since this detector is located near the top of the reactor core, it would be expected to see neutrons which originate in that region. The top of the core is expected to be the core region with the highest void fraction. The observed detector response is consistent with this expectation. Furthermore, comparison of the detector response with pre-test predictions of hydraulic conditions in the pressure vessel correlate well with the structure observed in the data obtained from this detector. Based on the detector response during this test and comparison with that obtained during the earlier LP-FW-1 test, the authors consider that the results support the ability of this system to monitor density changes during an accident.

This work was supported by the US NRC under grant No. NRC-G-04-81-024. This support is gratefully acknowledged.

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AN ASSESSMENT OF PRESSURIZED WATER REACTOR CORE EXIT THERMOCOUPLES DURING ACCIDENT AND POSTACCIDENT SITUATIONS

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The current Nuclear Regulatory Commission (NRC) guidelines and regulations dealing with pressurized water reactor (PWR) core exit temperature measurements are rigorous. Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," specifies Category 1 design and qualification criteria. NUREG-0737, "Clarification of TMI Action Plan Requirements" states that PWR core exit temperature measurements shall be part of the inadequate core cooling (ICC) measurement system, and also restates Category 1 qualification requirements. Both of these documents endorse the currently-installed Type K thermocouples (TCs) as an appropriate measurement device for this parameter and state the upper range as 2300°F.

The Nuclear Power Plant Instrumentation Evaluation (NPPIE) Program finds that the currently installed core exit thermocouples (CETs) do not meet the NRC requirements and guidelines for this measurement. Specifically, in a core uncovery accident, the CETs may give ambiguous information and may sustain enough damage to be considered as not surviving. The bases for this concern are as follows:

1. Mechanisms for thermocouple output
2. Method of installation for CETs
3. Error sources for currently installed CETs
4. Performance of CETs during the TMI-2 incident
5. No diagnostic methods to verify reliability of thermocouple output.

There may be other instrument systems besides the CETs that could meet the intent for measuring core exit temperature (detection of inadequate core cooling and long-term surveillance). However, it would be costly, both in terms of time and money, to develop and qualify new systems for accident/postaccident use. Therefore, NPPIE personnel have chosen to work within the limitations of the currently installed CETs and have developed some recommendations that would minimize signal ambiguity, as well as be cost-effective. Highlights of this work are as follows:

1. <u>Reduce the Category 1 Design and Qualification Requirement to Category 3 for the Thermocouple Portion of the System</u>
   Based on the TMI-2 experience, it is doubtful whether the CETs can survive a core uncovery accident without sustaining mechanical damage and/or decalibration. In other words, the core exit Type K thermocouples cannot meet Category 1 requirements. Therefore, it would be more cost effective to specify Category 3 requirements.

2. Develop Suitable Diagnostic Test Methods for CETs

NPPIE personnel have developed a sheathed-thermocouple model based on classical thermodynamics and irreversible thermodynamics. Based on the mode, a "damaged TC" is defined as "a thermocouple that has developed a shunt path between elements or between element and sheath."

By measuring thermoelectric voltages and resistances during normal operating conditions, it should be possible to obtain a history of "undamaged" thermocouple voltages and resistances. The thermocouple will be subjected to "aging" effects which could potentially alter the readings. However, these effects would be slow departures from baseline. If there is rapid departure from the baseline, then it can be concluded that the TC has sustained damage during the accident.

The actions proposed above will not necessarily solve the CET measurement problems during accident conditions. However, they will allow the operator to have increased confidence in the TC data without major modifications to the existing system, and will offer relaxation of very expensive qualification requirements in favor of historical in situ data.

SAND83-1965C

Performance and Effects of Terminal Blocks in a Loss of Coolant Accident Environment

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Historically, terminal blocks have been used by the nuclear industry to make cable junctions in both 1E and non-1E circuits inside and outside containment. Today many plants continue to use terminal blocks, even though, as a result of recent qualification requirements, there has been a trend to replace in-containment terminal blocks in 1E circuits with splices. There is, therefore, continued interest in the performance and effects of terminal blocks in LOCA environments. This paper addresses this issue.

Reviews of industry qualification tests of terminal blocks indicate that little attention is paid to the applications in which terminal blocks are used. Rather, a generic application approach is taken and qualification failure criteria are not correlated to actual applications. These tests do not specifically address the question of what is the terminal block performance during the Loss of Coolant Accident/High Energy Line Break (LOCA/HELB) environments. Also, measurements of terminal block insulation resistance (IR) are generally taken before and after the LOCA/HELB simulations but not during the simulation.

To quantify terminal block performance in a LOCA environment and to investigate the failure and degradation modes of terminal blocks, twenty-four terminal blocks (five models from four manufacturers) were tested in simulated LOCA environmental conditions. The environmental exposure profile followed the recommended qualification profile of IEEE 323-1974, Appendix A. The objective of the test was to determine the magnitude of leakage currents during simulated LOCA conditions for commonly used terminal blocks that are typically protected by NEMA-4 enclosures and powered with relevant voltages. The terminal blocks were powered at three voltages typical of in-plant applications: 4 Vdc (typical of RTD circuits), 45 Vdc (typical of instrumentation circuits), and 125 Vdc (typical of control circuits.) The terminal-to-terminal and terminal-to-ground leakage currents were monitored on a discrete time basis throughout the environmental exposure. Based on this data the effective IR for each of these paths was calculated.

Surface leakage currents through conducting surface moisture films are the primary mechanism by which terminal blocks contribute to instrumentation and control circuit degradation. During our tests, the formation of surface films reduced insulation resistance to 10^2 to 10^5 ohms from initial values of 10^8 to 10^{10} ohms. Leakage currents when we applied 45 Vdc were on the order of 0.1 to 10 mA, sufficient to affect by 15 to 90 per cent high impedance, instrumentation circuits. Insulation resistance at 4 Vdc was 10^3 to 10^5 ohms, values which are sufficiently low to affect indicated temperatures of RTD's by 3 to 8 per cent. At 125 Vdc, IR values were in the range of the 45 Vdc IR values, and in some cases, may have been slightly higher. Given the right circumstances, it is possible for leakage currents to become sufficiently large that permanent, open failure of the circuit can be induced by melting conductors. We observed this behavior for 1 terminal block. Sporadic breakdowns to very low values of insulation resistance (a few to several hundred ohms) lasting from less than a second to several seconds were observed. Also, increasing the applied voltage level dramatically affects IR. Immediately after repowering or rapidly increasing applied voltage, IR drops dramatically and then slowly (minutes to hours) recovers to higher nominal values. The change can be 1/2 to 2 orders of magnitude in range.

Terminal block insulation resistance returned to 10^6 to 10^8 ohms subsequent to the LOCA environment simulation, but not to pre-test levels. Some permanent degradation of the insulation surface occurs presumably due to leakage currents either graphitizing the terminal block surface, or carbonizing other organic materials in the vicinity. Another possible mechanism for the permanent degradation of surface IR is the formation of conducting deposits such as cadmium sulfide on the terminal block surface.

There was a noticeable dependence of IR on system temperature. Insulation resistance at temperatures less than 110°C tended to be 1/2 to 2 orders of magnitude greater than IRs at temperatures greater than 110°C. The IRs at 95°C showed the most improvement, but this result can be attributed to the residual heat in the terminal blocks vaporizing the film. Since saturated steam conditions prevailed throughout the test, a pressure dependence for IR was also observed. However, neither temperature nor pressure alone govern IR behavior. Rather, pressure in concert with temperature is important in that they govern the existence of conditions appropriate for film formation. If the system is superheated, films will not exist and terminal block performance will be relatively good. If the environment is subcooled and if the terminal block temperature is above the dew point for the existing conditions, the same situation will exist. If the temperature of the terminal block is below the dew point in the subcooled environment or if the terminal block is cooler than the saturated steam environment, films will form and the performance of the terminal block will be degraded.

Our tests show that the performance of terminal blocks during a LOCA is degraded to the point where low power instrumentation and control circuits may be affected. In determining the significance of terminal block performance, analysis of specific applications must be made. Thus, if generic qualification acceptance criteria are to be used, they must be representative of the most restrictive terminal block application. Otherwise, sufficient data to make application specific analyses need to be provided.

INTROUCTION, OVERVIEW OF APPROACH AND SCHEDULE FOR SOURCE TERM REASSESSMENT

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Accident source terms, that is the quantity, characteristics, and timing of radionuclides released to the environs following an accident, continue to be the subject of intense activity at the NRC. Only recently, the NRC established a program office, Accident Source Term Program Office (ASTPO), to focus and direct the Agency's efforts in this area, and to bring this issue to a resolution no later than Spring of 1984.

The present approach adopted for the initial calculation of radionuclides and aerosols release involves the development of mechanistic computer codes and their application to specific plants under conditions specified for selected accident scenarios, or sequences of events. The models developed treat the fission product release during the fuel melting process, the behavior and transport of fission products and aerosols in the reactor vessel, behavior and transport in the primary cooling system, behavior and transport in the containment, the effect of engineered safety features, and calculate the release from the containment to the environment.

The computer codes used for this purpose include mechanistic, phenomenological models of processes where available, and simplified approximations of other important natural, physical, and chemical processes involved, with particular emphasis on the radiologically important elements such as iodine, cesium and tellurium, in vapor or particulate state.

Five plants, PWR SURRY, ZION, BWR Mark I PEACH BOTTOM, BWR Mark III GRAND GULF, and PWR Ice Condenser SEQUOYAH are being analyzed.

A number of key technical issues have been identified during the on-going peer review by technical specialists of the Battelle (BCL) reports. Many of these issues are not new. Some of them are being addressed by the NRC source term contractors within the limitations of the state of technology. In several other areas special task groups have been formed by the NRC staff to resolve certain issues using contractor assistance, while in some cases the on-going NRC research program has been redirected or augmented to provide additional analysis or limited methodology improvement. Finally, the resolution of certain issues will not be completed until the data base from on-going research is further developed to confirm methodologies used in source term reassessment and the findings.

The final process for integrating the four basic elements of the source term reassessment will culminate in a report that will be prepared by the NRC staff, NUREG-0956. Following the completion of the various supporting tasks (containment performance, uncertainty and sensitivity, and special thermal-hydraulic analyses), the staff will initiate its appraisal of the risk and regulatory significance of reassessed source terms based on the NRC contractor analyses and findings, results of the individual technical peer reviews and the review of the IDCOR work.

Based upon the staff appraisal at the end of 1983, feedback from discussion with the broad-based scientific peer review by an independent scientific organization, and the review of ANS study, Battelle will be directed to proceed with refined source term analyses for specific plants and accident sequences starting in January 1984. The staff will continue with the final preparation of its appraisal, factoring in results from the new Battelle analyses, and leading to publication of NUREG-0956 (draft) by June 1984 for public comment. The final document is scheduled for the end of 1984 after receipt of the public comments and the findings of the independent, broad-based scientific peer review.

SUMMARY OF SOURCE TERM ANALYSES FOR FIVE LWR PLANTS

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Physical processes which affect the release of radionuclides from nuclear power plants under accident conditions are becoming more thoroughly understood and can provide a basis for re-evaluating source terms to the environment. Improved characterization of source terms would provide a basis for formulating impacts on and changes to licensing practice, emergency planning, safety goals, and indemnification policy. This study represents the identification and formulation of a systematic, mechanistic approach to estimating source terms and the implementation of this approach through calculations of fission product release to the environment for five LWR plants under selected sets of accident conditions.

The overall approach was based on selecting the specific plants and accident sequences for consideration, and then using consistent and improved analyses of fission product release from fuel, transport, and deposition to predict fission product release to the environment for these specific cases. The approach is comprised of a series of steps performed in sequence such that in the combined analysis, the results are specific to an individual set of accident conditions for each reactor and each successive transport step is based on results from analyses of the previous step.

The specific plants and sequences chosen for study are as follow:

- Surry: AB, TMLB', S_2D, V
- Peach Bottom: TC, AE, TW
- Grand Gulf: TPI, TQUV, TC
- Sequoyah: TML, TMLB', S_2HF
- Zion: TMLB', S_2D

The accident sequences were selected because they include those with high risk, large consequences and most importantly, a considerable range in physical conditions.

After selection of sequences, the stepwise analyses proceeded with the collection of plant design data and the performing of thermal hydraulic analyses for the sequences. Overall thermal hydraulic conditions on a time-dependent basis were estimated with the MARCH code and detailed thermal hydraulic conditions for the primary system estimated with the MERGE code which was developed specifically for use in this program. The CORSOR code was used to predict time and temperature dependent mass releases of radionuclides from the fuel within the pressure vessel and releases during core-concrete interactions of radionuclides remaining with the melt were provided by Sandia National Laboratories using their code VANESA.

Using the MARCH/MERGE predicted thermal hydraulic conditions and the CORSOR predicted radionuclide release rates as input, the TRAP-MELT 2 code was used to predict vapor and particulate transport in the primary coolant circuit. Transport and deposition of radionuclides in the containment were calculated using the NAUA-4 code. Aerosol scrubbing in pressure suppression pools was predicted with the SPARC code developed by Battelle-Northwest.

The calculations performed were of a "best estimate" type using input derived, to the extent possible, from experimental measurements. It is to be recognized that the computation of radionuclide release and transport using mechanistic models is subject to many uncertainties of various magnitudes and importance. It has not been a part of this study to produce quantitative estimates of uncertainties in individual parameters and hence the overall importance of such uncertainties has not been assessed. However, limited evaluations were made of the sensitivity of results to a few specific parameter variations.

The results of the calculations show a strong dependence of fission product release to the environment on the specific plant design and accident sequence. Also, assumptions influencing the time estimated for containment failure were of major importance. The calculated release fractions ranged from values near the WASH-1400 releases for the appropriate category to much below these values.

SUMMARY OF STATUS OF SOURCE TERM CODE VALIDATION

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It is clear that the technology for assessing severe accident source terms has been considerably advanced since WASH-1400. We now have more data, more analysis codes, and, I believe, a better understanding of the fundamental mechanisms involved.

It is difficult to overemphasize the importance of accurately assessing the "source terms" at this time. Whatever the outcome of any reassessment, it can have a profound influence on the perceptions of nuclear safety, on what we do for emergency planning, and on the general acceptability of nuclear power. It is essential, in my view, then that the source term determination be based on credible technology, that the results be defendable, and that they receive acceptance by the entire community.

Because source term determination relies so heavily on the exercise of large, complex computer codes, the credibility rests on the assessed validity of these codes. Therefore, one element of the reassessment of the source term by the USNRC's Accident Source Term Program Office (ASTPO) is to provide a review of the status of validation of the source term codes used in the BMI-2104 study. The true measure of validation of any computer program would be a quantification of the uncertainties in the calculated key output parameters. This requires a rigorous sensitivity/uncertainty analysis. Although another element of NRC's reassessment effort is to develop such a sensitivity/uncertainty analysis, it is emphasized that the code validation status effort being reported on here does not include such a rigorous sensitivity/uncertainty analysis within its scope. Rather, it is intended to provide the following for each of the codes used (ORIGIN-2, MARCH 2.0, CORSOR, MERGE, TRAP-MELT, CORCON, VANESA, and NAUA-Mod4):

- A description of the physical models in the code
- A discussion of the status of documentation
- A compilation of the required inputs and major outputs
- A discussion of the implied and explicit assumptions and any perceived limitations in the code
- An identification and assessment of the data bases for the inputs, prefixed parameters, and internal phenomenological models
- A description of the "verification" and quality assurance status
- An assessment of the validation needs and a description of NRC programs that are addressing those needs

- A subjective opinion of the validity of the code as used in the study.

The results of the validation status review will be published as ORNL/TM-8842 in the near future. Of course, the experts at the various institutions involved in the review have made specific findings and judgments regarding each of the codes under review which will be presented in the full paper. Only a general "personal perspective" of the findings will be presented below:

> A real problem in assessing the status of validation of any code is the fact that there does not exist a formally established definition to determine what constitutes a "validated" code nor are there any agreed upon and documented formal validation procedures.

Overall the codes have been judged to be definitely lacking in experimental validation and that they treat some phenomena in a crude simplistic fashion. Nevertheless, they were on the whole judged to be basically sound and to utilize good fundamental approaches. They were found to be lacking, not so much in what they do and how they do it, but in the things they don't do - the phenomena left untreated so to speak. Some specific areas felt to be inadequately treated by the codes were:

- The details of fuel melting and slumping behavior

- The specific development of fuel debris; its characteristics and influence on thermal hydraulic behavior

- The chemical and physical interactions of various fission product species with surfaces

- The thermal hydraulic response of the primary system - particularly for the upper plenum regions

- The general lack of simultaneous coupling of the aerosol transport behavior and the thermal hydraulic behavior

- The assessment of the chemical forms of all the fission products

- The lack of inclusion of the fragmentation of both fuel and water on energetic expulsion from the primary system.

OVERVIEW OF EXPERIMENTAL SUPPORT FOR FISSION PRODUCT TRANSPORT ANALYSES AT OAK RIDGE NATIONAL LABORATORY

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This paper presents an overview of the experimental work currently in process at Oak Ridge National Laboratory in support of the NRC severe accident source term evaluation work.

FISSION PRODUCT RELEASE FROM FUEL

This program was designed (1) to determine fission product and aerosol release rates from irradiated fuel under accident conditions, (2) to identify the chemical forms of the released material, and (3) to correlate the results with experimental and specimen conditions and with the data from related experiments. Tests are being conducted over a range of fuel operating histories, test times, temperatures, and steam flow rates, so that the data obtained are applicable to a spectrum of accident sequences in both boiling and pressurized water reactors.

The experimental apparatus includes an induction furnace for heating the fuel specimens to 2000°C in flowing steam. On-line release data are obtained with NaI(Tℓ) detectors, and quantitative data are obtained via posttest gamma-ray spectrometry of all apparatus components. Mass spectrometry, activation analysis, and metallographic examination of samples from selected locations provide additional information.

Currently, five release tests have been performed. Significant releases of Kr, I, and Cs were measured in all tests, and smaller fractions of Te, Sb, Ag, Ru, Ce, and Eu were measured in some cases. The release-rate results are being compared with available predictive methods which were developed for the "0772" study and which are incorporated into the current CORSOR code. Future experiments will explore the use of shorter-cooled fuel and higher temperatures.

CORE MELT PROGRAM

In this program, a 1-kg portion of an LWR core is heated to melting. Core samples have included simulated PWR and BWR fuel, PWR control alloy, and some fission product simulants. A specially designed furnace inductively heats, first, the metallic portions of the sample and, then, the fuel material. A steam/hydrogen environment is provided by the in-flowing steam and by reaction with the cladding to form hydrogen.

The material condensed on vessel surfaces, the deposit on the filter, and the melt residue are analyzed for amount and composition. To date, information has been obtained on the nature of control-rod failure and on possible interactions of control-rod material with cladding. In addition, it has been observed that tellurium has a strong tendency to alloy with cladding or steel materials, if such opportunities arise.

FISSION PRODUCT VAPOR DEPOSITION ON AEROSOLS

The transport of fission products in both the primary vessel and in secondary containment is related to aerosol behavior. In this experimental program, the nature and rate of fission product vapor interaction with aerosols have been studied.

Two experimental techniques (both using radiotracers) are being investigated. The first is a flowing system in which simulated aerosol and fission product vapors are fed into a pipe maintained at temperatures up to ~700°C, and the relative deposition of the vapor between the pipe surface and the aerosol is measured.

A second experimental technique employs a static system for determining the chemisorptive capacities of a range of typical aerosols with fission product vapors. The sorptive capacities of nonfuel aerosols (which will be generated by a plasma torch) for fission product vapors will be measured by radiotracer methods.

TRAP-MELT VERIFICATION PROGRAM

This program focuses on aerosol deposition rates and transport in the reactor vessel during LWR core-melt accidents. Specifically, the objectives are (1) to measure aerosol deposition and resuspension rates in idealized geometry using representative aerosols, contact times, and lift-off velocities; and (2) to compare test results with values obtained by using current computer models such as TRAP-MELT and QUICK.

Two types of aerosol deposition rate experiments are currently in progress. In the first type, a plasma torch is used to generate aerosols into a vertical pipe 12 in. in diameter by 12 ft. high. Aerosol plateout, settling, and airborne concentration level are measured as a function of air flow rate, wall temperature gradient, and solids mass-input rate. Two tests (a zinc aerosol and an iron oxide aerosol) have currently been completed. A second series of experiments is currently in progress to investigate the degree of aerosol resuspension that may occur as a result of high gas velocities.

NUCLEAR SAFETY PILOT PLANT

This facility is currently dedicated to developing an expanded data base on the behavior of aerosols generated during a severe accident. The facility consists of a 38.3-M^3 insulated vessel with provisions for (1) aerosol injection by a plasma torch, (2) steam injection for simulation of the environment expected in the containment vessel for some accident sequences, (3) on-line aerosol sampling for determination of suspended concentration vs time, and (4) fallout and plateout surfaces used in posttest examination.

The principal objective of the current test series is to determine the effect of steam condensation on typical aerosols produced by an overheated core and by core melt/concrete interaction. Two oxides, Fe_2O_3 and U_3O_8, and powdered, nuclear-grade concrete are being used to simulate these aerosols.

Twelve tests have been run using single-component materials under wet and dry containment conditions. Steam condensation appears to accelerate aerosol deposition on the walls due to the mass flux of the steam to the wall. In a single comparison to date with a predictive code (NAUA), calculations matched the data fairly well when the diffusiophoretic deposition mechanism was properly considered. Future tests will include mixtures of aerosol materials and possibly also aerosol simulants from control rod and cladding materials.

POST-ACCIDENT IODINE AND TELLURIUM CHEMISTRY

Numerous LWR accident sequences involve interaction of various types of water bodies with gases containing fission product vapors. The principal objective of our iodine chemistry work is to determine iodine volatility from water solutions for the various realistic conditions that would exist in a severe accident. We are taking two approaches toward this goal. In the first, we are making direct measurements of volatility for a range of iodine concentrations (10^{-4} to 10^{-6} M) a range of pH conditions (6 to 9), both buffered with boric acid and unbuffered. In addition, tests are under way to determine the effects of radiation and oxidizing or reducing impurities on iodine volatility.

The second approach, which is more fundamental, is needed for the interpretation of the volatility measurements. This work employs solution spectrophotometry to examine the rate of production or disappearance of the iodine species I^-, I_3^-, I_2, IO_3^-, and possibly HOI.

OVERVIEW OF CHEMISTRY AFFECTING CESIUM, IODINE, AND TELLURIUM TRANSPORT IN A REACTOR COOLANT SYSTEM*

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Transport of fission products released during fuel degradation is an essential process for these fission products to pose a public threat. The transport of the volatile fission products cesium, iodine, and tellurium in the primary system is greatly affected by the chemical form of these elements and the chemical interactions of these species with other materials. The High Temperature Fission Product Chemistry program sponsored at Sandia National Laboratories by the U.S. Nuclear Regulatory Commission was established to develop quantitative descriptions of the chemistry that affects fission product transport. During the last year this program has yielded results of fundamental significance to the estimation of the radiological effects associated with reactor accidents involving fuel degradation.

When only fission product species, steam and hydrogen are considered, the dominant forms of cesium and iodine in the primary system atmosphere are expected to be CsOH and CsI, respectively. The transport characteristics of CsOH at temperature of 700 -1000°C in steam/hydrogen mixtures has been studied experimentally. These experiments show there to be a weak interaction of CsOH with structural materials expected to be present in a reactor system such as stainless steel and Inconel. Deposition of CsOH on these materials leaves the cesium in one of three forms: (1) soluble particulate (2) physically adsorbed vapor, and (3) insoluble, chemically-bound, vapor. The research suggests that the particulate material may be formed by the vapor-phase reaction of trace species volatilized from the structural material to form condensed products. The deposition of these particles on the steel obeys aerosol kinetics rather than the empirical deposition velocity rate equation typically used to describe chemical interactions in the primary system.

*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.

Deposition of CsOH as particulate is important at temperatures of 700-850°C but becomes quite unimportant at 1000°C. The reaction of CsOH to form insoluble products may also involve reaction of the vapor with impurities such as silicon in the structural alloys. CsI, too, has been found to react at 1000°C with impurities in structural alloys. These results provide an explanation for the presence of cesium bound to steel parts in the TMI-2 core.

Nonradioactive vapor and aerosol species expected to form in a primary system atmosphere have been found to affect the chemical form and transport of fission products. Boric acid, produced either by steam corrosion of boron carbide control rod materials or vaporization of solutes in the primary system water reacts with both CsOH and CsI. The reaction with CsI yields iodine in a gaseous state:

$$CsI + HBO_2 \rightleftharpoons CsBO_2 + HI$$

$$2HI \rightleftharpoons H_2 + 2I\ (I_2)$$

This reaction could have drastic effects on the transport of iodine during a reactor accident and the behavior of iodine that emerges into the reactor containment.

Tellurium vapor has been found to react with structural alloys efficiently. The low volatility of the products means that these reactions can effectively remove tellurium from the severe accident source term. Tellurium is, however, also reactive with gas-borne species such as silver from control rods and tin from clad alloys. Reaction with these species leaves tellurium as a potential contributor to the severe accident source term. The data necessary to quantitatively evaluate the competitive reactions of tellurium with structures and aerosols have been produced in the High Temperature Fission Product Chemistry program.

RECENT PUBLICATIONS

(1) R. A. Sallach, C. J. Greenbolt, and A. R. Taig, CHEMICAL INTERACTIONS OF TELLURIUM VAPOR WITH REACTOR MATERIALS, NUREG/CR-2921, SAND82-1145, Sandia National Laboratories, Albuquerque, NM

(2) R. M. Elrick and R. A. Sallach, FISSION PRODUCT CHEMISTRY IN THE PRIMARY SYSTEM, Proc. ANS/ENS Topical Meeting on Severe Reactor Accidents, Cambridge, MA, August 28-Sept. 1, 1983.

(3) R. M. Elrick and R. A. Sallach, Chemical Interactions of CaOH and CsI with Reactor Materials in Steam and Hydrogen Atmospheres, (in press).

SUPPRESSION POOL MODELING

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Boiling Water Reactors in the United States are built with a safety feature, a suppression pool, designed to condense steam released during an accident. In a severe accident aerosols can be formed during core melt and can be formed from the concrete basemat/molten core interaction. This pool has the capability to remove a significant fraction of these aerosols. The Reactor Safety Study (WASH-1400) found little information available to produce a scrubbing model and therefore used simplistic assumptions for particle scrubbing efficiency.

This paper describes a mechanistic approach to particle scrubbing that is intended to supply realistic estimates of scrubbing efficiencies. The various pool scrubbing phenomena are modeled and combined in a computer code SPARC (Suppression Pool Aerosol Removal Code) that is under development at Pacific Northwest Laboratory sponsored by the U.S. Nuclear Regulatory Commission.

Three different areas of gas bubble existence are of interest: bubble formation at inlet depth, bubble rise to the pool surface and bubble breakup at the surface. During bubble formation particles can be scrubbed by inertial deposition, diffusiophoresis if steam condensation occurs and thermophoresis. If the entering gas is hot and dry it will cool to pool temperature and cause high evaporation rates that hinder particle scrubbing. During the bubble rise period swarms of bubbles of different degrees of elliptical eccentricity rise to the surface. Since particle scrubbing depends on particle size and bubble size and eccentricity, the code incorporates the statistics and hydrodynamics of swarms and the input data includes the particle size distribution. During the rise period, particles are removed by three mechanisms: inertial deposition due to drop circulation, gravitational settling and diffusion. Each bubble containing non-condensible gas picks up additional moisture by evaporation as it rises in order to maintain a near saturated condition. This evaporation hinders particle removal, especially in saturated pools, and is modeled with appropriate heat and mass transfer resistances. Particle growth due to water absorption by soluble particles is important. At the surface, bubble breakup can reentrain liquid droplets which form a new aerosol above the pool. Other phenomena under investigation but not yet included in SPARC are: the role of turbulence, effects of surfactants on restricting bubble circulation, more detailed gas phase thermodynamics and the possibility of supersaturated conditions occurring in rising bubbles.

SPARC results are best interpreted by examining the relationship between DF and particle diameter (DF = decontamination factor = ratio of into-pool particle mass flow rate/out-of pool particle mass flow rate). A plot of log DF versus log (particle diameter) is a "u" shaped curve. The minimum DF particle diameter (aerodynamic equivalent diameter) is estimated at 0.2 μm for a saturated pool at 100°C with hydrogen gas and insoluble particles. The minimum DF can range from 1 - 10 depending on bubble parameters, pool depth and temperature and fraction of steam in the inlet gas. DFs are larger for cooler pools and for soluble particles and lower for higher molecular weight non-condensible gases. An overall DF for the whole particle spectrum is of prime importance. Since the very tiny and very large particles have large DF's, the mass fraction of the minimum DF size controls the overall DF.

STATUS OF ASSESSMENT OF CONTAINMENT FAILURE

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In a reactor core melt accident the source term, what is released to the environment, depends largely on the mode, magnitude and timing of failure of the containment structure. In case of a PWR large dry containment, if the structure survives early pressure pulses (steam spike or hydrogen burn), the atmospheric release is reduced significantly due to attenuation of the releases in containment, settling, etc., during the hold up time, until further rise in temperature and pressure breaches the containment. A leakage model, therefore, is needed which can describe containment release from about 10-100 volume percent per day leakage up to much larger releases. In order to develop the leakage model, a realistic containment loading model must be developed. In case of a BWR, the containment may fail before the core melts. Since the suppression pool, at least in Mark I and Mark III containments, can provide significant scrubbing of releases, any failure in the dry well, whether ultimate strength failure or leakage, can bypass the pool leading to higher releases. If vents are provided on the vapor space of the wetwell, it can be opened prior to containment failure, thus breaching the containment, but ensuring the pathway of minimum release. Here, too, a leakage model and model for containment loading are needed.

To attack these problems, the Accident Source Term Program Office (ASTPO) has undertaken a program on the development of containment leakage models as well as realistic loading models for PWR and BWR containment types. The ultimate objective of this program is to generate probability distributions of containment loads under severe accident conditions, and of containment capacities (leak rate, etc.), so that probabilities of unacceptable containment behavior (excessive leakage, etc.) can be estimated.

Actual field data on containment capacity is sparse, and those on containment loading in a severe accident are nonexistent. Some experimental data are available, but are not adequate for developing the required probability distributions. In order to circumvent this problem, the collective wisdom and experience of a group of experts in these areas are being utilized to interpret the available information and apply their knowledge to come up with the necessary distributions.

The Containment Load Experts (CLE) panel consists of 18 experts. "Standard Problems" are developed for both PWR and BWR containment types. The standard problem specifies the initial conditions (e.g., for PWR, size of core, metal content of core, temperature of corium, size of reactor cavity and containment, type of concrete, water remperature and primary system pressure at the time of vessel breach). The experts determine the accident phenomena after vessel breach and estimate specific resulting pressure loads on the containment. Their estimates include the median, 5 and 95 percentile pressure loads, together with some sensitivity calculations.

The capacity aspects of containments are being studied by the Containment Performance Working Group (CPWG), formed within the agency, with contractor help. Some leak rate data are available from previous leakage studies and leak rate tests. However, a great deal of subjective information has to be factored in before a suitable capacity distribution is obtained. The median, 5 and 95 percentile leak rate estimates as functions of containment pressures are obtained. Using this information, a probability density function for specific leak rates is developed.

The final step in this program is to combine the load and capacity distributions to obtain probabilities of containment failure, and the appropriate uncertainty bounds. By keeping track of the leakage location and the corresponding attenuation, environmental releases and their probabilities (and consequences, if necessary) will be estimated.

Results of the study to date and preliminary conclusions will be presented at the meeting.

SOURCE TERM UNCERTAINTY*

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The NRC's Interim Source Term Study (ISTS) (1) has yielded near state-of-the-art source terms which are individually determined for each given plant and accident sequence. The usefulness of these source terms will be enhanced by having an estimate of the uncertainty associated with them. The Quantitative Uncertainty Estimation for the Source Term (QUEST) study is making that estimate in a limited fashion on a six-month time scale.

The QUEST study will determine the source term uncertainty in two categories. The first is the uncertainty due to alternate, but reasonable, inputs into the suite of codes used in the ISTS. These codes include MARCH, MERGE, CORSOR, TRAP-MELT, CORCON, VANESA, NAUA, and SPARC. The second category of uncertainty is that due to phenomena that are known but which are either omitted or poorly modeled in the present suite of codes.

The determination of the source term uncertainty will be similar for both categories. First, sources of uncertainty and key input parameters will be identified. Second, the range over which each source of uncertainty or input parameter might vary will be determined. Third, the sources of uncertainty will be combined and propagated to yield the net uncertainty in the radionuclide release.

Identification of sources of uncertainty will draw upon those noted by the peer review of the ISTS draft reports, as well as those noted in other uncertainty studies. In addition, sensitivity studies will be reviewed (or performed, as needed) for the ISTS codes. These studies will help determine which sources of uncertainties have the potential for a large influence on the amount of released radionuclides.

Once key sources of uncertainty are identified, the range over which they might vary will be determined. These ranges will be determined by the spread in available data and by the spread in predictions of defendable models. For example, the uncertainty range in the coefficients for the release of Cesium at a given temperature can be determined from the large amount of data available on Cesium release. However, the uncertainty in the time and temperature history of the fuel (which also affects Cesium release) can only be determined by calculations using various assumptions.

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

The range for each key source of uncertainty will by documented and subjected to peer review. A statement of the confidence level for each range will not be made, nor will subjective probability distributions be constructed. (The amount of time required to develop technically defendable subjective probability distributions or confidence levels and to combine and propagate them to obtain their quantitative effect on the radiological source term is more than that available for this study.) Results concerning the ranges in the input uncertainties should be available for presentation at the meeting.

Combining and propagating the input uncertainty ranges is a difficult task. The technique used must allow for synergisms and avoid inconsistencies. The Monte Carlo method (in which values for each input parameter or process are chosen at random from within their ranges) can meet the above criteria if checks are made that each set of values chosen is internally consistent. However, the Monte Carlo method requires many full calculations and is therefore not acceptable for this short study. The method which will be used might be called a "selective" Monte Carlo method. The set of parameter or process values for the full calculations will be chosen deliberately so as to yield a large range in the calculated radiological source term. Sensitivity studies and scoping calculations will help in choosing the input values. Only a very few full calculations will be performed. The resulting range in the calculated source term will not be the greatest possible, but it should be representative of the source term uncertainty and will not require a large number of full calcultations.

The same source term groupings as used in the ISTS will be used (i.e. iodine, cesium, tellurium, refractory fission products, and non-radioactive aerosol materials). The range of airborne materials within containment as a function of time will be displayed with containment failure time as an independent parameter. In addition, the range in the total amount of radiation released from containment will be determined for two specific failure times (early and late). This latter determination will include effects after containment failure such as resuspension or agglomeration and settling of aerosols. Due to time limitations, only a few plants and sequences will be considered. They will be Surry (a BWR with a large subatmospheric dry containment) with TMLB' and S_2D, and Peach Bottom (a BWR with a Mark I containment) with TW. These cases are considered in the ISTS and dominate the risk for those plants.

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Summary of Cladding Ballooning
Experiments Conducted in NRU

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A series of experiments utilizing full-length, Zircaloy clad PWR size fuel rods were performed in the National Research Universal (NRU) Reactor at the Chalk River Nuclear Laboratories to study cladding deformation and rupture, flow blockage, and coolability of nuclear-heated LWR fuel bundles under prototypic loss-of-coolant accident conditions.

The fuel assemblies consisted of a 32-rod, 3.65 m long array on a 6 x 6 pattern with the four corner rods removed. Each rod of the 17 x 17 PWR design was 9.29 mm (0.379 in.) O.D. with a cladding wall thickness of 0.54 mm (0.022 in.). The fuel bundle assemblies were surrounded by a stainless steel shroud to protect the loop pressure tube. For the Material Tests, the center grouping of twelve test rods was pressurized leaving the outer twenty nonpressurized rods to function as guard rods.

The fuel rods and the test train assembly were extensively instrumented to monitor the experiment conditions. Thermocouples were located at the centerline of selected fuel rods and attached to the inside surface of the cladding at different axial locations as well as on the fuel bundle assembly. Pressure transducers or pressure switches were attached to the center twelve test rods which were pre-pressurized to levels that caused rupture to occur in the high alpha temperature range of 1050K to 1150K, i.e., 3.21 MPa (465 psia) to 4.62 MPa (670 psia). In addition, neutron flux detectors (SPNDs and flux wires) and liquid level detectors (displacement or a TDR) were utilized in the test assemblies.

Seven LOCA-type tests, i.e., three Thermal Hydraulic Tests and four Material Tests, were performed over the period October 1980 to May 1982. Each test consisted of several individual tests for a total of 59. While maintaining the average rod power constant at about 12 W/cm (0.39 kW/ft) to simulate decay heat, reflood delay time and reflood cooling rate were varied for the different tests to simulate the adiabatic heatup (refill), reflood, and quench portions of a LOCA accident sequence.

The rods were preconditioned prior to the LOCA transients by power cycling three times to a peak power of about 328 W/cm (10 kW/ft) which caused the fuel pellets to fracture and relocate within the cladding tubes. With the exception of MT-4, which was conducted in a manner to cause rod rupture to occur during the adiabatic heatup phase before reflood was actuated, a two-phase flow condition existed in the fuel bundles during fuel rod rupture for the Materials Tests. Reflood delay times ranged between 9 and 57 sec. and reflood flow rates ranged

*Operated for the U.S. Department of Energy by Battelle Memorial Institute under contract DE-AC06-76RLO 1830. This work was performed for the U.S. Nuclear Regulatory Commission.

between 0 and 13.7 cm/sec. (0-5.4 in./sec.). The duration, i.e., the time from steam off to reactor trip, of the Materials Tests ranged from 3.2 to 18.7 min. and the average times for rod rupture ranged from 55 sec. for MT-4 to 133 sec. for MT-3. For MT-4, all twelve test rods ruptured within ± 2 sec. of the average rupture time. Maximum cladding temperatures ranged from 1126K (1567°F) for MT-3 to 1459K (2166°F) for MT-4. Average cladding temperatures in the rupture regions of the fuel bundles ranged from 1067K (1461°F) for MT-3 to 1156K (1623°F) for MT-2.

In addition to the data obtained from the instruments during the tests, the test assemblies and fuel rod bundles were designed to be easily disassembleable in the water basin after each test. By use of the Disassembly Examination Reassembly Machine (DERM), the outer contour of each of the test rods was accurately measured at all locations where strain occurred.

The diametral cladding strain and resulting reduction in flow area (blockage) that occurred in the fuel rods during the Material Tests were quite coplanar and occurred axially between the spacer grids which provided mechanical restraint and a localized increase in heat transfer. The average peak cladding rupture strain for rods irradiated in MT-1, MT-2, and MT-3 were 43, 43, and 47 percent, respectively for average cladding rupture temperatures of 1145K (1601°F), 1156K (1623°F), and 1067K (1461°F). The rupture strain for MT-4 was 72 percent at a temperature of 1094K (1511°F). The reason for the higher cladding rupture strain for MT-4 is not understood but may be related to the mode of cooling for this test compared with the other three Materials Tests, i.e., the rods in MT-4 ruptured during the adiabatic heatup phase before reflood cooling was activated. This may have caused a more uniform temperature distribution within rods, the effect of which accentuated the extreme temperature sensitivity of Zircaloy deformation. The maximum calculated flow blockages for the test rod and bundle regions for MT-4 were 100 percent and 25 percent, respectively. The diametral strain-to-failure data and the flow blockage results as functions of temperature are consistent with the regulatory criteria as defined in NUREG-0630.

Thermocouples securely attached to the inside surface of the cladding tubes by welding provided data on fuel rod heat transfer during ballooning and rupture. When compared with the results from comparable Thermal Hydraulics Tests in which no cladding deformation occurred, the results from the Materials Tests showed essentially no difference in heat transfer between the ballooned and ruptured test rods and the nondeformed guard rods and essentially no difference between blocked and nonblocked fuel bundles, i.e., fuel bundle blockages of 18 to 25 percent did not decrease heat transfer. If anything, the increased turbulence associated with ballooning and rupture may have caused increased heat transfer in the deformed regions of the fuel bundle. These effects are attributed to flow acceleration, droplet breakup, and improved mixing in the blockage region which override the effects of local flow reduction due to the blockage.

THE PBF OPTRAN EXPERIMENT RESULTS

Scott A. Ploger

EG&G Idaho, Inc.

Operational transients are temporary deviations from standard nuclear reactor operating conditions resulting from occasional plant component malfunctions. Many projected sequences for operational transients involve loss of secondary heat sinks and a corresponding rise in primary coolant system pressure. In a boiling water reactor (BWR), such a pressure surge would briefly collapse steam bubbles in the two-phase coolant--thereby causing a large positive reactivity feedback and a sudden rise in fuel rod powers. The associated thermal expansion of fuel pellets, compounded by transient release of corrosive fission products, creates a potential for cladding failure from pellet-cladding mechanical interactions (PCI).

A power excursion of this sort would almost always be terminated by a prompt reactor scram. However, an additional malfunction of the scram portion of the reactor protection system can also be postulated, in which case the reactor event would be termed an anticipated transient without scram (ATWS). During a BWR ATWS, the reactivity increase could proceed until the primary coolant pressure exceeds relief valve setpoints and local rod powers could increase by an order of magnitude. Onset of a boiling transition could increase cladding temperatures through reduced heat transfer and initiate oxidation/embrittlement cladding damage mechanisms.

The most severe BWR anticipated transients with and without scram are predicted to be a generator load rejection without steam bypass and a main steam line isolation valve closure ATWS, respectively. However, these operational events have not actually occurred during commercial reactor operation. Accordingly, during 1982 the Thermal Fuels Behavior Program of EG&G Idaho, Inc., was requested by the Nuclear Regulatory Commission to simulate these transients in the Idaho National Engineering Laboratory's Power Burst Facility (PBF) in order to assess the forms and extents of fuel rod damage.

Four scrammed transients of increasing severity were performed during the OPTRAN 1-1 test, with four previously irradiated test rods ($\sim$1 m long) of 8x8 General Electric design mounted within the PBF test train. (Two of the rods were withdrawn for damage investigations and replaced, after the first transient.) The first power excursion followed GE code predictions for a BWR-6 turbine trip without bypass, whereas the second transient simulated a generator load rejection without bypass. The third and fourth OPTRAN 1-1 transients were conducted at successively higher peak rod powers in an attempt to determine a PCI-induced failure threshold for BWR power surges of approximately 2 s in duration. The six Monticello test rods had burnups between 5 and 23 GWd/t and included two "PCI remedy" test rods with zirconium-liner and fuel-additive developmental designs.

The OPTRAN 1-2 test consisted of a single, 20-s power excursion that simulated a main steam line isolation valve closure without scram. Two high-enriched, unirradiated coolant heater rods were plumbed in tandem, through variable orifices, with two irradiated GE test rods. This scheme produced the desired coolant conditions wherein boiling transitions and elevated cladding temperatures could occur.

Nondestructive posttest analyses established that no test rod failures occurred during OPTRAN 1-1 and 1-2. Destructive hot cell examinations were then performed to investigate formation of incipient PCI cracks on interior cladding surfaces. Rod segments were cut, clamshelled, defueled, and flattened to enhance visual detection. One crack (penetrating 20% of the cladding wall) was found within an OPTRAN 1-2 segment, while a few shallow surface anomalies were exposed within OPTRAN 1-1 segments. Per scanning electron microscope examinations, only the OPTRAN 1-2 defect displayed the brittle fracture features typically associated with stress-corrosion cracking and PCI-induced cladding damage. Formation of the OPTRAN 1-2 incipient crack was apparently influenced by much larger rod elongation and fission product release (per rod gas analyses) than occurred during OPTRAN 1-1.

Metallography of the OPTRAN cladding confirmed nondestructive indications that no ridge (permanent hoop strain) formation occurred at pellet interfacial positions. Thus, peak cladding stresses and strains were confined to the elastic region. Metallography also determined that localized cladding temperatures rose to a peak of approximately 1200 K during the OPTRAN 1-2 test as a result of boiling transitions. However, neither cladding deformation nor evidence of oxidation embrittlement was detected at this position. Furthermore, the incipient PCI crack was distant from the high-temperature axial region. Therefore, OPTRAN 1-2 cladding damage directly attributable to reduced heat transfer was apparently limited to cladding phase transformations.

In conclusion, no evidence of PCI-induced fuel rod damage was found after the four successive, conservatively severe OPTRAN 1-1 transients. With the qualification that only six, 1-m-long rods were tested during OPTRAN 1-1, it now seems that commercial rod failure probabilities would be quite small during a BWR turbine trip without bypass, load rejection without bypass, and other promptly scrammed operational transients. It also appears that standard BWR fuel would tend to resist failure during a severe ATWS. Only one small, incipient PCI crack was detected within the two OPTRAN 1-2 test rods, and oxygen embrittlement from elevated cladding temperatures was insignificant. However, the OPTRAN 1-2 rods did experience cladding phase transformations and large fission gas releases as a consequence of reduced heat transfer to the coolant. This suggests BWR fuel downrating or reloading would be warranted following any commercial ATWS event wherein boiling transitions were induced.

EX-REACTOR PCI EXPERIMENTS

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The objective of this work is to study the strain behavior of irradiated Zircaloy cladding using simulated fuel rods in a pressurized water loop. Simulated fuel rod tests are conducted in the loop under flowing coolant conditions of 300°C and 1500 psig. The simulated fuel rod consists of 1) 0.430 in. OD x 0.373 in. ID. Zircaloy-4 cladding irradiated in water-tube positions to an equivalent burnup of 6 MWd/KgM, 2) an 18 in. high stack of nonirradiated, annular UO_2 fuel pellets with a 0.002 in. diametral fuel-cladding gap, and 3) a central tungsten heater rod. During the tests the diametral cladding strain is measured over the length of the rods at two azimuthal orientations 90° apart while the rod is being heated.

To date five irradiated cladding tests have been completed (Table 1). Test conditions were selected to expose the cladding to strain rates in the range that is considered to be most sensitive to PCI-SCC failure. LHGR levels to 8 kW/ft (electrical) which is equivalent to a nuclear rating of 10 kW/ft based upon differential expansion, have been attained for short periods and 6.5 kW/ft (electrical) was maintained for several hours. Both as-received and intentionally scratched cladding tubes have been tested. The simulated fuel rods were exposed to numerous power cycles. Although two of the five tests resulted in cladding failure, the failures were attributed to heater-cladding shorting and not PCI.

Based on cladding strain measurements, the LHGR required for fuel-cladding mechanical interaction is about 2 kW/ft. Both elastic pellet-pellet ridges and midpellet ridges formed in the cladding. Ridge heights increased with increasing LHGR up to values of 0.0007 in. (diametral) at an LHGR of 6.5 kW/ft and relaxed slightly over periods up to 26 hr. The magnitude of these ridges is similar to that observed in prototypic test rods (with larger fuel-cladding diametral clearance) at much higher powers (12-15 kW/ft). Therefore, the simulated rods are approximating the localized stresses in power reactor rods at high power levels. The relaxation was considerably less than what has been observed to occur in-reactor during similar PCI. This is a strong indication that the in-reactor relaxation is fission-enhanced.

Additional testing with iodine additions is planned for FY-1984.

*Operated for the U.S. Department of Energy by Battelle Memorial Institute under contract DE-AC06-76RLO 1830. This work was performed for the U.S. Nuclear Regulatory Commission.

TABLE 1. Summary of PCI Simulator Tests on Irradiated Zircaloy Cladding

Test Identification	Power Cycle No.	Power Ascent[1] Strain Rate, $X10^{-5}\ min^{-1}$	Maximum Power,[3] kW/ft	Maximum Power Holding Period, hr	Power Descent[1] Strain Rate, $X10^{-5}\ min^{-1}$	No. Axial Profilometry Traces	Result
PCII-1	1	Not controlled Varied between 1 and 100	8	0.5	--	30	Cladding breach apparently by electrical shorting
PCII-2	1	2	6.5	4	-2		
	2	2	6.5	3	-2	44	Cladding intact.
	3	2	6.5	0.25	-2		
PCII-3	1	12	6.5	13	--	16	Cladding breach apparently by electrical shorting
PCII-4[2]	1	12	6.5	6	-12		
	2	12	6.5	0.5	> -100		
	3	12	6.5	0.25	> -100	31	Cladding intact.
	4	12	6.5	0.25	> -100		
	5	12	6.5	0.25	> -100		
PCII-5[2]	1	12	5.5	27	-12	21	Cladding intact.

(1) Cladding strain rates were estimated by averaging a) the overall strain including ridges, and b) the ridge strain at several locations from PCII-1 between LHGRs of 2 and 6.5 kW/ft.

(2) The inner surface of the cladding for PCII-4 and -5 were scratched to a depth of about 10% of the wall thickness.

(3) The powers quoted are electrical powers in the simulated rods.

FRACTURE BEHAVIOR OF HIGH-BURNUP SPENT-FUEL CLADDING*

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The Zircaloy cladding of water reactor fuel rods is susceptible to local breach-type failure, commonly known as pellet-cladding interaction (PCI) failure, during operational and off-normal power transients after the fuel has achieved a sufficiently high burnup. An optimization of power ramp procedures or fuel rod fabrication to minimize the cladding failure would result in a significant decrease in radiation exposure of plant personnel due to background and airborne radioactivity as well as an extension of core life in terms of allowable off-gas radioactivity. As part of a program to provide a better understanding of the fuel rod failure phenomenon and to facilitate the formulation of a better failure criterion, a mechanistic study of the deformation and fracture behavior of high-burnup spent-fuel cladding is in progress under simulated PCI conditions. Zircaloy cladding tube specimens from power reactor fuel assemblies (burnup $\gtrsim$20 MWd/kg U) have been deformed to fracture at 292 to 325°C by either internal-gas-pressurization or expanding-mandrel loading techniques without the addition of fission product simulants (e.g., I, Cs, or Cd) to the test environment. The inner and outer surfaces of the fuel cladding were cleaned ultrasonically in alcohol and the mechanical tests were conducted in research-grade helium or argon environments. The fracture surfaces of 11 test specimens were examined by SEM, and six specimens were found to contain various degrees of the pseudocleavage-plus-fluting feature that is characteristic of in-reactor PCI failures. Metallographic cross sections of the failure regions of the six specimens revealed small diametral strains ($\lesssim$1%) and numerous branching cracks. By means of 100- and 1000-keV transmission electron microscopy of thin-foil specimens from regions adjacent to the failure sites, we have established that the brittle-type fractures are characterized by an absence of slip dislocations and an extensive amount of second-phase precipitates. Dense aggregates of ellipsoidal precipitate particles 100-200 Å in size were located primarily in cell wall regions of the stress-relieved material in association with dislocation substructures. From diffraction analyses, the precipitate was identified as Zr_3O phase, which is metallic and brittle in nature. Two different orientational relationships between the Zr_3O and α-matrix phases have been identified. From the results of microstructural examinations, it appears that the brittle-type failure is associated with segregation of oxygen atoms primarily in stress- and defect-concentrated regions, which facilitates formation of the Zr_3O phase and immobilization of dislocations. The mechanism is consistent with evidence of oxygen segregation associated with a strain-aging phenomenon in Zircaloy and Zr-O alloys, and irradiation-induced segregation of oxygen observed in association with a radiation anneal

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

hardening phenomenon in Nb-O and V-O alloys. It appears that a high density of irradiation-induced defects, intrinsic oxygen augmented by corrosion, and local stress fluctuations (due to thermal expansion of the fuel pellets) are conjointly conducive to oxygen segregation and Zr_3O precipitation in high-burnup cladding, which could lead to minimal plastic deformation and PCI failures. TEM microstructural evaluations of Zircaloy cladding from fuel rods that have experienced in-reactor PCI failure are required to further explore the brittle fracture mechanism associated with oxygen segregation and Zr_3O precipitation in regions of localized deformation.

FISSION GAS AND IODINE RELEASE MEASURED UP TO 15 GWd/t UO_2 BURNUP

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The radioactive fission products residing within the fuel-cladding gap constitute a potential source term in the event of cladding failure during reactor or fuel handling accidents. As part of the U.S. Nuclear Regulatory Commission's Fuel Behavior Program, EG&G Idaho, Inc. is conducting fission product release studies with the Instrumented Fuel Assembly 430 (IFA-430) in the Heavy Boiling Water Reactor in Halden, Norway. This paper presents a summary of the measured release of xenon, krypton and iodine up to 15 GWd/t UO_2 burnup for fuel centerline temperatures ranging from 950 to 1800 K, at average linear heat ratings of 15 to 35 kW/m. The IFA-430 is composed of four 1.28-m-long fuel rods containing 10% enriched UO_2 pellet fuel. Two of the fuel rods are connected, top and bottom, to a gas flow system that permits the fission gases released from the fuel pellets to be swept out of the rods during irradiation and measured via gamma spectrometry.

Selected results are summarized in Table 1. The results for the noble gases are presented in terms of the release-to-birth (R/B) ratio, which is the ratio of the measured release rate of an isotope to its calculated production rate. The results of the iodine measurements are presented as release fractions (F), the fraction of the total inventory of an isotope that is residing within the free volume of the fuel rod. In the limit as an isotope approaches radioactive equilibrium the release fraction equals the R/B ratio.

As illustrated in Table 1, the R/B ratio is very low at low burnup, generally $\sim 10^{-4}$, and the release increases by a factor of about 2 up to $\sim$8 GWd/t burnup. Measurements performed with different fill gases (He and Ar) that permit the fuel temperature to be varied at constant power, indicate the temperature of the fuel is the dominant parameter affecting release, and diffusion is the dominant release mechanism. The data up to $\sim$10 GWd/t burnup have previously been compared with the American Nuclear Society Fission Gas Release Model (ANS 5.4) and with a diffusion model (NRC Water Reactor Safety Information Meeting 1980). The ANS model was generally within a factor of 2 of the measured data, and the diffusion model was shown to be consistent in predicting the temperature- and power-dependence of the release and the relative release of the various isotopes.

The data also show that between $\sim$10 and 15 GWd/t burnup during a high power ($\sim$35 Kw/m) operating cycle, a significant change in the gas release occurred. Comparison of the R/B ratio measured at 20 kW/m before and after the high power cycle shows that the ratio increased by a factor of 5 to 7. However, comparison of the pre-high-power-cycle fuel centerline temperature at 20 kW/m with the post-high-power-cycle temperature at 20 kW/m showed no change, indicating the changes in the fuel that affected the gas release did not affect the centerline temperature. This increase in fission gas release independent of an increase in temperature indicates that the effective surface area for release (or the surface-to-volume ratio) has increased. Thus, it appears that the increase in R/B ratio is

TABLE 1. R/B FOR SELECTED ISOTOPES AT ~20 kW/m WITH He FILL GAS

	Burnup (GWd/t)				
	4.2	7.7	9.6	12.6	14.8
^{138}Xe	2.0E-5	5.1E-5	6.3E-5	2.8E-4	4.3E-4
^{135}Xe	1.1E-4	2.5E-4	2.9E-4	--	1.2E-3
^{85m}Kr	9.7E-5	2.3E-4	2.6E-4	6.4E-4	9.1E-4
^{87}Kr	6.1E-5	1.2E-4	1.5E-4	5.0E-4	7.3E-4
^{133}I[a]	1.3E-4	6.1E-4	--	--	1.4E-3

a. Release fraction.

due to physical changes in the fuel. Related measurements of the fuel behavior indicate that significant physical changes in the fuel structure have been occurring. The hydraulic diameter of the fuel-cladding gap has decreased, indicating continued fuel cracking and relocation, and the fuel centerline temperature sensitivity to fill gas pressure has changed significantly, indicating the development of circumferential cracks. Thus, it appears that restructuring of the fuel has occurred during high power irradiation from ~10 to 15 GWd/t burnup. Concurrent with these physical changes, the gas release increased.

Grain boundary interlinkage is the mechanism which is thought to have caused the increase in gas release. It is expected that during the higher power (35 kW/m) irradiation cycle, the fuel temperature and fission gas inventory were sufficient to permit interlinkage of the fission gas bubbles located along grain boundaries. This increases the effective surface area for release of the gases. Apparently, the interlinkage is relatively permanent, since release measurements performed at low power (20 kW/m) following the high power cycle show the factor of 5 to 7 increase compared with the previous (prior to high power operation) results.

The iodines exhibited the same type of behavior as the noble gases, indicating that the iodines follow the same general release path as the noble gases.

Although the R/B ratio has increased by a factor of 10, from 4 to 15 Gwd/t burnup, the quantitative fraction of the noble gases and iodines released from the fuel to the gap remains small, on the order of 10^{-3} to 10^{-2}. Thus, the NRC Regulatory Guide (1.25, 1.3, and 1.4) assumptions of the fraction of noble gases and iodines released as a result of cladding failure during terminated accidents in which fuel temperatures remain low (<~1500 K) or during fuel handling accidents would appear to be adequately conservative for fuel operated at conditions similar to the IFA-430 assembly.

CURRENT STATUS OF THE FASTGRASS/PARAGRASS MODELS FOR FISSION PRODUCT RELEASE FROM LWR FUELS DURING NORMAL AND ACCIDENT CONDITIONS

by

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The theoretical FASTGRASS model has been used for predicting the behavior of fission gas and volatile fission products (VFPs) in UO_2-base fuels during steady-state and transient conditions. This model represents an attempt to develop an efficient predictive capability for the full range of possible reactor operating conditions. Fission products released from the fuel are assumed to reach the fuel surface by successively diffusing (via atomic and gas-bubble mobility) from the grains to grain faces and then to the grain edges, where the fission products are released through a network of interconnected tunnels of fission-gas-induced and fabricated porosity.

Models are included for the effects of the following key variables: production of gas from fissioning nuclei, bubble nucleation and re-solution bubble migration, bubble coalescence, gas-bubble/channel formation on grain faces, temperature and temperature gradients, interlinked porosity on grain edges, microcracking, experimentally derived steady-state bubble mobilities, and phenomenological modeling of bubble mobilities during transient nonequilibrium conditions. These models are used to calculate fission product release and the swelling due to retained fission-gas bubbles in the lattice, on grain faces, and along the grain edges for steady-state and transient thermal conditions.

As the noble gases have been shown to play a major role in establishing the interconnection of escape routes from the interior to the exterior of the fuel, a realistic description of VFP release must a priori include a realistic description of fission gas release and swelling. In addition, as the VFPs are known to react with other elements to form compounds, a realistic description of VFP release must include the effects of VFP chemistry on VFP behavior. A mechanistic description of VFP behavior was developed by modifying the FASTGRASS fission gas analysis to include theoretical models for the effective production rates of the relevant VFPs, the chemical interactions between the various VFPs, the interaction of the VFPs with the fission gas bubbles, and the migration of the VFPs through the solid UO_2 fuel. In the present treatment, the VFPs I, Cs, and their major reaction products (CsI, Cs_2MoO_4, and Cs_2UO_4) have been included. The formation of Cs_2MoO_4 and Cs_2UO_4 can have a crucial effect on the reactions involving CsI, which are of major concern for deducing the form of the iodine released in LWR power plant accident scenarios.

Improvements in modeling fission product behavior include a more realistic model for calculating the diffusive flow of fission products, a more precise model for gas bubble re-solution, the use of a lenticular (instead of spherical) bubble geometry on the grain boundaries in the calculation of intergranular gas bubble swelling and saturation, and an improved model for iodine solubility. Because of existing uncertainties in both materials properties and mechanisms of fission product response, any verified mechanistic description of fission product release entails assumptions in these areas. The diffusivity of fission gas bubbles during transient conditions, and the effect of grain boundary bubble re-solution on intragranular diffusive flow rates, are aspects of fission product behavior that are currently clouded with uncertainty. The assumptions that have been made in the theory have been evaluated and are discussed in relation to the model verification, uncertainties in the existing data base, and other theoretical descriptions of fission product behavior.

PARAGRASS is an extremely efficient, mechanistic computer code whose models are based on the more detailed ones in FASTGRASS. The major differences between PARAGRASS and FASTGRASS are in the treatment of volatile fission products, and in models for the migration of fission products up the temperature gradient. PARAGRASS contains a simple VFP release model. This model is based on the more detailed VFP models contained in FASTGRASS. Sensitivity studies were carried out using the more elaborate FASTGRASS code, and estimates made regarding a correlation between volatile and noble gas behavior. In addition, based on FASTGRASS analysis, it has been concluded that the long-range migration of fission products up the temperature gradient provides a minor contribution to the overall behavior of the fission products in LWR fuel under most conditions of interest. By eliminating this mechanism in PARAGRASS, it has been possible to decouple the calculation radially as well as axially. This feature provides for faster execution times for cases where more than one radial node is modeled. PARAGRASS is ideally suited for incorporation into a whole-core accident analysis code which has the capability of providing PARAGRASS with the relevant fuel operating conditions (e.g., fuel temperatures). PARAGRASS updates have recently been designed and implemented for the FRAPCON, FRAP-T, and SCDAP computer programs. PARAGRASS has undergone verification with available steady-state and transient experimental data on fission-gas behavior.

General Overview of the Severe Fuel Damage Research Program

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As part of its response to the accident at TMI-2 the U.S. Nuclear Regulatory Commission has established a Severe Accident Research Program to furnish the Commission with information needed for decisions on severe LWR accidents beyond the design basis. A major part of this program is the Severe Fuel Damage (SFD) research program of the Fuel Systems Research Branch. The purpose of the SFD research program is to furnish a data base and verified analytical models for use in assessing the consequences of severe LWR accidents.

The technical issues addressed by the program are as follows. What is the fission-product release from the core during the course of a severe accident, including the timing, chemical form, and aerosol formation; and what are the in-vessel fission-product removal mechanisms? These are major questions in determining the accident radiological source term. What is the hydrogen release from the core? This is a major question in evaluating containment integrity. What are the physical and chemical states of the core and the termperature distributions during the major severe accident sequences, including the progression of core melt and the attack on reactor structure and the reactor vessel? What are the conditions at reactor vessel failure, including the mode of vessel failure and the mass and temperature distribution of the melt. This information on the state of the core-temperature distributions is a necessary basis for reliability modeling severe accident consequences. What are the characteristics of the severely damaged core debris in different accident sequences? This is a primary unknown in determining the coolability limits of severely damaged cores by reflooding, including the requirements on coolant supply and timing for successful accident termination by reflooding.

Severe accident safety assessment and risk assessment require models and safety analysis codes for the range of risk-significant accident sequences and parameters for the governing phenomena, and a data base for model development and verification. In order to have the verified models needed for severe-accident safety assessment, an integrated research program is needed. This program requires: large scale integral tests for multi-effect interactions, which are expensive but essential; separate effects phenomenological experiments to develop and verify models and to cover the necessary parameter range, which are very cost effective; and analytical model and code development and verification. It is the verified models and codes that are used in severe accident safety assessment and risk assessment, and not the test data directly.

The backbone of the Severe Fuel Damage research program is the series of integral multi-effect tests performed in the PBF test reactor. Phase 1 of this program consists of five tests performed under core-uncovery conditions to maximum temperatures of 2400K, two of which have already been performed. These tests use both fresh and pre-irradiated fuel, all preconditioned by five days operation for fission product build up with eight days pretest shutdown for cesium build up by fission-product decay. Fission product release, hydrogen generation, and temperature distributions are measured, and the post-test fuel characteristics are determined by neutron radiography and tomography and by Post Irradiation Examination (PIE). Two tests involve reflood quenching of the hot fuel bundle.

The Phase-2 PBF tests will have advanced fission product diagnostics for both fission product release and transport. These two irradiated fuel tests will be allowed to proceed to full fuel melt temperature (3100K), probably in the plant blackout and the small break LOCA accident sequences.

Integral tests are also planned in the Canadian NRU reactor to provide full-length verification of the severe fuel damage models that are based on PBF and ACRR data with 1.0 and 0.5m long fuel rods.

Separate effects experiments in the SFD research program are performed in the ACRR test reactor and in the laboratory, both at KfK in the Federal Republic of Germany and in the United States. A series of experiments is underway in ACRR on the processes involved in the development of fuel damage and on debris relocation under both core uncovery and reflood quench conditions. These fresh fuel experiments are a very cost effective way of covering the needed severe-accident parameter space, and they give time-continuous optical data on damage development, surface temperatures, and hydrogen generation that are used for model development.

Separate effects phenomenological experiments are also underway in ACRR on the dryout coolability limits of severely damaged fuel under reflooding. The purpose of these experiments is to verify the current advanced LMFBR models of debris coolability limits for the LWR specific conditions of very deep beds, high pressure, and relatively coarse debris.

Laboratory separate effects experiments are underway, mainly at KfK, on the thermodynamics and the kinetics of the reactions between UO_2, Zircalloy, and steam. The pioneering integral laboratory experiments on the development of severe fuel damage by Hagen at KfK are being extended, along with laboratory experiments and modeling on debris-bed coolability limits.

Examination of the TMI-2 core will provide unique and invaluable benchmark data on fuel behavior under severe accident conditions.

In addition to model development on the governing processes using data from the experimental program, several mechanistic codes for severe accident safety assessment are under development and verification. The Severe Core Damage Analysis Package (SCDAP) describes the state of the core during core-uncovery and quench, including oxidation, fission product release, fuel liquifaction, and debris formation and coolability following reflood quenching. The Melt Progression Model (MELPROG) describes the core-melt fuel relocation and attack on the reactor internal structure and the reactor vessel. It also gives the conditions at vessel failure as initial conditions for analysis of the core-melt threat to containment integrity. These mechanistic codes will serve to benchmark MARCH-2 and the MELCOR advanced risk assessment code, and they will also be the embodiment of the results of the Severe Fuel Damage experimental program.

"Integration of PTS Studies to Calculate Through Wall Crack Probabilities"

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In a program sponsored by the Division of Risk Analysis, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, the Oak Ridge National Laboratory has been studying three operating nuclear power plants to assess their susceptibilities to pressurized thermal shock. The issue of pressurized thermal shock pertains to the concern that some pre-existent crack in the steel reactor vessel of a nuclear plant might be induced to propagate completely through the wall. This issue has been addressed by the technical community previously; studies in Europe and the U.S. generally concluded that the probability of vessel failure due to PTS was less than 10^{-6} at a 99% confidence level if they were built to the specifications of the ASME Boiler and Pressure Code.

Since these previous studies, however, thermal hydraulic transients have occurred in operating commercial plants that are subjecting the pressure vessels to unanticipated loadings, TMI-2, Ranch Seco, Crystal River. Such transients have raised the concern that the probability of pressure vessel rupture may be larger than considered previously. These transients may have caused a rapid cooling of the vessel internal surface with resulting temperature distributions leading to significant tensile stresses. If such transients occur when the vessel is pressurized, the pressure stresses will compound the problem significantly. Also, low temperatures at the reactor vessel wall result in a lower fracture toughness.

Recent analyses by the staff of the NRC raised the concern that the fracture toughnesses of some vessels in operating nuclear plants have been reduced considerably already by radiation damage due to fast neutron fluences. The vessels of concern have relatively high copper and nickel concentrations, primarily in the welds. The combinations of: 1) lowering of fracture toughness due to radiation damage, 2) further lowering of fracture toughness due to low temperatures in observed transients, and 3) high tensile stresses generated at the inner surface during these transients have led to significant concern within the nuclear industry.

In addressing this concern, the NRC is sponsoring a research program at ORNL to provide a best estimate of the probability of through wall crack penetration in the pressure vessel of each of three operating nuclear power plants: Oconee-1, a B&W reactor owned and operated by Duke Power Company; Calvert Cliffs-1, a C.E. reactor owned and operated by Baltimore Gas & Electric Company; and H. B. Robinson-2, a W reactor, owned and operated by Carolina Power and Light Company. As a part of this study for each plant, ORNL will determine:

1) which transients or types of transients are most important contributors to the PTS problem,
2) sensitivity to key assumptions
3) source and magnitude of major uncertainties
4) effectiveness of possible corrective measures.

The status of the program is shown below:

	Work Element	Organization	Status
1)	Acquire all necessary plant information	ORNL	Complete for all three plants
2)	Develop scenarios for all overcooling events	ORNL	Complete for all three plants. Event tree formalism, truncated at 10^{-6} probability.
3)	Quantify all scenarios	ORNL	Complete for Oconee-1 and Calvert Cliffs-2.
4)	Develop thermal hydraulic and control system models	ORNL, INEL, LANL	Complete for all plants
5)	Calculate thermal hydraulic consequences of selected representative scenarios	INEL, LANL	Complete for Oconee-1 and Calvert Cliffs-1
6)	Estimate consequences of all other scenarios	ORNL	Complete for Oconee-1
7)	Perform probabilistic fracture mechanics calculations	ORNL	In process for Oconee-1
8)	Integration of results	ORNL	In process for Oconee-1
9)	Sensitivity/Uncertainty Analysis	ORNL	In process for Oconee-1
10)	Evaluation of potential corrective measures	ORNL	In process for Oconee-1

Program completion is scheduled for June, 1984.

ORNL would like to express appreciation for the considerable efforts expended by INEL and LANL in their calculations of the thermal hydraulic consequences of selected overcooling scenarios and appreciation for the quality assurance reviews of these calculations by BNL.

RELAP5 ANALYSES AND SUPPORT

OF OCONEE-1 PTS STUDIES

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EG&G Idaho, Inc.

The integrity of a reactor vessel during a severe overcooling transient with primary system pressurization is a current safety concern and has been identified as an Unresolved Safety Issue(USI) A-49 by the U.S. Nuclear Regulatory Commission (NRC). Resolution of USI A-49, denoted as Pressurized Thermal Shock (PTS), is being examined by the U.S. NRC sponsored PTS integration study. In support of this study, the Idaho National Engineering Laboratory (INEL) has performed RELAP5/MOD1.5 thermal-hydraulic analyses of selected overcooling transients. These transient analyses were performed for the Oconee-1 pressurized water reactor (PWR), which is Babcock and Wilcox designed nuclear steam supply system.

The RELAP5 model simulates in detail the Oconee-1 power plant. All major flow paths for the primary and secondary systems are modeled. The primary system includes power operated relief valves (PORVs), safety valves, and the emergency core cooling system (ECCS). The secondary includes turbine bypass and stop valves, safety valves, emergency feed water (EFW) system and the integrated control system (ICS). The model contains 220 volumes, 232 junctions, and 208 heat structures. The RELAP5/MOD1.5 computer code used for the analyses is an interim version of the RELAP5/MOD2 computer code being developed by the NRC at the INEL.

Nine transients were analyzed with the RELAP5 Oconee-1 model. Each transient was simulated until the analyses could be extrapolated to two hours of transient time. Table 1 summarizes the nine transients analyzed. Also shown are the two parameters of primary importance, downcomer pressure and temperature. The most severe of the nine transients were:

1. Failed open turbine bypass valves (4) with the reactor at hot standby
2. Two and one-half inch diameter reactor coolant pump suction break (reactor at full power)
3. Main steam line break with reactor coolant pumps restarted when the subcooling margin was obtained (reactor at full power).

The two inch pressurizer surge line break (transient 7) produced the lowest downcomer temperature. However, the primary system did not repressurize, thus decreasing the severity of the transient relative to pressurized thermal shock.

The results of the RELAP5 thermal-hydraulic analyses performed at the INEL provide input to further fracture mechanics analyses being performed at the Oak Ridge National Laboratory. The results of these analyses will ultimately determine the potential of the nine transients for pressurized thermal shock.

TABLE 1. SUMMARY TABULATION OF OCONEE-1 PTS RELAP5 ANALYSES

Transient	Sequence	Minimum Downcomer Fluid Temperature (K)	Minimum Downcomer Fluid Temperature (°F)	Maximum Subsequent Downcomer Fluid Pressure (MPa)	Maximum Subsequent Downcomer Fluid Pressure (psia)
Main steam line break	1. RC pumps restarted 10 min after subcooling attained	481	407	16.99	2465
	2. RC pumps restarted at time subcooling attained	494	429	16.99	2465
Steam generator overfeed	3. NFW pumps tripped on low suction pressure	505	450	16.99	2465
	4. Maximum sustainable without tripping MFW pumps	500	440	16.99	2465
	5. Failure open of 4 TBP valves at reactor hot standby	387	237	16.99	2465
Small break LOCA	6. Stuck-open PORV, RC pumps not tripped	545	521	11.38	1650
	7. Two-inch diameter pressurizer surge line break	355	180	1.48	214
	8. Two and one-half inch diameter RC pump suction break	446	343	5.17	740
	9. Steam generator tube rupture	505	450	16.99	2465

TRAC ANALYSIS AND SUPPORT OF OCONEE-1 PTS STUDIES*

by

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Pressurized thermal shock (PTS) in pressurized water reactors (PWRs) has been identified by the Nuclear Regulatory Commission (NRC) as an unresolved safety issue (A-49). Because of this, the NRC has a major multi-organization program to help resolve the PTS issue by determining the potential risk of older reactor vessels to severe overcooling transients that rapidly cool the primary system. The concern over PTS arises because the vessel wall material becomes embrittled and the nil-ductility temperature (NDT) increases after many years of neutron irradiation.[1] If during a severe overcooling transient, cold high-pressure injection (HPI) liquid cools the vessel wall below the NDT, flaws in the vessel welds might be transformed into larger cracks that could lead to a vessel rupture.

At Los Alamos we are using the multidimensional, two-fluid, non-equilibrium numerical simulation code, TRAC-PF1[2], to predict the thermal-hydraulic response of nuclear power plants to selected overcooling scenarios. These thermal-hydraulic conditions then are used for input to detailed stress-analysis codes. Duke Power Company's Oconee-1, a Babcock-&-Wilcox (B&W)-designed plant with once-through steam generators was the first plant studied under this program. Because the risk of initiating or propagating flaws in the vessel wall depends on coupling the thermal stresses produced by overcooling with the mechanical stresses from repressurization, detailed modeling of both the reactor-plant primary and secondary systems is necessary to properly analyze the PTS phenomena. The steam-generator (SG) secondary-side inlet conditions directly affect primary temperature, pressure, and the emergency core-coolant injection. Secondary-side inlet conditions are highly dependent on main feedwater-pump and SG control-valve operation and on the termination of the extracted steam supply to the feedwater heaters. Other

*Work performed under the auspices of the US Nuclear Regulatory Commission.

important systems modeled in the TRAC-PF1 input decks include models of the Integrated Control System (ICS) which monitors the primary flows and temperatures to determine the feedwater demand and regulates the main and startup flow control valves, the main-feedwater (MFW) pumps, and the turbine bypass valves (TBVs).

Several overcooling transients were identified by Oak Ridge National Laboratory (ORNL),[3] and additional transients may be specified after these initial results are evaluated. The initial transients included a main stream-line break (MSLB) with a delay in isolating the affected SG, small-break LOCA [full-open failure of the power-operated relief valve (PORV)] with failure of the ICS to throttle MFW flow and trip the reactor coolant pumps (RCPs), and TBV transients with SG overfeed. An actual plant transient (Oconee-3 turbine trip) simulated by TRAC-PF1 was also compared with actual plant data to verify the code models of the primary side. In addition, several small hot-leg breaks were analyzed to investigate the effects of accumulator and low-pressure injection flows on downcomer fluid temperatures.

Although a number of transients were calculated, only two transients for the Oconee-1 plant will be presented in detail. These transients are the MSLB and the TBV transient in which one bank of TBVs was assumed to fail open. For both of these transients, the calculated minimum downcomer fluid temperatures (MSLB $\sim$405 K; TBV $\sim$365 K) were near the NDT limit ($\sim$365 K) for Oconee-1. All Oconee-1 calculations showed significant primary-system depressurization followed by repressurization depending whether or not the HPI was throttled. Also, the system repressurization was dependent on the break size for the small hot-leg break cases. Some overcooling was obtained in all calculations as evidenced by highly subcooled liquid temperatures in the downcomer.

Other calculations should be performed to consider other operator actions that may affect the overcooling sequences and to determine whether or not system repressurization occurs for the small-break LOCA's with other break sizes and locations.

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TRAC ANALYSES OF POTENTIAL OVERCOOLING TRANSIENTS AT THE CALVERT CLIFFS-1 PWR FOR PTS RISK ASSESSMENT

By:

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Prolonged exposure of a nuclear reactor vessel to radiation from its core will increase the temperature at which the nil-ductility transition (RT_{NDT}) occurs in that vessel. As a result, transients in which the primary fluid rapidly cools the vessel below the RT_{NDT} and in which the system subsequently repressurizes have the potential of cracking the vessel. Pressurized thermal shock (PTS), therefore, is of serious concern and has been identified by the U.S. Nuclear Regulatory Commission (NRC) as an Unresolved Safety Issue USI-49. A major effort among the utilities, NRC, Oak Ridge National Laboratory (ORNL), Brookhaven National Laboratory, Idaho National Engineering Laboratory, and the Los Alamos National Laboratory (LANL) has been established to study the PTS issue.

Three power plants have agreed to participate with the NRC in this detailed study: (1) Oconee, a Babcock and Wilcox plant, (2) Calvert Cliffs, a Combustion Engineering (CE) plant, and (3) H. B. Robinson, a Westinghouse plant. The task assigned to LANL is to model each of the before mentioned plants using the transient thermal-hydraulic code TRAC and to perform a series of overcooling transients postulated by ORNL. This summary shows preliminary results obtained for Calvert Cliffs.

Calvert Cliffs is a CE power plant located on the Chesapeake Bay in Maryland. It is owned by the Baltimore Gas and Electric Company. Unit 1 began operation in January 1975. It has a 2 X 4 loop arrangement: two hot legs and two steam generators with four cold legs and four reactor coolant pumps. It presently operates at 2700 MW_{th}.

Baltimore Gas and Electric Company and CE have put extensive effort into supplying information about the plant and its operation. LANL used this information to prepare a TRAC model of the Calvert Cliffs plant. Both steady state and transient plant data supplied by Baltimore Gas and Electric Company were used to normalize the model.

Seven transients (specified by ORNL) have been completed. The transients were performed with the assumption that no operator intervention was allowed (with the exception of tripping the reactor coolant pumps). A summary of the minimum fluid temperature in the vessel downcomer and the repressurization behavior is tabulated in Table I.

TABLE I
Summary of Minimum Temperatures and Pressures

Transient	Minimum T (K)	Minimum P (MPa)	Repressurizes
1.0 ft^2 Main Steamline Break from hot, zero power	395	4.8	yes
Double-ended Main Steamline Break from hot, zero power with runaway auxiliary feedwater to the broken steam generator	377	3.7	yes
1.0 ft^2 Main Steamline Break from full power	450	6.0	yes
Stuck-open Turbine-bypass Valve from full power	530	10.8	yes
Stuck-open PORV and Atm. Dump Valve from full power	407	6.0	no
Hot-leg Break from full power	342	2.6	no
Runaway Feedwater from full power	480	7.0	yes

*Assumes no operator intervention.

QUALITY ASSURANCE OF PTS THERMAL HYDRAULIC CALCULATIONS AT BNL

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Rapid cooling of the reactor pressure vessel at high pressure has a potential of challenging the vessel integrity. This phenomenon is called overcooling or Pressurized Thermal Shock (PTS). The Nuclear Regulatory Commission (NRC) has selected three plants representing three types of PWRs in use for detailed PTS study. These are Oconee-1 (B&W), Calvert Cliffs (C.E.), and H. B. Robinson (Westinghouse). Oak Ridge National Laboratory (ORNL) is the lead contractor for this study and they have identified several groups of possible transients which could lead to severe overcooling in these plants. The thermal hydraulic calculations for these transients were to be calculated at the Los Alamos National Laboratory (LANL) and the Idaho National Engineering Laboratory (INEL) using the latest versions of TRAC-PWR and RELAP5 codes, respectively. The Oconee-1 transients were divided between LANL and INEL, with some transients common to both. The Calvert Cliffs and Robinson transients were assigned to LANL and INEL, respectively.

The Brookhaven National Laboratory (BNL) has been requested by NRC to review and compare the input decks developed at LANL and INEL, and to compare and explain the differences between the common calculations performed at these two laboratories. However, for the transients that will be computed by only one laboratory, a consistency check will be performed. So far only Oconee-1 calculations have been reviewed at BNL, and the results are presented here.

In the first part of the task, BNL checked and compared input decks for Oconee-1 as prepared by LANL and INEL. There were some differences between these decks in the reactor vessel and heat structure description. BNL also reviewed the models for control systems as developed by Science Application, Inc. (SAI) for RELAP5 and by LANL for TRAC-PF1. The comments based on these reviews were transmitted to the NRC and the PTS study participants.

Calculations for twelve transients selected by ORNL for Oconee-1 were divided between LANL and INEL. Some of these transients such as Main Steam Line Break (MSLB), 2-Inch Hot Leg Small Break Loss-of-Coolant Accident (SBLOCA), and Turbine Trip Transient for Oconee-3 were common to both the laboratories. The TRAC and RELAP5 results for these transients were compared at BNL. It was also observed that MSLB and 2 and 4 Turbine Bypass Valves (TBVs) stuck open at full power and at hot standby were relatively severe transients. Therefore, 4 TBVs stuck open transients were also investigated. The review of Oconee-3 transient indicated the differences between the TRAC and RELAP5 code calculations and the data. However, after this calculation, the codes were modified and the conclusions from this transient are no longer relevant.

The comparison of the Main Steam Line Break transient indicated that the difference in the minimum downcomer fluid temperature prediction was due to the control system model which regulated the MFW pumps and EFW pumps, and multi-dimensional effects which resulted in different temperature histories for the hot legs and RCPs restart times. The RELAP5 model of the control system which was based on the pressure drop in the tube region of the steam generator secondary side was closer to the plant conditions than the TRAC model. The other major difference was the way the upper head was modeled in two codes. TRAC had no dead end volume for the upper head so void accumulation did not occur there; instead they migrated to the candy cane resulting in the termination of natural circulation in the unaffected loop B. However, if this natural circulation was maintained a little longer in loop B, the differences in the two hot leg fluid temperatures would be less and the multi-dimensional effect would be less significant. Based on the comparison of two calculations, an appropriate procedure would be to delay the restart of RCPs in the RELAP5 calculation until TRAC's RCPs restart time of 526 seconds and this would result in a minimum downcomer fluid temperature of approximately 450 K.

The second transient compared was initiated by the failure of all 4 TBVs at the full open position after a turbine trip. This transient is like a small break in the steam line. Here the initial conditions (full power for TRAC and hot standby for RELAP5) and additional failures were different. The codes predicted the transients reasonably well. However, the important causes of differences in the calculations were the failure of the controller to throttle the EFW based on secondary side level and the failure of the operator to close the TBVs at 600 seconds in the TRAC calculation.

The third transient compared was the Small Break (2-inch) LOCA in a hot leg. The ICS worked as designed. Both the codes computed continuous drop in the primary side pressure during the transient. This made the transient less critical for PTS. However, a comparison of two calculations indicated that there were many differences. The codes modeled the reactor trip differently. RELAP5 correctly based it on the low primary side pressure while TRAC assumed it to occur at 0.5 seconds. Furthermore, after the natural circulation was lost due to candy cane voiding, TRAC computed oscillations in the cold leg flows while RELAP5 predicted stable circular flow between the cold legs connected to common steam generators. The loop oscillations in the TRAC calculation warmed up the cold leg and the downcomer fluids. However, it is not clear if these oscillations are real. Moreover, the TRAC calculation was not carried out far enough in time to determine the minimum downcomer fluid temperature with confidence. The RELAP5 calculation, on the other hand, was more complete and looked reasonable.

In summary, three of the several transients for Oconee-1 computed by LANL and INEL using the latest versions of TRAC-PF1 and RELAP5/MOD1.5 have been reviewed at BNL. Both the codes were reasonably successful in modeling these transients. The major differences in their results were due to the difference in modeling the plant, control systems and event sequences, and the one-dimensional modeling of the reactor vessel by RELAP5.

3-D THERMALHYDRAULIC CALCULATIONS USING SOLA-PTS

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The TRAC[1] and RELAP-5[2] codes are being used to examine various accident scenarios for specific reactor designs to determine whether conditions exist that might lead to crack propagation in the vessel wall of the reactor. However, in some cases these system studies must be supplemented by detailed multidimensional studies in order to examine the complicated thermalhydraulic mixing in the cold leg annd downcomer of the reactor. The SOLA-PTS[3] code was developed to perform these detailed studies, at isolated times during the transient, using the system code data for inlet boundary condition specification.

SOLA-PTS is a three-dimensional, incompressible, single-fluid code for examining turbulent mixing in complicated geometries. A system of eight finite-difference equations is solved for the momentum components, the incompressibility condition, the mean temperature, the turbulent energy, the turbulent energy decay rate, and the square of the temperature fluctuations. These equations are written in second-order form using the Tensor Viscosity Method[4], and modified by the FRAM method[5] to avoid dispersion errors. A modified form of the Launder-Spalding turbulence model[6] is used in the cold leg pipe away from the HPI inlet, but the three-equation model of Chen and Nikitopoulous[7] is used throughout the remainder of the computation region. A wall-heat transfer capability is included in the code, but adiabatic boundary conditions are used for the calculations that are done in conjunction with the system codes.

The SOLA-PTS code has been checked by applying it to Creare 1/5 scale experiments[8] that simulate the mixing of HPI and loop flow in the cold leg and downcomer of reactor geometries. The calculated results are generally in excellent agreement with the experimental data in both the cold leg and the downcomer. This agreement with data extends to such features as the cold leg flow jumping the gap to impact on the core barrel wall, the diverting of the cold water flow in the downcomer in the azimuthal direction (generally in the direction away from the hot leg obstacle), and the intermittent nature of the cooling of the vessel wall just below the cold leg inlet. To obtain this good agreement with data in the downcomer, an accurate calculation of the mixing of the cold HPI fluid with the hot fluid in the downcomer is required. Our experience has shown that the turbulence model of Ref. 7 can predict this mixing process very well.

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ONE HALF SCALE THERMAL MIXING TESTS

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The issue of Pressurized Thermal Shock [1] requires a multidisciplinary evaluation, as described for example by Sun and Chexal [2]. The Electric Power Research Institute (EPRI) and the U. S. Nuclear Regulatory Commission (NRC) are individually pursuing research programs to address this issue. One of the disciplines addressed in this program is so-called "thermal mixing" which involves the flow distribution and mixing of cold High Pressure Injection water as it is introduced into the primary coolant water of pressurized water reactors, and also involves heat transfer to the reactor vessel walls. This paper describes a jointly sponsored EPRI/NRC project of analysis and testing of thermal mixing which uses a facility at 1/2 of reactor scale in all dimensions.

Much of the understanding of the underlying thermal-mixing phenomena was first established by four series of experiments sponsored by EPRI at Creare [3,4,5,6] which used a transparent facility at 1/5 of reactor scale and emphasized flow visualization in conjunction with numerous temperature measurements. Various fundamental analyses and correlations have also recently been developed based on these and other data, as described for example by Oh, Sursock and Sun [7] and a scaling analysis was presented by Rothe and Wallis [8]. A detailed description of the 1/2-scale facility and test plan is available [9] and this summary will highlight its major elements.

The similarity principles guiding the facility and tests are:

1. Geometric similarity
   a) preserve all ratios of flow path physical dimensions and
   b) the model size scale will be 1/2
2. Kinematic similarity — preserve all ratios of boundary velocities
3. Dynamic similarity
   a) Froude and Reynolds numbers cannot be simultaneously preserved
   b) Froude scaling will be emphasized and Froude number will be preserved
   c) Reynolds number will also be preserved in special tests
4. Thermal similarity
   a) temperature differences will extend up to about 60% of prototype
   b) density ratio will be varied over a factor of four range extending up to about 50% of prototype
   c) thermally thick walls will give applicable data for heat transfer coefficient and fluid temperature

The detail design basis of the facility includes:

1. Prototypical pressurized water reactor components:
   a) cold legs of Combustion Engineering, Babcock & Wilcox and Westinghouse designs, including loop seals and pump simulators
   b) high pressure injection ports for the above cold legs.
   c) a 90° planar sector of a reactor vessel downcomer, including one cold leg (simulating a plant having 4 cold legs)
   d) a lower plenum with scaled volume and internal flow baffles
2. Test vessel pressure limit of 1.38 MPa (200 psia)
3. Thermally thick walls of at least 2 inch thickness
4. Replication of all geometric features that bear significantly on thermal mixing or heat transfer for all pressurized water reactors

5. Capability to add a thermal shield in the downcomer
6. Two different cold leg geometries (horizontal and inclined)
7. Multiple injection locations for each cold leg
8. Steady-state and transient testing capability
9. Extensive instrumentation summarized below:
    a) 90 thermocouples in the cold leg and loop seal (at least 10 with fast response) and 130 thermocouples in the downcomer/plenum
    b) 40 thermocouples on the downcomer wall surfaces
    c) 40 heat flux sensors in the downcomer walls
    d) 40 velocity probes in the cold leg and downcomer
10. Additional process instruments to monitor loop pressure, loop, HPI and vent valve flows and temperatures for controlled steady-state and transient tests.

In addition to describing the facility and test plan, the presentation may include sample data. As this summary is prepared, the facility and procedures are in the last stage of shakedown, prior to testing.

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DECAY OF BUOYANCY DRIVEN STRATIFIED LAYERS WITH APPLICATION TO PTS

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Results from Purdue's PTS program will be reported in three areas: (a) Further development and testing of our regional mixing model originally proposed in the Bethesda scaling meeting September 1982 (Proceedings in NUREG/CR-0043), (b) Development and testing of a criterion for stratification during HPI injection, and (c) Local mixing and turbulence data in a 1/2-scale simulation of a cold leg geometry under HPI injection in a stagnated loop flow.

(a) *Regional Mixing Model.* The model is based on the idea that at zero loop flow a well-mixed cold leg cannot exist because if it did the stability criterion for existence of no back-flow from the downcomer would be violated. The calculation for the temperature transient is based on a fundamentally oriented turbulence model for the decay of buoyant plumes and takes into account the effect of lower plenum and loop seal in providing a source of warm water for mixing with the decaying plumes in counter flow. Data for the decay of simple plumes at the relevant range of low injection Froude numbers will be reported and agreement with the turbulence model predictions will be shown. The applicability of this model to the more complicated reactor geometry is demonstrated next by showing excellent predictive capability of the 1/5-scale CREARE data at zero loop flow. Finally predictions for a typical Westinghouse PWR, including the effect of the metal wall heat will be shown.

(b) *Stratification Criterion.* Current system codes do not take into account stratification in the cold leg. It is useful, therefore, to determine the applicability of such calculated results as the loop flow decreases during a transient (i.e., by means of appropriate screening criteria). We will show the theoretical development of such a criterion that express the transition from the well-mixed to the stratified regime in the Fr_{HPI} - Q_L/Q_{HPI} plane (with D_{CL}/D_{HPI} as a parameter). The criterion is expressed as a simple algebraic calculation convenient for calculation and correctly predicts the behavior in the 1/5-scale CREARE tests. In addition experimental data will be presented from Purdue's 1/2-scale facility in support of the stability criteria utilized in the derivation.

(c) *1/2-Scale Experiments.* A 1/2-scale geometric model of a Westinghouse reactor cold leg was designed, including the whole accumulator (injection) line, the loop seal, the upper portion of the downcomer and a distorted simulation of the lower plenum. Salt resolution is used to model cold HPI water. The facility is built of acrylic and is extensively instrumented with conductivity probes and hot wire anemometry for local, instantaneous measurements of mixing and turbulence. The purpose is to complement the CREARE 1/2-scale facility in providing the detailed data required for testing the various predictive models (i.e., our Regional Mixing Model, LANL's SOLA-PTS etc.). The facility description and the first round of experimental data will be reported at the meeting.

Simplified Predictions of Pressurized Thermal Shock

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This paper describes a simplified model for the prediction of temperatures in the downcomer annulus of a pressurized water reactor (PWR) under conditions which could lead to pressurized thermal shock. These conditions typically occur following activation of the high pressure injection (HPI) system. For certain PWR small break accidents, the flow in the downcomer annulus has been predicted to go nearly to zero. HPI activation in this zero flow case can result in a rapid cool-down of the cold leg into which the HPI water is injected. After mixing with cold leg water, the HPI flow then enters the downcomer and cools the sector of the downcomer below the cold leg discharge. Localized downcomer cooling can also occur when the reactor is operating with one of its cold legs isolated at power with forced flow or during cool-down at natural circulation and the HPI system is activated. HPI injection water will mix with water in the isolated cold leg and then enter the downcomer annulus, cooling the sector below the isolated cold leg discharge. The downcomer temperature maldistributions which result from either of these cases depend on the mixing of the HPI flow in the cold leg and the mixing which occurs as the HPI flow enters the downcomer annulus.

A model has been developed to study PWR downcomer and thermal response under the conditions just described. Elements of the model include: modeling of the HPI jet as it enters the cold leg and entrains and mixes with water in the cold leg; modeling of the cold leg discharge into the downcomer annulus and mixing with water in the downcomer; and, finally, modeling of the downcomer flow which results when the HPI flow enters it at the cold leg discharge. While all of these elements have been greatly simplified in the model, they all play an important role and are necessary for prediction of downcomer temperatures.

Comparisons have been made between model predictions and scaled test data which represent the zero flow condition. Data from these scaled tests are in sufficient detail to define phenomena occurring in the cold leg and downcomer during HPI injection and provide a basis for the modeling simplifications which have been made. Finally, predictions by the model of PWR downcomer annulus temperatures under those conditions which could lead to thermal shock are presented. The effects of various modeling uncertainties on the temperature predictions are also discussed.

Light Water Reactor Safety Research Trends at Electric Power Research Institute

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The trends of the Light Water Reactor (LWR) safety research pursued by the Nuclear Division at Electric Power Research Institute (EPRI) have evolved with the evolving safety concerns. The current programs have emphasized areas of research, where information is needed in a relatively short time frame, to resolve safety concerns, which are generic to the nuclear industry. Thus, certain research areas have received and are receiving increased attention. These are: (a) fission product "source term", (b) hydrogen control, (c) containment failure modes, (d) the seismic loads, (e) common-mode failures, (f) safety control technology, (g) pressurized thermal shock, (h) secondary side transient behavior. In addition to these, a reactor analysis code package RASP is being developed for the analysis of reload cores for PWRs and BWRs. The principal element of the safety research methodology has been the acquisition of data in as prototypic geometries and conditions as possible, coupled with best-estimate analytical descriptions for reactor transient and accident conditions.

The major elements of the fission product "source term" R&D have involved construction and/or adaptation of major experimental facilities. These include, e.g., (i) the TREAT reactor, (ii) the Marviken reactor, (iii) the containment systems experiment (CSE) vessel at Hanford and (iv) a tank at Batelle Columbus simulating BWR suppression pools. The physical and chemical character and the transport of fission products is being (or will be) measured in these facilities.

The hydrogen control R&D work has involved combustion experiments at increasingly larger scales to establish the adequacy of combustion control of the hydrogen produced in degraded core accidents. Hydrogen distribution experiments have also been performed to establish that hydrogen "pocketing" does not occur.

A set of experiments are currently being performed to determine the failure modes of sections of containment concrete slabs. Leak rates as a function of pressure will be measured.

A seismicity owners group was formed recently to pursue research to reexamine the seismicity of the plant sites in the eastern United States and to reevaluate the seismic design of those plants.

Common mode failures will be analyzed in a project which is currently being developed. The intent is to determine the role of such failures in PRA methodologies.

The safety control technology work has attempted to go beyond the safety parameter display system (SPDS) to provide the plant operators and shift supervisors real time data and analyses to support the critical decision-making process. The DASS systems being developed for BWRs and PWRs will incorporate the SPDS with the emergency procedures currently being developed by the owners groups.

The R&D work in pressurized thermal shock was an integrated effort covering thermal mixing, plant specific analyses, neutron dosimetry, integrity assessment and evaluation of remedial actions. The initial results of the work have demonstrated good mixing and large margins available.

The secondary side transient behavior R&D has involved development of a code for predicting the conditions in a steam generator following faults in the secondary side of PWRs. Experiments on a scale model freon facility were performed to support model development.

The RASP project links several codes developed by EPRI to provide the utility industry a code package for (a) performing specific plant safety analysis, (b) supporting reload licensing, and (c) performing fuel cycle management analysis.

In conclusion, the R&D pursued by EPRI has consistently attempted to provide information for resolution of safety issues in a timely manner.

EPRI RESEARCH & APPLICATION EFFORTS ON REACTOR VESSEL PRESSURIZED THERMAL SHOCK

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The potential for long term neutron embrittlement of reactor vessels has been recognized for a number of years. Recent attention has focused on the performance of vessels during overcooling transients.

An overcooling transient can occur in a variety of ways. These include primary system breaks, secondary system breaks, and excessive feedwater flow. The likelihood that a transient can pose a threat to the reactor vessel integrity is related to the simultaneous occurence of several conditions. These conditions are:

- The reactor vessel has become highly embrittled due to neutron irradiation and thus the brittle-to-ductile transition for the vessel material has shifted to higher temperatures.
- There is a crack or flaw in the vessel of sufficient size to propagate.
- There are large induced thermal stresses in the vessel beltline region that are caused by cold water cascading by this region.
- The reactor remains pressurized or is repressurized during a time of decreased temperatures.

The pressurized thermal shock issue is unique because of:

i) Technical complexity (involves many disciplines, i.e., thermal-hydraulics, neutronics, materials, and fracture mechanics).

ii) Dynamic nature--vessel conditions change with age (neutron embrittlement).

iii) Reactor vessel failure consequences are not currently part of safety analyses.

iv) Economic incentive to assure reactor vessel integrity.

v) High political and media attention.

These have combined to give a high priority to the research and development effort on the pressurized thermal shock issue.

The Electric Power Research Institute (EPRI) has been supporting research on reactor vessel integrity for a number of years.

The pressurized thermal shock issue is being addressed at EPRI using matrix management to coordinate and apply the program results. In its program, EPRI is:

- Developing the capability to predict reactor vessel wall transient temperatures.
- Developing the capability to determine vessel wall radiation exposure.
- Developing the capability to determine reactor vessel toughness.
- Developing a linked set of analytical codes so utilities or their contractors can independently assess the effects of reactor vessel thermal shock.
- Investigating ways to extend reactor vessel life.
- Transferring these technologies to help utilities build their in-house capabilities.

Reactor vessel thermal shock is not a new concern. With a growing number of plants approaching their mid-lives, it is a concern that must be understood and dealt with. The technical aspects of the pressurized thermal shock (PTS) issue are understood; i.e., the R&D as well as application programs to address the various facets are either underway or complete. Industry is continuing to cooperate with the NRC to achieve a workable and flexible PTS rule so that the screening criteria limits are not turned into operating limits. If a plant approaches the screening limits, it is important to be able to make plant specific calculations to determine if remedial actions are necessary. The dimensions of the thermal shock problem and ways of dealing with it are understood. From what we now see, it is a moderate concern. It is unlikely that any plant should need to curtail operation because of the risk of thermal shock. However, some plants may need to mitigate the effects of a possible thermal shock event by applying the results of PTS R&D to their plant operation.

EPRI Sponsored TMI-2 Research Program

J.T.A. Roberts

The Joint Information and Examination Program at TMI-2 was officially started in March 1980 with the signing of the GEND Coordination Agreement among General Public Utilities Service Corporation, EPRI, NRC, and DOE. Planning that started in 1979 continued during 1980 to develop an R&D program which would respond to both the TMI-2 recovery effort and the general needs of the LWR nuclear industry. Experts from government laboratories, utilities, vendors, and academia had participated in a number of expert planning groups that produced recommendations on R&D for the Joint Program.

DOE has initiated projects on instrumentation and electrical equipment survival, fission product transport and deposition, and containment decontamination and associated waste processing and disposal. Planning for core removal and fuel examination is under way.

EPRI accepted responsibility for the planning and the primary R&D funding to support reactor coolant-system and auxiliary-system decontamination and dose reduction, mechanical component survivability, and primary-system pressure-boundary characterization.

A special subprogram, "TMI-2 Information and Examination Subprogram", was formed in EPRI's Nuclear Power Division in 1982 to contain all TMI-2 activities. This subprogram's budget is planned at $18.0 million over the period 1982 to 1987. DOE plan expenditures of $27 million in FY 1983 and $37 million in FY 1984.

A matrix management system has been set forth to implement the TMI-2 I&E subprogram, drawing upon staff from all four Nuclear Power Departments: Systems and Materials, Safety and Analysis, Engineering and Operations, and NSAC. To facilitate liaison with the GPU site staff, DOE's Technical Integration Office (TIO), and the various on-site contractors, an EPRI site office was established in November 1981, using space provided by DOE, and staffed by contractor personnel. In addition to project management, significant staff contributions are also being made to TMI-2 Technical Evaluation Groups (TEGs) formed by DOE.

The scope of research being conducted by EPRI in support of the TMI-2 Recovery is illustrated by the following examples of recent accomplishments.

Reactor Coolant System Decontamination and Dose Reduction

- Evaluation of Chemical Decontamination Methods.

Early in 1982, a thorough review of historical post-accident chemical decontamination operations on reactors was conducted. The operations were documented in detail, with particular emphasis on the effectiveness of the various chemicals used and on problems encountered.

Later in the year, some fourteen potentially useful chemical decontamination processes were evaluated, including four commercially-developed methods. All were compared on the basis of effectiveness and impact, using a specially developed set of comparison criteria. These processes are now being evaluated in detailed laboratory tests with TMI-2 materials to determine effectiveness of decontamination (DF's), process variables, waste production etc., prior to on-site demonstrations in 1985.

- Evaluation of Nonchemical Decontamination Methods.

In 1982, seventeen different nonchemical decontamination methods were identified and described by prior applications and specific advantages or disadvantages. All were rated for their suitability for use in decontaminating TMI-2 reactor cooling systems. In addition a proprietary decontamination method using abrasive grit in high-pressure water jets was described separately and evaluated for applicability to TMI-2. During 1983, the leading candidate approaches (pigs and brush/hone devices) have been subjected to vigorous testing on archive and actual materials removed from TMI-2. A flushing/decontamination system for the upper plenum and underhead reactor vessel surfaces which involves high pressure water jetting, was also developed, built and tested in a mockup.

Fission Product/Fuel Transport and Deposition

- o Characterization of TMI-2 Contamination.

Using available computer codes, theoretical analysis was made of the generation, release, escape, and dispersion of core debris during and after the TMI-2 accident.

Also, during 1982 and 1983, scrape samples and gamma spectroscopy were used to comprehensively map the radioactivity in the auxiliary buildings. Spectroscopic analyses also revealed the types and quantity of radioisotopes that were deposited on parts of a control rod lead screw removed from the reactor vessel's upper plenum.

Mechanical Component Examination and Information

- o TMI-2 Polar Crane Examination and Requalification

A crane specialist was funded to work with GPUN/Bechtel to prepare the plans and procedures for examination, decontamination, refurbishment, and requalification of the reactor building polar crane. Gear boxes, lubricants, shafts and bearings, trolleys, etc., were found to be affected very little by the accident, although outside surfaces were somewhat corroded and contaminated. Brakes had to be replaced. Exposed elastomers and plastics were demaged by the hydrogen burn and will have to be replaced. A lift test is still pending, but the crane is functional.

Primary System Pressure Boundary Characterization

- o Control Rod Drive Leadscrew Examination

Metallurgical-chemical-physical examination of a small section of the 8H control rod drive leadscrew revealed that the majority of the radioactivity, i.e. 90% of ^{147}Cs and 15% of ^{90}Sr, is trapped in a tightly adherent, seemingly impermeable layer (10-90 μm) thick on the surface. However, there is no evidence of intergranular attack of the base metal by cesium.

This and other work ongoing both on and off the TMI-2 site will be described in the paper and plans for future activities will be presented.

RECENT SOURCE TERM EXPERIMENTATION

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Since it is now an objective of the reactor safety community to attempt to do "realistic" accident evaluations as contrasted to the past "conservative" evaluations, it is important to undertake experimental programs in key areas to generate needed data. This paper will cover, in a very brief fashion, some of the EPRI programs which are addressing this task. At this point only a limited amount of data is available. Many of the programs are paced to yield their data in 1984.

In order to guide the experimental program, estimates have been made of the RCS thermal hydraulic conditions for key accident sequences. These estimates pertain to the period from the predicted start of core uncovery to the estimated time of core slumping. The metal and gas stream temperatures as a function of time for major structures and volume regions in the primary coolant system have been calculated. The gas compositions and flow rates were also calculated. These results are available in EPRI-NP-3120.

Results will be partially reported for the EPRI sponsored consequence analysis for the Surry reactor. Early results indicate source terms of about $10^{-5}\%$ for cesium and iodine and $10^{-3}\%$ for tellurium for TMLB'. In the case of the "V" sequence or interfacing LOCA, the source terms are not as low.

EPRI is also sponsoring work on the identification of the fission product compounds released from fully irradiated fuel by using a mass spectroscopic technique. These experiments are carried out remotely in a cave.

Another program directed at characterization of fission products released from preirradiated fuel is being carried out in Argonne's TREAT reactor. The status of this work will be discussed.

EPRI and NRC are working with eight countries in an approximately full scale series of tests on aerosol transport in the primary coolant circuit of the Marviken reactor in Sweden. The first test has been carried out.

Plans are also being developed to do a large scale test on the possibility of aerosol retention in the primary coolant system in a simulation of the "V" sequence and these will be described.

Another particularly important experimental program area is the investigation of the removal of fission product aerosols by scrubbing. This work is still underway, but tentative conclusions are that significant scrubbing of aerosols of even submicron particles is observed in pools even when the water is near the boiling point. It appears that the higher the steam/non-condensible ratio of the carrier gas the higher the decontamination factor. The parameters investigated include mass flow, pool temperature (20 to 100°C), steam fraction (0 to .95), submergence (.3 to 1.65 meters and eventually going to 4.8 meters) and aerosol type (CsI, TeO_2, and SnO_2).

In addition to the programs mentioned above, other EPRI programs will be mentioned.

RECENT DEVELOPMENTS IN ULTRASONIC PIPE INSPECTION

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Mohamad Behravesh[2]

James R. Quinn[1]

INTRODUCTION

The detection, identification, and sizing of intergranular stress corrosion cracks in stainless steel piping used in BWR's has received a great amount of attention during the last year. EPRI, on behalf of the Electric Utility Industry, has taken an active role in providing assistance in the in-service inspection area. A summary of the activities in the areas of Performance Capability Demonstrations, Training, IGSCC Sizing Round Robin, Equipment Evaluation, and Off-Line Data Analysis is given in the sections that follow.

Performance Capability Demonstrations for IGSCC Detection

After IGSCC was discovered and confirmed in the large diameter recirculation piping of the Nine Mile Point-1 plant, the NRC issued I&E Bulletin 82-03 Revision 1. This bulletin required that licensees "provide a reasonable level of assurance that inspections which were currently being performed or scheduled are sufficient to detect cracking in BWR thick wall recirculation pipe welds". This assurance consisted of representatives of the groups responsible for inspection of nine BWR's to demonstrate on blind samples that they could detect IGSCC. In order to provide a mechanism whereby these utilities could demonstrate the required capabilities, the EPRI NDE Center staff members, in cooperation with EPRI personnel, obtained and documented representative samples removed from the Nine Mile Point plant, arranged for and scheduled the testing exercises at the Battelle Columbus Laboratory hot cell facility, assisted the NRC in the organization and formatting of the activity, and provided general support and special training to the utilities to aid in their preparations for the qualification exercise. The NRC monitors graded each group's performance as pass or fail.

[1] EPRI

[2] EPRI NDE Center Operated by J.A. Jones Applied Research Co.

On March 3, 1983, I&E Bulletin 83-02 was issued to cover the remaining plants. By this time, a facility to handle radioactive material was added to the NDE Center, and the required performance capability demonstrations continue to be conducted there with NRC oversight.

Training

The experience gained from meeting the requirements of I&E Bulletin 82-03 indicated that training can be very beneficial. The EPRI NDE Center, working with a group of utility representatives, developed a one week training course for detection of IGSCC. It was first offered in late June, and is being offered on a twice-a-month schedule since August 1. The practical final examination is patterned after the blind performance capability demonstration required as part of I&E 83-02. A similar course is being developed for IGSCC depth sizing.

IGSCC Sizing Round Robin

The performance capability demonstrations have increased the IGSCC detection and length measurements to adequate levels. Concern by the utilities about IGSCC depth measurements lead to EPRI initiating a Round Robin to assess overall industry practice. The effort conducted by the EPRI NDE Center was completed in two months with the results documented and reported to the industry and the NRC on August 4, 1983. The results indicated that it is possible to size cracks with ultrasonic methods. However, the overall level of performance was less than that desired by the industry and improvement was warranted. As a result, three activities are underway. They are development of a training program for sizing, followed by implementation of a qualification program, and acceleration of evaluation and qualification of advanced automated systems.

Equipment Evaluation

A new class of inspection equipment is being developed that permits the operator to train it to detect and classify up to three classes of signals. The evaluation of one such device, the ALN 4060, has performed very well in the manual mode. It is now being evaluated in an automated mode of operation.

An ultrasonic data acquisition system that consists of a portable computer based electronics system to control a scanner and record data, (ALN 4000), a mechanical transducer scanner (AMAPS), a booted search unit and appropriate software (MARK III) has been assembled and is undergoing evaluation. It has successfully acquired data from a pipe joint in the Browns Ferry-1 plant. In addition, it was used to provide complete documentation on the samples used in the IGSCC sizing Round Robin. During this exercise, over 22,000 data points (ultrasonic signals plus position coordinates) were recorded on magnetic tape in about 200 hours of operation. In addition, after only a simple change of transducer and housing, the same system was used to acquire data on the round robin samples for later SAFT processing by PNL under their NRC sponsored program.

Off-Line Data Analysis

The successful development and use of the ultrasonic data acquisition system now permits recording data in the field on magnetic tape and processing in an off-site computer. This process was successfully demonstrated for IGSCC sizing during the round robin. Optimized software for both detection and sizing are now near completion of development. Their effectiveness will be analyzed in the off-line mode. When performance is judged adequate and reliable, the analysis routines will be incorporated into the same hardware system used to perform data acquisition. At this point, a complete automated field inspection system will be available.

RECENT DEVELOPMENTS IN BWR PIPE CRACKING

By

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Since the introduction of the first commercial boiling water reactor, incidents of intergranular stress corrosion cracking in the weld heat-affected zones of austenitic stainless steel piping have occurred. Early incidents of pipe cracking were confined to small diameter lines of 100 mm (4 inch) and less. In time larger diameter lines were also subject to the phenomenon, and in 1982 cracking was reported for the first time in 710 mm (28 inch) 316 stainless steel lines in Nine Mile Point. Subsequently, cracking was detected in the large diameter 304 stainless steel lines in Monticello, Hatch Unit 1, Browns Ferry Unit 1 and Dresden Unit 2.

In spite of the overall 380 incidents, there has never been a reported case of pipe severance. Pipe cracking incidents do not constitute a safety issue and an industry and Nuclear Regulatory Commission position is the "leak-before-break" concept. A leak will occur before the tough ductile austenitic stainless steel pipe will experience a severance. However, pipe cracking has created a serious problem of plant availability and economics to the utility industry. In addition, radiation exposure of skilled crafts involved in inspection and repair operations is of concern to the utilities.

In response to this utility problem, a joint Boiling Water Reactor Owners Group and Electric Power Research Institute research and development program was organized in 1979 to provide an engineering solution to the problem. The four-year program (1980-1983) budgeted for $43 million is nearing completion. Various reports of progress in resolution of the pipe cracking problem have been published. Therefore, the purpose of this paper is to discuss recent pipe cracking incidents and their resolution.

During the 1982 period several new and significant pipe cracking incidents were reported. The first incident occurred at the Niagara Mohawk Nine Mile Point Unit 1. This unit, a BWR model 2 with five 710 mm (28 inch) 316 stainless steel recirculation loops and furnace sensitized safe ends developed leaks in two of the safe ends after 12 years of operation. Subsequent examination of the welds in the recirculation loop disclosed indications in many of the joints and a number were confirmed as intergranular stress corrosion cracking. The significance of Nine Mile Point was the extent of cracking in the large 710 mm diameter pipes and the results of a study that showed the cracking to be generic in nature. Niagara Mohawk decided to replace all of the safe ends and recirculation piping with 316 nuclear grade stainless steel.

In Monticello, a BWR model that has been in operation for 11 years, indications were detected in a 304 SS 560 mm (22 inch) end cap-to-ringheader or manifold and four 305 mm (12 inch) discharge risers. The lines were repaired with a weld overlay technique that was accepted by the NRC.

Inspection of Hatch Unit 1, a BWR model that has been in operation for 7 years, revealed indications in four 304 SS 560 mm (22 inch) end cap-to-ringheaders and in one sweep-o-let in the ringheaders. A weld overlay repair was used for the end caps and continued operation with the existing indications in the sweep-o-let was proposed. This is the first reported indication in the sweep-o-let of the recirculation piping system in a domestic plant. Browns Ferry Unit 1 (BWR) in operation since 1977 also found indications in two sweep-o-lets in the 304 SS 560 mm (22 inch) ringheader. Continued operation without any repair was proposed and accepted by the NRC.

In Dresden Unit 2, indications were detected in a 710 mm (28 inch) furnace sensitized safe end, in four 305 mm (12 inch) discharge risers and a 560 mm (22 inch) end cap-to-ringheader. No repair will be performed on the safe end and some of the risers. The others will be repaired using a weld overlay. This plant is being used for continued long term hydrogen water chemistry (HWC) experiments, therefore the unrepaired indications will provide a measure of the ability of the HWC to arrest crack propagation.

Pipe cracking incidents in austenitic stainless steel piping in BWRs continue, and for the older plants significant cases of cracking are appearing in the larger diameter lines 560 mm to 710 mm (22 to 28 inch). Temporary repairs using a weld overlay and continued operation of flawed pipes with or without H_2 water chemistry has been accepted by the NRC. This has resulted in substantial reduction in outages for normal repairs.

Thermal-Hydraulic Analysis of Steam Generators - The ATHOS Code

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To address the needs of the nuclear power industry for the design and performance analysis of steam generators, a computer program capable of three-dimensional thermal-hydraulic analysis of different types of steam generators has been developed and is being used for various applications at the Electric Power Research Institute (1). This computer program, called ATHOS (Analysis of the Thermal Hydraulics of Steam Generators), is capable of simulating the geometry and internal structure including tube bundle, tube support plates, baffles, etc., in considerable detail. The thermal-hydraulics simulation is based on a description of the two-phase flow by means of a set of partial differential equations representing thermal equilibrium flow with slip between the two phases. An implicit numerical method (2) is used to solve the set of finite difference equations derived from the governing partial difference equations using a set of constitutive models for slip, friction, and heat transfer. The results of this solution include detailed distributions of the velocities of the two phases in R-Θ-Z coordinates, void fractions, and enthalpies.

A typical nuclear steam generator of the U-tube type is shown in Figure 1. Primary water at high pressure (about 16 MPa) and temperature (600°K) from the reactor flows through the U-tubes. It rises in the left half of the steam generator referred to as the hot side and flows down through the right half called the cold side. The secondary fluid, a mixture of water and steam, flows in the shell of the steam generator. Most of the feedwater (90%) is let in at the economizer section (lower part of the cold side) of the generator and the rest is let in at the top of the downcomer section. A steam separator located at the top separates the steam from water which flows down through the downcomer and joins the incoming stream of feedwater. There are several structural components in the shell such as the full and partial tube support plates, distribution plates, baffles, divider plates, etc. as indicated in Figure 1.

For computational purposes, the flow field considered corresponds to one-half of the steam generator extending from the tube sheet at the bottom to the separator deck at the top. This calculation domain which includes structural elements such as the tube bundle, support plates, etc., will be considered as a porous medium, that is, the solid phase will be considered as a distributed resistance in the secondary side.

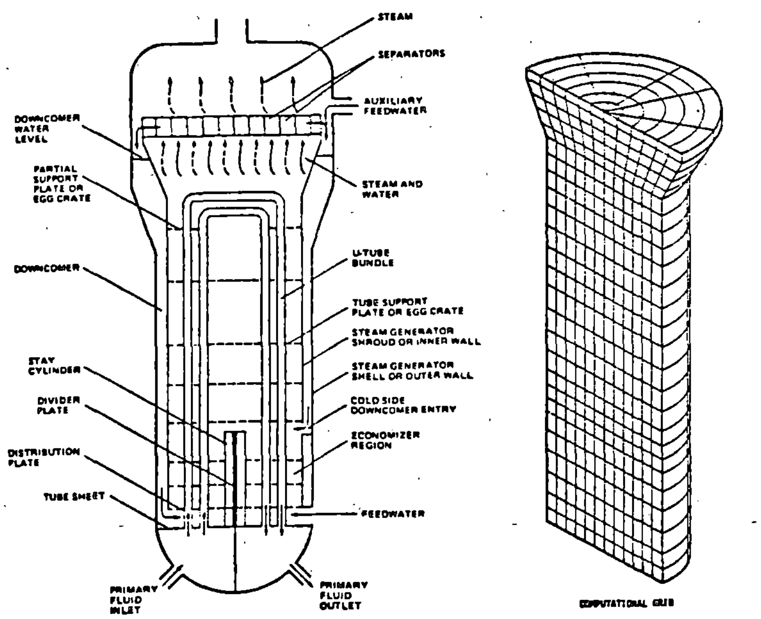

Figure 1: Typical U-tube Steam Generator and Computational Grid

Other assumptions in the mathematical formulation of the two-phase flow on the secondary side are:

(i) The liquid and vapor phases are in thermal equilibrium,

(ii) Slip is allowed between the two phases by means of an algebraic model,

(iii) The saturation properties of liquid and vapor phases are computed as functions of a single pressure,

(iv) All other physical properties such as viscosity, thermal conductivity, specific heat, etc., of the liquid and vapor are assumed constant,

(v) Diffusion and turbulence are negligible, so are interphase momentum transfer due to mass transfer between phases,

(vi) Dissipation and spatial component of the pressure work terms in the energy equation are negligible.

The current version of the code has been checked out for three different UTSG configurations. The code qualification and verification performed to date include numerical consistency studies, physical parameter sensitivity studies, comparisons with small and medium scale experiments, as well as some large scale plant tests. Verification against experiments and plant test data is continuing. A typical steady state calculation of a U-tube steam generator (UTSG) with 1000 control cells requires about 80 cp seconds on a CDC7600 computer and an operational transient requires about 30 cp seconds per timestep.

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Large Scale Molten Core/Magnesia Interaction Test*

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I. Objective of the Test

The main objective of the test was to study the core debris interactions with magnesia bricks. In addition the test also provided experimental verification of recent advances made in melting technology for producing several hundred kilograms of core melts.

II. Physical Description of the Test

The test was carried out at the Large-Scale Melt Facility (LMF) at Sandia National Laboratories [1]. The facility consisted of an induction furnace on a platform for melt preparation and test chamber below for the interaction test. The inductive power was coupled to a graphite susceptor which in turn heated the melt crucible and the melt charge inside. Explosively driven projectiles were used to tap the melt from the furnace into the test chamber. Because of material compatibility considerations, the melt crucible was a multi-walled assembly with an outer vessel made of tantalum-10% tungsten and an inner liner of tungsten. The melt charge was a mixture of UO_2 70% by weight and stabilized ZrO_2 30% by weight. The charge was manufactured by hot pressing of powder of UO_2 and ZrO_2; the final charge was approximately 85% of theoretical density. A total charge of 230 kg was used in the test.

The magnesia crucible was made of Harklase MgO bricks. The bottom was made of three layers of flat bricks with a total thickness of 18.4 cm. The sidewall was made of two layers of arch bricks with a total thickness of 22.9 cm. The cylindrical cavity formed had an inside diameter of 35.6 cm and was 66.0 cm deep.

III. Results

a) Melt Preparation

The heating and melting phase of the experiment lasted approximately twenty hours. The thermal response of the melt charge was found to be well predicted by pretest calculations [2]. At the time of tapping, the melt was at 2873 K (2600°C). Approximately four minutes were required to drain all the melt into the magnesia interaction crucible.

b) Core Melt Magnesia Interaction

(1) Thermal Response of the Magnesia Crucible and Upward Heat Flux

The thermal response of the crucible was measured by embedded thermocouples from 0.32 to 15.24 cm. below the surface. In general

*This work supported by the U.S. Nuclear Regulatory Commission and performed at Sandia Nat'l Labs. which is operated for the U.S. Dept of Energy under contract number DE-AC04-76DP00789.

the central portion of the bottom of the crucible was heated more than the edge of the bottom and the sidewall was heated less than the bottom. The maximum temperature measured 3 mm beneath the bottom of the crucible was 700°C. The upward heat flux as measured 33 cm above the pool was about 40 W/cm^2.

(2) Mechanical Response of the Crucible

The brick structure remained generally intact. Hairline cracks were observed in the bricks above the melt pool but none was large enough to threaten the structure integrity of the crucible. No deep penetration of the melt into the cracks was observed. Thick film crack detectors indicated that bricks in the melt pool might have suffered diagonal cracks at the corners.

(3) Gas and Aerosol Generation

Gas generation was minimum during the test. No hydrogen generation was detected as a result of the possible reaction of the melt with water in the magnesia bricks.

Aerosol samples were taken during the test using filter samplers, cascade impactors, cascade cyclones and various surface deposition samplers. The aerosol size distributions were found to be trimodal in the early stages of the pour and single mode, about 1 μm, at later stages of the interaction. The aerosol concentration reached a maximum of 1.95 g/m^3 about two minutes into the tapping of the melt. Total aerosol generated during the test was found to be less than 103 grams. Extrapolating to the reactor case using the ratio of melt mass (80×10^3 kg/200 kg = 400) suggests an aerosolized mass of 41 kg for a gravity driven pour. This is small compared to assumed releases of in-vessel aerosol and to aerosol generated by melt concrete interaction.

IV. Conclusion

Detailed analyses of the interaction are still being conducted. Preliminary results seem to indicate the magnesia bricks to be a suitable candidate for core ladle construction.[3,4]

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High Pressure Melt Ejection*

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Probalistic risk assessments have received much interest since Three Mile Island for evaluating power plant performance during severe accidents (1). They have used and extended the analysis techniques employed in the Reactor Safety Study (RSS) (2), but recognize differences in the core meltdown processes, vessel failure modes, and ex-vessel debris behavior.

The analyses of the expulsion of core debris from the reactor pressure vessel depend on the pressure in the primary system at the time of failure. The behavior following a large break LOCA (< 1.3 MPa) parallels that of the RSS. In the small-break LOCA and transient sequences the pressure may be as high as 15.2 MPa when vessel failure occurs. In these latter two cases, the predominant vessel failure mode is predicted to occur at one or more of the instrument tube penetrations in the lower head.

The analyses postulate that the high pressure ejection of molten debris and subsequent vessel blowdown through the breach in the reactor pressure vessel remove debris from the cavity region and disperse it throughout the containment building, where it is quenched. As a result of the dispersal process, the core debris/concrete interactions would be greatly attenuated and the quantities of non-condensable gases and aerosols reduced.

These analyses do not have the benefit of experimental data on the high pressure ejection of molten material and subsequent interactions within a concrete cavity. The objective of the High Pressure Melt Streaming (HIPS) program at Sandia National Laboratories is to experimentally investigate the ex-vessel debris behavior during high pressure melt ejection.

The HIPS test program uses 1/10 and 1/20th linear scaled models of the Zion cavity to investigate debris dispersal. The models are constructed of prototypic limestone-common sand concrete. A scaling analysis has shown that the criteria for the postulated debris dispersal mechanisms are met or exceeded in the experiments. The melt is developed in a pressure vessel using an iron/alumina thermite with either nitrogen or carbon dioxide as the pressurizing gas.

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

Initial unconfined free-jet experiments performed in support of the HIPS program have shown that the melt-stream does not behave as a coherent, stable jet and that a large aerosol source term accompanies the emerging jet.

A stable jet geometry is necessary to establish two of the debris removal mechanisms proposed (1). Flash x-ray photographs of the melt stream have shown large variations in behavior. When the melt is pressurized with nitrogen, a divergent, obviously two-phase jet is formed. A compact stream with Helmholtz surface instabilities is observed in tests pressurized by carbon dioxide. These characteristics have been interpreted as the effervescence of gas dissolved into the melt prior to ejection. Nitrogen and carbon dioxide are representative of upper and lower bounds, respectively, of the solubility of steam and hydrogen pressurizing molten core material under accident conditions (3).

High speed photography of the melt generator events shows that the emerging melt stream is preceeded by a luminous vapor cloud. As the cloud expands, it darkens as the vapors condense. Eventually, the aerosol cloud totally obscures the test apparatus. The aerosols are collected with inertial cascade impactors and characterized by scanning electron microscopy.

Aerosol size distribution data have been interpreted as multimodal with modes at 0.5, 5 and greater than 10 micrometers aerodynamic equivalent diameter. The smallest mode (0.5 micrometer) appears to consist of agglomerates of smaller 0.05 to 0.1 micrometer diameter particles. These particles appear to be formed by the condensation of vaporized melt species. Elemental analysis of the smallest mode shows that most of the material is iron, consistent with the higher vapor pressure of iron compared to aluminum oxide.

The large size modes (5 & > 10 micrometers) are attributed to mechanical breakup of the melt. Hydrodynamic formation mechanisms involve the breakup of ligaments and droplets until surface tension forces balance inertial forces (4). These processes are predicted to yield particles 10 micrometers or larger in size. Effervescence of dissolved gas may be responsible for aerosols due to bursting of bubbles. Bubble bursting in liquid metals is known to produce aerosols in the 1 to 10 micrometer size range (5).

Jet aerosols in an accident are significant in that they may provide a fission product source term distinct and separate from those source terms identified in the RSS (2). Aerosols produced during reactor accidents by the ejection of molten core debris are assumed to contain fission products. These aerosols represent a fission product source term potentially as great as the in-vessel aerosol source term (6). The hot particles may also be chemically reactive and will vigorously burn in containment (7) and act as a distributed ignition source for hydrogen.

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SAND83-1851A

CORE MELT/COOLANT INTERACTIONS: EXPERIMENTS*

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Research on vapor explosions predates the nuclear power industry, and probably dates back to the first attempts to quench molten metals in water more than five thousand years ago. Many experiments on vapor explosions have been conducted throughout the world over the past several decades. The experimental results have spawned a large variety of theories and models, some with lifetimes not much longer than the mean time between key experimental investigations. There is currently no unified theory capable of explaining all the core melt/coolant interaction (CMCI) phenomena that have been observed for various liquid pairs under a variety of initial and boundary conditions.

Experiments at Sandia have been primarily aimed at investigating CMCI phenomena associated with hypothetical accidents in light water reactors. These phenomena include steam generation (both explosive and non-explosive), hydrogen generation, and the characteristics of the resulting fuel debris. Table I summarizes some features of these experimental investigations.

Table I. Experimental Investigations

Methods:	Fuel Mass:	Contact Mode:
Arc Melter	10-15mg	flooding
Single Droplets	few mg	droplet falling through water
Field Tests	1-20 kg	pours and drops
FITS and EXO-FITS	2-20 kg	pours and drops
REFLOOD	10-20 kg	flooding

These experimental facilities span a range of about 400,000:1 in fuel mass, but can still be more than three orders of magnitude below possible reactor situations. The experiments have indicated that all CMCI phenomena depend strongly on: molten fuel mass and composition, water/fuel mass ratio,

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

water temperature, ambient pressure and trigger strength (when artificially triggered). As long as the fuel is molten when it contacts the water, the fuel temperature does not appear to play a major role. Water composition also appears to have a negligible effect for prototypic liquid pairs. The experimental matrix addressing water depth, interaction geometry (including structures), alternate contact modes and fuel entry velocity is currently too sparse even to estimate the qualitative effects of these initial and boundary conditions.

Of the variables known to have important effects on the characteristics of CMCI's, several have been shown to depend strongly on scale (masses or volumes of liquids involved), and in some cases, on details of experiments hitherto considered unimportant. This paper will summarize the most important experimental observations from past Sandia and other related experiments. Major emphasis will be placed on recent intermediate-scale experiments which have addressed the effects of water subcooling, melt composition (pure oxidic melts and mixed melt/oxide melts), and alternate contact modes. Measurements of hydrogen generation have been made for explosive and non-explosive CMCI's. Small-scale single-droplet experiments have investigated the explosibility of zirconia, urania, and mixed urania/zirconia compositions.

We believe that the current understanding of CMCI's is insufficient to reliably support the extrapolation of any model predictions to reactor scale. Where such predictions have been made, we expect that large uncertainties are involved. In particular, we do not believe that the current data conclusively support any of the following statements: Steam explosions at high ambient pressure are impossible; fragmentation of large fractions of a molten core are impossible; failure of the vessel or containment due to a CMCI is impossible.

CMCI modelling is discussed in another paper in this session.

SAND83-1852A

CORE MELT/COOLANT INTERACTIONS: MODELLING*

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In 1975, the Reactor Safety Study (WASH-1400) [1] estimated the probability of containment failure via missile generation caused by a steam explosion to be 0.01. Since that time, our research work [2-20] has indicated that core melt/coolant interactions (CMCI) are complex processes which can affect containment response in a variety of ways: fuel-coolant mixing and rapid steam and hydrogen generation, fuel debris formation, dynamic pressures from a steam explosion, and possibly generation of missiles. In this summary we focus on recent modelling and analysis which we have used to analyze the experiments and apply the results to accident concerns.

Phenomenological Models for Mixing and the Explosion

To better understand CMCIs, we are investigating several phases: mixing, triggering, propagation and expansion. Our approach is to identify the key physical phenomena and model them. One of our objectives is to be able to analyze portions of the single-droplet small-scale tests [6,16] and the FITS experiments [3,14] which provide insight to CMCI phenomena and provide the necessary input for empirical two-dimensional calculations using the CSQ hydrodynamics code. The second is to use the models developed for analysis of severe accident phenomena in support of PRA needs. One of our first results from this effort has been the development of a simple dynamic model, WISCI (WISconsin Coolant Interactions), for core melt-coolant interactions including mixing and explosions [18,19,20]. The model considers a fuel melt pour into a water pool. It calculates the fuel/water mixing, droplet diameter and temperature, and steam and hydrogen generation rates. At a preselected moment a steam explosion is triggered and the fuel is finely fragmented, exchanging energy with the coolant in the mixture. The explosion will eject the surrounding water as a slug and disperse the steam produced. To date we have applied the model to the FITS experiments to look at only fuel-coolant mixing. These WISCI results will be discussed in the presentation.

Two-Dimensional Calculations of CMCI Experiments

The 2-D, Eulerian, multimaterial, finite-difference wavecode CSQII has been used to investigate the transformation of thermal energy in the melt into kinetic energy in the water, steam and air. The initial shape and mass of the water and steam in the calculation approximated those of the FITS 9B experiment when the mixing region contacted the base and the steam explosion occurred. The propagation of the thermal energy transfer front through the steam was simulated. Parameters based on experimental measurements, (a propagation speed of 300 m/s, a front width of 0.4 m and an energy release rate of 70 kJ/kg) gave good agreement between measured and calculated pressure records in the water and on the sides of the vessel. The calculated water

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

velocity of 115 m/s was in reasonable agreement with the measured velocity of 95 m/s. The simulation of the FITS 9B experiment was continued until the maximum kinetic energy in the steam, liquid water and air was calculated, see Table I. The maximum kinetic energy in the system was less than the sum of the maximums of the water, steam, and air because the maximums occurred at different times.

Another calculation was performed where the lucite walls of the vessel that initially held the water were replaced with rigid walls. This prevented the water from expanding radially. The calculated motion of the steam and air was very similar to that of the first calculation. The maximum kinetic energy in the steam and air were similar to those of the first calculation. The maximum kinetic energy in the liquid water was substantially lower than that of the first, Table I.

Another calculation was performed where the steam region was completely surrounded by liquid water. This was done by adding a layer of water above the steam. The layer was as thick as the layer of liquid water on the sides. The maximum kinetic energy of the steam and air was lower than in the original calculation but the maximum kinetic energy of the liquid water was larger.

These calculations imply that a constrained geometry, similar to the reactor vessel may result in lower conversion ratios than unconfined experiments. This surprising result is being actively investigated.

Table I. Peak Kinetic Energy of Calculations

	FITS 9B	RIGID WALLS	WATER SURROUNDED
Maximum Kinetic Energy in liquid water (MJ)	0.85	0.32	1.1
Maximum Kinetic Energy in steam (MJ)	0.65	0.57	0.33
Maximum Kinetic Energy in air (MJ)	0.14	0.11	0.03
Maximum Kinetic Energy in System (MJ)	1.50	0.86	1.38

Application of Results to Severe Accident Concerns

One area of historical interest (e.g. WASH-1400) has been the probability of direct containment failure from a steam explosion due to missile generation. Past work focused on a best estimate of this probability using deterministic analyses [12,13] and a Monte Carlo statistical approach [15]. Our recent effort has been to attempt to quantify the uncertainty of this best estimate [17]. The results of this work indicate that the uncertainty range is quite large, and is dominated by our uncertainty of how much fuel and coolant can mix at large scale and what the melt thermal-to-mechanical energy conversion ratio would be as the scale increases.

In a different area we are using WISCI to analyze reactor cavity phenomena during specific severe accident sequences. We are working with the Accident Source Term Program Office (ASTPO) in a Containment Loads Working Group to determine the steam and hydrogen generation rate when the fuel melt is discharged into the reactor cavity in the presence of water. The purpose of this work is to determine if the "steam spike" phenomenon is important for certain dominant accident sequences and classes of reactors.

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THE LARGE SCALE EXPERIMENTAL SIMULATION OF LWR DEBRIS BED COOLABILITY

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Current debris bed coolability data are limited with respect to their applicability to LWR accident analysis situations in the following important aspects:

(a) Bed Size and Makeup. The usual volumetric heating methods (microwave, induction) restrict the size and make up of such beds. Available data extend to beds with a maximum diameter of $\sim$10 cm and a maximum depth of $\sim$20 cm while in typical LWR study applications one is interested in beds several feet deep and an extended lateral dimension. Thus multidimensional effects cannot be studied experimentally in a controlled fashion while at large particle sizes (cm-scale and above) undesirable wall effects may affect the experimental data. Further, in induction heating one is limited to metalic particulates, and the vast majority of data were in fact obtained with spherical metal particles. This is in contrast to the irregular particles expected from the quenching of molten corium in water.

(b) Pressure Effects. In many LWR study applications the containment pressure may approach the 100 psi range, yet no data are available to support the increased coolability limits predicted by all available correlations and models.

(c) Instrumentation. Available data are based on very limited diagnostics of the dryout phenomenon and no data are available for extended operations in a partially dried-out bed.

A new concept of an experimental facility was devised with the aim of overcoming the above limitations. Insulated resistance heaters are coiled into a spiral shape and threaded through 1/2-inch (or any other size desired) aluminum spheres. One such heater element resembles a slice of a packed bed of a thickness equal to the characteristic dimensions of the particles. By stacking these elements together a volumetrically heated bed is formed. By incorporating layers of any desired particulate (i.e., stone, gravel, etc.) between the heater elements a quasi-uniform volumetric heating with an independent control on the porosity, particle shape and size, etc. may also be obtained. Since direct electrical heating is utilized the size of such a bed is only limited by the available power. Furthermore, the bed can be contained within a pressure vessel, to allow the study of pressure effects, and it is very convenient for incorporating extensive instrumentation (i.e., the bed is free of electromagnetic fields).

The construction and the first two series of experiments in such a facility will be reported. In its present form the bed is 21.6 cm in diameter and 101.6 cm deep. It is heated by forty heater elements individually controlled for a total power capability of 100 kw. The pressure vessel is designed for 100 psig. Experiments may be conducted from high vacuum up to the maximum design pressure. Control is achieved by adjusting coolant flow to a reflux condenser coil and/or a steam relief valve. Two hundred thermocouples are positioned throughout the bed and can be scanned at a rate of one thermocouple per millisecond by a PDP-11 minicomputer. The temperature pattern is displayed on an oscilloscope screen that helps select the thermocouples whose outputs are to be stored in computer memory for later analysis. Data on the inception of dryout and the dryout pattern for powers well exceeding this inception limit will be reported and compared to existing data as well as available models and correlations.

OVERVIEW OF THE HDR-CONTAINMENT TESTS

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This presentation gives an overview about the experimental results for various integral and local subcompartment quantities following 4 different steam blowdowns (V42 through V45) and 2 different water blowdowns (V21.1 and V21.3) into the large-scale HDR containment. Furthermore, a brief and preliminary summary of the predictive qualities of blind pre-test predictions by various institutions with different containment computer codes will be provided and first explanations for obvious discrepancies between measured data and calculational results given (the follow-up presentation by Prof. Almenas discusses the discrepancies in the heat transfer models as one of the contributors to observed differences in greater detail). Although data were sampled up to 40 hours into the transients, the presentation will only emphazie the results for the short-term (0 - t - 2 s) and intermediate-term (2 s - t - 300 s) transient behaviors of the subcompartments.

The HDR containment with a total volume of 11.300 m^3 is enclosed within a cylindrical steel shell with a height of 60 m and a diameter of 20 m. Concrete walls subdivide this volume into 62 subcompartments, some of which contain a vast amount of metal surfaces (piping, equipment etc.). The main characteristics of the HDR-containment can be summarized as follows:

a) Very large ratio (2.6) of surface areas (total of 30.000 m^2) to volume
b) Vastly differing geometries of vent flow openings between subcompartments.

The breakroom location and its neighbors were chosen such and fixed for all experiments performed that unique break near phenomena could be closely examined. The test instrumentation consists of about 230 thermalfluiddynamic (pressure, temperature) and 80 structural dynamic sensors. In addition, 11 special so-called heat-transfer blocks of different materials and constructions as well as 3 steam/air concentration measurement devices using infrared-measurement techniques in the vent flow openings connecting the breakroom with its neighbors were applied.

A cross-section of the most interesting results for pressures, pressure differences, temperatures, local heat coefficients, droplet velocities and steam/air concentrations in vents will be presented by comparing the results of the 4 steam- and 2 water-blowdowns.

Prior to the HDR containment tests, the verifications and assessments of containment codes rested upon the data base provided by the 32 experiments performed in the Battelle-Frankfurt model containment with a total volume of 600 m^3. In order to assess the scaling factor towards the extrapolation of full-size containments, the HDR-tests were thought to confirm the primary findings from the Battelle-tests and to support the modeling procedures.

However, comparisons between experimental data and the results of pretest computations by a wide spectrum of codes suggest that the insights into HDR-containment phenomena resulting from the previous cycle of code assessment on the basis of the Battelle-tests are insufficient in various aspects. This holds for the steam - as well as (to a lesser extent) the water blowdown tests for both short-term (substantial underestimation of pressure differences between subcompartments) as well as intermediate-term (either substantial over-or underestimation of maximum pressure) transient containment behavior. In this respect, HDR data provide a new and substantially improved potential for adding in our knowledge about basic intra- and inter-subcompartment mass, momentum and energy transport processes.

Containment Related Tests at HDR

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The containment experiments conducted at the HDR facility represent an important contribution to a better understanding of the physical processes taking place within the containment after a LOCA. The experiments have greatly expanded the limited data base of measured energy and mass transfer rates within and between large, complex geometry volumes. The new data has:

A. In several domains bracketed the parameter range to be found in actual reactor containments. Thus interpolation rather than extrapolation can be used in licensing and design analysis of containments.

B. Provided severe tests of some modeling aspects. The tests have shown that in the "best estimate" sense shortcomings exist in the modeling of:

1. Energy transfer into structures.
2. Short term pressure gradients between containment zones.

This presentation deals primarily with the modeling of energy transfer. Comparison of experimental HDR data with computed results of pre-test and a limited number of post-test calculations lead to the following conclusions:

1. Integral and local energy transfer rates into structures must be considered separately. That is, heat-transfer coefficient models determining local or integral h values must have a different normalization or separate input assumptions.

   For the HDR experiments a factor of ∿3 difference existed between integral and locally measured heat transfer rates. The local measurements are performed for vertical surfaces which are centrally located and have no obstructions. They thus represent maximum energy transfer rates and can be used directly for equipment qualification purposes.

2. Few (1 or 2) node codes will in general underestimate energy transfer rates early in the transient and overestimate them later on. This will lead to an overestimation of the pressure peak and subsequently an overestimation of the pressure decay rate. These characteristics are traced back to the inability of a few node codes to calculate the nearly pure steam concentrations which will be present in the break zone.

3. In order to reproduce the range and variability of the locally measured h values, a mathematical model must consider the following local conditions:

   a) Steam-to-air ratio
   b) Atmospheric to surface temperature difference
   c) Atmospheric turbulence
   d) Condensing and convective components
   e) Geometry.

   The geometry should describe both the magnitude of the structure surface area and the characteristics of the flow field in front of the surface.

4. A unified correlation of total energy transfered up to the time of peak pressure is possible if the steel and concrete surfaces of the containment are considered separately. A simple correlation is presented which encompasses the CVTR, Battelle-Frankfurt and HDR experimental facilities.

COBRA-NC: AN ADVANCED CONTAINMENT CODE

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The COBRA-NC computer code has been developed by the Pacific Northwest Laboratory for the United States Nuclear Regulatory Commission. The Pacific Northwest Laboratory is operated for the Department of Energy by the Battelle Memorial Institute.

The code has been developed to analyze the transient response of LWR containment systems to transients that result in the release of steam, water, and noncondensable gases into the containment atmosphere. It is designed to predict room pressurization, pressure differentials between rooms, jet impingement forces, and assymetric loads during a loss-of-coolant accident. It is also designed to predict the distribution of hydrogen and noncondensable gases within the containment building, both during and following a loss-of-coolant accident where hydrogen may be generated by metal-water reactions or by radiolysis.

The COBRA-NC code provides a two-component, two-fluid, three-field representation of two-phase flow. It is a two-component model to allow the modeling of water and its vapor as well as a noncondensable gas mixture. The gas mixture may consist of any number of gas species. The properties of eight gases are currently coded.

The two-fluid capability is used to predict the two-phase flows associated with steam-water blowdown into the containment, condensation of steam in the containment atmosphere and on containment structural surfaces, flow through pressure suppression pools, containment sprays, etc. The three fields are (1) the vapor-gas mixture, (2) the continuous-liquid phase, and (3) the liquid-drop phase. The continuous-liquid phase is used to model liquid films on containment structures, pools on containment floors, and pressure suppression pools. The liquid-drop phase is used to model the two-phase jet, containment sprays and drop entrainment between containment rooms.

Three momentum, four mass and two energy equations are solved for the fluid. Momentum equations are solved for the vapor-gas mixture, the continuous liquid and the liquid-drop fields. Thus, each of these may travel at different velocities. A liquid film flowing down walls with vapor flowing across it can be modeled. The vapor may also contain drops that travel at a still different velocity than the vapor.

Mass equations are solved for the noncondensable gas, the vapor, the continuous liquid, and the liquid drops. Thus, the mass of each phase can be accounted for. In addition, a mass-transport equation is solved for each species of the noncondensable gas mixture, so that the concentration of each species can be determined. Energy equations are solved for the vapor-gas mixture and for the continuous liquid-drop mixture. It is therefore assumed that both the vapor and gas in a given computational cell will have the same temperature, and that the liquid film and liquid drops within a given mesh cell will be at the same temperature. These two mixtures, however, can have different temperatures. This model permits the modeling of nonsaturated air (air with a relative humidity of less than 100%) and superheated vapor in the presence of subcooled liquid.

The code is a three-dimensional, compressible-flow, finite-difference code formulated in Cartesian coordinates. However, it features an extremely flexible noding scheme that allows it to be run in a lumped parameter, one-dimensional, two-dimensional, or three-dimensional mode. It has a finite-difference slab conduction model for structural heat transfer. Any number of materials may be used in each slab, and the number of heat transfer nodes through the thickness of the slab may be specified by the user.

A mixing-length turbulence model has also been included to allow the user to model turbulent shear flows and the turbulent diffusion of gas species due to concentration gradients.

A general set of boundary conditions has been included to facilitate the modeling of containment sprays, blowers, blowdown flows, etc. The code does not contain specific models for hydrogen source terms. These must be specified as a boundary condition for the calculation.

COBRA-NC has been used to perform pretest and post-test predictions of several steam/water blowdown and hydrogen distribution experiments. The results of these code assessment data comparisons will be presented in this paper. Current research activities and future directions of the program will also be discussed.

SUPPRESSION POOL DYNAMICS RESEARCH AT MIT

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The past year's research at MIT has been in two areas: problems involving the dynamics of steam injection in and condensation on cold water, and problems involving dynamic interactions between a liquid and its flexible boundaries, whether the latter be solid structures or gas cavities (bubbles).

An issue of some licensing concern is the upper operational bound on BWR pool temperature which was set because there were questions about the stability of SRV steam discharges, even those of the quencher type, under conditions where the pool subcooling becomes very low and the wetwell begins to pressurize. We reexamined this problem via a two-step approach. First, we showed by laboratory-scale testing that Sonin's [Nucl. Eng. and Design 65 (1981) 17-21] method of simulating SRV discharges on a small scale is accurate. Secondly, using small-scale simulation, we investigated SRV-type discharges into a closed wetwell system where pool temperature was allowed to rise to a point where significant pressurization occurred, and demonstrated that peak dynamic loads occur at a finite subcooling (of the order of 20K) and that the wetwell pressurization which occurs at very low subcooling appears to be dynamically benign.

Direct-contact condensation of steam onto highly turbulent, cold water is or has been central to a number of LWR safety issues (SRV discharge, condensation oscillations and chugging, waterhammer events, PWR pressurizer, ECC cold-leg injection, etc.). Many or most of these defy accurate modeling because of our inadequate understanding of the condensation mechanism that's involved. Anderson (MIT PhD Thesis, 1982) recently concluded a careful study of the condensation process in chugging, measuring the instantaneous condensation heat transfer rate and making simultaneous high-speed film observations of the steam-water interface. He showed that chugging involves intense, short condensation bursts, apparently triggered by a sudden instability which depends on subcooling, and that during the bursts the condensation rate exceeds that predicted by the more accepted available correlations by more than a factor of 10.

This year we have embarked on a more fundamental study, the purpose of which is to develop a more unified understanding of, and correlations for, this whole class of condensation phenomena. A special apparatus has been developed where water with a controlled turbulence intensity can be exposed in steady state to steam, and the resulting condensation rate measured. A linear correlation between the condensation heat transfer coefficient and the turbulence intensity in the water has been obtained. At sufficiently high turbulence levels, however, a departure occurs: the condensation becomes unstable, and intense, short bursts of condensation similar to those observed by Anderson in his very different apparatus begin to occur.

The work on coupled gas-liquid-structure interactions has been reported recently by Joos-Gieschen (MIT PhD Thesis, 1982). The problem is simplifed by breaking it into two superposed parts: first, the hydrodynamic transient in which the liquid undergoes bulk motion but its boundaries displace slowly, and second, the fast oscillations produced by the structural and gaseous interactions at the liquid boundaries. The simplifications allow rapid numerical calculations of coupled transients. Comparisons with experiment are excellent.

SAND83-1862A

MITIGATION OF DAMAGING EFFECTS OF HYDROGEN COMBUSTION IN NUCLEAR POWER PLANTS*

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Hydrogen combustion during LWR accidents can pose a direct threat to containment or to the survivability of important safety equipment. The nature and extent of this threat depends strongly on the type of reactor and containment building. The NRC has issued rules with respect to several plant types, indicating the need for additional mitigation for ice condenser PWRs and all BWR plants. The objective of this program is to investigate possible systems for the mitigation or prevention of hydrogen combustion.

The primary emphasis of this research has addressed the following schemes:

1. Deliberate ignition;
2. Deliberate ignition with pressure suppression;
3. Deliberate flaring;
4. Catalytic combustion;
5. Partial pre-accident oxygen depletion (reduced oxygen concentration, and/or gas bag filled with nitrogen or carbon dioxide);

Deliberate ignition systems for ice condensers and BWR Mark III's have been evaluated using the accident analysis computer code, HECTR. Some results of these calculations will be presented. The functionability of igniters (glow plugs and coils) in spray and steam environments is being experimentally investigated. Tests on igniter performance in hydrogen:steam: air mixtures are planned for next year.

The combination of deliberate ignition with pressure suppression has been investigated. Preliminary experiments indicate that the maintenance of high fog densities is difficult to accomplish due to droplet agglomeration. This led to the investigation of water foams as a means of introducing large quantities of water into the containment environment. Although earlier small-scale experiments looked promising, larger-scale tests have indicated that the benefits of foams may not justify their use. The foam tends to reduce pressures at low hydrogen concentrations. At higher concentrations, however, significant flame acceleration occurs, with a reduction in the benefits of this scheme.

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

A preliminary set of experiments investigated the effects of aerosols on combustion, and the effects of combustion on the chemical properties of the fission product aerosols. Early results indicate that cesium iodide is dissociated into gaseous iodine-containing compounds (elemental iodine, hydrogen iodide, etc.) during a hydrogen combustion. These aerosol studies will be continued into next year. Combustion in the presence of water sprays which contain dissolved fission products will also be studied.

Experiments have been conducted in the hydrogen:steam jet facility to investigate deliberate flaring from the primary system. These results will be discussed. An analytic effort has been initiated to investigated thermal loads due to such flaring.

In terms of severe accident mitigation, deliberate ignition systems cannot mitigate accidents which involve total loss of power. Preliminary calculations and experiments indicate that it may be practicable to design a deliberate ignition system which would be passive. Platinum-coated surfaces have been shown to ignite hydrogen:air mixtures at room temperature. We are currently investigating such passive igniters.

We have investigated the modification of containment atmospheres to reduce or eliminate the risks from hydrogen combustion. Among the schemes being studied are the partial depletion of oxygen in a nuclear reactor containment to a low oxygen concentration suitable for safe breathing and continuous working. We also studied the use of inert gas-space in a nuclear reactor containment building. Upon rupture of a bag containing inert gas, the oxygen level would be lowered dramatically without increasing the total pressure of the containment. Moreover, the inert gas can be chosen to reduce the risk of detonation and also significantly lower the peak temperatures to which equipment would be exposed during a deflagration.

ACOUSTIC EMISSION FOR ON-LINE REACTOR MONITORING:
RESULTS OF INTERMEDIATE VESSEL TEST MONITORING
AND REACTOR HOT FUNCTIONAL TESTING

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The objective of the subject program is to develop and validate the use of acoustic emission (AE) methods for continuous surveillance to detect and evaluate flaw growth in reactor pressure boundaries. Technology developed in the laboratory for identifying AE from crack growth and for using that AE information to estimate flaw severity is now being evaluated on an intermediate vessel test and on a reactor facility. These are two major program milestones.

Intermediate Vessel Test

A vessel, designated ZB-1, is being tested under fatigue loading with simulated reactor conditions at Mannheim, West Germany. The work is in collaboration with the German Materialpruefungsanstalt (MPA).

The ZB-1 vessel with a 120 mm thick wall, is 1713 mm O.D. and 3750 mm long. An A533B steel section 120 x 700 x 1500 mm containing three machined flaws has been inserted in the vessel wall. Two of the flaws are on the inside surface and one on the outside.

Water is used as an internal pressurizing medium to stress the vessel in both static and cyclic loading. The test matrix consists of five hydrostatic tests separated by blocks of cyclic loading which total 28,000 cycles. Testing temperatures of 70°C and 288°C with an operating pressure of 250 bar (3675 psig) have been used.

Basic AE data (count, signal amplitude, source location, digitized waveforms, etc.) from three monitoring sensor arrays is recorded and also analyzed in real time.

Highlights of the test results to date are:

- Fatigue crack growth from the machined flaws has been consistently detected by AE.
- Very little AE has been detected during hydrostatic testing up to 1.1 times operating pressure.
- A spontaneous crack in a fabrication weld separate from the test insert was detected and located by AE before there was any other indication of the crack.

- Initial testing of the AE/flaw evaluation model on the 70°C test data showed a satisfactory determination of crack growth rate.

- Initial trial of the pattern recognition AE identification algorithm was not satisfactory. A modified version, however, appears to be producing improved results.

A very large amount of data has been recorded during the 288°C testing which has yet to be analyzed.

Reactor Hot Functional Monitoring

Sixteen channels of AE sensors have been installed at TVA's Watts Bar Unit 1 reactor to monitor hot functional preservice testing and initial operation of the reactor. The locations being monitored are the #2 inlet nozzle, the 10" safety injection line adjacent to the #2 cold leg, and a segment of the reactor vessel wall. The waveguide high temperature AE sensors are pressure coupled to the reactor surfaces. These sensors are tuned to operate in a frequency range expected to be above the coolant flow noise (450-500 kHz peak response).

Results expected from on-reactor monitoring are to evaluate reactor coolant flow noise effects on AE detection, calibrate the detection of AE signals, demonstrate that false alarms can be avoided, and evaluate the ability of an AE monitor system to withstand the reactor environment.

The portion of the hot functional test monitored was a stepwise increase in temperature and pressure up to operating conditions (557°F - 2235 psig) and a hold at operating conditions. The primary results achieved to date are:

- Reactor flow noise decreases as the temperature and pressure increase. At 450°F - 1200 psig and above, the flow noise is only slightly above electronic background noise. Detection of AE is feasible under those conditions.

- AE from a crack growth specimen mounted on the safety injection line was detected under operating flow conditions.

- Signals from the #2 inlet nozzle were detected during the final increase from 450°F - 1200 psig to 557°F - 2235 psig. Data is being analyzed to evaluate whether it is AE or signals produced by insulation movement.

- The AE sensing system was not affected by this short exposure to full temperature and the detection sensitivity was good.

The ZB-1 vessel test and reactor monitoring at Watts Bar are producing long sought information essential to definition of methodology and criteria for applying AE to continuous surveillance of reactor pressure boundaries.

ACOUSTIC LEAK DETECTION AND ULTRASONIC DETECTION OF CRACKS*

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I. LEAK DETECTION

A. Background

The currently available technology for detecting leaks in piping at specific sites includes acoustic monitoring and use of moisture-sensitive tape. However, the present systems of leak detection do not provide reliable quantitative information on flow rate, the nature of the leak source (i.e., pipe crack or valve), or the location of the leak. Improvements are needed to locate and characterize leaks, and quantify and monitor leak rates, more accurately to eliminate false calls; and to minimize the number of installed transducers needed to "complete" a system. These improvements can be accomplished by increasing design reliability and system sensitivity. Detection of reactor piping cracks before leakage occurs is also difficult. Reliable detection of intergranular stress corrosion cracks (IGSCCs) and inspection of cast stainless steel (SS) are especially difficult tasks for ultrasonic inspection teams.

B. Objectives

The objectives of this effort are to (1) develop an independent capability for assessing the effectiveness of current and proposed techniques for acoustic leak detection (ALD) in reactor coolant systems, (2) develop a strategy for hardware realization, and (3) examine potential improvements in ultrasonic methods for detection of IGSCC and inspection of cast SS. The program will determine whether meaningful quantitative data on leak rates and location can be obtained from acoustic signatures of leaks from IGSCCs and fatigue cracks in low- and high-pressure lines, and whether these can be distinguished from other types of leaks. The program will also establish calibration procedures for acquiring acoustic data and will determine whether advanced signal processing can be employed to enhance the adequacy of ALD schemes.

C. Progress

In the past year the ALD test facility was modified. Two fatigue cracks and two thermal-fatigue cracks (TFCs) have been welded into the pipe run. Tests have begun with these laboratory-grown cracks. The first IGSCC was installed in 1981; a second, field-induced IGSCC was recently welded into the pipe run, and leak data were acquired. The acoustic characteristics of the IGSCCs were established and the minimum leak rate that is detectable at a distance of 50 cm under laboratory conditions was identified to be 0.001 gal/min. for a frequency window of 200-400 kHz. Cross-correlation techniques were also demonstrated for locating leaks and distinguishing among leak types. A comparison of acoustic spectra of the two leaking IGSCCs at flow rates of 0.005 gal/min. revealed nearly identical signals in the frequency range 200-400 kHz. Below 200 kHz, the differences in geometry between the two cracks can be reflected in a signal difference. These data suggest that geometrical effects may be less significant at frequencies above 200 kHz. It appears that leak flow rates may be more reliably related to acoustic signals at higher frequencies.

Acoustic-emission noise generated by the smaller of the two TFCs was measured. The spectra were similar in shape to those obtained from an IGSCC, except that the noise level from the IGSCC was 5 dB or 1.77 times the level produced by the TFC at 400 kHz even though the flow rate of the IGSCC was only half that of the TFC. This is consistent with previous observations that IGSCCs produce more noise (at equivalent flow rates) than slits.

Two field-implementable waveguide systems have been completed and tested under laboratory conditions. One system is a "quick connect" type which uses spring loading to press a rounded waveguide tip to the pipe surface. In the other system the waveguide is screwed through a plate strapped onto the outer surface of the pipe. A preliminary analysis indicates that the acoustic signal levels of both designs will be similar if gold foil is used to couple the waveguide to the pipe in the spring-loaded design ($\sim$10-lb load).

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

After ultrasonic testing revealed cracks in the Georgia Power Co. HATCH-1 BWR recirculation header, an ALD system was installed by Georgia Power and Nutech personnel. The transducer and waveguide are similar to the types being evaluated at ANL for leak detection. Data from HATCH-1 have given us an indication of the background noise level at a BWR recirculation header sweepolet weld. The HATCH leak detection system was reproduced and tested using the ANL ALD system to determine the sensitivity and dynamic range of the HATCH system. The results (obtained indirectly) suggest that at a recirculation header of a BWR operating at full power, background noise in the range from 200 to 400 kHz is only a few decibels above the electronic noise. At a distance of 50 cm the system should be capable of detecting leaks of less than 0.01 gal/min.

Other background data have been acquired at the Watts Bar Nuclear Reactor in Tennessee. Here (in cooperation with PNL) an ANL waveguide system, including transducer and electronics, was installed on an accumulator safety injection pipe. The pencil-lead-breaking technique was successfully used to calibrate the system. Data on background noise from 50 to 400 kHz were acquired during the hot functional test, during which temperatures varied from 110 to 556°F and pressures varied from 300 to 2235 psi.

Major effort this past year was directed to hardware realization. GARD Inc. first established a system configuration and then proposed a breadboard system. A local microcomputer was implemented to control data acquisition and to provide operator interface. Communication with a minicomputer at GARD was to be accomplished via two modems and a dedicated telephone line. Further scrutiny of the breadboard's configuration suggested the possibility of using a transportable minicomputer with an appropriate front end. This configuration would avoid the use of the telephone lines and modems with their associated costs as well as the efforts associated with establishing a remote computer interface.

To help compare the adequacy of moisture-sensitive tape with that of ALD systems, a facility has been assembled to simulate leaks in an insulated (reflective insulation) 304 SS pipe. The 2-m-long pipe is heated from the inside. Steam is injected under the insulation to simulate leakage from a crack. Preliminary data have been acquired with this system.

II. ULTRASONICS

The possibility of using ultrasonic wave scattering patterns to discriminate between IGSCCs and geometric reflectors has been explored. Thirteen reflectors (field IGSCCs, graphite wool IGSCCs, weld roots, and slits) were examined. The data indicated that the "trough" in the plot of echo amplitude vs. skew angle is not as sharp for an IGSCC as for a geometrical reflector. This work led to the design and fabrication of a 7-element skew-angle probe for 28-in. piping, assembled for ANL by Magnaflux Corp. This probe is now being evaluated for its ability to distinguish cracks from geometrical reflectors.

The work with cast stainless steel included acquisition of data on sound velocity and attenuation in isotropic and anisotropic cast SS. Reducing anisotropy does not help reduce attenuation in large-grained material because the main factor in attenuation is correspondence of ultrasonic wavelength to grain size. Other work indicated that large artificial flaws (e.g., a 1-cm-deep notch with a 4-cm path) could not be detected in isotropic centrifugally cast SS (1 to 2-mm grains) by longitudinal or shear waves at frequencies of 1 MHz or greater, but could be detected with 0.5-MHz shear waves.

III. FUTURE EFFORTS

Future efforts will concentrate on (a) acquiring additional data for the design of ALD systems (other IGSCCs and fatigue cracks, time domain structure, coupling schemes, background noise, etc.); (b) assessing a preliminary prototype ALD system for field applications; (c) beginning the analysis of hardware in the laboratory; (d) establishing an estimate of leak location capability; (e) developing a prototype hardened acoustic-emission leak detection system to be evaluated for performance in a nuclear reactor monitoring situation; (f) determining the final system configuration, including operator interface and graphics display; (g) testing the multielement skew-angle probe in the laboratory; (h) acquiring data to help correct calibration curves for ultrasonic testing of anisotropic cast SS; and (i) transferring ultrasonic NDE technology to industry.

INTEGRATION OF NONDESTRUCTIVE EXAMINATION RELIABILITY AND FRACTURE MECHANICS

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INTRODUCTION

The primary pressure boundaries (pressure vessels and piping) of nuclear power plants are inspected inservice according to the rules of the ASME Boiler and Pressure Vessel Code, Section XI. The NDE/FM Program at PNL was established to determine the reliability of current ISI techniques and to develop recommendations that will insure a suitably high inspection reliability. The activities of the program are divided into three primary tasks: Pipe Application, Vessel Applications, and Fracture Mechanics. Highlights from each of these tasks are detailed below.

PIPE APPLICATION TASK

Two activities were the focus of the efforts in this task: the pipe inspection round robin (PIRR) data analysis and the development of a document on qualification. A draft PIRR report was prepared and sent to the NRC for review. The PIRR samples have been involved in advanced technique assessment and will be destroyed over the next few months. However, preliminary conclusions for the PIRR data are:

- Inspection of clad ferritic main coolant pipe can be quite effective (approaching 100%) if sufficient sensitivity is used. Section XI minimum sensitivity is not adequate.

- Inspection of clad ferritic pipe is effective with or without weld metal in the sound path.

- Inspection of CCSS pipe using conventional manual UT techniques is ineffective.

- Section XI minimum requirements do not provide adequate sensitivity for effective UT inspection of wrought SS pipe welds. Increased sensitivity and selection of optimized transducers improves detection reliability.

- When the sound beam must pass through SS weld metal, manual UT inspection using current field techniques is ineffective.

- Crack detection capability for inspecting SS pipe welds should be qualified by demonstration.

- Current field amplitude drop crack depth measurement techniques are not effective.

The qualification document describes criteria and requirements for qualifying ultrasonic testing for inservice inspection of the Class I nuclear reactor components. The criteria and requirements are in addition to, but not a replacement for, the applicable requirements in ASME Section XI and are applicable to UT personnel, equipment, and procedures.

VESSEL APPLICATION TASK

The initial thrust of the Vessel Application Task has been establishing the effectiveness and limitations of using UT for detecting flaws of interest in Pressurized Thermal Shock (PTS). The analysis of pressurized thermal shock shows it is necessary for NDE to demonstrate high probability of detecting cracks 3 mm deep and deeper at the clad/base metal interface. Techniques developed in Europe using 70^{o} compressional waves have been shown to be effective on European vessels in detecting underclad cracks 3 mm deep.

Flaw detectability experiments were carried out to determine the inspection technique's ability to detect defects under conditions of rough surfaces, including manually applied clad.

Experimental data shows surfaces with roughness greater than 12.6×10^{-3} RMS are not readily inspected. Surfaces between 12.6×10^{-3} and 5.6×10^{-3} RMS are marginally inspectable. Surfaces that are smoother than 5.6×10^{-3} can be inspected with a high degree of confidence. Our conclusion is, for sufficiently smooth clad surfaces, the dual element 70^{o} compressional wave technique can be effective for detecting underclad cracks.

FRACTURE MECHANICS TASK

The objective of this task is to apply deterministic and probabilistic fracture mechanics to guide the development of ISI requirements. Critical factors of concern are NDE sensitivity requirements, inspection intervals, and weld inspection sampling plans. The primary accomplishments include calculations performed with a PNL model of weld inspection sampling plans. This analysis is integrated with published crack growth data and NDE effectiveness based on the PIRR results. The evaluation is couched in terms of a "factor of improvement" in system reliability. The effectiveness of the ASME Code sampling plans was compared with alternative plans that require fewer or greater numbers of inspections. The estimated improvement factors showed considerable sensitivity to the assumed weld-to-weld leak probabilities, to the assumed random or systematic nature of these probabilities, and to the extent of augmented versus scheduled inspections.

DEVELOPMENT AND VALIDATION OF A REAL-TIME SAFT-UT SYSTEM FOR INSERVICE INSPECTION OF LWRs

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The PNL effort is the continuation of a program started several years earlier by the NRC at the University of Michigan. A final report of the work performed by the University of Michigan is being presented by S. Ganapathy at this conference and the reader should refer to his paper for the appropriate information. This paper will present only the work being performed at PNL.

The objectives of the program are: (1) to engineer and evaluate a real-time flaw detection and imaging system based on SAFT-UT for inservice inspections of all required LWR components, (2) to establish calibration and field test procedures, (3) to demonstrate and validate the system through actual field reactor inspections, and (4) to gain ASME Code acceptance of the real-time SAFT-UT technique.

The initial task of the program was to transfer the SAFT technology developed at the University of Michigan to PNL. PNL staff went to the University of Michigan and worked with them to learn and thoroughly understand the technology. Next, a 3-year program plan was developed.

The emphasis of the first year was placed on developing the specifications for a real-time fieldable SAFT-UT system. The system specifications were approved and the front end of the system was assembled. The front end is a stand-alone automated data acquisition system consisting of an Amdata AMAPS pipe scanner, LeCroy digitizer, power supplies, and scanner controller which are all under minicomputer control. This SAFT data acquisition system was assembled by the end of August and will be under evaluation during early September. Once the system is operational and checked out, an extensive evaluation of IGSCC and thermal fatigue cracks in 10 inch, Schedule 80 pipe and thermal fatigue and mechanical fatigue cracks in centrifugally cast stainless steel will be conducted.

Preliminary results for IGSCC and CCSS look extremely encouraging and further results from these studies will be presented at the 11th LWR meeting. With this data acquisition system, we now have the capability to collect SAFT data at field locations. However, the SAFT data will have to be returned to PNL for processing and evaluation.

A document that evaluates the literature and discusses the physics of SAFT was prepared by PNL staff. In addition, this document contains an analysis using a simple model to explain the images that should be expected from defects of primary concern;

specifically, service induced cracking. This document will be published at the end of FY83 and will be used to educate people on the Code committees to fully understand SAFT.

During September the PNL staff will be working with the staff from GARD, Inc., to evaluate the real-time SAFT processor prototype that they are developing.

The second year of the program will primarily focus on integrating the data acquisition system with aids to assist the operator in viewing and interpreting SAFT images. Additional work will be conducted to prepare for the extensive field testing planned for the third year of the program. This will include going to other sites on a selected basis to acquire SAFT data and shake down the system.

IMPROVED MULTIFREQUENCY EDDY-CURRENT TESTING OF STEAM GENERATOR TUBING*

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Eddy-current inspection is the most suitable method for rapid boreside evaluation of steam generator tubing. However, small flaws can be masked by the effects of harmless variables such as tube supports. To identify the critical properties accurately and reliably in the presence of extraneous signals caused by variations of unimportant properties, sufficient information is needed to identify harmful variations and to reject harmless ones. For this reason we have developed instrumentation capable of measuring both the amplitude and phase of the eddy-current signal at several different frequencies as well as computer equipment capable of processing the data quickly and reliably.

In addition to detecting the presence of flaws, we also want to know their location, size, and orientation, as well as larger scale changes in the tubing such as intergranular attack and buildup of deposits such as copper or magnetite on the tubing. To distinguish all these properties as well as reject other variables that can affect the readings, we need even more information from the sample. At present, there is a practical limit of three or four to the number of frequencies that can give significantly

*Research sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, under Interagency Agreement DOE 40-551-75 with the U.S. Department of Energy under contract W-7405-eng-26 with the Union Carbide Corporation.

different information, but additional information can be obtained by the use of coil arrays or by utilizing the changing information as a coil is scanned past a flaw. We have shown theoretically and experimentally that the size and location of a defect can be obtained by computer processing of the data obtained as a coil is scanned past the defect.

Our computer studies and experimental tests have also shown that small, flat, "pancake" coils pressed against the inside wall of the tubing are an order of magnitude more sensitive to small flaws than are the conventional large circumferential coils, as well as much less sensitive to variations outside the tube, such as tube supports. The small coils achieve these advantages by examining a much smaller region of the tubing. The penalty is that an array of small coils is needed to scan a whole tube at the same speed as a single large coil. The additional cost and complexity of the array is justified by the great increase in reliability of flaw detection.

The eddy-current instrument contains a microcomputer that must be "trained" to recognize at the different frequencies response patterns that correspond to significant flaws and reject harmless response patterns.

One of the most difficult problems has been to recognize flaws at the edge of a tube support or the tubesheet. We have found that flaw detectability can be maintained by programming a "tube support channel" in the computer, to recognize when the probe is inside, outside, or at the edge of the tube support. Then the computer can use different formulas to recognize the flaw patterns in the different regions. Actually, it turns out that the same formula works for the probe completely inside or outside the tube support, and a different formula is needed only for the transition region at the edge of the tube support.

STATUS AND PROGRESS OF RESEARCH ON A
REMOVED-FROM-SERVICE STEAM GENERATOR

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Under an NRC directed group sponsored project (including French, Italian, Japanese, and EPRI participation) a steam generator removed from service at Virginia Electric and Power Company's (VEPCO) Surry nuclear station is the subject of extensive research. This U-tube type steam generator was in service six years, and was removed with ~22% of its tubes plugged due to various defects or for engineering judgement. The generator now serves as a research vehicle for studies involving validation of the accuracy and reliability of current nondestructive testing (NDT) characterization during inservice inspections, determination of remaining integrity of service defected steam generator tubes, determination of failure consequences (leak rate) of defects, and demonstration of cleaning and decontamination techniques. The unit is also available for demonstration/development of NDT and repair techniques. Program objectives are to provide inputs to regulatory guides on inservice inspection and tube plugging criteria. Operational and reliability benefits will also accrue from demonstration and development of inspection, maintenance, and repair techniques.

This past year efforts have included decontamination of the channelhead region, removal of 970 explosive plugs, multifrequency eddy current inspection of all accessible tubes, and extensive characterization of the secondary side. Two dilute chemical reagent techniques were used, one on either side, for the channelhead decontamination. Both of these techniques were successful in meeting programmatic goals for decontamination factors. Neither method appeared to damage the unit. Demonstration of the two techniques has allowed their entry into the nuclear service sector. The methods potentially offer exposure savings and minimization of secondary waste. Tube unplugging was necessary to open the majority of interesting tubes, those plugged because of defects, for

NDT validation studies. A very successful effort realized removal of 970 plugs in 20 days of operation. The subcontractor acquired additional experience with large scale unplugging, as well as development time for automated equipment. A number of plugged tubes contained water, secondary or primary, and 30 contained significant amounts of sludge. These latter tubes, in particular, are being evaluated for post plugging degradation.

Subsequent to unplugging, a multifrequency eddy current examination was conducted along all accessible tube sections. This baseline establishes the condition of the generator and will aid in the choice of specimens for further reliability studies using round robin techniques. Secondary side inspections have been conducted through several generator shell penetrations. Examinations of inner row U-bends, the tubesheet upper surface and support plates have been conducted. Innovative photographic techniques and devices have been developed which greatly enhanced this effort.

This coming year, efforts will concentrate on primary side NDT followed by specimen removal for validation studies and integrity testing. Secondary side characterization and corrosion product sampling will continue. Other specimens will be removed for laboratory scale examinations of secondary side cleaning and primary side decontaminations techniques.

Quantitative Aspects of Factors that Influence Stress Corrosion Cracking of Alloy 600 in High Temperature Water

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Stress corrosion cracking data are being generated for Inconel 600 steam generator tubing using U-bend, constant load and slow extension rate tests. Arrhenius plots are made of failure times or crack rates vs. inverse temperature for crack initiation and propagation; the effect of applied load is expressed in terms of log-log curves of failure times vs. stress; variations in environment and cold work are included. Microstructure and composition of oxide films stripped from Inconel 600 surfaces are also being examined. These corrosion tests address two simulated conditions, i.e., where deformation occurs but is no longer active, such as when denting is stopped, and where slow plastic deformation of the metal continues, as would occur during denting. Laboratory test media consist of pure water as well as solutions to simulate service environments. Further details of autoclave testing, specimen preparation, materials and their composition have been described in earlier publications of results under this program (Ref. 1-3).

U-bend Tests

There is good correlation between crack velocities calculated from U-bends and CERT tests at the same temperature: U-bends at 365°C $\doteqdot$ 6 x 10^7mm.sec^{-1};CERT $\doteqdot$ 8 x 10^{-7} mm sec^{-1} at 365°C. (See also the section on CERT data for more details on variations in crack propagation rate).

U-bend tests over a limited temperature range in pure H_2O suggest that carbon level could possibly influence crack initiation activation energy, which seemed to increase with increasing carbon content in the range tested (0.01-0.05%). At lower temperature (315°C) SCC has only recently been observed, confined so far to the 0.01%C heat; for this material, the Arrhenius plot corresponds to about 33000 cal/mole, and no discontinuity is observed in the curve from 365° to 315°C. These exposures continue, and have reached 2 years at present.

G. Theus[4] found that the Arrhenius curves become steeper in the region of operating PWR steam generator temperatures. Our data for higher carbon tubing are yet insufficient to compare with these findings, although the 0.01% carbon heat (above) did not show the inflection in the curve.

Slow Extension Rate Tests

This method does not eliminate fully the initiation stage of SCC; cracks in tubing start after yielding[1,2]. Initiation times are much shorter than for U-bends, and vary with temperature.

A family of straight line Arrhenius plots of CERT data, obtained at strain rates of about 2 to 3 x $10^{-7}sec^{-1}$ has been generated, and an activation energy of about 33 Kcal/mole calculated for material as cold worked, or aged (365°C), or mill annealed, suggesting that these treatments do not affect the crack propagation mechanism.

To correct for initiation times, plots of crack length vs. strain have been extrapolated to zero crack length and used in calculating crack velocities. This procedure is believed to be reasonable, but may not be accurate enough for reliable estimates of percent strain required to induce SCC initiation. More direct measurements are being explored for the latter purpose.

Temperature exerts a much greater influence than strain rate on crack velocity. Most of the present tests are conducted successfully at about $2\text{-}3\times10^{-7}sec^{-1}$, but for test conditions where cracks grow slowly, a lower strain rate is needed.

Cold work appears to increase the SCC velocity in CERT, but we find the same temperature dependence as for as-received material. For cold worked material, the effect of environment seems to be masked by the increase in crack velocity due to prior cold reduction.

In as-received tubing there seems to be an acceleration of crack propagation due to hydrogen, but a lowering of crack velocity due to an increase in pH such as would result from the addition of lithium hydroxide to the test medium. As a result, it appears that simulated primary water is not substantially different from pure water in this particular test, and in all cases the curves appear to be of about parallel slope.

Constant Load Tests

Some tests in progress use simulated dents in Inconel 600 production tubing, but so far they have been exposed for too short a time to give SCC. Other, constant load, tests in pure water at 365°C, show a band within which failure times could be plotted on a logarithmic scale against log of stress, for as-received material. The slope of the log-log curve gives failure time proportional to the -4th power of stress. Data obtained from other environments* with as-received Inconel 600 production tubing at 365°C fall generally in or very close to this pure water band.

Using points for the same applied stress but at 2 temperatures, 365°C and 345°C, an Arrhenius equation yields a calculated activation energy for SCC of 34 Kcal/mole. Although only tentative, this value is remarkably close to the CERT and U-bend values for 0.01%C Inconel 600. At low carbon levels, therefore, the mechanisms of crack initiation and propagation could be very closely related.

In constant load tests, 5-20% cold worked specimens failed in periods below and up to the same time as the as-received tubing in pure H_2O at 365°C; some of the 5% cold worked specimens cracked remarkably quickly, i.e. in one or two days.

Comparison of Methods

Some heats have cracked easily only as U-bends and with difficulty or not at all in one (or both) the constant load test and CERT when using simple tensile specimens. It is likely that a more complex stress is needed for SCC in some heats of tubing.

Oxide Film and its Role in SCC

Extraction replicas of oxides and precipitates were studied after film removal in a start to understand the role of microstructure: part of this ongoing work is described in the work to be presented.

*Simulated primary water, and water with only some of the primary ingredients.

Evaluation of Stainless Steel Pipe Cracking:
Causes and Fixes

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Introduction

Leaks and cracks in the heat-affected zones (HAZs) of weldments in austenitic stainless steel piping and associated components in boiling water reactors (BWRs) have been observed since the mid-1960s. Cracking has continued to occur and indications have been found in all parts of the recirculation system including the largest-diameter lines.

Proposed remedies include procedures primarily intended to produce a more favorable residual stress state, materials which are more resistant to stress corrosion cracking, and changes in the reactor environment which decrease the susceptibility to cracking. In addition to evaluating these remedies it is also important to understand the influence of key variables, such as residual stress, crack growth rates, and the margin for leak-before-break in flawed piping, which may impact regulatory decisions on operating plants.

Technical Progress

Finite element calculations of the stresses in 12-24 in. pipes which have received Induction-Heating-Stress-Improvement or weld overlay procedures have been carried out for piping with flaws. For shallow cracks these processes still produce compressive residual stress states on the inner surface, and the stresses at the crack tips remain compressive under design loads. Additional analyses are needed to determine the maximum crack depths for which the processes are effective. The leak-before-break margin for piping with a part-through crack completely around the circumference and a throughwall crack over a portion of the circumference has been assessed. Even severely degraded piping is shown to have a significant leak-before-break margin, although the available margin decreases rapidly with increases in depth of the part-through crack.

Most of the available data on crack growth rates for Type 304 stainless steel was obtained on furnace sensitized materials in high purity water with 8 ppm dissolved oxygen under constant load. Crack growth tests are currently in progress to investigate the effect of impurities, loading history, and the degree of sensitization on crack growth rates in more realistic BWR environments. However, since crack growth tests are extremely time consuming, only a limited number of these parameters can be examined, and constant-extension-rate-tensile (CERT) tests are being carried out for a much wider range of conditions.

The CERT tests have shown a strong synergistic interaction between the dissolved oxygen content and the concentration of impurities. For oxygen contents between 0.1-8 ppm even low levels of impurities reduce the differences in susceptibility observed at different oxygen levels in high purity environments. Similarly, impurities appear also to diminish differences in susceptibility due to differences in sensitization level. Materials which are susceptible to cracking in both high purity and impurity environments show higher crack propagation rates in the impurity environments ($\sim$2-3 times faster).

At the much lower oxygen levels (<50 ppb) possible in BWRs with hydrogen additions ($\sim$2 ppm) to the feedwater it appears that susceptibility to intergranular stress corrosion cracking (IGSCC) is diminished significantly. However, careful control of impurity levels will also be required. Although our current results illustrate the interaction between oxygen content and impurity concentration, we are not able at present to define tolerable levels for the impurities in low oxygen coolant environments.

The most widely used alternate piping material is Type 316 "Nuclear Grade" stainless steel with controlled carbon and nitrogen levels. Laboratory tests have demonstrated its resistance to IGSCC in high purity water but much less testing has been carried out in environments with realistic impurity levels. Our results have confirmed the resistance of Type 316NG to IGSCC in impurity environments for realistic heat treatments. Some transgranular cracking has been observed in CERT tests in environments with chlorides and sulfates. This may be an artifact of the high strains produced in these tests, but preliminary information indicates that the cracks appear to initiate at fairly low strains. Even in these cases the crack growth rates are at least an order of magnitude lower than the IGSCC crack growth rates observed in conventional Type 304 stainless steel under the same conditions.

Laboratory testing of materials intended to have 30 year lifetimes invariably requires accelerated testing. To confidently extrapolate from these tests to in-reactor loading situations, a better understanding of the effects of loading history on susceptibility and of the mechanisms actually involved in the cracking process is needed. A model which uses an estimate of the crack tip strain rate to develop correlations between the measured parameters in the slow strain rate test such as crack propagation rate, time-to-failure, maximum stress, etc., and the nominal strain rate has been developed. Tests have also been performed to examine IGSCC susceptibility under pure torsion (mode III) and tensile loading (mode I). Large differences in failure times (>50) have been observed for Type 304 stainless steel at 284°C with 0.2 ppm dissolved oxygen and 0.1 ppm sulfate as H_2SO_4. These very tentative results suggest that hydrogen may play a role in IGSCC of Type 304 stainless steel in BWR-like environments.

AGING DEGRADATION OF CAST STAINLESS STEEL: STATUS AND PROGRAM*

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Cast duplex stainless steels (SS) are used extensively in the nuclear industry to fabricate pump casings and valve bodies for light-water reactors (LWRs) and primary coolant piping in pressurized water reactors (PWRs). The ferrite phase in the duplex structure of austenitic-ferritic SS increases the tensile strength and improves weldability, resistance to stress corrosion, and soundness of castings of these steels. However, the presence of ferrite in these steels introduces several disadvantages regarding their metallurgical stability. The precipitation of additional phases within the ferrite phase leads to variability in properties, increased susceptibility to σ-phase embrittlement at high temperatures, and a reduction in low-temperature ductility due to a phenomenon known as "475°C embrittlement;" the temperature of maximum embrittlement.

Data on the aging behavior of ferritic or austenitic/ferritic stainless steel show no evidence of σ-phase formation at temperatures below 540°C, except in severely cold-worked material where the σ phase is observed at temperatures as low as 482°C. Consequently, precipitation of σ phase is not a concern at the operating temperatures of LWRs, i.e., 288 to 316°C. At temperatures below 500°C, embrittlement of the ferritic or duplex stainless steels is attributed to the precipitation of α' in the iron-rich α matrix.

Initial studies on the kinetics of 475°C embrittlement suggested that serious embrittlement damage at the operating temperatures of LWRs would occur only after ~40 yrs. Embrittlement was assumed to occur via nucleation and growth of α' particles. The time required for significant embrittlement was estimated by using the activation energy of 54,900 cal/mole for chromium diffusion in the ferrite matrix. Recent investigations of the aging behavior of CF-8 and -8M cast duplex SS show substantial reductions in room-temperature impact strength after 10,000 to 70,000 h at temperatures as low as 300°C. The ferrite content of the cast structure has a pronounced influence on the embrittlement behavior. In the temperature range of 300 to 400°C, the data yield an activation energy of 24,000 cal/mole; a value that is much lower than expected for a mechanism controlled by solute bulk diffusion. These results suggest that the precipitation process may be controlled by another mechanism, e.g., spinodal decomposition, or that processes other than α' precipitation contribute to embrittlement. The available information on the microstructure

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

of aged, cast duplex SS is not sufficient for correlating the microstructure with the mechanical properties or for determining the mechanism of low-temperature embrittlement. Changes in composition of cast duplex SS also influence the aging behavior and add to the uncertainty of predicting the long-term embrittlement behavior.

The objectives of this program are to (1) characterize and correlate the microstructure of in-service reactor components and laboratory-aged material with loss of fracture toughness and identify the mechanism of embrittlement, (2) determine the validity of laboratory-induced embrittlement data to predict the toughness of component materials after long-term aging at reactor operating temperatures, (3) characterize the loss of fracture toughness in terms of fracture mechanics parameters in order to provide the data needed to assess the safety significance of embrittlement, and (4) provide additional understanding of the effects of key compositional and metallurgical variables on the kinetics and degree of embrittlement.

Material was obtained from various experimental and commercial heats of ASTM A351 and A451 grades of CF-8, -8M, and -3 cast SS in different product forms and section thicknesses. The relationships of time and temperature to the initiation of precipitation and the onset of embrittlement will be determined by microstructural examination, hardness measurements, and charpy-impact tests. Measurements of impact strength and ductile-to-brittle transition temperature will be used to define the aging histories, chemical compositions, and metallurgical structures that lead to significant embrittlement and to better characterize the embrittlement phenomenon. Measurements of fracture toughness will be carried out to determine the degree of embrittlement that can be expected as a function of service time and the compositional variables.

The initial experimental effort is focused on characterizing the microstructure of long-term, low-temperature, aged material. Specimens from three heats of cast CF-8 and -8M SS aged for up to 70,000 h at 300, 350, and 400°C were obtained from George Fisher, Ltd., Switzerland. Initial analyses reveal profuse formation of an unidentified second phase in the ferrite matrix of the steels after 70,000 h at the three temperatures. This second-phase precipitate shows some preference for forming on dislocations. The diffraction patterns indicate that the precipitate is not α'. A microstructural evaluation of the aged material is presented in the text.

CHARPY TREND-CURVE DEVELOPMENT BASED ON PWR SURVEILLANCE DATA

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A least-squares analysis has been used to determine functional relationships connecting (a) the irradiation-induced increase in the 41 Joule Charpy transition temperature, (b) the pressure vessel steel chemistry, and (c) the irradiation fluence. The data base consisted of transition temperature increases, chemical composition and fluence values for 41 weld metal points and 106 base metal points. All but two of the data points are from pressurized water reactor (PWR) surveillance irradiations. The remaining two data points came from test reactor irradiations of reactor pressure vessel weld materials. The data were originally supplied by Dr. P. N. Randall of the U.S. Nuclear Regulatory Commission. Most of the fluence values are those determined by R. L. Simons.[1]

Functional forms investigated were of the type

$$\text{Delta Temp} = F(\text{Chemistry})*(\phi t)**N, \tag{1}$$

where ** denotes exponentiation and the fluence is given in units of 10^{19} n/cm^2, E > 1.0 MeV.

A least-squares code was used to minimize the weighted sum of squares of two types of residuals:

1. The difference between the measured and calculated Charpy shifts.

2. The difference between the reported and adjusted (by the computer) fluence values.

Earlier work[2] had shown that failure to include the fluence errors produced a false low (in error by as much as 0.05) value for the fluence exponent.

Following a suggestion by G. R. Odette, N was expressed as $N = A + B \cdot \ln(\phi t)$ where A and B are adjustable parameters. This allows the fluence exponent to vary slowly with the fluence.

The computer program used separate functional relations for the plate and weld metal, and adjusted the parameters in the two relationships simultaneously, while it also determined "best" values (in the least-squares sense) for the irradiation exposures. The exposure adjustments were constrained to adopt a single fluence adjustment for specimens irradiated in adjacent positions in a common capsule.

For both the plate and the weld equations, the chemistry-dependent part of equation (1) was made up of various linear combinations of (1) an additive constant, (2) the copper content (Cu), (3) nickel content (Ni), (4) Cu^2, (5) Ni•Cu, (6) $(Cu \cdot Ni)^{0.5}$, and (7) Cu•tanh (A•Ni/Cu) where A is an adjustable parameter. Favorable combinations of functions (1) through (7) were chosen by forward linear regression techniques, combined with separate observations of correlations between the functional forms and the normalized

Charpy shift.The normalization was based on previously determined laws for fluence dependence, e.g., $\Delta T \propto (\phi t)^{**}N$ where N was 0.25 or 0.3. Favorable functional forms were investigated further by use of the nonlinear least-squares program.

For the plate metal relationships, standard deviations on the order of 16°F were obtained using linear combinations of the functional forms (1) through (7), so no attempt was made to expand the number of chemical variables beyond copper and nickel.

For the weld relations, standard deviations based on linear combinations of functions (1) through (7) had values which ranged from 26°F up. Consequently, some attempt was made to reduce the standard deviations by including additional chemistry variables. The attempt was only marginally successful. For welds, it was found that, among the single elements the nickel concentration correlated most strongly with the normalized temperature shift, followed by copper. Phosphorous showed a weaker and reversed correlation ($\sim$-.45). A Cu•Ni product term correlated more strongly than either copper or nickel singly. Some silicon-nickel combinations were investigated and forward regression studies indicated that the term Ni•Cu•Si**0.5 showed some promise as an addition to the chemistry factor of Eq. (1).

The following equations give the weld and plate Charpy shifts, using only the nickel and copper concentrations as independent chemical variables:

$$F_W(°F) = 582.\cdot Cu - 322.\cdot (Cu\cdot Ni)^{0.5} + 261.3\cdot Ni \tag{2a}$$

$$N_W = .287 - .0472\cdot \ln(\phi t) \tag{2b}$$

$$F_p(°F) = -37.8 + 539.8\cdot Cu + 522.1\cdot Cu\cdot \tanh(.304\cdot Ni/Cu) \tag{3a}$$

$$N_p = .272 - .0457\cdot \ln(\phi t) \tag{3b}$$

These equations follow the form of Eq. (1). The elemental concentrations are to be entered in weight percent.

The standard deviations for the two formulas are 15.6°F for the plate equation and 26.4°F for the weld equation. The covariance matrix for the parameters is available and can be used to estimate uncertainties in calculated shifts using standard error propagation methods. It is recommended that the use of the plate formula be restricted to the (Cu, Ni) weight percent range of (.05 to .25, .17 to 1.1) due to the limited range of the data used to fit the parameters. The similar restriction for the weld formula is (.11 to .36, .05 to 1.1).

For application at deep wall positions the fluence entered into the expression for the Charpy shift must be normalized to a standard dpa spectral index ratio of 1.6×10^{-21} dpa/(n/cm^2; E > 1.0 MeV). This is an average ratio for the surveillance capsules used in this data base. Thus for example, at a deep position where the dpa/(n/cm^2, E > 1.0 MeV) ratio is 3.2×10^{-21}, and where the actual fluence is 1.0×10^{19}n/cm^2 (E > 1.0 MeV) the fluence used in the calculation must be doubled to 2.0×10^{19}n/cm^2.

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Validation of Neutron Transport Calculations in Benchmark Facilities for Improved Vessel Fluence Estimation*

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An accurate determination of the damage fluence accumulated by reactor pressure vessels (RPV) as a function of time is essential in order to evaluate the vessel integrity for the pressurized thermal shock (PTS) problem.(1) The desired accuracy for estimated neutron exposure parameters, such as ϕ (E > 1 MeV), ϕ (E > 0.1 MeV), and displacements per atom, is on the order of 10-30%;(2) however, these types of accuracies can only be obtained realistically by validation of calculational methods in benchmark facilities.(3) This paper reviews method approximations and nuclear data that require validation, addresses the validation of current and proposed benchmark experiments, summarizes conclusions from completed benchmarks, and suggests areas where further benchmarking is needed. The Nuclear Regulatory Commission (NRC) and the Electric Power Research Institute (EPRI) have sponsored complementary benchmarking programs aimed at addressing the most crucial validation requirements for RPV analysis.(3)

The current benchmarking programs are sufficient to validate most of the transport methods and data needed for RPV analysis. The preliminary results that are being obtained from these benchmarks indicate that acceptable accuracy (better than 20%) can be obtained for midplane calculations in which the core source is well characterized. The investigations to date lead to the following preliminary conclusions:

1. It has been shown that the ENDF/B-V dosimetry cross sections can be consistently adjusted to produce better agreement among many differential experiments. The major effect of one such modification is a reduction of about 9% in the ^{63}Cu(n,α) cross section, which produces more consistent results in the ORNL Poolside Facility (PSF) benchmark analysis.

2. The iron total inelastic cross section in ENDF/B-V should be reduced by about 1 σ (6%) above 3 MeV, and the ^{235}U fission spectrum increased by about 0.7 σ (10%) above 6 MeV.

3. Midplane calculations of dosimeters in simulated RPV configurations, such as mocked up in the Pool Critical Assembly (PCA) and the PSF, can be computed to an accuracy of 10-20% using discrete ordinates transport theory. By using LEPRICON least squares adjustment procedures to improve the calculations, the uncertainty of adjusted reaction rates at the midplane T/4 position in a simulated vessel can be reduced by about a factor of two.

*Research sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission under Interagency Agreements 40-551-75 and 40-552-75 with the U.S. Department of Energy under contract W-7405-eng-26 with the Union Carbide Corporation.

4. The effect of the surveillance capsule perturbation is well predicted by including a representation of the capsule in the transport calculation. For Westinghouse capsules, the perturbation to dosimeter reaction rates was found to be up to 33% for a ^{237}Np dosimeter.

5. Comparisons of calculations with measured fission rates in the VENUS reactor core indicate the shape of the neutron source can be computed to an accuracy of about 5% near the important core-baffle interface using transport theory. However, no validation of diffusion theory calculations has been done.

6. The 3-D flux synthesis method has been benchmarked for the ORR-PSF, a small research reactor. This approximation was able to accurately predict axial and horizontal reaction profiles within 120 mm of the midplane, even with asymmetries in the core source. This distance corresponds to about 40% of the distance to the top of the active ORR core. However, the synthesis method has not been validated near the top of the active core, nor for large reactor cores, nor for cavity calculations in which streaming may affect the accuracy.

7. Cavity measurements and calculations in the 2568 MW_{th} ANO-1 PWR indicate that ex-vessel dosimeters at the midplane elevation can be computed to an accuracy of about 10%, but the agreement at an elevation corresponding to the coolant inlet nozzle is only about 30%. It is not presently known if this discrepancy is caused by difficulty in computing cavity streaming or by a breakdown of the 3-D synthesis method or by some problem with the dosimeter measurements.

Further benchmarking could significantly enhance the validation of RPV damage fluence calculations. The first of these is an extension of the present VENUS calculation program to include benchmarking the accuracy of "standard" core analysis methods, such as two-group diffusion theory, as mostly used by industry to compute the power shape near the core-baffle interface. The second is the 3-D flux synthesis approximation in the region within the RPV above the top and below the bottom of a PWR core. Often it is these locations which are critical for PTS analysis, and yet it has not been validated that the synthesis procedure will produce reliable results at such extreme axial locations. These latter two benchmarking efforts should receive high priority in the future transport methods validation programs.

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THE NESTOR SHIELDING AND DOSIMETRY IMPROVEMENT PROGRAMME (NESDIP): THE REPLICA EXPERIMENT (PHASE 1)

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This paper summarises the progress, results and analysis obtained to date from the UK Shielding and Dosimetry Improvement Programme on the NESTOR reactor at AEE, Winfrith. This programme is a broadly based series of experiments aimed at investigating generic radiation physics and shielding problems associated with the operation of a PWR plant. Of interest in the present context is that part of the programme which is seeking to provide benchmark-quality data against which modern techniques for the estimation of reactor pressure vessel damage fluence can be assessed. The background to the NESDIP and in particular its complimentary connection with other international dosimetry programmes, notably the US-NRC/SDIP and the Belgian VENUS programmes, have been fully explained in a paper given at the 10th Water Reactor Safety Research Information Meeting in 1982. It is the intention at this meeting to present the results of experiments and analysis carried out during 1983 including current work on the further analysis of the PCA.

Phase 1 of the NESDIP, now complete, has concentrated upon the measurements made in a "REPLICA" of the Pressure Vessel Simulator array built for the original US experiments at the Oak Ridge National Laboratory using the Pool Critical Assembly (PCA) as a neutron source. In the UK experiments however, the core source was replaced by a rectangular fission-plate source. The objectives of the REPLICA experiments were:

1. to provide data to demonstrate the comparability, in neutronic terms, of the UK and US experimental configurations,
2. to obtain additional experimental data, particularly of the neutron spectrum, to attempt to confirm certain anomalies observed in the calculational analysis of the US experiment,
3. to obtain specific data for the testing and validation of adjustment techniques currently under development for the estimation of pressure vessel damage fluence.

It should be noted that the REPLICA programme has afforded another excellent opportunity for an international intercomparison of measurement and counting techniques involving UK, US and Belgian participants.

The paper contains a brief summary of the experimental data obtained including neutron spectrum measurements within the pressure vessel and the "Void Box". Also included are the results of an initial UK calculational analysis using Monte Carlo techniques.

The agreement between calculated and measured reaction-rates within the pressure vessel is excellent, but it is inconsistent with the results of the similar analysis of the PCA experiment. In this case all organisations participating in a calculational "blind test" of the US data reported underestimates of the measured reaction-rates. In order to investigate this apparent calculational difference, the UK analysis of the PCA experiment is currently being extended with a view to improving the statistical accuracy of the fluxes and reaction-rates scored in the Monte Carlo calculation.

The REPLICA results have also been used in the development and validation of an advanced method of assessment of pressure vessel neutron damage using a new technique in which the parameters in the calculation are adjusted. The technique allows the estimation at points of interest such as the "PV¼T" location to be made by way of measurements in the pressure vessel "cavity" and includes a comprehensive treatment of the uncertainties involved in the process. The method has been applied successfully to the REPLICA data and the results are summarised in the paper.

FINITE-FLAW EXTENSION UNDER THERMAL SHOCK:
TSE-7 TEST AND EVALUATION*

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Fracture-mechanics (FM) models used for investigating the integrity of PWR vessels during overcooling accidents (OCA's) have generally been restricted to two-dimensional (2-D) flaws (long axial and continuous circumferential) even though finite-length flaws presumably are much more likely. This was the case because (1) 2-D flaws were much more amenable to accurate analysis, (2) the approach appeared to be conservative, yet not excessively so, and (3) there were indications that at least in the absence of cladding short flaws would effectively become long flaws.

At the present time there is a need to remove excessive conservatism from our FM models, and fortunately analytical techniques have been developed to the point that the analysis of at least some finite-length flaws is practical. A remaining question is whether short flaws will extend on the surface to effectively become long flaws.

In keeping with the need to remove excessive conservatism from the FM models, it is necessary to include the effect of cladding on the surface extension of short flaws. If cladding can prevent surface extension, it is likely that initially short flaws will not be able to penetrate the vessel wall as the result of an OCA.

A first step in the experimental investigation of the behavior of short flaws was to conduct an experiment without cladding. If the flaw extended on the surface, a follow-up experiment with cladding would reveal the influence of cladding.

Thermal-shock experiment TSE-7 was designed for the investigation of the behavior of a short, axially oriented flaw on the inner surface of an unclad, thick-walled, steel cylinder that was to be subjected to a severe thermal shock. Calculations indicated that the flaw would extend on the surface, possibly the full length of the cylinder in a single event, and subsequent events would extend the depth of the flaw to ~fifty percent of the wall thickness.

The test cylinder was fabricated from A508 class-2-chemistry material. It was tempered at 704°C for 4 h and was machined to the following dimensions: 686-mm ID x 991-mm OD x 1220-mm length. The initial flaw was generated with

*Research sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission under Interagency Agreements 40-551-75 and 40-552-75 with the U.S. Department of Energy under contract W-7405-eng-26 with the Union Carbide Corporation.

the electron-beam weld technique and was essentially semielliptical with a surface length of ∿38 mm and a depth of ∿13 mm. The desired thermal shock was achieved by effectively dunking the test cylinder, initially at 93°C, in liquid nitrogen. The inner surface, which was the only surface in contact with liquid nitrogen, was coated with a thin layer of rubber cement to suppress film boiling and promote nucleate boiling. The ends and outer surface were thermally insulated. Instrumentation included thermocouples to measure the temperature distribution in the wall, COD gages to indicate the time of events and to track the surface extension of the flaw, and ultrasonic instrumentation (UT) to provide an indication of crack depth at the site of the initial flaw.

During TSE-7, there were three major initiation-arrest events, and it appears that during the first event the flaw extended over the entire inner surface, bifurcating many times and extending to or nearly to both ends of the cylinder. The time of this event was 1.53 min, and the arrest depth indicated by the UT instrumentation was ∿32 mm. Subsequent events took place at 2.43 and 3.00 min, and the corresponding arrested crack depths (based on UT) were ∿46 and 52 mm. Posttest measurements indicate a final crack depth of ∿57 mm in the area of the initial flaw and for axial sections of the flaw that extended to the ends of the cylinder. For other sections the final flaw depth was somewhat less.

The results of TSE-7 agree reasonably well with predictions, although the possibility of bifurcation had not been considered. One of the effects of bifurcation was to relieve more of the thermal stress in the cylinder wall than would have occurred otherwise, and this resulted in less radial propagation than would be expected for a single 2-D axial flaw. In a PWR vessel bifurcation would not be expected to have as much effect in this regard because in most cases the propagating crack would tend to be restricted to the rather narrow weld region in which the initial flaw is assumed to reside.

In conclusion the results of TSE-7 indicate once again that under severe thermal-shock loading conditions a short flaw will extend in length to effectively become a long flaw. Furthermore, TSE-7 is a satisfactory prelude to a similar experiment designed to help investigate the effects of cladding on the surface extension of short flaws.

COMPUTATIONAL METHODS FOR FRACTURE ANALYSIS OF HSST PRESSURE VESSEL EXPERIMENTS*

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This presentation summarizes the capabilities and applications of the general-purpose and special-purpose computer programs that are being developed at ORNL for use in conjunction with HSST pressure vessel experiments. The programs perform static analyses of brittle or ductile fracture in two-dimensional (2-D) or fully three-dimensional (3-D) geometries. An emphasis is placed on the OCA/USA code, which is being used for assessments associated with the ORNL pressurized-thermal-shock experiments (PTSE) and on the ORMGEN/ADINA/ORVIRT system which is used for more general analysis.

Extensive computational analyses have been performed with representative material parameters to aid in establishing pressure-temperature transients compatible with proposed pressurized-thermal-shock (PTS) test scenarios and with the capability of the PTS test facility at ORNL. Both linear and nonlinear material models have been employed, as well as 2-D and 3-D finite element representations of crack geometries. Computational economy required application of certain techniques suitable for parametric studies involving the analysis of a large number of transients. These techniques, which include the use of a 3-D superposition method, an inelastic ligament stability assessment, and an upper-shelf arrest analysis, have been incorporated into the previously developed OCA-I code to form the OCA/USA computer program. The fundamental features of the OCA/USA program are discussed, including sample results from applications to the ORNL PTS test configuration.

The finite-element analysis system, ORMGEN/ADINA/ORVIRT, is a three-program system that addresses linear or nonlinear static fracture in 2- or 3-D crack geometries. The ORMGEN program automatically generates a 3-D finite element

*Research sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission under Interagency Agreements 40-551-75 and 40-552-75 with the U.S. Department of Energy under contract W-7405-eng-26 with the Union Carbide Corporation.

model of a user-defined crack configuration in a plate or cylinder. The model includes a core of special wedge or collapsed prism elements at the crack front to produce an appropriate stress singularity at the crack tip. Date files defining the finite-element model have formats which are compatible with the ADINA structural analysis program. Program ORVIRT functions as a post processor to ADINA and employs a virtual crack extension technique to compute energy release rates at specified positions along the crack front. The thermo-mechanical formulation used in ORVIRT is valid for general fracture, including nonplanar fracture, and applies to thermo-elastic as well as deformation plasticity material models. Applications of this system are presented that include (1) pretest planning and posttest analyses of the ITV V-8A test that was concerned with ductile tearing in a low upper-shelf toughness weldment and (2) analyses to evaluate the reliability of OCA/USA models as applied to the PTS test facility. In the case of ITV-8A, comparisons are presented between predictions and test results.

EVALUATIONS OF THE IRWIN β_{Ic} ADJUSTMENT FOR SMALL SPECIMEN FRACTURE TOUGHNESS DATA*

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When small specimens are used to measure the cleavage fracture toughness of pressure vessel steels in the transition range of temperature, specimen thickness size effects and large amounts of data scatter are often observed. The size effects are manifested by an increase in the average value of fracture toughness with decreasing specimen thickness, eventually resulting in a change in fracture mode from cleavage to ductile tearing. It has been shown that a semiempirical adjustment for the interacting effects of specimen thickness, yield stress and toughness originally proposed by Irwin is capable of reducing the calculated values of toughness and data scatter to levels consistent with large specimen test data. This is true for dynamic as well as for static initiation toughness values.

Data recently obtained on HSST-Program materials demonstrate the usefulness of the β_{Ic} adjustment procedure for characterizing the toughness of pressure vessel steels with specimens small in both size and number. Toughness data obtained from 1TCT specimens in the transition range, when adjusted by the β_{Ic} procedure, helped to confirm the selection of a tempering temperature for the cylinder used for thermal-shock experiment TSE-7. Similar data facilitated the pretest estimate of toughness and the formulation of the pressure temperature transient for the first HSST-Program pressurized thermal-shock experiment, PTSE-1.

Additional evidence of size effects can be seen in the comparison of the static 1TCT data in the EPRI data base for HSST Plate 02 with the original large-specimen valid data. Evidence of the difference in fracture mode between large and small specimens is provided by considering the

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cleavage fracture toughness values obtained from two 6TCT specimens of HSST Plate 03 tested in the upper transition range. It is clear that unadjusted values of cleavage fracture toughness in the upper transition range can considerably exceed the toughness value corresponding to the onset of ductile tearing. This poses a problem with respect to the formulation of empirical monotonic representations of toughness, such as the tanh method, because, the increase in tearing resistance with stable crack growth is not considered. Consequently the cleavage fracture toughness may be underestimated as the upper shelf temperature range is approached.

If cleavage instability occurs after maximum load, but maximum load is used as the toughness measurement point, the calculated toughness values still increase with temperature, but a size effect may not be evident within the range of specimen sizes tested. The increase in maximum-load toughness values with temperature implies the occurrence of stable cleavage microcracking, while the lack of a prominent size effect implies that crack extension up to maximum load is due mainly to ductile tearing, the resistance to which is less size dependent than the cleavage fracture toughness.

The occurrence of stable cleavage microcracking in the plastic zone ahead of a macroscopic crack provides an important clue to the physical basis for thickness size effects. Microcracks most frequently occur in ferrite grains when grain boundary iron carbide formations, tightly bonded to the ferrite grains, crack at about two percent strain. The continued propagation of a microcrack requires a tensile stress equal to about four times the yield stress, which can only be developed, at two percent strain, by some combination of high triaxial constraint and a high strain rate. Hence, these two variables exert a controlling influence on the cleavage fracture toughness.

FRACTURE TOUGHNESS CHARACTERIZATION OF IRRADIATED, LOW UPPER SHELF WELDS

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Previous studies have shown that after irradiation steels used in the beltline region of commercial, light water reactor pressure vessels can show both a significant increase in the temperature of the brittle-to-ductile transition and a reduction in Charpy-V (C_V) upper shelf energy level. Certain A 533-B and A 508 submerged arc weld deposits containing a high copper impurity level can exhibit C_V upper shelf energy levels below 68 J (50 ft-lb). For vessels which attain such low toughness, Federal Regulations (10 CFR Part 50) require, as an option, the performance of a fracture mechanics analysis that conservatively demonstrates the existence of adequate margins of safety for continued operation. In the upper shelf region, the vessel material is assumed to exhibit elastic-plastic behavior so that a linear elastic fracture mechanics approach is inappropriate. The question of what is a suitable approach is addressed through Generic Safety Issue A-11 on Reactor Vessel Materials Toughness. In resolving this issue, the NRC has suggested that under elastic-plastic conditions the vessel can be properly evaluated in terms of the tearing instability concept of Paris and others. Currently, this concept is being verified through intermediate vessel tests at Oak Ridge National Laboratory (ORNL) under the Heavy Section Steel Technology (HSST) Program.

The fracture toughness property required for a tearing instability analysis is the J-R curve. The NRC is therefore establishing a data base of J-R curve trends for irradiated pressure vessel steels of low shelf toughness. Of primary interest is A 508 submerged arc weld deposit made with Linde 80 flux and containing a high copper impurity level. This high sensitivity to irradiation associated with the copper, coupled with a low preirradiation toughness associated with the Linde 80 flux, can result in a low upper shelf behavior. Seven welds (61W-67W) of this type were irradiated in the HSST Program. All of the welds are essentially identical to those in operating plants in which the material may exhibit a low upper shelf behavior. Experimental capsules containing compact toughness (CT) specimens of several sizes (e.g., 0.5T-, 0.8T-, 1.6T- and 4T-CT) as well as C_V and tensile specimens were irradiated to fluence levels of 0.6 to 1.5×10^{19} n/cm^2 > 1 MeV to produce C_V upper shelf levels of 54 to 81 J (40 to 60 ft-lb). The J-R curve characterization of these welds was initiated by the Naval Research Laboroatory (NRL) with completion of the program this past year by Materials Engineering Associates (MEA).

J-R curves were obtained in this program by means of the single specimen compliance (SSC) technique, with the CT specimens sidegrooved by 20% to achieve straight crack-front extension. With the 0.5T- and 0.8T-CT specimens, an alternate procedure termed the "double clip gage technique" was developed to determine the J-R curve. This procedure was validated by a round robin program, in which eight laboratories participated. Results have been expressed both in terms of J_D (deformation theory) and

J_M (modified J devised by Ernst). All of the J-R curves were determined within the upper shelf temperature regime to ensure ductile behavior. Tests were conducted up to the reactor operating temperature of 288°C.

From this program, the SSC technique was demonstrated to be an effective method for characterizing the J-R curve of irradiated steels. For these reactor vessel steels exhibiting low upper shelf energy levels, the J-R curves were shown to follow a power-law relationship for small crack extensions, i.e., less than 2 mm. This, in turn, has led the authors to propose a new indexing procedure for J_{Ic}, the J-value at the initiation of crack growth. In some cases a possible size dependence of the R-curve has been seen but the results are inconclusive on this point.

A correlation between J-R curve parameters (J_{Ic} and average value of tearing modulus, T_{avg}) and C_v upper shelf energy has been suggested at 200°C. This finding could enhance the significance of C_v reactor surveillance data with respect to structural integrity. However, J_{Ic} and T_{avg} have demonstrated an inverse relationship with temperature which is not reflected by the C_v upper shelf trend. Therefore, the correlation between C_v energy and R-curve must be adjusted to account for temperature.

EVALUATION OF IN-PLACE THERMAL ANNEALING OF REACTOR PRESSURE VESSELS

W. L. Server

EG&G Idaho, Inc.

Radiation embrittlement of ferritic pressure vessel steels results in an increase in the ductile-brittle transition temperature and a decrease in the upper shelf level of toughness as measured by Charpy impact tests. Both effects of radiation damage have safety consequences recognized by the Nuclear Regulatory Commission (NRC): the temperature shift relates to the pressurized thermal shock issue, and the decrease in Charpy V-notch energy relates to the low upper shelf toughness issue.

A thermal anneal cycle at a temperature well above the normal operating temperature of the vessel can restore most of the original Charpy V-notch energy properties. The degree of recovery may depend upon fluence levels, residual element concentrations (in particular, copper), and alloying element concentrations (such as nickel). Of particular importance are some of the submerged arc welds in a number of older pressurized water reactor vessels that have a high copper content and high radiation damage sensitivity.

In 1967, the Army SM-1A reactor vessel was successfully thermally annealed. This reactor operated at a low temperature of 430°F and was made of a highly radiation-sensitive material. After a short operational time, the embrittlement of the steel was considered too high, and a wet annealing treatment was undertaken at 560-572°F using only nuclear heat. While this anneal cycle recovered much of the Charpy V-notch properties, the approach is not practical for commercial power reactors. Their operating temperature is around 550°F, and the required temperature differential between operation and annealing pushes the annealing temperature well above the 650°F pressure design limit. Therefore, an annealing method using dry heat to soak the beltline region is necessary.

The Electric Power Research Institute (EPRI) has funded an extensive Westinghouse Electric Corporation program to develop in situ thermal annealing procedures and evaluate actual weld metal properties changes. The NRC has funded annealing recovery research conducted by the Naval Research Laboratory (NRL) and Materials Engineering Associates, Inc. The NRC has also funded a project conducted by EG&G Idaho, Inc. at the Idaho National Engineering Laboratory to further evaluate the feasibility of in-place thermal annealing and to develop criteria for regulatory evaluation of a utility annealing program, if one should ever be developed. Both the NRC and EPRI are continuing to support research in this area.

The project at EG&G Idaho can be broken into five major areas:

- Annealing feasibility and possibilities for a full-scale demonstration
- Review of critical weld metal Charpy energy and fracture toughness recovery and reirradiation data
- American Society for Testing and Materials (ASTM) and American Society of Mechanical Engineers (ASME) Standards and Code Committee involvement
- System engineering problems and actual heat treating approaches and procedures
- Residual stresses and possible distortions resulting from an annealing transient.

Significant progress has been made in the first four areas, and work has begun in the last one. The following discussion will briefly address each of these areas.

An industry survey and review of possible candidates for an irradiated vessel annealing demonstration were completed at the end of calendar year 1982. The results from the EPRI/Westinghouse program were reviewed, and discussions were conducted with European nuclear personnel concerned with potential annealing of reactor vessels. The results from this task indicate that annealing of an in-place reactor vessel is feasible, but solvable engineering problems do exist. None of the potentially available irradiated pressure vessels (Indian Point-1, Shippingport, and Humboldt Bay in the United States, KRB-A in West Germany, and BR-3 in Belgium) are similar enough in size, geometry, vessel support, and radiation embrittlement to the commercial pressurized water reactors of concern to warrant an irradiated vessel demonstration effort. In addition, the demonstration could cost \$30 to \$50 million. Thus, such a demonstration is not cost effective. Work is now in progress to define the usefulness and cost of performing an annealing demonstration on a mockup of an unirradiated vessel.

The materials with highest radiation sensitivity in the older, critical reactor vessels are submerged arc weld metals with high copper concentrations. Existing Charpy V-notch and limited fracture toughness data for five such welds were reviewed. Two of the welds came from an NRC/NRL study, and the other three were from an EPRI/Westinghouse investigation. Because of the limited data, different annealing temperatures, irradiation fluence levels, and different weld fluxes, direct comparison of the two sets of data was hindered. However, the results suggest that the two sets of data are generally consistent. The sparce fracture toughness data on the upper shelf do not show the same recovery response as the Charpy V-notch upper shelf energy results. However, significant recovery occurs when annealing at $850^{o}F$ for one week. The actual degree of recovery and the rate of re-embrittlement are not predictable at this time; this information can only be obtained by further experimental and surveillance programs.

An ASTM task group is in the process of upgrading and revising guide ASTM E509-74 entitled "In-Service Annealing of Water-Cooled Nuclear Reactor Vessels." Emphasis in this effort is on the materials and surveillance aspects of annealing rather than system engineering problems. The system safety issues would most likely be covered by ASME Code subcommittees. Currently, the Section XI subcommittee is establishing a Subgroup on Requalification which will consider the issue of vessel annealing. Other groups within Section XI also impact upon this issue. Therefore, EG&G Idaho involvement in both ASTM and ASME continues.

The EPRI/Westinghouse study presented a heat treating approach for dry annealing of a typical Westinghouse vessel. An EG&G Idaho subcontractor, Cooperheat, one of the world's largest heat treaters, has developed a state-of-the-art review and an alternative heat treating procedure. The development of this procedure has required further consideration of health physics and system engineering problems.

One of the main concerns with a localized heat treatment is the degree of distortion that may occur after the annealing cycle. The extent of residual stresses is also an important consideration. The EPRI/Westinghouse program only superficially addressed this aspect of annealing. EG&G Idaho is currently placing a subcontract with Combustion Engineering, Inc. to perform a thermal and structural analysis of the reactor vessel specifically evaluating distortions and residual stresses. This work should be completed by the end of calendar year 1983.

EVALUATION OF REEMBRITTLEMENT RATE FOLLOWING ANNEALING AND RELATED INVESTIGATIONS ON RPV STEELS

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Earlier studies by Materials Engineering Associates (MEA) of Irradiation (I) - Anneal (A) - Reirradiation (R) behavior, using two commerically produced 0.35% Cu content weld deposits, showed a significant promise of the method for limiting radiation embrittlement accrual in reactor vessels. Experimental conditions included a first cycle exposure of 1.3 x 10^{19} n/cm^2, E > 1 MeV, and an intermediate annealing heat treatment of 399°C for 168 hours. Encouraged by the findings, a new set of investigations was mounted to explore the question of the path, i.e., of reembrittlement following intermediate annealing. In addition, the present study was designed to test the significance of welding flux type to subsequent I and IAR behavior.

Two 210 mm thick submerged arc weldments were produced for the study, using Linde 80 and Linde 0091 welding flux types and a common lot of high MnMoNi welding wire. As-deposited copper contents through the weld thickness averaged 0.39% Cu (0.35 to 0.43%, range). Notch ductility tests after postweld heat treatment showed Charpy-V (C_V) 41-J transition temperatures of -23°C and -62°C and upper shelf energy levels of 80 J and 157 J for the respective welds. Investigations of the I and IAR conditions were made using standard C_V specimens and 0.5T-CT compact tension specimens. First cycle irradiation and annealing conditions were patterned after those used for the initial study; however, reirradiation properties were established at fluence intervals of about 0.25 x 10^{19} n/cm^2 to a total (reirradiation) fluence of 1.0 x 10^{19} n/cm^2. Reference properties for the I condition were established at ~ 1.0, 1.5 and 2.0 x 10^{19} n/cm^2. Observations on reembrittlement path and the correlation of notch ductility and fracture toughness (J-R curve) indications of radiation induced embrittlement are described. Differences in irradiation response between the present test materials and those evaluated earlier are also discussed.

Recent findings from related investigations on variable radiation embrittlement sensitivity are also presented. The investigation employed plates from two A 533-B laboratory melts (4-way split) representing statistical variations in copper, phosphorus and tin content. Copper contents were either 0.002% or 0.30%, tin contents were either 0.004% or 0.24% and phosphorus contents were 0.005%, 0.015% or 0.025% (nominal compositions). C_V notch ductility comparisons werre made ~ 2.3 x 10^{19} n/cm^2 at 288°C. Experimental results for these nickel-alloyed plates are compared with findings for an earlier A 302-B plate series (<< 0.4% Ni) to explore the interaction of nickel and phosphorus content in radiation sensitivity development.

INDEPENDENT ASSESSMENT OF TRAC AND RELAP5
CODES THROUGH SEPARATE EFFECTS TESTS

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Independent assessment of TRAC-PF1 (Version 7.0), TRAC-BD1 (Version 12.0) and RELAP5/MOD1 (Cycle 14) that was initiated at BNL in FY 1982, has been completed in FY 1983. Six categories of separate effects tests were simulated with the above codes. These categories are:

1. Critical flow tests (Moby-Dick nitrogen-water, BNL flashing flow, Marviken Test 24),

2. Counter-Current Flow Limiting (CCFL) tests (University of Houston, Dartmouth College single and parallel tube tests),

3. Level swell tests (G.E. large vessel test),

4. Steam Generator tests (B&W 19-tube model S.G. tests, FLECHT-SEASET U-tube S.G. tests),

5. Natural circulation tests (FRIGG loop tests),

6. Post-CHF tests (Oak Ridge steady-state test).

Note that TRAC-PF1 and RELAP5/MOD1 were applied to all of the above categories, whereas TRAC-BD1 was applied only to the CCFL and post-CHF tests.

The conclusions from the BNL code assessment program are given below:

1. <u>Critical Flow</u> - Both TRAC-PF1 (Version 7.0) and RELAP5/MOD1 (Cycle 14) tend to underpredict the <u>subcooled</u> critical flow rate. Although the RELAP5 critical flow rate was in better agreement with the Marviken Test 24 data, neither RELAP5 nor TRAC-PF1 could satisfactorily predict <u>both</u> the break flow rate and the vessel inside pressure. Further improvement of the subcooled critical flow model is recommended for both codes.

2. <u>CCFL</u> - For the single-tube CCFL tests, TRAC-PF1 and TRAC-BD1 yielded much better results than RELAP5/MOD1. However, TRAC-BD1 tends to overpredict the liquid downflow rate for the same gas flow rate, and TRAC-PF1 sometimes predicted anomalous results (e.g., decreasing liquid downflow rate with decreasing gas flow rate for the Dartmouth College 6-inch tube test).

For the Dartmouth parallel tube test, only TRAC-BD1 predicted qualitatively reasonable results. TRAC-PF1 was unable to produce any stable result, and RELAP5/MOD1 was <u>not</u> applied to this test because of its poor prediction of the single tube tests.

3. Level Swell - TRAC-PF1 tends to overpredict the level swell rate and the void fraction below the mixture level, although it predicts the depressurization rate quite well. Higher interfacial shear in the bubbly and bubbly-slug regimes seems to be the reason.

RELAP5/MOD1, on the other hand, tends to underpredict the level swell rate and exhibits some irregularities in the axial void fraction profile. Some errors or lack of smoothing in the interfacial shear package could be the reason.

4. Steam Generator Thermal Performance - For the B&W IEOTSG test, both TRAC-PF1 and RELAP5/MOD1 yielded reasonable average results, although the latter showed some numerical instabilities. Manual control of maximum time step is necessary to avoid these instabilities.

For the B&W OTSG test, TRAC-PF1 underpredicted the exit steam flow rate during a loss-of-feedwater transient. This was caused by the lower initial water inventory due to the lower rate of aspirator steam condensation. An increase in the condensation rate improved the TRAC-PF1 results. For the same test, RELAP5/MOD1 yielded a correct trend for the steam flow rate and primary exit temperature. However, the steam temperature was underpredicted by RELAP5.

For the FLECHT-SEASET U-tube steam generator tests, both TRAC-PF1 and RELAP5/MOD1 codes overpredicted the secondary-to-primary heat transfer rate. One of the main reasons for this discrepancy seems to be the higher interfacial heat transfer rate in the droplet flow regime in both codes, particularly in RELAP5 which did not produce any steam superheat until all the droplets were evaporated. There were also some numerical instabilities in RELAP5 which disappeared when the calculations were repeated with manually controlled time step.

5. Natural Circulation - Both TRAC-PF1 and RELAP5/MOD1 overpredicted the flow rates for the FRIGG-Loop natural circulation tests. However, slightly increased values of wall friction factors and/or form losses would yield reasonable agreement with the data.

The CHF correlations used in both codes, i.e., the Biasi and the W-3 correlations, were unable to predict the CHF condition in the FRIGG test. However, the RELAP4/MOD7 CHF correlation predicted the same condition quite well.

6. Post-CHF Heat Transfer - TRAC-BD1 (Version 12.0) produced the best overall result for a steady-state post-CHF test conducted at ORNL. It predicted the correct CHF location and correct trend for the rod surface temperature. TRAC-PF1 and RELAP5/MOD1, on the other hand, predicted an early CHF and the RELAP5 prediction of rod surface temperature was poor. Improvements in the CHF correlation and the post-CHF heat transfer models are recommended for both the TRAC-PF1 and RELAP5/MOD1 codes.

In summary, for the CCFL and post-CHF tests where all three codes (TRAC-PF1, TRAC-BD1 and RELAP5/MOD1) were applied, TRAC-BD1 yielded the best overall results. Between the TRAC-PF1 and RELAP5/MOD1 codes, TRAC-PF1 produced better results in most cases.

TRAC INDEPENDENT ASSESSMENT FOR PWR ANALYSIS

by

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The Los Alamos National Laboratory is developing the Transient Reactor Analysis Code (TRAC) for application to pressurized-water reactors (PWRs). Several code versions have been released; each new version introduced improvements to existing models and numerics and added new models to extend the applications of the code. The first goal of the code was to analyze large-break loss-of-coolant accidents (LOCAs), and the TRAC-P1A and TRAC-PD2 codes primarily addressed the large-break LOCA. The TRAC-PF1 code contained major changes to the models and trips and to the numerical methods. These modifications enhanced the computational speed of the code and improved the application to small-break LOCAs. The TRAC-PF1/MOD1 code added improved steam-generator modeling, a turbine component, and a control system together with modified constitutive relations to model the balance of plant on the secondary side and to extend the applications to non-LOCA transients.

During the past year we have assessed TRAC-PD2, TRAC-PF1, and TRAC-PF1/MOD1. This work supports applications of the codes to large-break LOCAs, small-break LOCAs, and non-LOCA transients. We have used several experiments from the Loss-of-Fluid Test (LOFT) and Semiscale facilities.

We analyzed LOFT L2-3 and L2-5 with TRAC-PD2; both tests simulated large, double-ended cold-leg breaks. Test L2-3 operated the primary-coolant pumps at approximately constant speed whereas Test L2-5 utilized an early pump trip and a very rapid pump coastdown. The code correctly calculated the hydraulic behavior in both tests. In particular, the calculations agreed qualitatively with the core flows that can be inferred from a number of instruments (no direct measurements of the core flows were made). For Test L2-3 the code correctly calculated the early bottom-up rewet of the core cladding thermocouples except at the lower elevations of the center fuel module. We attributed the discrepancy to the minimum film-boiling-temperature correlation in the code that did not allow the cladding to rewet from the high temperatures as the data indicated. The code calculated the early top-down rewet of the top of the core that Test L2-5 exhibited; we found the extent of the top-down rewet to be sensitive to the balance of flows into the core and to the axial power profile. During the analyses we discovered a deficiency in the modeling associated with the emptying of the accumulator; we made a code update to alleviate the problem.

* This work was funded by the USNRC Office of Nuclear Regulatory Research, Division of Accident Evaluation.

We analyzed with TRAC-PF1 a series of Semiscale natural-circulation and reflux-cooling tests. These analyses demonstrated the capability of the code to calculate single- and two-phase natural circulation, reflux cooling, and the transition between natural circulation and reflux cooling. The results showed that TRAC-PF1 calculated well both the magnitude of the natural-circulation flows as a function of system inventory and the system inventory at which the cooling mode switched from natural circulation to reflux. We also analyzed Test S-NC-6, during which a series of nitrogen injections were made to investigate the effect of a noncondensable gas on the reflux cooling. The code calculated the correct behavior of the system following the nitrogen injections. During the analyses (but not as a result of the analyses) we discovered several errors in the horizontal stratified-flow logic; therefore, we also tested and validated in TRAC-PF1 the modified logic appearing in TRAC-PF1/MOD1. We demonstrated that the wall-condensation heat transfer in TRAC-PF1 was low and showed that the modified correlation in TRAC-PF1/MOD1 was adequate for these analyses.

Our TRAC-PF1 calculations of LOFT L9-1/L3-3 and L6-7/L9-2, which simulated respectively a loss-of-feedwater transient and the cooling phase of the Arkansas Nuclear One Unit 2 turbine-trip transient (each test had compounding additional failures), demonstrated that the code could be applied successfully to non-LOCA transients. Except for low wall-condensation heat transfer, we did not discover any significant problems with code models. However, the analyses did demonstrate the necessity to represent all flow paths, including leakage paths that had been ignored in previous analyses, and all the structural mass and heat-transfer surfaces. This increased detail in the facility model is required to obtain the correct energy inventory and energy distribution, both of which can impact non-LOCA transients. These results point to the need for additional generality in the steam-generator component (provided in TRAC-PF1/MOD1) and in the heat slabs and for a plenum-type component with more connections than currently allowed with a tee component.

We are analyzing with TRAC-PF1/MOD1 Semiscale Test S-UT-8, a small-break LOCA simulation. Although these analyses are continuing, the preliminary results indicate that this code provides significant improvements over TRAC-PF1 in critical-flow modeling, certain constitutive relations, the calculation of the primary-system inventory distribution, and the flexibility of the input.

The ongoing assessment effort at Los Alamos indicates that in general the quality of the code improves as new code versions are released. And while the work continues to indicate needed improvements in the code, the TRAC series of codes currently provides a very flexible analysis tool for treating a wide variety of transients pertinent to PWRs.

CONCLUSIONS FROM THE INDEPENDENT ASSESSMENT OF

TRAC-BD1/VERSION 12

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The boiling water reactor (BWR) Transient Reactor Analysis Code (TRAC) is an advanced best-estimate computer code being developed for the thermal-hydraulic analysis of BWR loss-of-coolant accidents (LOCAs). The developmental work is being conducted at the Idaho National Engineering Laboratory (INEL) by EG&G Idaho, Inc. for the United States Nuclear Regulatory Commission (NRC). The code is structured to address single-and two-phase, nonequilibrium, multidimensional hydraulic conditions. An intermediate version of the code (TRAC-BD1/Version 12) underwent a limited release to interested agencies in April 1982. Then, from April 1982 to March 1983, a series of nine independent assessment studies, consisting of 14 test simulations, was performed at the INEL by EG&G Idaho. The major conclusions from these studies relate to code technology strengths and deficiencies, recommended modeling practice, and run statistics.

The types of facility assessment studies reported are demonstrated by the experimental data bases summarized in Table 1. The scaling of these facilities ranged from 1/676 to full scale. All transients identified in Table 1 are associated with LOCAs and cover a wide range of LOCA behavior.

Among its strengths, the code (TRAC-BD1/Version 12) generally calculates the global behavior of the experiments. Parameters normally simulated will include system pressure, radiation heat transfer in vapor filled volumes, and saturated, single- and two-phase break mass flow. Of particular interest is the code's demonstrated ability to reproduce multidimensional behavior in the core, that is, the calculation of parallel channel effects in those experiments where this multidimensional behavior existed.

The assessment studies have discovered a number of areas where further code development will improve the simulation capability. These areas include: (a) interfacial heat transfer and drag models, (b) subcooled break flow model, (c) reflood and ambient heat transfer models, (d) condensation model, (e) water packing model, (f) jet pump model, and (g) natural circulation wall friction model.

Because of the added complexity, codes having multidimensional capability are significantly more expensive to use than are one-dimensional codes. Effective use of the multidimensional codes requires a good estimation of the potentially required computer resources that are functions of the code, model complexity, type of transient, and the specific computer in use. Several parameters, which reflect the influence of the independent variables on cost, have been developed for each of the subject studies.

As a general rule, the quality of results from advanced safety analysis codes is strongly influenced by the analyst. Assessment studies, where code performance is compared directly with experimental data, are instrumental in determining both good and poor modeling practice. The areas where important modeling practice experience was gained include break flow modeling, minimum nodalization required to capture significant multidimensional behavior, ambient heat loss representation, recommended fuel or heater rod grouping relative to radiation heat transfer effects, and recommended control system characterization for initialization of transient calculations.

TABLE 1. TRAC-BD1 CODE ASSESSMENT DATA BASE

Facility	Types of Experiments
Two Loop Test Apparatus	Integral effect, design basis accidents with full, degraded and no emergency core cooling systems (ECCS). Small breaks with full and degraded ECCS.
30 Degree Steam Sector Test Facility	Integral effect, multidimensional parallel channel behavior during refloods of a BWR/4 and a BWR/6.
Rig of Safety Assessment-III	Integral effect, small break with failure of the high pressure core spray system.
FRIGG	Separate effect, natural circulation versus power, to burnout. Core void distribution versus forced mass flow.
GOTA	Separate effect, steady state radiation heat transfer. Constant pressure core spray cooling.
Marviken	Separate effect, heated vessel blowdown through large and small, length over diameter, nozzles.

SAND 83-1834A

RELAP5/MOD1 ASSESSMENT CONCLUSIONS*

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The RELAP5 independent assessment project at Sandia National Laboratories is part of an overall effort funded by the NRC to determine the ability of various systems codes to predict the detailed thermal/hydraulic response of LWRs during accident and off-normal conditions. The RELAP5/MOD1 code has been assessed at SNLA against a variety of test data from both integral and separate effects test facilities. The completed analyses include

-- a PKL natural circulation test series,
-- a FLECHT SEASET steam generator separate effects test,
-- three LOBI large break tests,
-- LOFT small break test L3-6/L8-1,
-- two Semiscale Mod-2A natural circulation tests,
-- LOFT turbine trip transient L6-7/L9-2,
-- four Semiscale Mod-3 small break tests,
-- LOFT loss-of-feedwater transient L9-1/L3-3,
-- LOFT intermediate break transients L5-1/L8-2, and
-- a B&W OTSG separate effects test.

Results from a number of integral tests show that the primary system response (e.g., pressure and break flow) is well-predicted in a wide variety of transients; most observed discrepancies can be attributed to known problems already being addressed by the code developers, or to difficulties with initial and/or boundary condition specifications. The secondary system response (e.g., steam generator pressure and temperatures) is usually not correctly calculated in either integral or separate effects test analyses, with the predicted depressurization always being too rapid because of calculated saturation conditions throughout the secondary (rather than subcooled/saturated layers).

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

Core heatup response (e.g., dryout and PCT) is generally also well-predicted, particularly during the blowdown phase of large break LOCAs (± 30 K and ± 50 K for large and intermediate break LOCAs, respectively); however, problems have been encountered in both intermediate and small break analyses with unphysical heatup interruptions due to MOD1's simplistic dryout/rewet criterion. Given the lack of specific reflood algorithms in MOD1, it is not surprising that core quench is not calculated accurately in large breaks; we have, however, seen a number of problems with the accumulator injection which would preclude calculating reflood correctly even when such a reflood package is implemented in later versions of RELAP5.

All results show that good steady state initial and/or operating conditions are readily obtained, given an adequate facility description and some user experience or guidelines, although problems are usually encountered in the steam generator secondary sides. Several classes of unphysical oscillations (temperature, mass flow and pressure) have been found that can often be eliminated by reducing the time step used, implicating the code time step control algorithm, and the quantum nature of the allowed time step in the code algorithm has been seen to result in inefficient run times. A number of MOD1 calculations indicate that calculated two-phase natural circulation flow rates are always high, and that code mass and energy conservation problems can be encountered through improper user modelling of bypass and leakage flow paths. Both these last items are being addressed in new or altered code models for future versions of RELAP5.

The MOD1 code is not readily portable to computers other than the CYBER 176 on which it was originally developed, and significant amounts of programming time and effort are needed for any such desired conversion. The code documentation has been found to contain a number of errors and/or omissions which have not been corrected to date.

TRAC CODE IMPROVEMENTS*

by

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The Transient Reactor Analysis Code (TRAC), being developed at Los Alamos, has been modified and improved extensively since the first version TRAC-P1A was released in 1979. The initial versions, TRAC-P1A and TRAC-PD2, were primarily large-break loss-of-coolant accident (LOCA) codes although TRAC-PD2 and TRAC-PD2/MOD1 could simulate other types of transients. The early versions of TRAC employed a five-equation drift-flux model in the one-dimensional components and a six equation two-fluid model in the three-dimensional vessel. TRAC-PF1 (released in 1981) had a multi-fluid model (steam, liquid, and noncondensable gas) both in the one-dimensional and in the three-dimensional components to model better the countercurrent stratified flows present during reflux cooling and certain small-break accidents. TRAC-PF1 had implicit two-step numerics in the one-dimensional algorithm to permit large time steps to be taken for slow transients as well as numerous other changes that made this code version better suited than previous editions to run non large break LOCA transients.

TRAC-PF1/MOD1, which is currently available, has many new features that permit an even wider range of accidents to be simulated. TRAC-PF1/MOD1, in addition to the improvements added to TRAC-PF1, has

1. a complete balance-of-plant capability (including turbine, feedwater, and condenser components),
2. boron tracking with feedback,
3. full trips and controller capability,
4. improved steam generator, and
5. improved condensation modeling.

These additions permit TRAC-PF1/MOD1 to model almost any type of accident in

*This work was funded by the USNRC Office of Nuclear Regulatory Research, Division of Accident Evaluation.

any type of pressurized-water reactor (PWR). Three national laboratories, Los Alamos, Brookhaven, and Sandia, independently will assess this code against several Loss-of-Fluid Test (LOFT) and Semiscale experiments during fiscal year 84.

The major part of TRAC development is complete with the introduction of TRAC-PF1/MOD1. In the next fiscal year we will develop an implicit multistep procedure for the three-dimensional module to reduce further the running time for complex problems as well as continue to implement constitutive equation improvements. Because over 60 organizations world wide possess recent versions of TRAC, a large part of our effort will be devoted to maintenance. TRAC-PF1/MOD1 is a mature code, well assessed, and capable of analyzing most thermal-hydraulic problems of interest to reactor safety.

TRAC-BD1/MOD1 - A Best Estimate Analysis Code for Boiling Water Reactor Systems

Walter L. Weaver

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The TRAC-BWR Code Development Section of the Nuclear Safety Methods Division at the Idaho National Engineering Laboratory (INEL) is developing several versions of the Transient Reactor Analysis Code (TRAC) to provide the Nuclear Regulatory Commission (NRC) and the public a best estimate capability for the analysis and modeling of postulated accidents and transients in boiling water reactor (BWR) systems. This program is unique among advanced code development projects in that it focuses on the hardware, thermal-hydraulic, and heat transfer phenomena that distinguish BWR systems and BWR system response. In addition to providing a best estimate analysis capability for BWR systems, the code can also be used to address current licensing concerns such as anticipated transients without scram (ATWS) or the scram discharge small break loss-of-coolant accident (LOCA). It also provides analytical support to NRC experimental safety programs. The success of the program is attributable in part to the continuing participation of the General Electric Company as part of the Full Integral System Test (FIST) Experimental Program cosponsored by General Electric, the NRC, and the Electric Power Research Institute (EPRI).

These codes are based on a developmental version of TRAC supplied to the INEL by the Los Alamos National Laboratory (LANL) containing a two-fluid hydrodynamic mode in both one- and three-dimensional flow components. New models required for BWR analysis have been developed by the INEL in cooperation with the code development group at the General Electric Company. TRAC-BD1, the first publicly released code version, provides a basic capability for the analysis of LOCA in BWRs and related experimental facilities. The mission of the recently released code version, TRAC-BD1/MOD1, has been expanded to include operational transients and anticipated transient without scram (ATWS) as well as an improved LOCA analysis capability. The models developed to accommodate this increased scope include a reactivity feedback model with a boron tracking capability for use in the existing point kinetics neutronic model and a control systems model. A new interfacial shear model, based on the work of Ishii, a subcooled boiling model and a direct moderator heating model as well as a two-phase level tracking model, have been implemented to improve the prediction of the void distributions. A moving mesh reflood model similar to that in TRAC-PD2 has been implemented for quench front tracking on both fuel rods and on the BWR fuel channel walls. In addition to the development of the models that allow a more accurate modeling of the reactor core, balance of plant component models such as turbine, feedwater heater, and condenser models have been developed. A simple containment simulation model with noncondensible gas tracking capability completes the models required to simulate a complete BWR reactor system with all of its engineered safety systems.

Each of the models in the TRAC-BWR codes was developed under a comprehensive quality control procedure that ensures that each model is thoroughly tested and documented during its development as well as when it is integrated with other models to form a new code version. Each released code version undergoes extensive developmental assessment to ascertain whether the individual models are working correctly and do not have detrimental interactions. The developmental assessment test cases include both separate effects and integral systems tests which exercise the heat transfer and hydraulic models in the code. Selected results of the developmental assessment of TRAC-BD1/MOD1 will be presented.

Work currently in progress toward the final code version, TRAC-BF1, include a one-dimensional neutron kinetics model based on the analytic nodal method and a Courant limit violating hydrodynamics model similar to the Two-Step method contained in TRAC-PF1. The addition of Courant violating numerics should greatly reduce the cost of analyses, particularly for transients of long duration such as an ATWS.

RELAP5/MOD2

Victor H. Ransom

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RELAP5/MOD2 is a pressurized water reactor (PWR) system transient analysis code that has been designed for use in the U.S. Nuclear Regulatory Commission (USNRC) Safety Research and Regulatory Programs. The code is an extension of the RELAP5/MOD1 capability to include simulation of PWR transients up to the point of fuel damage. For analysis of transients involving fuel damage, RELAP5 is being coupled with the SCDAP code, which will permit analysis up to the point of vessel melt through.

The improved capability of RELAP5/MOD2 compared to the earlier version is a result of continued improvement of the hydrodynamic models, added user convenience features, and addition of new component models. To a large extent, the new capability has been obtained through logical extension of the RELAP5/MOD1 modeling. As a result, the basic code structure is the same and system model input decks developed for RELAP5/MOD1 are compatible with the RELAP5/MOD2 input requirements, with minor exceptions.

The development of RELAP5/MOD2 was completed during May of 1983. The code has been tested using a matrix of 54 developmental assessment problems and a draft of a two volume users manual was completed during September of 1983. Results of the developmental assessment will be documented during FY-1984.

The hydrodynamic modeling improvements that have been added in RELAP5/MOD2 include: (a) revised interphase drag and wall friction formulations that are based on dynamic flow regime maps and include stratification effects in horizontal components; (b) nonequilibrium wall heat transfer and interphase mass transfer models have been added that are coupled to the basic flow regime maps and include the nonequilibrium regimes of subcooled boiling, post critical heat flux (CHF) film boiling, and condensation on subcooled emergency core cooling (ECC) injection; and (c) addition of a second energy equation to eliminate the need for the a priori assumption that one phase exists at the saturation temperature. This latter extension was required for accurate modeling of nonequilibrium states that result from system repressurization. This extension was achieved without significant modification of the numerical solution scheme and has had a negligible effect on the code execution time.

Improvements in other system models include: (a) addition of the 1978 ANS decay heat standard and boron, moderator, and doppler feedback to the reactor kinetics model; (b) addition of delay, lead/lag, and shaft components to the system controls model; (c) addition of a two-dimensional conduction and fine mesh rezoning model to the heat conduction package for use in quench front tracking during reflood; (d) addition of a dynamic gap conductance model for fuel thermal behavior modeling; and (e) an improved separator with user specified carryover and carryunder functions.

New component models have been added for turbines, generators, coupling capability between turbines, pumps, or motors, and a crossflow junction component for modeling quasi two-dimensional flow paths; such as at tees and crossflow between core channels.

During FY-1984 an extended developmental assessment will be conducted utilizing integral experiments to further assure correct functioning of the code. At the conclusion of this effort the code will be available through the USNRC for application work. Additional new models will be installed in the code using updates/new cycles as new requirements are identified. The models that are planned at present include a CCFL model for modeling of steam generator flooding, a level tracking model for more accurate modeling of repressurization, and a relief valve input processing model for increased user convenience. An optional more implicit numerical scheme will be added for steady state and slow transients.

Code configuration control and maintenance will continue during FY-1984 with increased emphasis on portability of the code. User support will be provided to the NRC assessment and applications tasks and for publicly released versions of RELAP5.

COBRA/TRAC APPLICATIONS PROGRAM

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SUMMARY

The COBRA/TRAC computer program has been developed at the Pacific Northwest Laboratory by the U.S. NRC to predict the thermal-hydraulic response of nuclear reactor primary coolant systems to small- and large-break loss-of-coolant accidents and other anticipated transients. It was derived by implementing COBRA-TF as the vessel component in TRAC-PD2.

The main objectives of the program in FY83 were:

- apply COBRA/TRAC to PWRs equipped with upper head injection (UHI)
- apply COBRA/TRAC to a non-UHI PWR
- assess the small break prediction capability.

Previously, COBRA/TRAC simulated 200% cold-leg break LOCAs in a reactor using a three-dimensional mesh with over 550 fluid cells. This year more emphasis was placed on improving the small-break analysis capability. Since small-break transients last longer, a simpler model was needed. A one-dimensional model of the PWR/UHI vessel was created directly from the input for the three-dimensional model by lumping the flow areas and volumes of the three-dimensional mesh cells into a one-dimensional mesh for each region of the reactor vessel.

A comparison was made with the three-dimensional calculation for a 200% cold-leg break LOCA to assess the limitations of the one-dimensional mesh. Both calculations were run using the same version of COBRA/TRAC. Although there were differences between the two calculations, the one-dimensional calculation predicted the same peak clad surface temperature (within 5°F) as the three-dimensional calculation.

A large-break LOCA in a non-UHI PWR was also performed. This calculation used the same one-dimensional mesh as the UHI calculation but incorporated the differences of the non-UHI plant.

Two experiments in the Semiscale test facility were simulated to assess the small-break analysis capability. Test S-UT-2 was a 10% cold-leg break with UHI. The COBRA/TRAC results gave a reasonable match with the measured data. The important phenomena, such as pump suction clearing, upper head liquid level, downcomer liquid level equalization, and core liquid level recovery were predicted. Results from Semiscale test S-UT-5, a 2.5% break with UHI, will also be presented.

STATUS OF CCTF TEST PROGRAM

Yoshio MURAO, Tadashi IGUCHI, Jun SUGIMOTO
Hajime AKIMOTO, Tsutomu OKUBO, Kazuharu OKABE

Japan Atomic Energy Research Institute

1. Introduction

The Cylindrical Core Test Facility (CCTF) is one of the facilities of the Large Scale Reflood Test Program which was initiated in April, 1976. The first series of the CCTF test (CCTF CORE I TEST) was completed in April, 1981 and the second series (CCTF CORE II TEST) was initiated in April, 1982. In the test, the following has been intended to be examined: (1) The conservativeness of the assumption of the safety analysis with the Evaluation Model (EM) code. (2) The refill and reflood phenomena for analytical modeling of thermo-hydrodynamics in the core and the system. (3) The validity of the models in the EM code and the application to the BE code development.

The previous tests of the cold leg injection type ECCS showed the following results: (1) The system model assumed in the EM analysis was similar to or more conservative than the observed in CCTF CORE I TEST except for the downcomer effective head. (2) Core hydrodynamics can be treated one-dimensionally. (3) Much water was accumulated above quench front,and the flow above quench front was recognized as slug flow. (4)A good prediction of the core heat transfer was made by using the slug flow model above quench front and the heat transfer correlation for the slug flow.

In this paper, presented are the quantitative evaluation of the REFLA code and the discussion of some CCTF CORE II TEST results. The REFLA code consists of REFLA-1D core code developed with the results of small scale reflood tests and a simple system model developed with the results of the CCTF CORE I TEST. The CCTF CORE II TEST was performed for developing more realistic model for the cold leg injection type ECCS and model for the alternative ECCS.

2. Quantitative evaluation of REFLA code

In order to evaluate core code, the core flooding rate and the liquid temperature and pressure at the core inlet were given at each calculational time step through the whole transient of the calculation. The core flooding rate was obtained from the mass balance calculation.

Figure 1 shows the calculated and measured clad surface temperatures at five elevations. The overall good comparisons are obtained except for the initial core cooling during the period of about 100 s after the reflood initiation.

The evaluation of the system code was performed. Figure 2 shows the calculated flooding rate and collapsed water levels in the core, downcomer and upper plenum compared with the measured data. The measured flooding rate was obtained by using the smoothed differential pressure readings. The calculated results agree well with the measured on an average except for the water accumulation in early period. The prediction of clad surface

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The work performed under contracts between Atomic Energy Bereau of Japan and JAERI.

temperatures was also good.

3. Experimental investigation of CCTF CORE II TEST for further modeling

In the previous tests, a significant large pressure drop was observed at the broken cold leg nozzle. It was believed that this caused the enhancement of the core heat transfer due to the pressurization of the pressure vessel and increase of the core flooding rate. In order to examine the effect of pressure drop at the broken cold leg nozzle, a high Low Pressure Coolant Injection (LPCI) flow rate test was performed, since the pressure drop was estimated to be reduced under high LPCI flow rate. Figure 3 shows the results of the test. The core cooling is found to be lower in the higher LPCI test. Because,in the EM code, the pressure drop is considered to be small, a single failûre assumption of LPCI pump does not introduce underestimation of peak clad temperature in the licensing calculation.

In order to know the model for the alternative ECCS, upper plenum injection tests ,downcomer injection, and vent valve simulation tests were performed. Figure 4 shows the results of the upper plenum injection test. The core cooling behavior seems to be identical with the cold leg injection. In detail,the core cooling is slightly different from the cold leg injection. Similar differences are found in other alternative ECCS tests. Therefore some improvements of the model are neccesary for the system code.

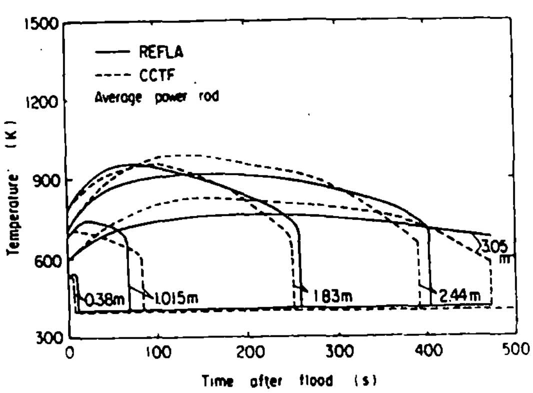

Fig. 1 Comparison of clad temperature histories for base case test between REFLA prediction and CCTF data

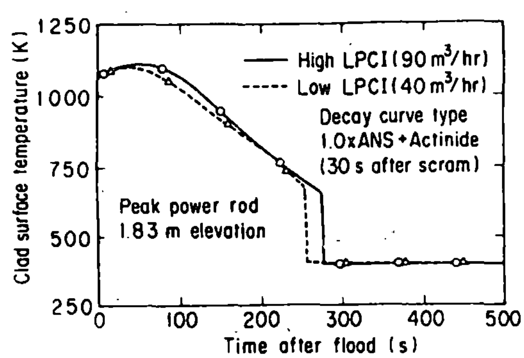

Fig. 3 LPCI rate effect on the clad surface temperature at the midplane of a peak power rod

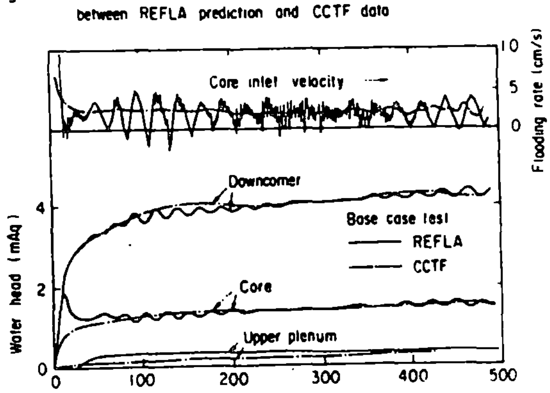

Fig. 2 Comparison of water heads and core flooding rate between REFLA prediction and CCTF data

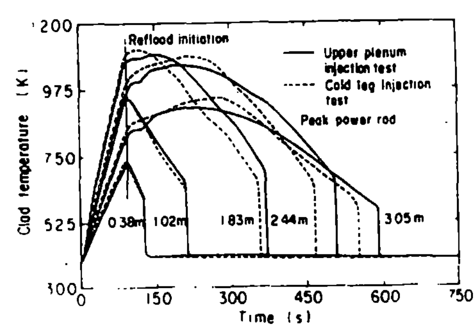

Fig. 4 Upper plenum injection effect on the clad surface temperature

SCTF CORE-I REFLOOD TEST RESULTS

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The Slab Core Test Facility (SCTF) Test Program is being performed at Japan Atomic Energy Research Institute (JAERI) as one of the subprograms of the Large Scale Reflood Test (LSRT) Program. In the Cylindrical Core Test Facility (CCTF) Test Program, another part of the LSRT Program, simulation in system behavior for the last part of the blowdown, refill and reflood phases of a postulated loss-of-coolant accident (LOCA) in a pressurized water reactor (PWR) is the primary concern. On the other hand, the major objectives in the SCTF Test Program are to clarify the following items :

(1) Two-dimensional thermal-hydraulics in a wide core (chimney effect, sputtering-induced fluid behavior, blockage effect, etc.),
(2) Flow interaction between core and upper plenum (fall-back, core entrainment, etc.), and
(3) Hot leg carryover characteristics (upper plenum entrainment/de-entrainment, counter-current flow in hot leg, etc.).

After investigation of the SCTF Core-I test results, the following conclusions were obtained :

(1) Core Cooling Enhancement by Chimney Effect
  i. The chimney effect enhances the cooling of the high power bundles in the central region of the core and degrades that of the peripheral low power bundles. As a result, the core thermal response is flattened radially in spite of the radial power distribution and thus the peak clad temperature (PCT) becomes lower.
  ii. Heat transfer coefficient at a given elevation before the quench is dependent on water fraction, $1-\alpha$, as shown in Fig. 1.
  The direct effect of the various test parameters such as core heating power and its radial distribution, core flooding rate and core inlet water subcooling on the core heat transfer characteristics is weak.
  iii. Core cooling enhancement at the high power bundles is considered not to be directly caused by increase in local water mass velocity but indirectly caused through the increase in water fraction with respect to the distance from the approaching quench front.

(2) Horizontal Cross Flow in Core
  1. Direction of horizontal cross flow in the middle core is from the central region to the peripheral region before the quench and from the peripheral region to the central region after the quench as shown

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The work was performed under contract with the Atomic Energy Bureau of Science and Technology Agency of Japan.

in Fig. 2.

ii. Significant oscillations in the horizontal cross flow can be seen especially below the quench front.

iii. Horizontal cross flow is so strong that it is comparable with the vertical main flow even in the flat power distribution.

(3) Blockage Effect

i. Blockage effect on the turnaround temperature and quench time is very small even in the case of two bundle size local blockage with 60% blockage fraction.

ii. Although quench time becomes slightly shorter at just downstream of the blockage sleeves when the core flooding rate is higher than a certain value, it becomes slightly longer when the core flooding rate is lower than the value. This result suggests the core flow bypassing around the large local blockage.

(4) Effect of Oscillations in Core Inlet Flow Rate

i. When U-tube oscillations occur between the core and the downcomer, the core heat transfer is enhanced.

ii. The heat transfer enhancement is considered to be caused by the oscillations through the increase in water fraction above the quench front due to the water splash-up effect.

iii. Relationship between heat transfer coefficient and water fraction above the quench front is not so much affected by the oscillations.

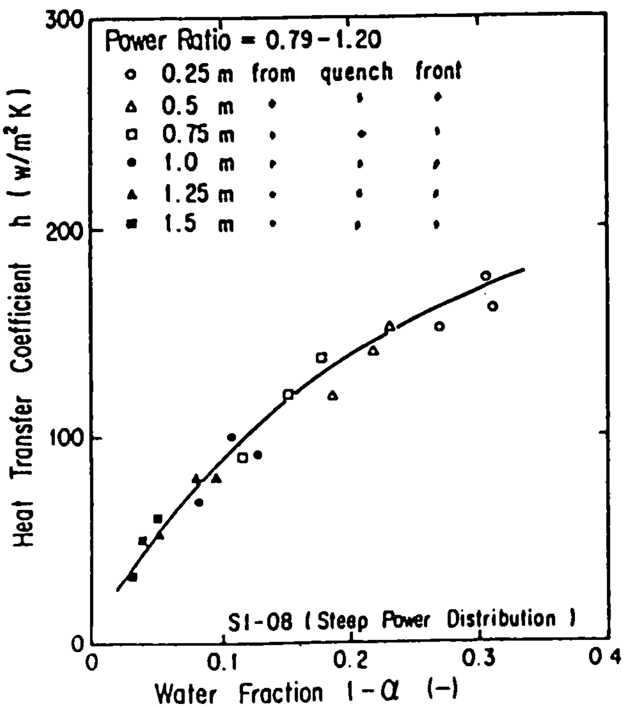

Fig. 1 Relationship between heat transfer coefficient and water fraction above the quench front

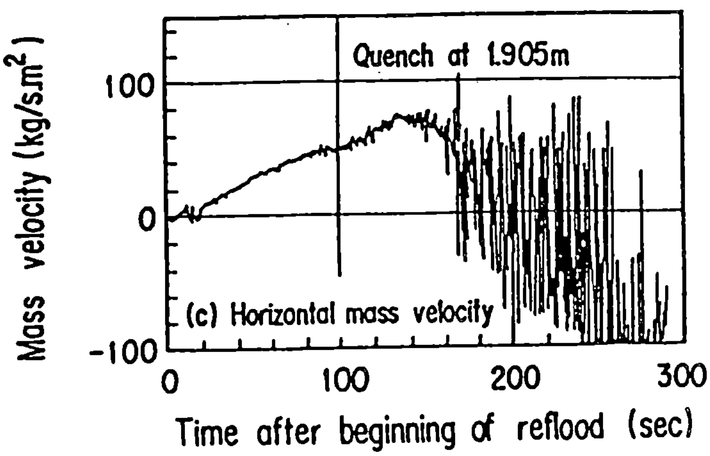

Fig. 2 Estimated horizontal mass flow rate with homogeneous two-phase flow model

TRAC ANALYSES FOR CCTF AND SCTF TESTS AND UPTF DESIGN/OPERATION*

by

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The 2D/3D Program is a multinational (Germany, Japan, and the United States) experimental and analytical nuclear reactor safety research program. Its main purpose is the investigation of multidimensional thermal-hydraulic behavior in large-scale experimental test facilities having hardware prototypical of pressurized water reactors (PWRs). The Japanese are presently operating two large-scale test facilities as part of this program: the Cylindrical Core Test Facility (CCTF) and the Slab Core Test Facility (SCTF). The CCTF is a 2000-electrically-heated-rod, cylindrical-core, four-loop facility with active steam generators primarily used for investigating integral system reflood behavior. The SCTF is a 2000-electrically-heated-rod, slab-core (one fuel assembly wide, eight across, and full height), separate-effects reflood facility. Both facilities have prototypic power-to-volume ratios, preserving full-scale elevations, and are much larger than any existing facilities in the United States (including LOFT). The German contribution to the program is the planned Upper Plenum Test Facility (UPTF), a full-scale facility with vessel, four loops, and a steam-water core simulator. All of these facilities have more instruments than any existing facilities: conventional instrumentation data channels alone are in excess of one thousand in each facility. The United States contributions to the program are the provision of advanced two-phase flow instrumentation and analytical support.

The Los Alamos National Laboratory is the prime contractor to the NRC in the latter activity. The main analytical tool in this program is the Transient Reactor Analysis Code (TRAC), a best-estimate, multidimensional,

*Work performed under the auspices of the US Nuclear Regulatory Commission.

nonequilibrium, thermal-hydraulics computer code developed for the NRC at Los Alamos. Through code predictions of experimental results and calculations of PWR transients, TRAC provides the analytic coupling between the facilities and extends the results to predicting actual PWR behavior.

Results from this program already have addressed, and will continue to address, key licensing issues including: scaling, multidimensional effects, downcomer bypass and refill, reflood, steam binding, core blockages, alternate emergency core cooling systems (ECCS), small-break phenomena, and code assessment.

During the previous year, the application of TRAC-PF1 to the 2D/3D program was highlighted by fine-node, large-break loss-of-coolant-accident (LOCA) calculations of both the US/Japanese and German PWR reference reactors. The calculations utilized new input models that more correctly represent plant geometry and operating conditions; for the US/Japanese PWR the input model was based upon the newer Westinghouse 17x17 type plants. These LOCA analyses included a double-ended cold-leg break for both the Westinghouse plant and the German KWU combined ECC injection plant. A hot-leg break LOCA calculation was also completed for the KWU plant. The UPTF design/operation calculations were a full-system representation including TRAC-PF1 modeling of the core simulator feedback control systems. This included a base case system calculation and parameters effects calculations. The SCTF analyses and tests covered gravity driven reflood operation for the first time. TRAC predictions were provided for these tests, and the code aided in the selection of operating conditions for the SCTF gravity tests. For the CCTF Core-II, analyses were provided for the base case test (Run 53) as well as for some of the parametric effects tests. These included the low core stored energy test (Run 51) and the flat radial power test (Run 64).

In conclusion, the Los Alamos analysis effort is functioning as a vital part of the 2D/3D program. The CCTF and SCTF analyses have demonstrated that TRAC-PF1 can correctly predict multidimensional, nonequilibrium behavior in large-scale facilities prototypical of actual PWRs. Through these and future TRAC analyses the experimental findings can be related from facility to facility; and more importantly, the results of this multinational research program can be directly related to licensing concerns affecting actual PWRs.

NRC PLANT ANALYZER DEVELOPMENT AT BNL*

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The objective of this program is to develop an LWR engineering plant analyzer capable of performing realistic and accurate simulations of plant transients and Small-Break Loss of Coolant Accidents at real-time and faster than real-time computing speeds and at low costs for preparing, executing and evaluating such simulations. The program is directed toward facilitating reactor safety analyses, on-line plant monitoring, on-line accident diagnosis and mitigation and toward improving reactor operator training.

The AD10 of Applied Dynamics International, Ann Arbor, MI, a special-purpose peripheral processor for high-speed systems simulation, is programmed through a PDP-11/34 minicomputer and carries out digital simulations with analog hardware in the input/output loop (up to 256 channels). Analog signals from a control panel are being used now to activate or to disable valves and to trip pump drive motors or regulators without interrupting the simulation. An IBM personal computer with multicolor graphics capabilities and a CRT monitor are used to produce on-line labelled diagrams of selected plant parameters as functions of time. The IBM personal computer supplements a Tektronix storage oscilloscope terminal and it serves to establish the requirements for a future acquisition of a graphics terminal.

This minicomputer technology for high-speed systems simulation is combined with advanced modeling techniques for neutron kinetics, thermal conduction, nonhomogeneous, nonequilibrium two-phase flow coolant dynamics, acoustics in the steam lines and for the flow through turbines, condensers, heat exchangers and pumps in the balance of the plant. Point kinetics with corrections for space-time effects during scram and with all feedback mechanisms simulate power generation. A lumped-parameter conduction model produces, with one ordinary differential equation, seven fuel element temperatures of the radial temperature distribution in one axial location. A four-equation two-phase flow model predicts vapor slip and phase separation as well as non-equilibrium boiling or condensation.

A BWR power plant simulation has been implemented. The simulation encompasses the pressure vessel with three channels in the core and the total of 55 computational cells for coolant dynamics and 24 for conduction, further the steam line (10 cells), the turbines, the condensers, the feedwater preheaters, the feedwater drive turbines and pumps, a recirculation loop with motor-generator and motor-pump assemblies and the suppression pool. Controllers have been implemented for regulation of feedwater flow, turbine entrance pressure and recirculation flow. Safety and relief valves, scram and trips for

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

feedwater heaters and pumps, for recirculation pumps and emergency coolant injections are also represented. The resulting code is called HIPA-PB2 for High-Speed Interactive Plant Analysis of Peach Bottom II.

Seven transients, including the severe licensing base transient with compound malfunctions, but not including the SB-LOCA transient, have been simulated successfully as of August 1983. SB-LOCA simulations are planned for the next fiscal year. Figures 1 and 2 below show results for ATWS simulation. Figure 1 shows pressure and steam line vapor mass flow rates at the first five seconds, Figure 2, the pressure, the vapor mass flow rate and the total fission power during the first 200 seconds after MIV closure. The graphs were produced on-line with the Tektronix Oscilloscope-Printer at ten times real-time simulation speed. All transients are carried out at computing speeds up to ten times real-time, using the maximum base time step of approximately 50 milliseconds and two substeps for stiff equations. HIPA-PB2 results have been compared with results from large systems code calculation. Good agreement has been obtained.

It has been demonstrated that the AD10 can outperform the CDC-7600 by two orders of magnitude in computing speed at much lower cost.

Reference

NUREG/CR-2331, BNL-NUREG-51454, Vols. 1 and 2.

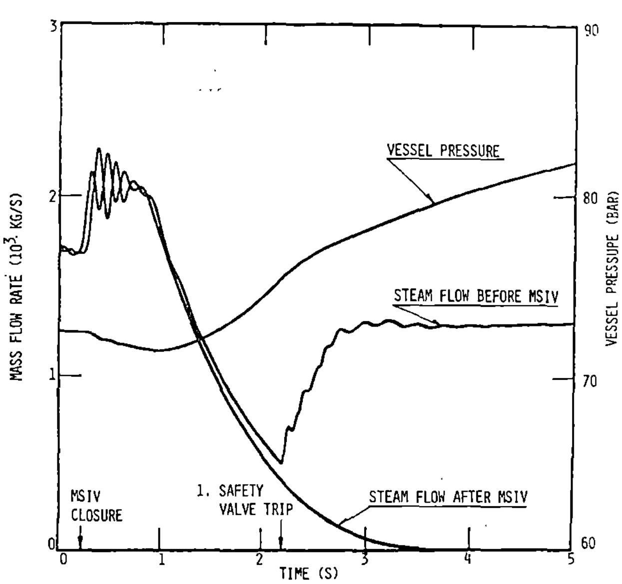

Figure 1 Early System Response to ATWS Transient.

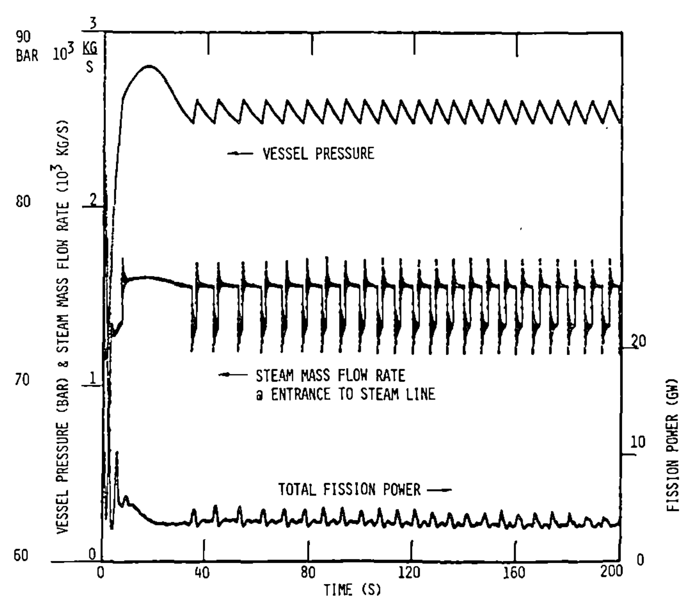

Figure 2 Long-Term Response to ATWS Transient.

NUCLEAR PLANT ANALYZER DEVELOPMENT AT INEL

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Kenneth D. Russell
Howard D. Stewart

EG&G Idaho, Inc.

The Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission (NRC) has sponsored development of a software-hardware system called the Nuclear Plant Analyzer (NPA). This paper describes the status of the NPA project at the INEL after one year of development.

When completed, the NPA will be an integrated network of analytical tools for performing reactor plant analyses. The software utilized by the NPA will consist of three major components: (a) the common user interface (CUI), (b) the reactor behavior modeling codes (such as RELAP5, TRAC-PF1, and TRAC-BD1), and (c) data libraries (such as the Nuclear Plant Data Bank and the NRC/DAE Data Bank). The hardware will consist of a nationwide network of mainframe computers and user consoles interconnected by way of the public switched telephone network.

Development of the NPA in FY-1983 progressed along two parallel pathways; namely, conceptual planning and software development. Regarding NPA planning, an extensive effort was conducted to define the function requirements of the NPA, conceptual design, and hardware needs. The general requirements determined for the NPA were: it must have an accurate calculation capability--accuracy cannot be compromised; it must be fast running, permit dynamic control of the calculation, and enable easy but thorough comprehension of analytical results; it should provide a ready capability to analyze abnormal transients in any plant; it must give the analyst access to the Nuclear Plant Data Bank (NPDB) for reference data and for modifying or building plant analytical models; it must be user oriented in order to reduce analysis labor and time requirements and assure quality. The detailed functional requirements, conceptual design, and hardware needs were then determined in support of the general requirements.

A report documenting the results of this study was issued for comments in September 1983. Upon the basis of this report, the initial and future direction of the software and hardware was defined.

Regarding software development conducted in FY-1983, all development was aimed toward demonstrating the basic concept and feasibility of the NPA. Nearly all software was developed and resides on the INEL twin Control Data Corporation 176 mainframe computers. A "Type-1" non-intelligent work station (an alphanumeric control terminal and an intelligent color graphics device) was assembled as the user console to initiate, perform, and display results of the reactor system analyses of the NPA. The RELAP5 code was used as the calculational tool to simulate reactor coolant system behavior. Using the Type-1 workstation configuration (Figure 1), the NPA was run in both an interactive mode, where the scenario modeled by the simulation code (RELAP5) was redirected as desired during the simulation, and in a playback mode, where previously generated RELAP5 calculations were colorfully displayed for review. These NPA capabilities were successfully demonstrated in early October of this year to the NRC and to the co-developers of the NPA.

Future software enhancement will first include the continued development of the Type-1 workstation configuration with the integration of the data libraries that are not now part of the NPA, and then the development of software to provide users some local intelligence in the form of a minicomputer as part of the work station (i.e., Type-2 workstation). It is also envisioned that other types of reactor behavior simulation codes will be integrated into the NPA, such as the SCDAP and CRAC codes, which will expand the analysis capabilities into the severe damage area.

In summary, the basic concept of the NPA was well defined in FY-1983, and was successfully demonstrated using a non-intelligent Type 1 workstation to access and utilize the common user interface and the RELAP5 systems analysis code. Future software development will be directed toward integrating more analysis codes, adding intelligent work station capabilities and incorporating the data libraries.

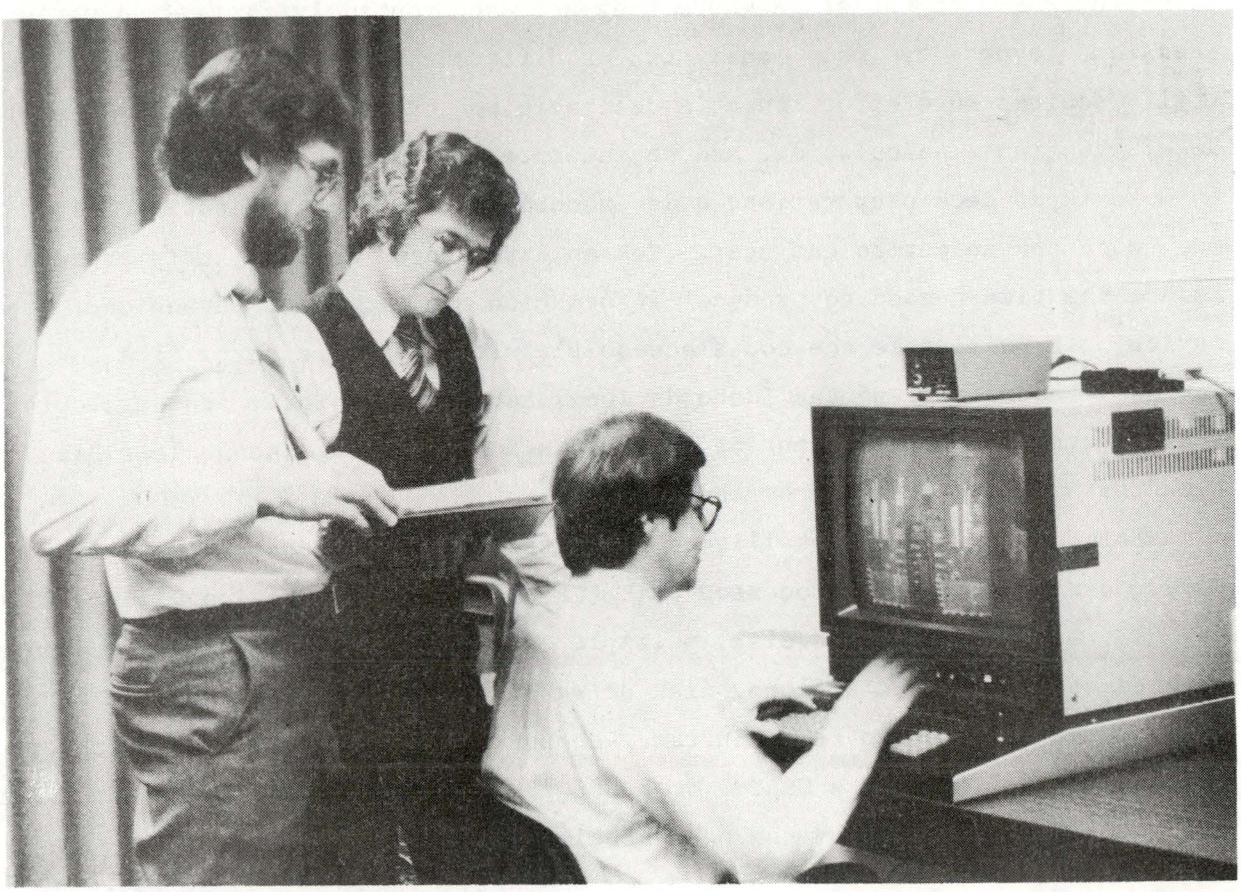

Figure 1. Three analysts seated at the Type 1 workstation running the RELAP5 computer codes interactively.

NUCLEAR PLANT ANALYZER DEVELOPMENT AT LOS ALAMOS*

by

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The Nuclear Plant Analyzer (NPA) is a Nuclear Regulatory Commission (NRC)-sponsored project to improve greatly the operating convenience of the major thermal-hydraulic Transient Reactor Analysis Code (TRAC) and the Reactor Leak and Power Safety Excursion Code (RELAP 5). These system computer programs have progressed to the stage where they are used routinely for a wide variety of transient types by many analysts at different facilities. Nevertheless, it still requires more effort than is desirable to create and qualify an input deck, to run a calculation, and to interpret the results. The NPA is designed to make input deck preparation, code execution, and interpretation of output much more convenient to the user. For an experienced analyst, this process will reduce the time needed to produce information on a plant transient; for a novice, it will make the codes accessible with minimal effort. The Los Alamos National Laboratory and the Idaho National Laboratory jointly are developing the NPA. Technology Development of California is developing the Nuclear Plant Data Bank that will be accessed by the NPA.

The NPA will be an intelligent work station based probably on a Motorola 68000 series central processor unit (CPU). Each work station will consist of the CPU, mass-storage hard disk, multiple cathode-ray tube (CRT) color displays, a hard-copy unit, and a floppy disk drive. A 9600-baud telephone tie line will connect the station to a mainframe computer. The basic thermal-hydraulic calculation will run on a CRAY-1 or a CDC 7600 computer whereas the input/output (I/O) will be performed by the work station.

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*This work was funded by the USNRC Office of Nuclear Regulatory Research, Division of Accident Evaluation.

The initial input deck can be prepared and modified locally using an editor. The Nuclear Plant Data Bank will contain geometric data on various nuclear facilities and will make that portion of input deck preparation almost automatic.

The plant calculation can be executed either in a batch or in an interactive mode. If the computation is run interactively, it can be stopped and restarted at any time by most operator actions available to a nuclear plant operator. In this mode the NPA will function as an engineering analyzer. For example, timing studies have shown that TRAC-PF1 can run a reasonable, one-dimensional Three Mile Island (TMI) calculation 16 times faster than real time on a CRAY-1 computer. Thus, scoping studies can be performed very quickly.

During the calculation or at some later time, the results can be displayed on a CRT to enhance the analyst's understanding. The results can be copied on a floppy disk that can be mailed to other sites for NPA evaluation.

The Los Alamos program has demonstrated an NPA successfully using a non-intelligent color-graphics terminal. During fiscal year (FY)84 an intelligent work station will be purchased and NPA development and improvement will continue. The final NPA will be released to the NRC in early FY86.

This software will improve greatly the analyst's ability to perform system calculations rapidly and accurately. The NPA represents the logical extension of extensive TRAC and RELAP code development.

AUTOMATING TRAC DECKS USING THE NUCLEAR PLANT DATA BANK

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The Nuclear Plant Data Bank development activity has the following two objectives:

1. To provide the rapid display of plant specific technical descriptions; and

2. To facilitate the initial generation and subsequent modification of input decks for NRC analysis programs.

The initial scope of this development activity was with respect to the Best Estimate thermohydraulic model requirements existing in 1981 but will be extended to accomodate control system and secondary system representations currently being developed at LANL and INEL.

The data contained in the Nuclear Plant Data Bank describes the physical plant and the characteristics of its components. To date, only a limited amount of data has been entered into the data bank for the ZION-1 power station. However, the design of the data bank and its corresponding display capabilities have been developed to be applicable for all U.S. PWR plants. A book containing screen display images that require the user to 'fill-in the blanks' has been generated to permit the collection of additional data by the National Laboratories for entry into the Nuclear Plant Data Bank.

Aside from the accomodation of several additional types of data, the principal upgrading to the display capability has been the addition of source references to all displays, the ability to rotate pipe isometrics interactively, and the capability to select specific pipe runs for modelling and/or display by interactively pointing to the pipe run on a Piping and Instrumentation Drawing (P&ID).

The focus of activity has been on the generation of input decks for the TRAC-PF1 and RELAP5 codes. Currently available on the INEL CYBER 176 is a version of the NPDB software that can generate a complete input deck for the TRAC code. A similar capability is expected to be available for the RELAP5, MOD2 code in the first quarter of 1984.

Although the NPDB contains only physical plant descriptions, it also provides the computer software necessary to construct an input deck for selected themohydraulic analysis programs. Results of the modelling operations performed in the deck construction process are retained by the NPDB, thereby permitting very rapid modification to existing decks.

Significant time and labor saving are anticipated once the computer software has been subjected to user testing at the National Laboratories. Demonstration TRAC decks have been prepared and LANL is initiating the user testing phase. Fiscal year 1984 activities will be focused on supporting initial use at LANL and INEL and accomodating the feedback obtained from the first practical applications of the deck generation capability of the NPDB.

Most of the data entered as input to a thermohydraulic analysis program can be obtained directly from the data contained in the NPDB thus permitting specific studies on any system or component in the plant. Approximately 10% of the entries to the TRAC and RELAP5 codes involve user judgements. Consequently, the preparation of an input deck can be accelerated by a considerable factor once the plant description has been entered into the Nuclear Plant Data Bank. The basic data can be used time and again for each model of the plant that is developed. Moreover, the NPDB software eliminates the need for an intimate knowledge of the details of the input specifications to the codes it supports. Code specific input is supplied in response to prompting by menus.

Estimated labor saving can be summarized by the following table:

	WITH NPDB	CURRENT METHOD	# PER PLANT
Data Collection	1 Year	1 Year	1
NPDB Data Entry	4 months	------	1
New Model	5 days	4 months	6
Minor Changes to Model Input	10 minutes	2 weeks	80

The labor cost of model deck preparation for a full set of studies can be reduced by 4.5 years/plant as shown above. However, the savings in time in the event of an accident may be much more important than labor savings. Quality assurance and traceability factors built into the NPDB system are additional benefits.

A major objective of the NPDB with respect to the Nuclear Plant Analyzer is to facilitate the modification of models by competent engineers who need not have an intimate knowledge of the details of the code being used but only of the desired change in modelling. NPDB software now being tested by LANL is expected to satisfy this requirement for the anticipated types of input deck changes required for the Nuclear Plant Analyzer.

To summarize, the NPDB has matured to the level of the active user testing and its primary development thrust will be to extend the code interface capability to include RELAP5 and to accomodate the feedback received from the LANL and INEL in order to hone it to a fully 'production level' and 'user friendly' state. Planned additions are to accomodate the types of thermohydraulic models currently being studied by these laboratories. This will require the incorporation of additional modelling features being developed at the National Laboratories into the display and modelling capabilities of the NPDB. The NPDB display and code interface capabilities will also be extended to accomodate the requirements of BWR plants.

REACTOR VITAL EQUIPMENT DETERMINATION TECHNIQUES

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1. INTRODUCTION

The vital area fault-tree methodology uses the SETS computer code to determine cut sets and provide usable information on vital areas. The construction of the fault tree is central to the entire program; an accurate representation of the plant is essential for credible results. Vital area analyses (VAA) have been performed on all operating nuclear plants and several plants nearing completion of construction. The thrust of current VAA work is to provide a consistent analysis of all nuclear plants. These analyses are being used by the Office of NMSS in its Regulatory Effectiveness Review (RER) program.

2. VITAL AREA DETERMINATION RESEARCH PROJECT

Some vital area analysis assumptions currently are being reevaluated. The assumptions to be examined were identified by an independent NRC working group formed for the review of the VAA program. The working group identified 11 topics for reexamination. The first phase of the vital area determination research was to survey existing literature for information relating to these topics. Ongoing research projects producing information of interest also were identified. The 11 topics were categorized into three groups based on the availability of information or ongoing research projects. The next priorities were set for performing required research topics. These priorities were based on ease of resolution and effect on the fault-trees. The current phase of the project consists of following ongoing work, using existing work or performing research to resolve questions on VAA assumptions.

3. CURRENT WORK

No topics were resolved totally based on current literature; therefore, all topics require monitoring of ongoing projects or some new research. However, the following four topics, in priority category 1, were resolvable within a short time and currently are being pursued.

A. Viability of Identification of Individual Safety-Related Cables (Topic 1). The potential for identifying individual safety-related cables in cable trays was examined by contacting plant engineers, electrical maintenance personnel, electrical contractors, and architect-engineers. The general consensus was that such a procedure is nearly impossible and requires extraordinary knowledge and determination under the best circumstances.

B. Maintaining Hot Shutdown Condition (Topic 2). One area of concern in maintaining stable hot shutdown conditions for pressurized water reactors (PWRs) is loss of reactor coolant pump seal (RCP) integrity during a station blackout. The severity of RCP seal leak is being scoped with a TRAC-PF1 calculation of a large Westinghouse PWR that is similar to other small-break calculations. The calculation should indicate whether inducing RCP seal leakage is a direct, high-success-probability scenario or a long, slow path to core damage.

C. Reactor Vulnerability While Shut Down or Refueling (Topic 3). In current fault trees, it is assumed that the reactor operating at power is in its most vulnerable configuration, so vital areas identified based on this analysis include as a subset vital areas under other reactor conditions. Work is underway to construct fault trees modified for shutdown conditions for a representative PWR and a boiling water reactor (BWR) to determine the impact of these conditions on the VAA.

D. Feasibility of Core Damage Because of Destruction of Cable Trays (Topic 4). Currently, only areas where all cable trays are located are considered in the VAA (for example, cable spreading rooms). The accuracy of the current fault-tree model is being investigated by first identifying scenarios requiring destruction of entire cable trays and then the feasibility of identifying a given tray.

4. CONCLUSIONS AND FUTURE WORK

Future work in this program includes tackling the more extensive research projects and following ongoing programs that seem to offer guidance on VAA topics. Some or considerable extension of some ongoing research may be required to address specific VAA concerns. We feel that this work provides a format for probing areas of reactor safety that usually are not explored because the VAA is not limited to single-failure criteria. Thus, complex interactions between several elements of the reactor system are explored. As this program progresses, we feel that the results will be of interest not only to safeguards concerns but in the safety arena as well.

HUMAN FACTORS IN NUCLEAR POWER PLANT SAFEGUARDS AND RELATED WORK

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Brookhaven National Laboratory (BNL) was contracted by NRC/RES/Safeguards Branch to develop long-term research plan for studying the effects of human factors on nuclear power plant (NPP) safeguards. Since the accident at Three Mile Island, NRC and the nuclear industry have become more aware of the significant effects human errors and performance deficiencies can have on plant safety. What has resulted is a large research effort to study the nature and effect of human factors in plant operations.

In 1982, the Human Factors Society developed, under NRC contract, a long-term research plan for studying human factors in NPP operations. That plan, published in NUREG/CR-2833, specifically excluded from consideration fuel cycle, waste disposal, health physics, and plant security activities. The purpose of the BNL research effort described in this paper is to address human factors in plant security. The BNL research effort did not address human factors associated NRC activities, such as the use of mandatory reporting systems, or areas of research outside of plant operation, such as probabilistic risk assessment (PRA). Instead, our research focussed on the performance of security activities by safeguards personnel at operating NPPs. For the purposes of this paper, the terms "safeguards" and "security" can be considered synonomous.

Our first task was to identify and rank human factors affecting the quality of NPP safeguards in terms of their importance. The opinions of over 85 experts were solicited and 30 responses were received. These responses were rigorously analyzed to ascertain what human factors could be considered important to NPP safeguards. In addition, the Safeguards Summary Event List (NUREG-0525) was systematically analyzed for human factors influences. Also relevant government and industry literature was reviewed. These data sources were then aggregated and an overall importance ranking was developed. This part of the research effort is fully documented and described in the BNL report entitled "Identification and Ranking of Human Factors Affecting Safeguards at Nuclear Power Plants," BNL-NUREG-33261, March 1983.*

The second part of our effort involved determining the feasibility of conducting research in the areas found to be important to NPP safeguards. A determination of research feasibility was based on the practicality, usefulness, and acceptability of conducting research and using the results in a regulatory context. This part of the effort is fully documented in a BNL report entitled "Feasibility of Research Approaches for Examining Human

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*All BNL reports are available through the National Technical Information Service.

Factors Affecting Nuclear Power Plant Safeguards" which will be published by BNL as a BNL-NUREG during September 1983.

Research efforts addressing human factors in safeguards were then developed and prioritized according to the importance of a human factors area derived in the first part of the study and the feasibility of research determined in the second part. Research was also grouped to take advantage of common research approaches and data sources where appropriate.

Four main program elements emerged from the analysis, namely (1) Training and Performance Evaluation, (2) Organizational Factors, (3) Man-Machine Interface, and (4) Trustworthiness and Reliability. Within each program element, projects have been proposed with results and information flowing between program elements where useful. An overall research plan was developed to be conducted over a four-year period leading ultimately to regulatory activities including rulemaking, regulatory guides, and technical bases for regulatory action. The entire plan is summarized in a NUREG/CR document scheduled to be published October 1983.

In terms of related research, BNL is presently studying the interaction of safety and safeguards at NPPs during normal and off-normal situations. In October 1982, NRC's Executive Director for Operations appointed a five-member Committee made up of high level NRC management to review security requirements at nuclear power plants with a view toward impacts on operational safety. After visiting five NPPs and conducting about 100 interviews with individuals representing 16 utilities and industry organizations, they issued their report, "Report of the Committee to Review Safeguards Requirements at Power Reactors," NUREG-0992. BNL is currently examining the report to systematically analyze its recommendations and its extensive appendices in order to develop specific actions which can be taken to lessen the impact of security activities on plant safety. A review panel consisting of security professionals, operational safety experts, and human factors specialists is being assembled in order to review the progress and results of our analysis. The results of this study, which began recently, will be published in mid-1984.

In addition to research related specifically to safeguards, BNL is studying human factors in operational safety including several efforts aimed at assisting PRA and developing more accurate ways of examining the impact of human error on plant safety. BNL is also examining the impact of NRC licensing of individuals working in NPPs on plant safety.

MEASURING RADIOIODINE IN THE ENVIRONMENT AFTER A NUCLEAR POWER PLANT ACCIDENT

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Following an accident at a light water reactor, the radioiodines are potentially one of the most significant radiological threats to the health of the surrounding populace. Thus, rapid and simple monitoring methods are required to accurately quantify radioiodine contributions to the population dose.

Support provided to the NRC by Exxon Nuclear Idaho Company, Inc., (ENICO) of the Idaho National Engineering Laboratory (INEL) has been directed to the development of emergency response monitoring methods. Specifically entailed was the evaluation of equipment and techniques for determining radioiodine exposure through two predominate pathways: inhalation of the airborne radioactivity during cloud passage and ingestion of deposited activity introduced into the human food chain via the grass-cow-milk pathway.

Initial efforts for quanitifying potential exposures via inhalation encompassed the study of a proposed radioiodine air monitor, which consists of an air mover, an adsorption media and a G-M detector. Testing of the different system components included: the mechanical reliability and lifetime studies of the air mover under varying environmental conditions; iodine species, krypton and xenon retention efficiency measurements of the silver silica gel media; and instrument response determinations of the specially shielded G-M survey meter. The major conclusions of the work were: 1) the original AC/DC motor exhibited a relatively short lifetime and an inability to operate reliably at temperatures of $\leq 0^{\circ}F$; 2) the noble gas retention efficiency of the silver silica gel could cause false positive indication of the presence of iodine when noble gases are present; and 3) the iodine species retention efficiencies of the silver silica gel were satisfactory.

Based on the results of the initial investigations, additional research needs were revealed and the support was expanded to include: the testing of several commercially available motors with the potential to operate reliably under extreme temperatures and with reasonably long lifetimes; the measurement of the noble gas retention efficiencies of alternate adsorption media, such as silver zeolite and silver alumina; and the development and fabrication of calibration sources for the CDV-700/6306 radiological survey instruments. In summary, the conclusions of these studies were: 1) a DC voltage motor with a generally acceptable lifetime was identified and performed satisfactorily at low temperatures; 2) adsorption media exhibited a wide variability in their noble gas retention efficiencies dependent primarily upon the vendor or treatment in preparation; 3) the calibration check sources demonstrated acceptable accuracy and precision with the agreement between instruments limited by the calibration program.

ENICO's programs for quantifying potential radioiodine exposures via the ingestion pathway initially included studies to determine instrument responses to ^{131}I, ^{137}Cs and ^{134}Cs deposited on grass, contained in liquids, on anion exchange resins, and in bovine and human thyroid phantoms. These measurements were made using several instruments equipped with G-M and NaI(Tl) detectors. The intention of these studies was to allow projection of human thyroid dose commitments from the various pathways. The results indicated the NaI(Tl) equipped instruments were the most sensitive for projection of human dose commitments from the various pathways. Following the NaI(Tl) detectors in sensitivity was the specially shielded CDV-700/6306 G-M instrument.

More recently, efforts associated with exposures from ingestion included the development of a rapid field method for the concentration of radioiodines from milk. Investigated was a batch anion exchange resin technique using six analytical or industrial-grade anion exchange resins. The most appropriate resin, DOWEX 1-X8, was used to finalize the procedure using 250 gm of 20-40 mesh resin. In the procedure, the resin is added to 3.5 liters of milk and shaken for three minutes at 30 revolutions per minute. The resin is filtered from the milk using a screened canister configured in a desired counting geometry. The milk container and resin is rinsed with 200 ml of distilled water and counted with a suitable instrument.

Progress on Qualification Testing Methodology Study of Electric Cables

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Many instrumental, control and power cables are installed in nuclear power plants, and these cables contain a large amount of organic polymers as insulating and jacketing materials. They are exposed to radiation at high dose rate, steam at high temperature and chemical(or water) spray simultaneously when a LOCA occurs. Under such conditions, the polymers tend to lose their original properties. For reactor safety, the cables should be functional even if they are subjected to a LOCA at the end of their intended service life.

In Japan, cable manufacturers qualify their cables according to the proposed test standard issued from IEEJ in 1982, but the standard still has many unsolved problems or uncertainties which have been dealt with tentatively through the manufacturer-user's agreement.

The objectives of this research are to study the methodologies for qualification testing of electric wires and cables, and to provide the improved technical bases for modification of the standard. Research activities are divided into the Accident (LOCA) Testing Methodology and the Accelerated Aging Methodology.

1. Research on LOCA Testing Methodology

An experimental apparatus named SEAMATE-II was built in August, 1979 to simulate the various LOCA environments as described in IEEE Std. 323-1974. Using this apparatus, investigations have been made on many issues in the current qualification testing. The issues involve synergism in simulated LOCA testing, a post-LOCA acceleration, oxygen effect, spray effect, influence of heating and cooling rate in LOCA profiles and others.

Degradation behaviour of materials in the simulated LOCA environments composed of the simultaneous exposure of radiation and steam/chemical spray has been also investigated to estimate the degradation under the possible LOCA condition and to evaluate the adequacy of the current qualification testing.

Most experiments were conducted on materials for reactor cables, and some tests were performed on the cables themselves. A part of results are summarized as follows:

(1) Synergism in the simulated LOCA testing

Comparison of degradation by a simultaneous and a sequential testing was performed to estimate the synergistic effect between radiation and steam/chemical spray exposure. In the sequential testing, test materials were exposed separately and sequentially to steam/chemical spray after irradiation. Both exposures were combined in the simultaneous testing.

A degree of degradation differs with a type and a formulation of materials as well as a testing profile. When the LOCA environments are simulated by saturated steam, most materials show slightly severe damage in the sequential testing compared with the simultaneous testing. However, when air is contained in saturated steam, the simultaneous testing gives a slightly severe damage.

(2) Effect of air in LOCA environments on degradation behaviour of the materials

In simulated LOCA environments with a constant temperature (100°C, 120°C and 140°C were tested), air is supplied as an overpressure onto saturated steam. Somewhat different degradation behaviours are observed compared with those in saturated steam. The tensile strength of EPR decreases markedly in air containing steam, whereas a decrease in elongation is suppressed at low dose. This suppression effect disappears at high dose. In case of Hypalon, the acceleration effect by air is observed in elongation as well as strength at high doses tested.

2. Research on Accelerated Aging Methodology

An accelerated aging method to degrade the materials to the extent expected at the end of reactor life(40 years) in the normal-operating condition is studied. The effort is concentrated in the research on the oxygen overpressure technique, synergetic effect of radiation and heat, and thermal aging under oxygen overpressure.

(1) Oxygen overpressure technique

Degradation at a low dose rate and at room temperature is possible to accelerate reasonably by the irradiation at a high dose rate in pressurized oxygen. However, at high temperature, the effect of heating time must be accounted.

(2) Synergetic effect of radiation and heat

If the thermal and radiation stress are applied to polymeric materials separately, the order of application will show a strong effect on degradation. The radiation exposure followed by heat treatment degrades materials more seriously than the reverse order. If the both stresses are applied under oxygen pressure, the difference will become more significant.

SAND83-1825A

A SURVEY OF EQUIPMENT QUALIFICATION RESEARCH ACTIVITIES WITH RESPECT TO LOCA ENVIRONMENTS*

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The intent of the full paper is to set the historical perspective for, and the current status of, the international activities involving safety-related equipment qualification (EQ) research efforts.

Historical Perspective: The use of "qualification" to delineate a quality product is merely a formalistic extension of good design practice, which has always been with us. That is, the demonstration of equipment qualification has always been a concern of the nuclear industry. In retrospect, it seems almost coincidental that the IEEE, almost alone, should embrace this area through their extensive electrical-equipment standards development activity when, in fact, qualification is such a multidisciplinary activity.

We mentioned only "qualification" above. But in the last ten years, great emphasis has been on qualification "research" activities. We believe this can be traced to IEEE 323-1974, and even more specifically to that first time when nuclear plants were committed to IEEE 323-1974. That is when it was recognized that there is a large gulf between philosophical and practical interpretation. Of course, here "accelerated aging" is almost always the example given.

In this ten-year period, there have been four significant milestones (or five if you feel compelled to include TMI-2), and they seem to have come with increasing frequency. In 1977, the UCS petition concerning electric equipment qualification and fire protection matters pushed the EQ-issue, judicially, to the forefront. This was followed early in 1979 with IE Bulletin 79-01B, which required an exhaustive review of prior EQ activities. Then, the development of NUREG-0588 (and climaxed with the Rulemaking in 1983), covering the 1979-81 time span, sought to formalize the USNRC approaches to old and new plants on the EQ issue. And most recently, and without such a clearly defined beginning, the issues of mild environment and mechanical equipment qualification have been raised.

Perhaps conspicuous by absence of mention, is the lack of reference to other nations' activities so far. That was intentional and systematic. Until very recently, they have been in "delayed-lockstep" with U.S. developments. This was a natural result of their nuclear program development in which the NSSS systems were built under licensing agreements with U.S. manufacturers and in effect followed U.S. requirements. Since then the more restrictive licensing agreements have expired, bringing in certain national design peculiarities and responsibilities. And once the national nuclear

*This paper was supported by the U.S. Nuclear Regulatory Commission, Office of Reactor Safety Research, as part of the Qualification Testing Evaluation (QTE) Program (FIN #A-1051) being conducted by Sandia National Laboratories, under Interagency Agreement DOE-40-550-75.

programs were satisfied, the international export market competition demanded further refinement of specifications and regulations.

Finally then, the role of regulation in dictating areas of EQ research needs to be mentioned. It has been most recently deemed incumbent upon the regulator to set requirements based upon data justifying the requirement. It is no longer sufficient, apparently, to leave all the open issues to the licensee to prove or disprove. And the licensees have sufficiently organized themselves to resist "blue-sky" EQ issues. So with regulation needing to be data-based and with the data largely not available, it is clear to see that regulation is a major motivator for EQ "research".

With this perspective of "why we're where we are," the objective of the full paper is to survey and summarize the worldwide EQ research activities. We choose to do this by country.

United States - Enough has been said about its leadership role. In discussing EQ research we will recognize three major contributors: government-sponsored; EPRI/NSAC-sponsored; a variety of common-interest groups and utilities. In fact, EQ issues are pervasive, many activities can be directly or indirectly related. In fact, it is not possible to be inclusive in this survey, because many programs are hidden by proprietary, or commercially-sensitive, trappings.

France - In France (and Japan) particularly, a close industry-government liaison exists. Whatever its shortcomings of independence may be, it does allow for a tight central control of all phases of nuclear development, including EQ research. Here the research is largely directed towards already-recognized issues, with lesser effort on exploratory concerns. Since France is a NSSS-exporting country, some of their effort is dictated by international marketing considerations.

Japan - The French and Japanese efforts are similarly structured. In general, the Japanese are taking a thorough, serial, approach to EQ research.

Sweden - Some early polymer (cable) research was done, but since the Swedish nuclear program effort has been reduced, so too has the research.

Italy - The Italians have recently built some LOCA test facilities which could be directed towards research.

West Germany - No coordinated program is apparent, in some part due to the licensing authority being invested in the individual states. Some related research at Hoescht Chemical, KWU, and Karlsruhe is being done.

England - With the selection of a (future) PWR-based nuclear program, the English have shown some recent interest in EQ issues.

The Import Countries - Individually, those countries which import reactor technology have had some say in the qualification to be done for their plants. Most often, the IEEE standards (and NRC Regulatory Guides) have been the common basis, but there is evidence that some country-specific requirements are being included, particularly as the licensing authorities mature to independent decision making.

International Auspices - Although the International Electrotechnical Commission (IEC) provides a forum for international standards development, the primary mover in this category is the International Atomic Energy Agency (IAEA). The IAEA does produce a series of semi-related Guides, but more importantly and more recently, it has organized a series of Specialists' Workshops on the subject of aging research activities and needs. While no direct research has been sponsored by the IAEA, the effect is to guide, suggest, and coordinate the effort to some extent.

ELECTRICAL PENETRATION ASSEMBLIES PROGRAM

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INTRODUCTION

PURPOSE - The Electrical Penetration Assemblies (EPAs) test program is an outgrowth of an earlier study on containment integrity [1]. That study investigated the potential for leak paths in EPAs under severe accident conditions. The objective of the EPA test program is to assess the physical integrity and electrical functionability under severe accident conditions. This study differs from the Equipment Qualification Research Test Program, in that the environments being simulated are a result of severe accident sequences. A severe accident is defined as an evironment that exceeds the design basis accident (DBA) conditions and is for example a result of a degraded core event that may eventually lead to loss of physical integrity of the containment building.

BACKGROUND - The leakage behavior of containments beyond design conditions and knowledge of failure modes is required for evaluation of mitigation strategies for severe accidents, risk studies, emergency preparedness planning, and siting. These studies are directed towards assessing the potential consequences of severe accidents. Other studies that are supporting these efforts are attempting to understand the functional failure or loss of physical integrity of containment buildings and penetrations. There are four NRC programs that are concerned with containment integrity beyond design conditions, the Containment Integrity Program [2], the Isolation Valve Program (at INEL), the Integrity of Containment Penetrations Under Severe Accidents Program, and the Electrical Penetration Assemblies Program [1].

An accident sequence analysis conducted on a Mark I Boiling Water Reactor (BWR), indicated very high temperatures in the drywell region, which is the location of the majority of the electrical penetration assemblies [3]. Because of these high temperatures, it was postulated in that study that the sealants would fail and all the electrical penetration assemblies would leak before structural failure would occur. Since other containments had similar electrical penetration assemblies, it was concluded that all containments would experience the same type of failure. This conclusion resulted in a reduction in time available for planning and management of accident strategy. The study also concluded since all containments had similar EPAs that leakage would occur in those penetrations as well. In order to evaluate these conclusions, the Electrical Engineering Branch of the Division of Engineering Technology, U. S. Nuclear Regulatory Commission, has sponsored this study to provide a review of the performance of EPAs.

Although the conclusions in Reference 1 indicated that EPAs have a low potential leakage, the available experimental data to substantiate that conclusion is very limited. Therefore, a test program has been initiated to provide experimental verification.

EXPERIMENTAL PROGRAM

EPA SELECTION BASIS - The primary considerations in selecting specimens for testing are based on EPAs that have the highest potential for leakage, availability of EPAs, plants near large population centers, and maximum severe accident environments. Some of these considerations are interrelated. The highest potential for leakage will be found in EPAs with organic seals and gaskets or EPAs with a low pressure capability. Three important parameters are needed in determining the capability of organic sealants and gaskets; these are temperature, pressure, and time. As environmental temperatures and pressures increase, the useful lifetime of organics are normally reduced. The availability of EPAs for testing is a major consideration in selecting test specimens. There have been eight major suppliers of EPAs during the last 15 years, in addition, some of the earlier EPAs were field manufactured. Five of the eight major vendors are no longer supplying EPAs and none of their EPAs are available for testing. However, the remaining three active vendors provide a good representation of the types of sealants in general use: the organic seals are contained in the Conax and Westinghouse designs and the glass-to-metal seal in the O'Brien designs.

The third major consideration is to test EPAs that are used in containment buildings that are located near large population centers. Four sites that fall into this category are Indian Point, Limerick, Salem, and Zion. These four sites also have containment EPAs supplied by active vendors. A fourth major consideration in the selection of test specimens, is the matching of EPAs with representative severe accident environments. The severe accident studies indicate that EPAs in the drywell of Mark I BWRs experience the highest temperatures. The majority of the Mark I BWRs were constructed in the early seventies and most of the EPA vendors are in the inactive category. In the other types of containments, the EPAs are not exposed to as high a temperature as in the Mark I BWRs. The maximum temperature considered appropriate for the other types of containments; i. e., large dry PWR, ice condenser PWR, and Mark III BWRs, is approximately 350 F. Some accident progressions indicate higher short-time temperature peaks, but the thermal mass of the EPAs effectively smoothes out the peaks. This same reasoning holds for hydrogen burns.

The proposed test matrix for the EPAs is; a D. G. O'Brien design exposed to a PWR environment, a Westinghouse EPA exposed to Mark II or III BWR environment, and a Conax EPA in a Mark I BWR environment. The assemblies acquired for these tests will be representive of field installed units and they will be designed and fabricated to the same standards as existing EPAs.

TEST SEQUENCE - The test sequence will include preconditioning of each EPA to simulate the end of life condition. The preconditioning consists of simulating radiation and thermal aging. The test sequence also includes appropriate inspections, circuit measurements, and leak measurements before and after each test sequence. The LOCA simulations are not conducted separately because the severe accidents are initiated by LOCAs and their effects can be accounted for in the beginning of the severe accident loads. The aging exposures are 200 Megarads for radiation and 150 C for 168 hours for thermal. The simulated accident progressions are represented by simple bilinear curves. The O'Brien test for a PWR accident consists of an initial rise to 293 F in ten minutes; then increasing to 360 F in ten hours; then holding constant for ten days and the test pressure will follow the temperature to maintain a saturated steam state. The Westinghouse test is for Mark III BWR accident and consists of an initial rise to 365 F in 38 minutes; holding at 365 F for ten days and the pressure will also correspond to the saturated steam condition. The Conax test is for the Mark I BWR accident and the initial rise is to 750 F in 25 minutes; holding 750 F for ten days, and the corresponding pressure will be 175 psia.

Electrical continuity will be monitored during the tests.

TEST FACILITIES - These tests will be conducted at the Sandia National Laboratories test facilities. These are the same facilities currently being used in the Qualification Test Evaluation (QTE) Program. The QTE program has previously tested an EPA in a design basis accident test and all the appropriate data acquisition hardware is available. The additional capability needed to support these tests is the development of leak rate measurements. This a major effort because very limited technology exists for measuring the leak rate of steam in a dynamic environment. The flow rates are substantially higher than encountered for the more common leak rate regime (10-6 scc/sec). The expected flow rates will be in the range of 10 to 1000 cc/sec, and the fluid that will be leaking will be multi-phase, i.e., steam. Pressure and temperature measurements will be made at number of locations in the test chamber and test specimen.

SUMMARY

The overall objective of the tests is to provide leak behavior and leak rate measurements of containment EPAs under severe accident conditions.

TMI-2 ACCIDENT: HOW THE INSTRUMENTATION HAS PERFORMED

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Research sponsored by the Department of Energy is being conducted at Three Mile Island Unit 2 to assess the performance of the instrumentation and electrical (I&E) components within the reactor building. EG&G Idaho, in cooperation with General Public Utilities Nuclear Corporation and its contractors, has undertaken this effort to analyze the effects of the accident environment on I&E components. The primary interest has been in safety related components, but the program is also conducting a systematic examination of all problem areas encountered since the accident occurred. This effort has been under way for two years and is 50% complete. Research is coordinated with recovery efforts whenever possible.

The accident related environment was generally not severe when compared to NUREG-0588. The ambient temperature did not exceed 182°F and the pressure spike from the hydrogen burn was only 28 psig. The total radiation dose is approximately 10^5 R as determined by effect on instruments. Peak radiation levels were about 10^4 R per hour, as determined from the dome area radiation monitor. Thermal effects of the hydrogen burn have been negligible except for the polar crane. Moisture/water seems to be the greatest problem encountered.

The research is divided into:

- Instruments
- Electrical components
- Interface/interconnection.

The instruments research is presently attempting to determine the entry mechanism for water into Bailey pressure transmitters that were above the high water level and to determine the cause of response time degradation in resistance temperature detectors installed in the primary coolant system. These areas are approximately 70% and 90% complete, respectively. Research on the in-core instruments has been concluded with an assessment of location of damage and estimates about the type of damage. A nagging problem here has been the inability in the laboratory to reproduce the large number of shorted thermocouple and neutron detector elements found in the core. Research on the radiation instruments has also been concluded, with several electronic problems reported back to the manufacturing industry where they are being corrected.

The electrical components research is only about 50% complete and is tightly coupled to the recovery schedule. A screening of channels through in situ tests has identified a number of anomalies, but many of them will not be resolved until the core is removed. The chief concern is safety equipment and no significant problems have been detected here. Even the reactor building fan motors are in relatively good electrical condition. DOE participated jointly with EPRI and GPU Nuclear in examining the polar crane and returning it to operation. The only significant problems detected were hydrogen burn related.

The research in the interface/interconnections area is less than 50% complete, but promises to be one of the largest contributors of important data to the nuclear community. In situ tests are being conducted on approximately 20% of the electrical channels in the reactor building. The detail obtained thus far shows many anomalies, with suspect moisture being the primary cause. Laboratory tests are being conducted concurrently to determine which of these anomalies are real or incipient operational problems.

RESEARCH ON EQUIPMENT SURVIVAL IN A HYDROGEN BURN*

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1.0 INTRODUCTION

One of the potential consequences of severe nuclear power plant accidents is the generation of hydrogen in amounts large enough to present a significant threat to containment integrity. In some circumstances, if the hydrogen is allowed to accumulate in the containment building and is ignited, pressures can exceed the design limits of the building structure.

In response to this concern, plant designs now often include deliberate ignition systems which, during an accident, intentionally initiate controlled hydrogen burns in the containment building before dangerous concentrations can accumulate. While these systems substantially reduce the threat of containment failure, they precipitate a new threat to equipment inside containment, i. e., any safety-related equipment located inside the containment building must survive the environment resulting from a hydrogen burn, perhaps from a series of such burns.

The Hydrogen Burn Survival (HBS) Program at Sandia National Laboratories in Albuquerque, NM, (SNLA) was initiated to help develop the technological basis for dealing with equipment survivability in hydrogen burns. The overall objective of the program is to develop an understanding of the hydrogen burn environment and its impact on exposed equipment. More specifically it seeks to provide the NRC an independent means of estimating equipment survivability and evaluating analyses submitted by license applicants and to develop a technological basis for determining whether consideration of the hydrogen burn environment must be included in qualifying equipment for use in nuclear facilities and, if qualification is necessary, for specifying qualification test procedures.

2.0 PREVIOUS RESULTS

The HBS Program has previously demonstrated that the hydrogen burn environment may, in some circumstances, cause component surface temperatures well in excess of current LOCA (loss-of-coolant accident) qualification guidelines. Analysis has shown that there is an effect of scale which indicates that the time of

*This work performed at Sandia National Laboratories supported by the U.S. Department of Energy under contract number DE-AC04-76DP00789 for the U.S. Nuclear Regulatory Commission, Offices of Nuclear Regulatory Research and Nuclear Reactor Regulation.

exposure to elevated gas temperature will be substantially longer in a full-scale containment than in test facilities. This effect is important in evaluating the results of test facility experiments and is the reason that simulations are required to represent full-scale exposures. Another conclusion from previous work is that multiple burns are likely for many accident scenarios, and that there is probably insufficient time between burns for components to cool to their pre-burn temperature. Thus, each successive burn produces progressively higher component temperatures.

3.0 RECENT ACTIVITIES AND RESULTS

Recent HBS Program activities have included medium-scale hydrogen burn experiments in the SNLA Fully Instrumented Test Site (FITS), simulation of the thermal environment associated with full-scale hydrogen burns at the SNLA Solar Central Receiver Test Facility (CRTF), and analysis of equipment survivability in ice condenser PWRs.

The FITS experiments provided data on hydrogen burns including the effects of steam in the gaseous mixture. The data is now available for use as an aid in understanding the hydrogen burn environment and as a basis for comparing analyses and codes to experiment. Its most severe limitation is that the vessel is of a fixed size, substantially smaller than reactor containments. To address physical mechanisms which may be functions of vessel size, the data must be considered along with that from other test facilities.

The primary result of the CRTF tests has been the development of a unique test capability to make possible the experimental evaluation of the operation and temperature response of typical safety-related equipment subjected to the thermal environment resulting from hydrogen burns in reactor containment buildings.

The equipment survivability analysis has indicated that the analysis is very sensitive to perturbations in the parameters describing the accident scenario. The sensitivity is enough that calculated surface temperatures may be either tolerable or excessive (relative to LOCA qualification guidelines) with variations within the accuracy of analytical models describing the physical phenomena.

These activities and their results will be discussed more completely in the full paper. Some of the near-term extensions of the effort will also be presented.

The Relationship of Fire Protection Research to Plant Safety

by

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Unlike many postulated accidents for which nuclear power plants are designed, fires have occurred and are expected to occur during the life of every plant. A plant may escape a major flood, tornado, hurricane, or even loss of coolant accident, but it most likely will experience some type of fire. Because of this, we should assure ourselves that, given any credible fire scenario, a sufficient set of safety equipment will survive to safely shutdown and cool a reactor.

From a licensing standpoint, the U. S. Nuclear Regulatory Commission (NRC) has issued Appendix R to 10 CFR 50[1], with the objective of providing the rules by which utilities must protect plants from fire. Unfortunately, Appendix R has created significant controversy because many utilities and NRC staff disagree with each other regarding the details of compliance with Appendix R. In addition, some fire protection researchers question the technical basis for certain portions of Appendix R.

For several years preceding and following the issuance of Appendix R, Sandia has been responsible for numerous tests of fire protection systems. Tests of fire retardant cables, cable coatings, cable tray covers, penetration seals, and fire barriers have been reported and summarized.[2] Other tests involving the adequacy of spatial separation, the effectiveness of suppression systems, and the vulnerability of electrical cabinets have been completed with reports in preparation. Despite this work, several major concerns remain, prompted by utility exemption requests to Appendix R, equipment qualification research findings, and probabilistic analyses of fire risk. In particular the following questions make up the central theme of current fire research by Sandia and NRC:

1. Under what conditions is spatial separation of redundant safety systems adequate?

2. What are the temperature, smoke, humidity, or corrosive vapor damage thresholds of cable and safety equipment exposed to fire or suppression activities?

3. What is the safety significance of fires involving control room cabinets or remote shutdown panels?

4. What is the relative importance of fires to nuclear power plant safety, as compared to other types of anticipated or postulated accidents?

Based on what we know today, we can say that:

- Spatial separation alone (even 20 feet or more) may not always prevent damage to redundant safety systems.

- Some cable and equipment may be expected to fail during a fire from conditions other than burning.

- Control room type cabinets may burn severely with an unpredicted effect on an operator's ability to maintain plant control.

- Fires in some plants seem to contribute significantly to estimates of overall public risk.

The test results or analyses which support the above statements will be presented, together with a description of the work underway at Sandia and Brookhaven National Laboratories to resolve the four research questions related to these statements.

1. Fire Protection Program for Operating Nuclear Power Plants, 10 CRF 50, Appendix R.

2. Fire Protection Research Program for the U.S. Nuclear Regulatory Commission 1975-1981, NUREG/CR-2607, April 1983.

THE DEVELOPMENT OF SLIM-MAUD: A MULTI-ATTRIBUTE UTILITY APPROACH TO HUMAN RELIABILITY EVALUATION*

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The Success Likelihood Index Methodology (SLIM) is a procedure which uses structured expert judgments of individuals or groups for quantitatively evaluating human reliability in nuclear power plants and other systems. This paper describes work on the development of SLIM and an interactive computer-based version of SLIM, called Success Likelihood Index Methodology-Multi Attribute Utility Decomposition (SLIM-MAUD), which was carried out during the past year.

The SLIM approach assumes that expert judges estimate the likelihood of successful completion of a task as a function of a number of attributes or Performance Shaping Factors (PSFs) present in the task situation. SLIM and SLIM-MAUD were developed to improve the reliability and validity of subjective judgments of human reliability by providing an explicit and structured procedure for deriving a Success Likelihood Index (SLI). A SLI is obtained by assigning weights to the PSFs to indicate the degree to which they affect the likelihood of success for the scenario being evaluated. The "quality" of these factors for the task of interest is then evaluated independently of the weights by assigning a value of 0 to 100 to each PSF, where zero indicates that the PSF maximally degrades success likelihood, and 100 that it maximally enhances it. The SLI is calculated as the sum of the products of the normalized PSF weights and the quality assessments. It is assumed that there is a relationship between the SLI and the perceived probability of success of the form: log (Probability of Success) = a(SLI) + b; where a and b are constants.

An empirical evaluation of a modification of the basic SLIM procedure and test of the assumed logarithmic relationship was carried out. A team of eight experts with diverse experience generated SLIs for three groups of seven tasks which represented examples of skill-based, rule-based, and knowledge-based

*This work is being done under the auspices of the U.S. Nuclear Regulatory Commission under FIN A-3219.

tasks. Empirical probabilities of success and failure were independently obtained for these tasks. The experts assigned generic weights to the following PSFs for each category of task:

- o Relevance and comprehensiveness of training
- o Time available to perform the task
- o Degree of motivation of the subjects
- o Information available to assist in the performance of the task
- o Quality of procedures
- o Degree of checking and supervision.

The SLI was calculated by multiplying the generic PSF weights with individual rating of the PSF for each task and then summing the result.

The correlation coefficient between the SLI and log of the empirical probability of failure for each task was -.47 (non-significant). However, when the correlation coefficient between the SLI and the log of the empirical probability of failure was calculated without using the generic PSF weights, it was -.60 ($p < .001$). When three tasks were eliminated from the task set because the judges considered that they were insufficiently described to allow effective assessment, the correlation increased to $r = -.71$.

The results supported the assumption of an underlying logarithmic relationship between the empirical probability of success and the SLI. The original non-significant correlation indicates the need to ensure that judges using SLIM are well-calibrated.

SLIM was used in a field study to estimate operator action probabilities within degraded core scenarios as part of the Industry Degraded Core (IDCOR) program. This field evaluation helped define areas of research which need to be addressed in the further development of SLIM and suggests that it will be found to have high user acceptance.

Ease of use, cost-effectiveness, breadth of application, capability to consider socio-technical factors, scrutability, and less stringent data requirements are advantages of the SLIM approach as compared to other expert judgment techniques.

These advantages made the SLIM and SLIM-MAUD approach a viable alternative procedure for the subjective assessment of human reliability.

SAND83-1701C

Estimation of Human Error Probabilities From Expert Judgment for Use in Probabilistic Risk Assessment of Nuclear Power Plants[1]

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A human reliability research program that is sponsored by the Nuclear Regulatory Commission (NRC) is being performed by Sandia National Laboratories (SNL) and SNL contractors. Primary objectives of the program are to develop human performance models, human reliability analysis (HRA) methods, and estimates of human error probabilities (HEPs) for nuclear power plant (NPP) tasks. This information is needed to perform HRAs in probabilistic risk assessments (PRAs) of NPPs. The human reliability research program is being funded by the Office of Nuclear Regulatory Research and managed by the Division of Facility Operations.

The major problem for HRA in PRA is the scarcity of actuarial data on HEPs, performance times, and associated uncertainty bounds (UCBs)[2] for NPP tasks. Publication of the Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications (Swain and Guttmann, 1983) made the human performance models and HRA methodology available. The Handbook also contains estimates of HEPs, performance times, and UCBs. However, due to the shortage of actuarial data, these estimates were obtained primarily by extrapolations from similar tasks or from informal evaluations of the expert opinions of the Handbook authors.

In an attempt to develop methods for obtaining high quality estimates of human performance data in a cost-effective and timely manner to support current PRAs, the NRC is sponsoring a psychological scaling project. Psychological scaling procedures are systematic methods for using expert judgment to assign numbers to objects, events, or their attributes so that the numbers represent relationships among the scaled items. Estimates obtained from expert judgment could be used on an interim basis until actuarial data become available.

Work during the first year of the psychological scaling project was performed

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[1]This work being performed at Sandia National Laboratories is supported by the U.S. Department of Energy under Contract DE-AC04-76DP00789 for the U.S. Nuclear Regulatory Commission.

[2]UCBs are estimates of the upper and lower values that an HEP can take due to variations in factors that can affect task performance.

by Decision Science Consortium (DSC). One product was the publication of a literature review of the psychological scaling and probability assessment research (Stillwell, Seaver, and Schwartz, 1982). A major conclusion from the review was that psychological scaling techniques show promise for obtaining consistent and valid probability estimates given that the procedures are used in a systematic manner. DSC also published a second document that includes a description of recommended procedures, the strengths and weaknesses of each, and instructions and requirements for their use (Seaver and Stillwell, 1983).

Although the judgment procedures recommended by DSC show promise for obtaining estimates of human performance data in a cost-effective and timely manner, they have not been frequently evaluated in applied settings and have never been used in the NPP industry. Therefore, in the second year of this effort, two of the recommended procedures will be tested and evaluated using NPP tasks and boiling-water-reactor (BWR) trainers as subject-matter experts. This test and evaluation is being performed by the General Physics Corporation and The Maxima Corporation.

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Maintenance Personnel Performance Simulation (MAPPS) - A Model for Predicting Maintenance Performance Reliability in Nuclear Power Plants*

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Since the Three-Mile Island accident and through subsequent investigations of human errors committed during the operation of nuclear power plants (NPPs), the nuclear power industry has become increasingly cognizant of the importance of maintenance functions in influencing the risks associated with plant operations. The U.S. Nuclear Regulatory Commission (NRC) supports many on-going efforts in assessing and arresting risks in NPPs, and specifically within this project, seeks to better understand and predict maintenance personnel errors in support of probabilistic risk assessment (PRA).

The primary objective of the program is to develop, validate, and disseminate a practical, acceptable, and useful methodology for the quantitative assessment of NPP maintenance crew reliability. Because of the relative paucity of research directed toward the quantitative evaluation for maintainer performance and because of the need for diagnostic and prescriptive data within PRA as well as the industry in general, this program is endeavoring to produce a practical, acceptable, and useful source for such data.

The Oak Ridge National Laboratory (ORNL) in association with Applied Psychological Services, Inc. (APS), is midway through four project phases. Phase One was a scoping study consisting of: (1) a user survey to identify the reliability data needs of potential users; (2) a literature survey of human behavioral methodologies; (3) job analyses of four maintenance positions; and (4) the formulation of a comprehensive program plan for model development, validation, and dissemination. One of the primary results of

*Research sponsored by the U. S. Nuclear Regulatory Commission under DOE Interagency Agreement 40-550-75 with Union Carbide Corporation under Contract No. W-7405-eng-26 with the U. S. Department of Energy.

the scoping study was the selection of simulation techniques as the type of methodology to be developed within this program. Considering the maintenance context, simulation modeling seemed most suited in matching the results of the literature review with the data needs identified through the front-end users survey. In addition, simulation techniques have been effectively utilized to dynamically model various military scenarios involving multi-task, multi-person requirements; characteristics which are also relevant for the NPP maintenance context.

Phase Two of this program involves the formulation and development of the Maintenance Personnel Performance Simulation model (MAPPS). Existing human behavioral methodologies and theories as well as applicable reliability and probability theories were operationalized into a simulation framework. The model focuses on the differences between the abilities required for adequate task performance and the current ability levels of the maintenance team. It also addresses a number of performance shaping factors including stress, radiation level, effects of protective clothing, etc., that are used to modify the ability levels of the maintenance team. The model includes a decision-making module and a separate trouble-shooting module. In addition, the model is being developed to be minimally dependent upon the availability of data base information.

The development of MAPPS and subsequent sensitivity testing is scheduled to be complete by the end of calendar year 1983. At that time, the state of the model will be fixed and the third phase of this program, validation, will be initiated.

Validation of the MAPPS model will seek to demonstrate content, construct, and criterion validity. Content validity is supported by the inclusion of critical variables identified in the front-end user survey. Construct validity will be assessed through an examination of correlations among module and model outputs. Criterion validity will involve comparisons of model predictions with criterion performance data collected through acceptable and feasible data sources. If necessary, the model will undergo any needed recalibration to improve simulation accuracy prior to its general release, which is tentatively scheduled for January, 1985.

Phase Four of this program involves the dissemination of the MAPPS model to potential users. For this effort, appropriate instructional training materials will be developed and workshops will be conducted to assist users in both the proper applications of the model and the effective interpretation of model results.

The development and validation of the MAPPS model is endeavoring to supply an acceptable, practical, and useful source of maintainer performance reliability data for a number of applications, including PRA. When properly utilized, the human reliability data generated by MAPPS should prove to be an important contribution to efforts addressing the improvement of the nuclear power plant maintenance structure as well as to studies that are aimed at minimizing the risks associated with nuclear power plant operation.

CONCEPT DEVELOPMENT OF THE HUMAN RELIABILITY DATA BANK*

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The U.S. Nuclear Regulatory Commission and Sandia National Laboratories initiated a three-phased research program in 1981 to develop a plan for a human reliability data bank. This research initiative was in response to the data needs of the nuclear power industry's probabilistic risk assessment community. The three phases are: A) Develop the data bank concept, B) Develop an implementation plan and conduct a feasibility test, and C) Assist the sponsor in implementing the data bank. This paper briefly describes some of the results of the work performed during Phase A and outlines the program elements scheduled for Phase B.

PHASE A RESULTS

In the initial stages of the program, an extensive review of existing data bases was conducted. This review had two primary goals:

(1) To ascertain whether human reliability data that could support nuclear power PRAs had already been collected and stored in an existing data base.

(2) To survey characteristics of past and existing data bases that might prove useful in the design of a new data base.

The review consisted of a survey and comparative analysis of the attempts to quantify and predict human operator and maintainer performance as a function of design, training, procedural or situational factors. An assessment was then made of these methods and techniques as to their potential applicability to PRA and as a supplement to the techniques and error estimates in NUREG/CR-1278 (Swain and Guttmann, 1983). A description of the review, as well as reproductions of four of the data bases is contained in NUREG/CR-2744, Volume 1 (Topmiller, et al, 1982).

Using the results of the review as a start, a concept for a human reliability data bank was developed. The intent of this data bank was to provide one central location for human performance data that could be used

*This work performed at Sandia National Laboratories is supported by the U.S. Department of Energy under Contract DE-AC04-76DP00789 for the U. S. Nuclear Regulatory Commission.

to perform a human reliability analysis (HRA) portion of a PRA. As such, the conceptual development of the data bank was divided into four areas: data collection, data treatment, data storage, and data retrieval. Phase A work concluded with a concept and system description for a human reliability data bank specially tailored for PRA application within the nuclear power industry. This work is described in detail in NUREG/CR-2744, Volume 2 (Comer, et al, 1983).

PHASE B APPROACH

The results of Phase A work have led to the initiation of a feasibility test of the data bank concept. There are five separate steps planned as part of this feasibility testing:

Step 1 is the development of an implementation plan for the data bank. This plan will contain sets of step-by-step instructions for those who will set up and maintain the data bank. Firm guidance and, in some cases, rules will be specified to assist those people in each of the data processing areas: data receipt, data treatment, data storage, and data retrieval.

Step 2 is the design of a feasibility test plan. Each of the areas described above will be thoroughly tested. This will entail testing each of the procedures developed in Step 1.

Step 3 is the actual conducting of the feasibility test. Current plans are to use govenment, industry, and national laboratory personnel as test subjects.

Steps 4 and 5 provide final documentation of the work performed. Step 4 will document the results of the test, and Step 5 will revise the implementation plan prepared in Step 1.

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NUCLEAR POWER SAFETY REPORTING SYSTEM
FEASIBILITY ANALYSIS & CONCEPT DESCRIPTION

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The Aerospace Corporation is assisting the U.S. Nuclear Regulatory Commission (NRC) in the evaluation of the feasibility and potential attributes of a voluntary, nonpunitive data gathering system, a Nuclear Power Safety Reporting System (NPSRS) for identifying and quantifying the factors that contribute to the occurrence of significant safety problems involving humans in nuclear power plants. The objectives of the Aerospace study have been twofold: (1) to analyze the existing Federal Aviation Administration (FAA)/National Aeronautics and Space Administration (NASA) Aviation Safety Reporting System (ASRS) in order to determine whether it would be feasible to apply the ASRS concepts for collecting data on human factor related incidents to the nuclear industry; and (2) to establish the feasibility and identify and define the basic elements and requirements of a NPSRS. NPSRS feasibility depends upon a number of issues such as: (a) practicality (e.g., costs and logistical requirements); (b) acceptability to government, industry, and operational personnel; and (c) the usefulness of data developed by such a system (e.g., its relevance, biases, and applicability to NRC and nuclear community needs).

In 1976, the FAA instituted the Aviation Safety Reporting System. The ASRS was designed to encourage flight crew members, air traffic controllers and others in the national aviation system to voluntarily report incidents that are related to air safety. Two provisions have been included in the system as an inducement to motivate voluntary reporting. First, a neutral and independent third-party organization (NASA) manages and operates the program, thereby isolating the report (and the reporter) from the FAA, and consequently providing anonymity for the reporter. Second, the FAA extends a limited waiver of disciplinary action to those individuals who submit reports of incidents that may have been induced in part by (or associated with) violations of Federal Air Regulations, as long as criminal offenses or actual accidents were not involved in the incident.

The most recent performance evaluation for the ASRS was conducted in April 1982 by the NASA sponsored ASRS Advisory Committee. The Committee observed that in the seven-year period since the ASRS was established, over 32,000 reports have been submitted, analyzed and cataloged. ASRS analysts have routinely searched and evaluated the information in the data base for trends that might identify existing or potential problems within the U.S. aviation system. On the basis of their review, the Committee concluded that the System was practical, useful, and widely accepted within the aviation community. The successful performance of the ASRS provides substantial support for the conclusion that a similar concept could be utilized and provide substantial benefits within the nuclear community.

In the Nuclear Power Safety Reporting System, like the ASRS, nuclear plant personnel involved in a safety-related incident would be encouraged to submit a simple, single-sheet report form to the NPSRS describing their experiences. The ASRS experience indicates that guarantees of anonymity would be an important feature of the NPSRS in assuring reporters they will not incriminate themselves by sending information on safety incidents to the data collecting system. Therefore, as soon as possible after receipt of the report by the NPSRS, the name of the reporter would be separated from the information that was submitted to the system. To further ensure reporter anonymity, it may also be necessary to deidentify the specific nuclear plants involved in the incident and perhaps their parent utilities as well.

In the U.S. aviation system, individual reporters have been motivated to support the ASRS by the FAA's warranty of a limited waiver (for one incident per five year period) of disciplinary action for regulatory violations. If the NRC were to invoke similar measures for NPSRS reporters, the NRC's regulatory procedures would probably not provide such an attractive "carrot" to nuclear power plant operational personnel. Because the NRC customarily takes disciplinary action against utilities rather than individuals, the reporters probably would not feel immediate concern over potential NRC sanctions. In order to raise their motivational level to report safety incidents, it may be necessary to extend the limited warranty of immunity from disciplinary action to utilities and power plants as well as individuals based upon the extent to which the personnel of the power plant have participated in the NPSRS. In order to qualify for the warranty of immunity, the utility management would have to demonstrate that a successful program existed for encouraging participation of plant personnel in the safety incident reporting program.

An independent, neutral, third-party managing organization for the NPSRS would apparently be essential to its success. Specifically, the NRC, both the maker and enforcer of regulatory requirements, would almost certainly need a NPSRS manager of this type to alleviate the fears of power plant operational personnel and plant management with respect to potential self-implication consequences for reporting safety incidents.

Finally, the success of a NPSRS would depend upon the broad support of the members of the nuclear power community. A forum would be needed to represent their interests in order to make a NPSRS effective, mutually acceptable to all members of the community, and to monitor and evaluate the performance of the system. This forum would be provided by an advisory committee to the NPSRS that was similar to the ASRS Advisory Committee.

A NPSRS would provide a number of benefits to the nuclear community. It would provide an unrestricted source of data and reports on safety-related incidents involving humans in nuclear power plants. These reports would provide a basis for assessment of trends in characteristic events of significance to human factors designers in plants and would aid PRA systems analysts in their efforts to model human impacts on safety-related incidents and to estimate the probability of human error related sequences in PRAs. The NPSRS taxonomy would be designed to identify the influences of performance shaping factors (such as control room design, the effectiveness of operating procedures, the influences of physio-psychological factors) on human actions. The data would also provide a rich source of information on problem solving mechanisms used by humans when they exercise their initiative and intervene in a positive manner to reduce the probability and potential severity of accidents.

In summary, The Aerospace Corporation's study concluded that the NPSRS appeared to be feasible, that it represented an important potential asset to the nuclear community, and that it also appeared to be practical, useful, and generally acceptable. In accordance with Aerospace recommendations, a preliminary NPSRS implementation plan and a test plan are currently being prepared in order to provide the NRC with the basis for making reasoned decisions about the potential for future development of a NPSRS under their auspices.

Criteria for Safety-Related Operator Actions*

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Introduction

The Safety-Related Operator Actions (SROA) Program, completed in FY-1983, was designed to provide information and data for use by NRC in assessing the performance of nuclear power plant (NPP) control room operators in responding to abnormal/emergency events. The primary effort involved collection and assessment of data from simulator training exercises and from historical records of abnormal/emergency events that have occurred in operating plants (field data). These data can be used to develop criteria for acceptability of the use of manual operator action for safety-related functions. The program also included studies of training simulator capabilities,[1] of procedures and data for specifying and verifying simulator performance,[2,3] and of methods and applications of task analysis.[4,5] This paper summarizes the major results of the program pertaining to the development of criteria for safety-related operator actions.

Simulator and Field Data

The initial impetus for the SROA Program was the need for data to assess proposed design criteria[6] for the choice of manual versus automatic action for completion of safety-related functions during design basis accidents. After a preliminary assessment of available data,[7] a program of data collection during "quasi-controlled" exercises was initiated in March, 1980. A parallel program was initiated to collect field data which could be used to "calibrate" simulator results. The approach taken in the proposed design criteria was that if the designer chose to rely on manual operator action, he had to allow certain time margins, depending on the severity of the event, complexity of actions, etc. If those time margins were not available, the actions should be automated. Consequently, the emphasis in the SROA Program has been on collecting data on the time required for operators to take correct action, despite the recognition that a more comprehensive approach to allocation of functions is desired and that other measures of performance may be equally or more important in many cases. This simple approach was felt to be reasonable for interim use in a design standard until some basic changes are made in the approach to NPP control room design and a more comprehensive research and data base exists.

*Research sponsored by the U. S. Nuclear Regulatory Commission under DOE Interagency Agreement 40-550-75 with Union Carbide Corporation under Contract No. W-7405-eng-26 with the U. S. Department of Energy.

A sizeable base of data on operator performance has been accumulated including primarily operator response times but also some error data. Data on operator response times, i.e., the time from the activation of an alarm or observable cue until the time of initial correct operator action, have been recorded for a series of preliminary simulator exercises on a pressurized water reactor (PWR) simulator[8] and boiling water reactor (BWR) simulator,[9] and a more extensive series of exercises on a PWR simulator was completed in FY-1983.[10] Response times are quite variable but tend to be correlated more to "operational" characteristics of the event, e.g., how rapidly it develops and how specifically it is annunciated than to the severity of the event. Initial comparison of field data to simulator data[11] suggested that for highly experienced operators, response times in the simulator will be "on the average" considerably less (as little as one-sixth to one-seventh) of typical response times in the field. However, there is obviously a question of the possible effects of stress during an actual event, which is probably not reproduced in the simulator.

In FY-1983, the field-data-collection methodology was modified to provide a much richer analysis of events that have occurred, and this methodology is being employed in a new program of simulator experiments which was initiated in late FY-1983. An extended form of the task analysis methods developed under this program and the NRC Crew Task Analysis[12] Program is used to document operating sequences which are verified in training simulators, calibrated to field data at the NPP and then used as controlled scenarios during simulator experiments.

Proposed Model for SROA Criteria

Based on results of the simulator and field data collection, through early FY-1983, the applications of task analyses methods described in Refs. 4, 5, and 12, and on reviews of existing human performance models,[13,14] a simplified and preliminary model of NPP operator performance during abnormal/emergency performance has been developed and tested. The model is computerized and programmed into the SAINT[15] simulation language to take advantage of an existing computer-based structure. It is proposed for further development and use to predict operator response time and performance during critical safety-related sequences. Given a quantitative system reliability goal, these predictions could then be used to evaluate existing systems (or in the design of new systems) to determine the suitability of assignment of safety-related actions to the operator. The model could also provide a focus for continued research on operator performance.

Conclusion

The Safety-Related Operator Actions Program has included a number of separate but related studies concerned with NPP operator performance, task analysis techniques, and the use of simulators in operator training. The program is one of the earlier NRC research programs in the human factors area, having begun prior to TMI-2, and has in some ways "evolved" with the

NRC research effort. The central task - development of criteria for safety-related operator actions based on simulator and field data - was completed in FY-1983 and this terminated the program as scheduled. An initial but substantial base of performance data has been accumulated, and a model of operator performance has been developed and tested which is proposed as a tool to help evaluate the acceptability of assignment of safety-related actions to the operator.

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NUCLEAR POWER PLANT PERSONNEL ENTRY LEVEL QUALIFICATIONS AND TRAINING*

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This paper summarizes the results and current status of a research program at Oak Ridge National Laboratory (ORNL) which initiates the use of the Systems Approach to Training (SAT) in the evaluation of training programs and entry level qualifications for nuclear power (NPP) control room personnel. The program is to some extent an outgrowth of previous work at ORNL under the Safety-Related Operator Actions Program.[1,2] The FY-1982 funded effort (March 1982-July 1983) was focused on adaptation of the SAT process to provide a practical structure for NRC to evaluate training systems and entry level qualifications. Variables (performance shaping factors) of potential importance to entry level qualifications and training were identified, and research to more rigorously define an operationally useful taxonomy of those variables was recommended. A high-level "model" of the SAT for use in the nuclear industry, which could serve as a model for NRC evaluation of industry programs, was constructed. The model is consistent with current publicly stated NRC policy, with the approach being followed by the Institute for Nuclear Power Operations (INPO), and with current training technology. Checklists to be used by NRC evaluators to assess training programs for NPP control room personnel were proposed based on this model. Further assessment of the proposed checklists to assure practicality, utility, and acceptability is recommended. In addition, other issues related to training effectiveness evaluation were identified, and a programmatic plan for research to address them was outlined. Results of this work are reported in Ref. 3.

An additional task accomplished was the development of a technique that can be used on an interim basis (until full implementation of the SAT in the nuclear industry) to rank possible plant malfunctions for their importance to training, especially simulator training. A separate report on that task is in draft and should be released in the near future.

*Research sponsored by the U. S. Nuclear Regulatory Commission under DOE Interagency Agreement 40-550-75 with Union Carbide Corporation under Contract No. W-7405-eng-26 with the U. S. Department of Energy.

In the FY-1983 funded work, which was recently initiated, the emphasis is on development of specific methodologies (evaluation tools with user guides) to operationalize the evaluation requirements developed in the earlier work. Four tasks are planned: (1) the development of an analytical framework or "structuring model" for describing skilled task performance, (2) further development of the interim malfunction selection technique to develop a task selection methodology to identify which safety-related tasks should be included in training programs, (3) the analysis of factors contributing to imprecision in task performance measurement, and (4) the development of a training simulator evaluation procedure. Task 1 will be carried out through a task descriptive model linked to taxonomic performance shaping factors. Each factor used is elaborated through a performance standard, measurement test, and training development principle. Task 2 then uses the information extracted from task structuring model descriptions to sort tasks into candidate training categories. Current planning is that this process will be based upon a multi-attribute utility theory weighting scheme. Task 3 will relate total measured skill performance to incremental measurement uncertainties generated as a course design process progresses toward a plant operator's application of instructional content. Finally, Task 4 will develop a simulator training-evaluation procedure to assess probability of positive transfer of training from high-fidelity simulators to on-the-job operations.

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NUCLEAR POWER PLANT PERSONNEL OPERATING PERFORMANCE DURING HIGH STRESS EVENTS

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NRC General Design Criteria 2 for Nuclear Power Plants in Appendix A to 10 CFR Part 50 requires that nuclear power plant (NPP) structures, systems and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. One of the results of the TMI accident and the general realization of the great importance of human factors in the safe operation of nuclear power plants is the objective to ensure that adequate attention is given to and provisions are made for the satisfactory performance of reactor operating personnel under severe stress, such as could be caused by a seismic event.

Ongoing and planned NRC research is intended to obtain a technical basis for such an objective. An initial, in-house investigative phase was started in FY 1982 through a review of the literature to determine the capability of NPP operating personnel to make important decisions correctly and to take appropriate actions under the stress of an earthquake. In May 1983, Idaho National Engineering Laboratory was contracted to perform a research project to (1) survey the procedures and training programs used by a representative set of licensees in response to the General Design Criteria 2 in order to evaluate the effectiveness of these procedures and training programs in supporting operating personnel during and after a Safe Shutdown Earthquake; (2) extend the literature review to complete the analysis of the major effects of stress on personnel performance in severe environmental conditions; (3) experimentally compare decisionmaking performance under stress conditions; (4) prepare recommendations for NPP personnel qualifications, training, and procedures, and suggestions related in ANS 3.1 and RG 1.8 documents that would improve the probability that NPP operating personnel would perform properly during an earthquake sequence.

Preliminary results to date from a recent review of the literature suggest that the adequacy of human response is sometimes compromised due to the effects of various types of stress. Generalized stress is a complex variable which may be an initiating event or may be a response to that event or may be an interaction of variables which in themselves are not stressors. Stress is a perceived imbalance between a demand and a capacity to respond under conditions where failure to meet the demand has important consequences. It is reasonable to assume that psychological stressors will, likewise, influence the ability of NPP personnel to respond after the occurrence of a seismic or high physical stress event.

The adequacy of procedures, the amount of workload encountered, the amount of contradictory or confirmatory information available as well as the operating crew's own individual personalities will impact decisionmaking performance. While many personality variables were reviewed, the most amenable to assessment for the experimental study are Type A and Type B behavior as measured by the Jenkins Activity Survey, Internality and Externality as measured by Rotters locus of control, management of previous life stresses as measured by the Life Stress Index, and predispositions toward changes in anxiety level as measured by the State Trait Inventory. The experimental design is a three factorial with repeated measures design with the three factors being Procedures, Conflicting Information, and Workload with two levels of each. The blocking variable is Conflicting Information by virtue of its being identified as the least important of the three variables. Computationial procedures are similar to those found in Winer (1966). Thirty male subjects will be randomly assigned. Multiple Regression Analyses procedures will be employed to determine the nature of the relationship between personality measures and decisionmaking performance as determined by the IN-basket technique, which will be the stimulus.

INEL/EG&G will analyze the data and information from the training and procedures survey with the experimental test results to prepare recommendations for NPP operating personnel qualifications, selection, training and procedures, information displays and job aids.

FAULT DIAGNOSIS USING ARTIFICIAL INTELLIGENCE

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An experiment was performed during 1983 that measured performance of nuclear plant operators with and without a computer-based operator aid. This paper discusses the results of that experiment and implications for design and regulation of advanced computer aids in nuclear control rooms.

Computers are now being used in nuclear reactor control rooms. The most prominent applications are the safety parameter display systems being installed by utilities. Computers certainly can do much more than display data. Fault diagnosis and response, alarm handling, procedure prompting, and plant status monitoring are some of the likely future applications for computers in commercial reactor control rooms. Therefore, the Nuclear Regulatory Commission has sponsored experimental evaluation of artificial intelligence applications which can do some of the operator's thinking for him. The research provides data to NRC which can form part of the technical basis for regulatory guidelines and criteria related to computer aiding. Issues for the NRC in this area include:

1. How can artificial intelligence based aids be evaluated?
2. What is the proper role of the operator in a system including aids?
3. What is implied for regulatory requirements by cognitive aid evaluations?

The operator aid which has been tested is called a response tree. This helps the operator to properly align a piping system despite pump failures, stuck valves, empty tanks, and failed electrical and pneumatic support systems. The computer-driven response tree tells the operator what the best available system lineup is based upon current piping system failures, as input by the operator. The response tree lineup recommendation highlights components which should be operated. The operator controls valves and pumps through a touch panel which overlays a color crt display of the piping system mimic.

In the experiment subjects were required to lineup the displayed piping system to provide water to cool a reactor. The presumed situation was that the core was heating up at 6 degrees per second. Each of the 28 subjects faced 18 scenarios in which varying components and support systems were failed. All subjects were current or previously qualified reactor operators.

An experiment session with a subject consisted of about 2 hours of training, 1 hour of experimental trials, and 1 hour of the subject filling out a questionnaire. Training was done using a script developed on PLATO, a computer-aided instruction system. Subjects read the script on a black and white terminal, typed in correct answers, and performed directed pratice with the color display and touch panel.

The experiment involved four groups of test subjects and two levels of scenario difficulty. Each level of difficulty had nine scenarios. Each test subject group had six members and one pilot test. Three test subject groups were trained on the simulation, the display, the task and grading scheme, and on the response tree aid. Of these three groups, one had the response tree aid available for use at their discretion during the scenarios. A second group had the aid available and was directed to use it to solve the problem. The third group that was trained on the aid did not have it working when they faced the scenarios i.e., it was as if the computer aid had failed. The fourth test group was trained on everything except the aid, and the computer display which they used did not include the response tree.

Data recorded in the experiment include all of the subject actions during the training and experiment runs and what the subject faced at each point in time. Also brief questionnaires were completed by each subject to record their opinions of crt/touch panel control and of the operator aid.

The response tree evaluation experiment was designed to determine whether the aid significantly improved operator performance. Measures of performance included maximum core temperature reached and how many of the five operating rules were violated. Based upon preliminary data analysis, no improvement in operator performance when using the aid was observed.

For NRC purposes, the specific value of the response tree aid is less important than observations from the experiment which can be useful in regulating artificial intelligence applications in nuclear power plants. The following options are based upon observation of the experiment runs and the subject responses to the questionnaire.

1. Subjects did very well at following the five rules. Subjects did so well at following rules that violations were not a useful measure of performance. This indicates that the effort to carefully formulate operating rules, prioritize them, and drill operators on them works, at least for five rules. Such prioritization contrasts with giving operators several rules, telling them to obey them all thus forcing quick operator decision when one or more rule violations is unavoidable.
2. The use of a touch panel overlaying a system mimic displayed on a color crt was well accepted by operators. Subjects used the mimic to trace out flow paths and this seemed to help them. The touch panel worked well for most subjects although it did take practice before they became comfortable with it. Visual feedback of actions taken was provided on the crt but more was needed.
3. Requiring operators to inform the computer of equipment failures does not work well. Often failures were found but not input to the computer before the response tree was invoked. Experiences like this caused many subjects to give up on the response tree. The need to perform a separate action to input failures seemed difficult even though it was covered extensively during training. In this case, a greater degree of automation is needed i.e., the operator aid should know when tanks are empty or components have failed on demand without operator input.
4. Bookkeeping help provided to operators by a computer can hurt as well as help. The ability to declare and display components as inoperable reduces the operator's need to keep track in his head. Problems occurred when components were wrongly or inadvertantly declared inoperable. Once the computer displayed something as inoperable, the subjects seldom reconsidered and tried to operate it again.
5. Subjects varied in the way they operated the system. After 28 subjects we were still seeing new things from the subjects in operating strategies, errors, and ways to uncover computer problems. This implies that many subjects are required for testing computer aids.

ALLOCATION OF FUNCTIONS

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NRC is sponsoring a continuing program of research concerning the allocation of functions between man and automation, and has just published a methodology which can improve the allocation of functions during control system design. Phase I of the program studied past efforts to optimize allocation decisions during design, and reported a number of methods which have been used with varying success. It was observed that none have been widely applied. The contractor (BioTechnology, Inc.) reported some lessons learned, and recommended that the allocation process include separate steps of hypothesis and formal test, within the iterative invent-and-test cycles of engineering and organizational design. It was observed that good design, and good allocation of functions, must depend on a good design documentation base.

The Phase II effort then developed an improved methodology for allocating functions, designed specifically for NPP control systems, but widely applicable in other process control. A logical model for the allocation decision was embedded into standard procedures of system design, and allocation decisions were linked to related decisions in the engineering and human factors design process. The methodology recognizes allocation of functions as the logic which decides the interface between an engineering subsystem and a human organizational subsystem, and which defines the functions which each must perform. During the design process many cycles must occur, of allocation, engineering design, human system design, test, and revision. During this iteration the design of the system grows in definition and in elaboration of detail. The methodology is detailed in NUREG-CR/3331.

This methodology provides the means by which a new plant or control system can be developed with a more effective (and safer) allocation of tasks to man and to automation. But it does not provide a means by which to evaluate an existing control room, in regard to the appropriateness (and safety) of functions performed by automation or by man. Because NRC's immediate concern is the safety of existing plants, the next step was to test whether the methodology could be used in evaluation. The contractor developed a composite plant design, derived from details of actual preliminary BWR designs. Since the methodology consists of separate hypothesis and test phases, it was logical to expect that the test logic could be applied to evaluate an existing design. Indeed, a limited test suggested that the methodology could in fact detect allocation errors in an existing control room design.

NRC now has a credible but undemonstrated methodology for allocating functions during NPP design. It is next intended to test the methodology in designing controls for DOE's Large Scale Prototype Breeder (LSPB). The resulting control system design will be installed on a control console simulator held by General Electric Company, It will be used to control an engineering simulation of the LSPB, and the design will be tested by simulated operations using a live crew. From this will be derived (1) criteria for evaluating NPP control rooms, and (2) improved documentation for the methodology.

PERCEPTIONS OF LWR RISK FOR DECISION MAKING

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INTRODUCTION

The nuclear power industry is sponsoring a program to investigate issues associated with potential rulemaking for degraded core accidents. This program, called the Industry Degraded Core (IDCOR) Program, includes tasks to develop information in a probabilistic risk assessment (PRA) framework. The purpose of these tasks is to develop risk reduction rational for decision making regarding issues related to severe accident response. The rationale is to be based upon the results of existing PRA studies and the current understanding of accident phenomenology and plant risk along with the estimated impact of potential preventive and mitigative modifications.

The PRA-related activities of IDCOR are designed to provide information to address the following questions:

- What is our perception of the risks associated with severe accidents at nuclear power plants?
- What types of conclusions can be drawn regarding light water reactor (LWR) risk?
- What is the potential impact on risk of various potential changes to plant design and/or operation?

The PRA information gathered, examined and evaluated for the IDCOR program comes from fourteen (14) published PRAs, nine studies of pressurized water reactors (PWRs) and five studies of boiling water reactors (BWRs). Four plants (two BWRs and two PWRs) are being utilized as IDCOR reference plants for the detailed examination of the impacts of current understanding on perceptions of plant risk and potential risk reduction of modifications.

PRA SURVEY RESULTS

The examination of PRA results identified the contributors to predicted core damage and major fission product release frequency as described in existing plant PRA studies from the past ten (10) years. The purpose of this information is to support decisions regarding the potential for risk reduction through measures to prevent core melt and/or mitigate release. The results of the survey also provide the starting point for detailed accident process, fission product transport and consequence analysis utilizing the tools developed by the IDCOR program.

The survey results indicate that core damage accidents result most often from loss-of-coolant accidents at PWRs and transient events at BWRs. From a loss of function viewpoint, core damage accidents result as follows:

Functional Failure	Averaged Occurrence Frequency in Core Damage Frequency
PWR	
Reactor Integrity, early	57%
Core Inventory Makeup, early	50%
Core Heat Removal, early	44%
Core Inventory Makeup, late	36%
BWR	
Core Inventory Makeup, early	56%
Containment Heat Removal, late	43%
Core Heat Removal, early	39%

The types of accidents predicted to contribute the most to core damage frequency are:

	Accident Type	Averaged Contribution to Total Core Damage Frequency
PWR		
	LOCA with failure of recirculation (S_2H)	32%
	Station blackout (TMLB')	20%
	Loss of feedwater (TML)	13%
	LOCA with failure of injection (S_2D)	12%
	Loss of feedwater with failure of feed and bleed; ATWS	9%
BWR		
	Transient with failure of long-term heat removal (TW)	35%
	ATWS (TC)	24%
	Station blackout	11%
	Transients with failure of makeup (TUV)	10%
	LOCA with failure of long-term heat removal (SI)	9%
	LOCA with failure of injection (SE)	6%

Identification was made of important system failure and fault type contributors to these accidents. Generally, system importance depends upon the specific design being examined. For fault types, human errors contributed to about half of the core melt and major release frequency.

This information, from existing PRA studies, can be used to support decisions on selection of accidents for analysis with new tools, identification of potential areas for risk reduction, and potential impact of changes in plant design or operational philosophy.

GENERAL PERCEPTIONS OF RISK

Based upon the dominant accident sequences leading to core melt and major release, the IDCOR program developed risk profiles for four specific plants. These profiles consisted of a set of numerical values for each plant representing the predicted core damage frequency, expected acute fatalities, expected latent fatalities and expected man-rems dose per reactor year resulting from severe accidents. The estimated risk profiles also included IDCOR program information of plant configuration, initiating event frequency, system success criteria, accident processes, fission product transport, and site-specific consequence analysis. These IDCOR profiles were then utilized to assess the risk reduction potential of various mitigating features.

Although the results are preliminary at this writing, the PRA information developed to date (July 1983) implies the following:

1) The perception is that, in most cases, potential nuclear power plant accidents pose a significantly reduced risk than previously described.
2) Most of the proposed plant modifications have negligible impact on risk.
3) Those few changes that are estimated to have a significant impact on risk only do so starting at an already low level of perceived risk.

These conclusions come from the IDCOR projections that although the frequency of core damage at the four plants remains essentially the same, accident time spans are longer than previously thought, and the potential releases of fission products are less.

INSIGHTS GAINED FROM CONDUCTING AN EPRI SPONSORED REVIEW OF FIVE PRA STUDIES

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Several PRA (probabilistic risk assessment) studies were recently completed for specific nuclear plants at specific sites, and more are in progress. Motivated by a variety of objectives, these studies range in scope from extrapolations of the methods and the results of the Reactor Safety Study to full-scale PRAs that have advanced the state of the art by developing and introducing new methods and approaches.

Although a knowledge of the results, insights, applications and efficacy of various PRA methods could be of significant value to nuclear utilities and regulators, PRA studies usually produce multi-volume reports that are difficult to comprehend and assess without extensive and dedicated scrutiny. Realizing that there is considerable interest in, and controversy about, results and their validity, the Electric Power Research Institute initiated a review of five probabilistic risk assessment (PRA) studies conducted by the NUS Corporation. These studies were sponsored by the utilities for the following plants: Big Rock Point (CPCO), Zion (CECO) and Limerick (PECO), and by the NRC for Grand Gulf and Arkansas Nuclear One - Unit 1. To initiate the review, a detailed questionnaire was developed by NUS to solicit information on all key ingredients viewed to be essential for a PRA study. Responses to the questionnaire were provided by the respective PRA study teams (study sponsors and/or their PRA consultants).

Using the questionnaire responses as a guide, NUS reviewed pertinent sections of each PRA study to supplement the information. The material from the five studies was organized, compared and interpreted. The report in draft form was reviewed by the study sponsors and/or their consultants, independent reviewers selected by EPRI and the EPRI Risk Assessment staff. The report presents a summary and interpretation of the review material including principal observations and insights and recommendations for future PRA development.

The principal observations and insights were presented in three groups: (1) management and overview items, (2) comparison of study results with those of the Reactor Safety Study, and (3) methods. These groups were intended for two general classes of readers: the first two for technical management and the third for PRA practitioners.

An illustrative example is reproduced below:

- o Both PWR and BWR plants of considerable diversity were included in the five PRAs studies examined. The studies were sponsored either by a utility company or the NRC. The primary motivation for the utility-sponsored studies was to provide a thorough assessment of impact on the public risk associated with real or potential regulatory induced backfits. The NRC sponsored studies focused on better understanding of dominant accident sequences and development of state-of-the-art plant systems models.

- o It was an enormous challenge to compare the results on some common basis. Although it was not possible to quantitatively pin-point reasons for variability among the study results and for departures from the results of the Reactor Safety Study (RSS), it was, however, possible to uncover general reasons attributable to plant design, plant operation, site characteristics, PRA methods and analytical assumptions postulated.

An example from recommendations for future PRA development is offered below:

Human Interactions. This area is in the developmental phase and programs are under way in the industry to improve both modeling and data collection, which include the information obtained from control room simulations of accident conditions and stresses. Recommendations are:

a. Advances in the treatment of human interactions should be employed in future PRAs: An EPRI project addressing a systematic human action reliability procedure (SHARP) is shortly going to become available.

b. Each PRA should provide a quantitative statement including sensitivities about the overall human impact in accident sequences which delineates the contributions to initiating events, system unavailabilities, dependent failures, and recovery.

INTERIM RESULTS OF THE ACCIDENT SEQUENCE EVALUATION PROGRAM (ASEP)

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This summarizes a paper which will highlight the interim results of ASEP. ASEP is one of many programs sponsored by the Nuclear Regulatory Commission (NRC) to study various aspects of severe core damage accidents. The results of ASEP will ultimately focus on identifying the most likely accident sequences and the high risk accident sequences for most light water reactors (LWRs) accounting for major design and operational differences among plants. The factors contributing to the likelihood or risk dominance of the above sequences will also be identified.

Two documents present the current interim results of ASEP. First, the "Catalog of PRA Dominant Accident Sequence Information," NUREG/CR-3301 (soon to be published), provides a summary of information associated with the dominant accident sequences given in published probabilistic risk assessments (PRAs). Information highlighted in this report includes core melt frequency, dominant sequences and their frequencies, success criteria used, initiating event information, factors contributing most to each sequence's frequency and the relative importance of each factor, and containment failure information. This catalog shows that while certain classes of initiating events and plant functional failures may tend to dominate the core melt frequency for most plants, the specific system and component failures are sometimes very plant specific.

The second document, "Interim Report on Accident Sequence Likelihood Reassessment," takes many of the insights from the previous document as well as those from other studies, and applies these insights to rebaseline existing sequence frequency calculations in PRAs. This effort updates our knowledge concerning accident sequence frequencies and factors that contribute to these frequencies. The document lists the important insights and major phenomenological and human-error-related uncertainties which appear to significantly affect our current understanding of accident sequence frequencies. The results of the rebaseline effort show that, in general, the following accident sequences appear most likely to occur in LWRs:

(1) transients (and small loss of coolant accidents for pressurized water reactors) with early core cooling failure;

(2) transients with long term heat removal failure (for boiling water reactors only);

(3) and transients with failure to scram.

Currently, ASEP efforts are focused on gathering plant data and developing plant class models for the set of general accident sequences identified by the rebaseline effort. These models differ depending on system configuration or operational differences which are typical of existing plants. Where similarities exist among plants, these plant designs are grouped into the plant classes alluded to above. The models will be used to gain insights into the accident sequences that are likely to occur for each plant class and the driving factors for those sequences. These insights will be an extension of the insights gained by the rebaseline effort in that they will apply to a wider range of specific classes of plants (more than just those for which PRAs have been performed) and the accidents associated with those classes.

In addition, plans are being formulated and initial work is in progress for development of inductive models (event trees) and supporting analyses to define a comprehensive set of accident sequences. These models will incorporate state-of-the-art analytical methods as well as the most recent findings in risk or reliability-related research. Ultimately, ASEP results will provide current insights--by plant class--into the most likely accident sequences and the high risk accident sequences including the reasons why such sequences are dominant.

SARRP - Risk Rebaselining and Risk Reduction Analysis*

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Sandia National Laboratories is conducting a program entitled Severe Accident Risk Reduction Program (SARRP) as part of the NRC Severe Accident Research Plan (SARP). The intent of this program is to provide NRC with a sound technical basis for severe accident decision making. Toward the achievement of this intent, SARRP has the following basic objectives:

(1) To incorporate insights gained from severe accident research toward a rebaselining of reactor risks and of the overall uncertainty,

(2) To investigate the use of generic plant categories as a means for generalizing the results of plant-specific risk analyses, and

(3) To evaluate the benefits and costs of proposed new safety features designed to reduce the frequencies and/or consequences of severe accidents.

The first objective (incorporation of insights) is being achieved through interfaces with ongoing programs and task forces. The Accident Sequence Evaluation Program (ASEP) is providing best estimates and bounds for accident sequence frequencies. The SARRP Phenomena Assessment Task Force is performing a state-of-the-art assessment of the threats to containment caused by hydrogen burning, ex-vessel steam spikes, in-vessel steam explosions, and containment leakage. A short-term program entitled Quantitative Uncertainty Estimation for the Source Term (QUEST) is examining uncertainties associated with fission product releases from containment. These investigations are further being coordinated with NRC's task forces on containment loading, containment response, and fission produce source terms.

The second objective (generalization of results) involves a cooperative effort with the ASEP program to derive generic plant categories and to evaluate their risk. Plant categories are derived by evaluating the sensitivity of risk to variations in system design, containment design, and siting factors.

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

Examples of system variations under consideration are the number and type of auxiliary feedwater and high pressure injection trains, dependencies in the service water sybsystems, and differences in actuation criteria. Examples of containment variations include the generic containment type, the failure pressure and free volume (taken together as a product), and the amount of communication between the containment sump and the reactor cavity. Siting factors include demographic and meteorological differences as well as the frequencies of external initiating events.

For the third objective (cost-benefit analysis), the risk reduction benefits of a variety of safety options are being compared to the costs of implementation. The safety options include both plant additions which would affect wide classes of accidents (e.g., hydrogen control systems, filtered venting systems, and core retention devices) and plant modifications which would affect specific accident sequences (e.g., onsite ac power improvements, check valve test and maintenance improvements, and assurance of feed-and-bleed capability). The risk reduction calculations initiate from the rebaselined plant-specific and generic risk assessments outlined above and include estimates of the overall uncertainty.

The approach to uncertainty analysis is based on a Monte Carlo sampling procedure in which the accident sequence frequencies, containment failure mode probabilities, fission product release fractions, and effects of weather on consequences are represented as random variables. The procedure samples from the distributions of these random variables to obtain mean values and uncertainty distributions for the expected costs of severe accidents over the life of a plant.

The SARRP program is evolutionary in nature and involves several iterations. Results from the first complete iteration will be available at the end of Phase 1, in June 1984. This paper will outline the status of the project, describe the approach being utilized, and present some illustrative results.

LESSONS LEARNED FROM PRA ANALYSIS: EXTERNAL EVENTS ANALYSIS

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The 'external-events' part of PRA analysis for nuclear power plants treats accident sequences that begin with any of the various initiators from outside the plant: earthquakes, floods, high winds, airplane crashes, explosions, external fires, and so on. Two types of internal initiators are also generally included in the category of 'external events' initiators: fires and floods within the plant. These are included because the methods used for their analysis are similar to those used for external initiators. The common feature that characterizes accidents initiated by all of these effects is the potential for the initiator not only to start an accident, but also to compromise simultaneously the efficacy of multiple safety systems designed and needed to halt or mitigate the accident. This strong vulnerability to dependencies among failures, or to failures from a common external cause, has required the development of a separate set of analytical methods within the framework of the overall PRA approach.

Of the various accident categories, the one whose development is most advanced is seismic analysis. Methodological development, involving several different approaches to some of the key technical problems, has proceeded to the point that full-scope seismic PRAs have been completed for a few plants, and some elements of seismic PRA analysis have been completed for many more reactors. The probabilistic analysis of internally-initiated fires has also been developed extensively, although its state of maturity is not as great as for earthquakes. Again, there have been a few full-scope PRA fire studies and several less-extensive analyses. For none of the other categories has methodological development or application been extensive: for example, there have only been a very few attempts at probabilistic analysis of internal flooding, high winds, and external flooding; and essentially no attempts for other initiators such as aircraft impacts, external fires, chemical spills, explosions, barge or ship impacts, pipeline leaks, etc.

In this short paper, it is not possible to discuss in detail the current state-of-the-art of PRA analysis for these various accident types. Suffice it to say that, although the conceptual framework for analysis is well founded (see, for example, the "PRA Procedures Guide",NUREG/CR-2300), limitations in the actual analyses themselves have turned up in essentially every category due to insufficient data, inability to model the phenomena in enough detail, and/or lack of resources to carry out a sufficiently full-scope analysis. Rapid evolution has occurred, however, over the past three to five years: significant methodological and data problems have been overcome, at least in the fire and earthquake areas, so that insights from the analyses are becoming more reliable. Nevertheless, in all of these analyses there remain major uncertainties in the numerical results when calculating frequencies of undesired end-points such as core-vulnerability and offsite risk.

One major lesson learned from the analyses performed so far is precisely this lesson about the validity of the results: bluntly, the numerical results of these analyses typically have such large uncertainties that applications requiring even modest numerical precision cannot reliably use these results. For example, a typical quoted uncertainty for frequency of core-melt would be between one and two orders of magnitude, plus-and-minus. It is thus very hard to make comparisons with core-melt frequencies calculated for internal initiators, or with 'safety goals' numbers.

Despite these large numerical uncertainties, engineering insights from these analyses have been very valuable, and much more reliable than the numbers. Probably the most important engineering insight is that the main contributions to undesired endpoints (core vulnerability, offsite risks) come from events significantly beyond the design basis or the licensing basis of the plants. This general conclusion, true in almost every external-events PRA analysis so far with only the rarest of exceptions, provides a remarkable affirmation of the quality of the design and operation of the plants vis-a-vis their design basis ! (Of course, whether the plants are 'safe enough', which means in practice whether their design and operating basis is conservative enough, is another issue,outside our scope here.)

A second insight is that, for almost all of the external initiators, the important accident sequences involve plant-specific idiosyncracies of design, configuration, construction, or operation. This has been almost universally true of the vulnerabilities discovered so far in the seismic and fire PRAs, and is a conclusion that, in retrospect, seems not to have been widely foreseen: only a few years ago, the general feeling in the safety community was that the main insights to be gained from these analyses would be to ascertain whether the broad levels of safety achieved were adequate, or whether significant and sweeping upgrades in the external-events design or licensing bases would be needed. As it turns out, the main contributors seem to be so plant-specific as to provide almost no insights transferrable to other plants, even to seemingly 'sister plants'. Again, this general insight is undoubtedly correct despite the large uncertainties in the numerical values calculated.

Still another major lesson learned is that there are entire classes of initiators for which full-scope PRA analysis is generally not necessary: these include initiators from transportation (barges, aircraft, pipelines, etc.) and from external fires and explosions. Relatively straightforward analysis of the frequency of these types of initiating events has generally shown that their frequency is low enough to eliminate the need for calculating plant response (fragility) given the initiator. The conclusion that the contribution of these initiators can generally be bounded acceptably was expected, but is a comforting conclusion.

In summary, progress in analyzing external events within PRA has been significant in the last few years, and major engineering insights have been gained into vulnerabilities of specific plants. The vulnerabilities have turned out to be highly plant-specific in most cases. Unfortunately, uncertainties in the numerical results of the PRA calculations (core melt and offsite risk frequencies) remain very large, sufficiently large that in general any comparisons with the numerical results of the other parts of PRA, or with 'safety goals' numbers, are not very useful.

ACCIDENT SEQUENCE PRECURSOR LESSONS FOR PRA

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The Accident Sequence Precursor program is concerned with the identification and assessment of precursors to severe core damage accident sequences reported in Licensee Event Reports of historic occurrences at commercial reactor plants. Events identified in the program are typically parts of sequences which, if complete, would result in the unavailability of core cooling in the short term, twenty to thirty minutes, and potentially result in severe core damage.

An initial phase of the program, completed in 1982, considered operational occurrences from 1969-1979 and identified 169 precursor events. These precursors included failures of functions required to mitigate an off-normal event or accident, degradation of two or more functions, and certain initiating events such as losses of off-site power and small break LOCAs.

The precursor events were used to estimate average failure probabilities associated with functionally based event trees onto which each event was mapped. A conditional probability of potential severe core damage (unavailability of core cooling given the occurrence of the precursor) was then calculated for each precursor. These conditional probabilities were used to rank events, exclude less important events from trends analyses, and provide a rough estimate of the frequency of potential severe core damage (unavailability of core cooling) in the 1969-1979 time period.

A second phase of the program dealing with the 1980-1981 time period is being completed at this time. Using the same selection criteria as for the 1969-1979 time period, approximately sixty precursor events have been identified.

Comparing the results of the two time periods, it appears that the number of precursors per plant year is essentially the same in both periods. In many cases, there appears to be an improvement in the 1980-1981 period in function reliability, although the small amount of available data precludes confidence in this result. In addition, the conditional probabilities associated with the 1980-1981 precursors are generally lower - a result of the lower failure probabilities discussed above, the availability of alternate mitigation paths (such as bleed and feed), and removal of dependencies (for example, removal of AC dependency from at least one auxiliary feedwater train).

While this appears promising, the events identified in this program pose important questions for Probabilistic Risk Assessment. Many of the events are complex and include interrelated failures not modelled in contemporary PRAs. In addition, the plant status prior to many events includes combinations of equipment-out-of-service and alternate system operating modes remote from the "clean slate, full power" condition assumed in conjunction with PRA initiators.

Although it is possible that the precursor events are bounded by PRA sequences or have been accommodated by equipment or operational modifications, this must be demonstrated. If PRA models are to be used to gain insight into plant operation and public risk, we must have confidence that these models are representative. This can only be achieved by comparison with actual operation and inclusion of necessary modelling detail to ensure all operational events are adequately bounded.

PRA UNCERTAINTIES

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The PRA Procedures Guide (1) identified three major types of uncertainties which occur in a Probabilistic Risk Analysis (PRA): completeness uncertainty, modeling uncertainty, and data uncertainty. PRA uncertainties can be categorized into more specific types according to their ramifications in a PRA. Table 1 gives such a categorization. In Table 1, "experimental uncertainties" refer to physical variations and "knowledge uncertainties" refer to uncertainties due to lack of knowledge. As experience is gained, knowledge uncertainties will decrease but experimental uncertainties will remain.

Uncertainties in current PRA's are large and most PRA uncertainty analyses only address a fraction of the uncertainties which exist in a PRA. Arguments of Bayesian versus classical statistical analyses which have occurred tend to obscure the fact that neither approach addresses the majority of uncertainties which exist in a PRA. The error spreads and Bayesian distributions which are calculated in current PRA's are thus incomplete and have sizeable uncertainties themselves due to the assumptions involved.

Because of the uncertainties, PRA results can only be meaningfully interpreted as providing imprecise, fuzzy indicators of the risk. The errors which are calculated also only provide imprecise, fuzzy indicators of the true errors which in general are larger. In spite of their uncertainties, PRA's can provide extremely useful information, even critical information, provided those types of results are utilized which are insensitive to these uncertainties.

References

(1) PRA Procedures Guide, NUREG/CR-2300, January 1983.

TABLE 1. CLASSIFICATION OF PRA UNCERTAINTIES

Category	Subcategory	General Type
Data Uncertainties	Variation in parameter values from one population to another	Experimental Uncertainty
	Imprecision in estimated parameter values	Knowledge Uncertainty
	Vagueness in parameter values or parameter ranges	Knowledge Uncertainty
	Indefiniteness in applicability of data	Knowledge Uncertainty
Analyst Uncertainty	Variation in results from analyst to analyst	Experimental Uncertainty
Modeling Uncertainties	Indefiniteness in the comprehensiveness of the model	Knowledge Uncertainty
	Indefiniteness in the characterizations used in the model	Knowledge Uncertainty
Completeness Uncertainties	Indefiniteness as to whether all significant contributors are included	Knowledge Uncertainty
	Indefiniteness as to whether the contributors are included in the proper context and in the correct relative manner	Knowledge Uncertainty

THE ORNL PROBABILISTIC FRACTURE-MECHANICS CODE OCA-P*

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The computer code OCA-P[1] was developed at the request of the USNRC for the purpose of helping to evaluate the integrity of PWR pressure vessels during overcooling accidents (OCA's). The code can be used for both deterministic and probabilistic fracture-mechanics calculations, and consists essentially of OCA-II[2] and a Monte Carlo routine similar to that developed by Strosnider et al.[3] In the probabilistic mode OCA-P generates a large number of vessels (10^6 more or less), each with a different combination of the various values of the different parameters involved in the analysis of flaw behavior. For each of these vessels a deterministic fracture-mechanics analysis is performed (calculations of K_I, K_{Ic}, K_{Ia}) to determine whether vessel "failure" takes place. The conditional probability of "failure" is simply the number of vessels that "fail" divided by the number of vessels generated.

OCA-II is used for the deterministic analysis. Basic input to OCA-II includes, among other things, the primary-system pressure transient and the temperature transient for the coolant in the reactor-vessel downcomer. With this and additional information available OCA-II performs a one-dimensional thermal analysis to obtain the temperature distribution in the wall as a function of time and then a one-dimensional linear-elastic stress analysis.

The stress intensity factor (K_I) is calculated using superposition techniques in combination with influence coefficients, and this provides a mechanism for considering both two-dimensional (2-D) and three-dimensional (3-D) flaws. Influence coefficients are available for 2-D axial and circumferential flaws on the inner and outer surfaces and for two specific inner-surface 3-D flaws. The latter two flaws are a shallow 6/1 semielliptical flaw that can be used for the

*Research sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission under Interagency Agreements 40-551-75 and 40-552-75 with the U.S. Department of Energy under contract W-7405-eng-26 with the Union Carbide Corporation.

first initiation event and a deep, axially oriented, semielliptical flaw with a surface length of 1.83 m (2-m flaw) that can be used for the first arrest and subsequent events.

In the probabilistics portion of OCA-P the simulated parameters are crack depth, K_{Ic}, K_{Ia}, $RTNDT_0$, $\Delta RTNDT$, fluence, and concentrations of copper and nickel. A flaw-depth distribution function is included that was taken from the Marshall Report,[4] and truncated normal distributions are currently used for the other simulated parameters. Vessel failure criteria in the code include $K_I > K_{Ia}$, $K_I = K_{Ia} \geq 220$ MPa $\sqrt{m}$, and the onset of plastic instability.

Output from the OCA-P probabilistics analysis includes plots of the conditional probability of failure vs the number of trials (used as a check on convergence), and histograms of crack depths, times and values of $T - RTNDT$ corresponding to crack initiation and failure.

OCA-P has been checked against similar codes and is presently being used in the Integrated Pressurized Thermal Shock Program for estimating the conditional probability of vessel failure for specific PWR plants.

References

1. D. G. Ball and R. D. Cheverton, *OCA-P, A Probabilistics Fracture-Mechanics Code for Application to Pressure Vessels*, NUREG/CR-XXXX (ORNL-5991), Oak Ridge National Laboratory, Oak Ridge, TN (in preparation).

2. D. G. Ball, R. D. Cheverton, J. B. Drake and S. K. Iskander, *OCA-II, A Code for Calculating the Behavior of 2-D and 3-D Surface Flaws in a Pressure Vessel Subjected to Temperature and Pressure Transients*, NUREG/CR-XXXX (ORNL-5934), Oak Ridge National Laboratory, Oak Ridge, TN (in preparation).

3. Personal communication with J. Strosnider, USNRC.

4. W. Marshall, *An Assessment of the Integrity of PWR Pressure Vessels*, United Kingdom Atomic Energy Authority, Second Report, March 1982.

PFM - THE WESTINGHOUSE PROBABILISTIC FRACTURE MECHANICS COMPUTER PROGRAM

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Westinghouse Electric Corporation has developed a multi-purpose probabilistic fracture mechanics computer program designated PFM which is especially oriented toward the evaluation of reactor vessels subjected to transient loads and irradiation induced material toughness degradation. The capabilities of the program are described, applications are briefly summarized and the continuing development is outlined.

The basic computer program was originally developed by G. M. Jouris, Westinghouse R&D Center under sponsorship by the Westinghouse Nuclear Technology Division. Verifications and application/production revisions were carried out by K. R. Balkey, E. L. Furchi and the author.

Parameters which may be random variables are fluence, RT_{NDT}, RT_{NDT} shift with irradiation, K_{Ic}, K_{Ia}, flaw size, K_I, E (modulus of elasticity), α (coefficient of thermal expansion), phosphorus, and copper. The inherent variability of K_{Ic} and K_{Ia} may be considered in addition to the functional variability. K_I may be taken as a point by point random variable. These parameters may also be made constants as desired.

The fluence variability through the wall may be discrete, exponential or cubic and covers both E>1 MeV and dPa representations. Regulatory Guide 1.99, Westinghouse or Guthrie Trend Curves may be selected. Four choices of fracture toughness curves are available - mean K_{Ic} and K_{Ia} (Westinghouse TANH Fit to ASME Section XI, Appendix A data base); mean K_{Ic} and K_{Ia} fit described in Appendix H of NRC Staff Evaluation of Pressurized Thermal Shock (Enclosure A to SECY-82-465, 11/23/82); lower bound K_{Ic} and K_{Ia} curves of ASME Section XI, Appendix A and lower bound K_{IR} of ASME Section III, Appendix G for both K_{Ic} and K_{Ia}. The postulated flaw depth may be a fixed number or a random variable. If the flaw depth is a random variable, the distribution may be described by a discrete distribution (histogram) or by a two parameter exponential. Thus, distributions as identified by

Octavia, Marshall and Becher may be used. Flaws can be postulated in either the longitudinal or circumferential directions.

Multiple initiation and arrest events are considered including a scenario of first initiation as a finite flaw and all subsequent arrests and initiations as a continuous flaw. The upper shelf initiation and arrest toughness may be arbitrarily chosen. Thus degrees of elastic-plastic fracture resistance in terms of the failure criterion are available. Non arrest may be defined by $K_I > K_{Ia}$, $K_I > \min$ (K_{Ia}, upper shelf toughness) or the propagating crack exceeds in depth some chosen percentage of the thickness. For monotonically decreasing thermal transients, two warm prestressing options are available. The first initiation of a postulated flaw or the reinitiation of an arrested flaw is postulated not to occur if (option 1) the applied K is in a decreasing K field or (option 2) K is in a constant or an increasing K field but does not exceed some arbitrary percentage of the prior maximum K value experienced by the flaw.

Monte Carlo simulation is applied either with or without importance sampling. The number of simulation trials is an input parameter. Comparisons have been successfully made with VISA for pressurized thermal shock evaluations. Validation studies involving importance sampling have been performed. Special pre-processors have been developed including an arbitrary multirun capability.

PFM has been used to analyze loss-of-coolant (LOCA), pressurized thermal shock and overpressurization scenarios. The thermal hydraulic parameters for these events can be input either in actual time history form or as stylized (i.e., exponential) characterizations, including appropriate heat transfer input data. Several sensitivity studies have been performed on both generic and plant specific applications yielding a better understanding of the behavior of the results for most of the above input parameters.

Data base studies and variability evaluations are among the continuing activities. Various random variable descriptions are being updated and/or included in the program. Elastic-plastic capabilities are being further integrated into the program. Random flaw descriptions will be upgraded.

Theoretical Predictions of Failure Probabilities for PWR Pressure Vessels Subjected to Accident Conditions

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The second Marshall Report (1982) presented a detailed analysis of the integrity of PWR pressure vessels. As part of that study a theoretical model to calculate failure probabilities for both normal operational and accident conditions was produced. Since the publication of that Report further probabilistic analyses have been made of the most severe accident conditions, the large LOCA and the steam line break.

The purpose of these studies was to extend the relatively simple probabilistic model to consider the effect of using more accurate representations of the crack distribution, the stress intensities and the crack growth laws. The effect of permitting a limited amount of stable crack growth has also been considered. Two of the main areas are reviewed below.

a) The crack distribution

The number of defects of a given depth (in the through-thickness dimension) is determined by the number of defects present after manufacture and the effectiveness of the pre-service non-destructive examination (NDE). Both these functions have been reviewed and in particular a detailed study of NDE capability has been made. This study has included the derivation of a new theoretical model for the reliability of ultrasonic inspection, and an analysis of the UKAEA Defect Detection Trials. The outcome of this work has been the derivation of a new function for the probability of not detecting a defect. This function considers the defect by its length and depth, and shows how the reliability improves with decreasing aspect ratio.

b) Stable crack growth

The failure criterion used is related to the probability that the stress intensity value around the defect exceeds the fracture toughness. Initially this criterion is used to give the probabilities of crack initiation. After initiation, and assuming that the vessel remains on the "upper shelf", the defect may grow stably. A limited amount of this stable growth is still "J controlled", and can be modelled as causing an increase in the fracture toughness. The effect of including this model on the predicted fracture toughness has been studied.

The application of these and other model developments to the analysis of the large LOCA and steam line break has produced revised estimates of the failure probability of the vessel. These indicate that the main contributors to the failure probability are the nozzle regions followed by the beltline. For the most severe transient considered, the large LOCA, the probability of vessel failure is estimated to be one to two orders of magnitude lower than previously predicted. These results, which are relevant to a vessel made of A508 Class 3 steel, assume that the vessel remains on the "upper-shelf" during the period of highest stress, and are applicable only to the particular transient considered.

INPUT DISTRIBUTIONS IN VISA

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An NRC sponsored project is currently underway at Pacific Northwest Laboratory to develop improvements in the probabilistic analyses used to model pressurized thermal shock (PTS) incidents in reactor pressure vessels, and further, to incorporate these improvements into existing Vessel Integrity Simulation Analysis (VISA) Code. Changes in the way the flaw size distribution is handled in VISA and changes in the error structure of the variables in the equation used to compute the shift in reference temperature for nil ductility transition (ΔRT_{NDT}) have been investigated. Some of the effects of making these changes are discussed.

At present, the simulation loop in VISA selects a flaw size at the beginning of each iteration, with flaw size values being selected according to the OCTAVIA distribution. Because flaw sizes are randomly selected, it is necessary to specify a flaw size distribution at the beginning of a simulation run. It is proposed that the code be modified so as to compute the <u>conditional</u> probability of failure, <u>given</u> a particular flaw size. This modification has at least two important advantages.

- Simulation can be eliminated for flaw sizes which are found to have a negligible effect on the probability of vessel failure.
- Once the probability of failure is estimated by simulation for each of several flaw sizes, the overall probability of failure from these flaws can be computed for a wide variety of flaw size distributions without additional simulation. Further, no important difficulties arise if the number of parameters used to describe the flaw size distribution is increased.

An example is presented to show how the computations are done. The example is based upon data reported by Dufresne and Lucia, where the frequency of undetected flaws is tabulated by length and depth.

Work performed for the U.S. Nuclear Regulatory Commission under a Related Services Agreement with the U.S. Department of Energy, Contract DE-AC06-76RLO.

In the present version of VISA, the shift in RT_{NDT} is modeled as a function of fluence, copper content and nickel content. In its most general form, this function is

$$\Delta RT_{NDT} = f(\phi + \eta_1, Cu + \eta_2, Ni + \eta_3) + \varepsilon \quad ,$$

where ϕ is fluence in neutrons ($E > 1$ MEV)/cm^2,
Cu is copper content in weight percent of weld,
Ni is nickel content in weight percent of weld,

and η_1, η_2, η_3 and ε are errors in the determinations of ϕ, Cu, Ni, and ΔRT_{NDT}, respectively. In VISA, the function f has the form

$$\Delta RT_{NDT} = (-4.83 + 476*Cu + 267*Cu*Ni)(\phi/10^{19})^{0.218}$$

The errors η_1 and η_2 as assumed to have independent normal distributions, while η_3 and ε are taken to be zero. Other probability models relating the η_i's and ε are possible, indeed some may be more realistic than the model being used. Results are presented which show how various probability distributions affect ΔRT_{NDT}.

MARGINAL DISTRIBUTIONS OF MATERIAL PROPERTIES IN RELATIONSHIP TO PRESSURE VESSEL INTEGRITY*

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INTRODUCTION

From a probabilistic viewpoint the integrity of a nuclear reactor pressure vessel rests on two sets of probability distributions. The first set relates to the stresses placed on the vessel, which are reassessed from time to time. For example, current concern about pressurized thermal shock was not part of the original design basis for the vessels. The second set relates to the subject of our work, the ability of the materials of which the vessel is made to resist the stresses imposed on them. The material property of most concern is its ability to resist crack initiation and growth, fracture toughness, which must be defined by another set of probability distributions. The material fracture toughness of the pressure vessel materials is poorly defined even before the vessel has been operated, and the development of distributions for it is further complicated by the uncertain way in which it changes under the influence of radiation. Additional complications are introduced by the rapid way in which understanding of all these features is changing with time.

IMPORTANT ASPECTS OF MATERIAL BEHAVIOR

Two features of the fracture toughness behavior of ferritic steels dominate; the existence of a transition from brittle to ductile behavior, and the ability of the steel to resist crack initiation and growth at the higher temperatures where the steel is behaving in a ductile fashion. Lacking well developed theoretical insights into either of these two phenomena, we are forced to rely heavily on empiricism to infer the probability distributions which must be used in the study of pressure vessel integrity. The key distributions appear to be as follows:

1. The temperature range over which the brittle - ductile transition takes place is usually approximated by a single temperature. The effect of irradiation on this temperature has been described by statistical equations. The probability distribution of actual observations about the best fit equation remains to be defined. The approach would be enhanced if the single temperature were replaced by the parameters of an equation defining the transition (such as the coefficients of a hyperbolic tangent function) so that details of the complex changes which occur during irradiation might be properly modelled.

2. If a single temperature has been used, the change in position of the upper shelf with irradiation must then be defined, also, and the manner in which fracture toughness varies about the mean value specified in a probabilistic fashion.

3. Items 1 and 2 above fail to address the need to predict valid fracture toughness from available data. The Pressure

*Project funded by EPRI under contract RP2180-9, under the direction of Dr. D. Norris, Project Manager.

Vessel Codes require that the quantity RTNDT is used to relate the Charpy V-notch (CVN) measurements taken in surveillance capsule studies to fracture toughness. Thus, the probability distribution for the fracture toughness of irradiated steel is a function of the variance of the RTNDT of the unirradiated material, the variance of the initial CVN energy, the variance about the predictive equation for radiation shift, and the variance of the relationship between these quantities and valid fracture toughness. This web of dependencies introduces severe problems. For example, RTNDT is based on the most conservative of either the NDT test or the results taken from the CVN test. The probability distribution of the RTNDT measurement of the unirradiated material thus depends upon the manner in which it was taken. If based on NDT there is one distribution, on the CVN data there are other distributions, if unspecified then a hybrid distribution must be developed depending upon the manner in which the population of RTNDT measurements is set.

4. The current methods require the computation of joint probability distributions of the RTNDT measurement, the fracture toughness brittle - ductile transition temperature, and the relationship between a shift measurement using the CVN test and the shift of fracture toughness. A simplification can be introduced if the RTNDT referencing step is omitted. This leaves the probability distributions of the unreferenced ductile - brittle transition and that of the CVN test to infer the shift of fracture toughness. A further simplification results from the utilization of the recent work of the ASME/MPC Working Group on Reference Toughness, who have developed and tested a procedure for directly referencing fracture toughness from the CVN test.

DISCUSSION

The probability distributions for the material properties described above are being developed. It appears at this time that the heavy dependence of probabilistic assessments of the fracture toughness of pressure vessel materials on the results of the reactor surveillance programs, coupled with the recent extensive research programs on the relationship between the CVN test and fracture toughness, make the overall CVN referencing approach the method of choice in a probabilistic study. The currently most vexed question in this work concerns the upper shelf fracture toughness. The CVN referencing work refered to above used J-R curve data, which are the only upper shelf fracture toughness measurements generally available at this time. A considerable volume of experimental work is in progress related to crack initiation in the ductile regime. At the moment, it appears that the J approach used in the work referred to gives conservative results. However, if a better procedure becomes available, a consensus agreement on a new approach is urgently needed.

NRC FORM 335 (7-77) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER *(Assigned by DDC)* NUREG/CP-0047
4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)* Transactions of the Eleventh Water Reactor Safety Research Information Meeting	2. *(Leave blank)* 3. RECIPIENT'S ACCESSION NO.
7. AUTHOR(S) Conference papers by various authors; compiled by Stanley A. Szawlewicz	5. DATE REPORT COMPLETED MONTH Sept. YEAR 1983
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)* Office of Nuclear Regulatory Research U.S. Nuclear Regulatory Commission Washington, D.C. 20555	DATE REPORT ISSUED MONTH Sept. YEAR 1983 6. *(Leave blank)* 8. *(Leave blank)*
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)* Same as Item 9 above.	10. PROJECT/TASK/WORK UNIT NO. 11. CONTRACT NO.
13. TYPE OF REPORT Transactions of conference on safety research	PERIOD COVERED *(Inclusive dates)* October 24-28, 1983
15. SUPPLEMENTARY NOTES	14. *(Leave blank)*

16. ABSTRACT *(200 words or less)*

This report contains summaries of papers on reactor safety research work to be presented at the 11th Water Reactor Safety Research Information Meeting. The meeting will be held at the National Bureau of Standards in Gaithersburg, Maryland, October 24-28, 1983. The summary reports highlight the programs and results of nuclear safety research work sponsored by the Office of Nuclear Regulatory Research, USNRC. Summaries of invited papers are also included. The latter represent work on reactor safety research conducted by the electric utilities through the Electric Power Research Institute, the nuclear industry, and various government and industry organizations in Europe and Japan. The summaries have been compiled in one report to facilitate discourse and the open exchange of information during the course of the meeting.

17. KEY WORDS AND DOCUMENT ANALYSIS	17a. DESCRIPTORS
17b. IDENTIFIERS/OPEN-ENDED TERMS	

18. AVAILABILITY STATEMENT	19. SECURITY CLASS *(This report)*	21. NO. OF PAGES
	Unclassified	375
Unlimited	20. SECURITY CLASS *(This page)* Unclassified	22. PRICE S

NRC FORM 335 (7-77)

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