NUREG/CP-0161

Transactions of the Twenty-Fifth Water Reactor Safety Information Meeting

To Be Held at
Bethesda Marriott Hotel
Bethesda, Maryland
October 20-22, 1997

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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PREFACE

This report contains summaries of papers on reactor safety research to be presented at the 25th Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel in Bethesda, Maryland, October 20-22, 1997. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, U.S. NRC. Summaries of invited papers concerning nuclear safety issues from U.S. government laboratories, the electric utilities, the nuclear industry, and from foreign governments and industry are also included. The summaries have been compiled in one report to provide a basis for meaningful discussion of information exchanged during the course of the meeting, and are given in order of their presentation in each session.

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

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CONSIDERATIONS IN THE USE OF NDE AND REVISED FRACTURE TOUGHNESS CURVES IN REACTOR PRESSURE VESSEL INTEGRITY ANALYSIS

E.M. Hackett, C.J. Fairbanks, D.A. Jackson, S.N. Malik, M.G. Vassilaros and M.E. Mayfield
U. S. Nuclear Regulatory Commission, Washington, DC

Structural integrity analyses are used to assure the safe operation of nuclear reactor pressure vessels (RPVs). These structural integrity analyses are used in setting allowable operating pressure and temperature limits, in demonstrating that pressure vessels can survive pressurized thermal shock (PTS) conditions, and in determining whether flaws that might be detected during an in service inspection can remain in the vessel during subsequent operation or if they must be repaired. These applications of structural integrity analysis methods differ in some of the specifics of the application -- flaw sizes considered, rigor in the fracture mechanics models, and explicit margins, for example -- but they are all similar in general approach.

Two areas that are of particular importance in regard to RPV structural integrity analyses are: (1) use of standardized fracture toughness curves and; (2) use of NDE to determine RPV flaw distributions. With respect to (1), a traditional approach, endorsed by NRC regulations, involves the use of ASME Code fracture toughness curves which are indexed for specific levels of embrittlement by impact toughness test results for the materials of interest. This methodology can be subject to significant uncertainties which relate to material and test method variability. The uncertainties, in turn, require the application of additional margin terms in the regulatory approach. In certain cases this can result in overly conservative conclusions with respect to RPV integrity.

A more recently proposed approach involves the use of smaller specimens to directly measure the fracture toughness of the material of interest and the assumption of a universal or "master" curve to define the fracture behavior of the material in the transition region. This approach holds the promise of greater accuracy in the determination of the fracture toughness for the vessel material of interest but is still subject to certain limitations. The NRC is supporting ongoing research and standardization efforts in this area. The NRC perspective on advantages and limitations of this approach will be presented.

With respect to (2), probabilistic fracture mechanics analyses for vessel failure under conditions like PTS have typically involved the use of assumed flaw distributions such as that attributed to Marshall (1976, 1982). Recent NRC-sponsored work in this area has non-destructively evaluated actual unirradiated reactor vessels to determine the specific flaw distributions and their variability and generic applicability. Preliminary results and conclusions will be presented.

In addition to the above, the prediction of RPV embrittlement due to neutron irradiation is a key element of RPV integrity evaluations. The current approach, described in NRC regulatory guide 1.99, revision 2, relies on an empirical methodology which is supported by extensive statistical evaluation of available data and by physical and mechanistic insights. Due to the largely empirical nature of the correlations that are used, the methodology is subject to considerable uncertainties which require compensation by additional margin terms. It would be desirable to use a non destructive methodology that could yield a direct measurement of vessel embrittlement. The NRC has therefore undertaken a research initiative in this area with the initial goal of evaluating non-destructive measurement technologies which have the capability for discerning and quantifying fine scale microstructural damage in RPV steels. The technical approach and preliminary results for this effort will be presented.

Resolving Embrittlement Issues - Industry Activities

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Brittle fracture of reactor vessel materials during Pressurized Thermal Shock (PTS) events must always remain incredible. Toward that end, utilities and NRC have devoted considerable resources over the past twenty years in evaluating reductions in resistance to brittle fracture due to interactions between steel and neutrons. Historically, new programs to address embrittlement have been established in response to acute perceived shortcomings in vessel embrittlement safety analyses. Such programs tend to have short term objectives that do not comprehensively address longer term or broader issues. In an effort to achieve greater stability in the evaluation of embrittlement issues, the industry (individual utilities, Owner's Groups, EPRI and NEI) has transitioned to a proactive and coordinated approach for managing embrittlement issues. The approach involves identification of new or improvable technology areas and subsequent planning and development work to establish the viability of the new or improved methods. Selection of areas for focus was based on the goal of significantly improving the accuracy of determination of vessel material embrittlement state. Direct measurement of fracture toughness and improvement of embrittlement correlations were the two technology areas targeted for development.

Material resistance to brittle fracture is determined by the material property called fracture toughness. Material resistance to fracture is currently evaluated by estimating toughness using measurements from Charpy Impact tests. Effects of embrittlement are determined by measuring changes in Charpy Impact behavior. Advances in fracture toughness measurement and analysis technology have made it possible to measure the fracture toughness directly on surveillance material and to evaluate the accuracy of the measurement. Eventual implementation of the Master Curve Approach to measure toughness would permit determination of toughness explicitly rather than estimation of change in toughness based on addition of a shift in a different material behavior added to an initial measurement of still another different material behavior. While the current method is certainly extremely conservative, it cannot be analytically related to the material property of interest; therefore, hidden conservatisms in the correlation between the various non-toughness material property measurement correlations with actual toughness cannot ever be determined. Direct measurement of toughness permits determination of those hidden conservatisms. The Owner's Groups and EPRI are conducting tests on a variety of vessel materials using the Master Curve Approach.

When surveillance data are not available, vessel material toughness is estimated based on measurements of Charpy behavior of unirradiated behavior, which is adjusted for effects of irradiation based on material composition. The correlation between composition, neutron exposure, and original toughness estimates has been periodically revised over time. The current correlation is based on 160 data points from surveillance capsules analyzed through approximately 198?. The Owner's Groups and EPRI have assisted an ASTM effort to revise the correlation using the additional 600 datapoints developed since 198?. Industry has reviewed an initial database and provided updated fluence evaluation, correction of erroneous and duplicate entries, provided missing chemical analysis and flux information, and developed a significantly more accurate evaluation of irradiation temperature. The improved, expanded database has served as the basis for developing improved correlations. Preliminary analysis of the newer correlations indicate significant improvement of accuracy of the predictability of embrittlement is possible. While efforts to improve embrittlement correlations and develop direct methods of measuring fracture toughness were ongoing, the industry supported a broad effort to collect and analyze vessel material property data. This program was developed to address an acute perceived shortcoming in vessel embrittlement safety analyses due to potential greater than anticipated material copper variability. A significant product of that effort is the identification of a variety of material property databases and the recognition of certain inefficiencies associated with maintenance of duplicative databases.

As short term and longer term efforts are completed. additional productive efforts have been identified and initiated. In particular, a program to improve the accuracy of flaw distribution estimates has been developed. In the longer term, deterministic evaluations of susceptibility to brittle fracture during PTS events should be pursued. Alternatively, the bases of Reg. Guide 1.154 could be redeveloped, and generic analyses performed to re-benchmark conventional PTS risk analysis. In this latter effort, a program to more accurately estimate event frequency should greatly improve the ability to quantify margins of safety.

Nondestructive Characterization of Embrittlement in RPV Steels
A Feasibility Study

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The Nuclear Regulatory Commission recently initiated a study by NIST to assess the feasibility of using physical measurements for evaluating radiation embrittlement in RPV steels. Ultrasonic and magnetic measurements provide the most promising approaches for nondestructive characterization of RPV steels because elastic waves and magnetic fields can sense the microstructural changes which embrittle materials. Our approach to establishing feasibility is to use samples of unirradiated materials to explore three main issues:

- *Are physical measurements sensitive to the ductile to brittle transition?*

We are using ultrasonic and magnetic measurements to characterize A533B steel over a temperature range centered on the ductile-to-brittle transition temperature. The test materials are A533B plates and welds cut from the SNUPPS reactor pressure vessel. The material has not been exposed to radiation and will not be heat treated or otherwise processed to modify its properties.

- *Can physical measurements be correlated to precipitation hardening?*

We are correlating ultrasonic and magnetic measurements with hardness of a copper-precipitation-strengthened steel. The test materials are copper-precipitation-hardened steel plates in the following conditions: solution treated, peak aged, and over aged.

- *Can measurements of sufficient accuracy be obtained through stainless steel cladding?*

We will assess the accuracy of ultrasonic velocity and attenuation measurements and of magnetic hysteresis loop measurements taken on clad steel sections removed from an RPV. The test materials are two sections, each about 1 m^2, of 200mm thick A533B, clad with approximately 10mm of stainless steel, taken from the Shoreham RPV. The steel sections have never been irradiated. One piece has an axial weld along the mid-width. The other piece is not welded.

The physical measurements being evaluated in the feasibility study include:

Ultrasonic velocity and attenuation: The complete elastic-stiffness tensor is measured by resonant ultrasound spectroscopy (RUS) to determine the elastic constants and internal friction. This is considered a baseline characterization which would reveals any sensitivity of ultrasonic measurements to the ductile-to-brittle transition and to precipitation hardening.

Nonlinear ultrasonics: NIST uses high-power, low-distortion electronics and an infrared Michelson interferometer to perform harmonic generation experiments. The interferometer enables measurement of absolute ultrasonic-wave displacements, including the harmonics arising from nonlinear elastic behavior. Nonlinearity coefficients determined in this way can be related to lattice distortion caused by precipitates and other microstructural features.

Internal friction (stress biased): NIST recently developed a measurement technique to trap ultrasonic resonant modes in a local region of a cylinder. Cylinders are loaded in a tensile machine to allow the internal friction due to dislocation motion to be measured as a function of stress and temperature. The temperature dependence of dislocation mobility will be compared to the Charpy transition temperature.

Magnetics: A vibrating-sample magnetometer is used to measure the hysteresis curve and the associated parameters such as the coercive force, remanence, saturation magnetization and the hysteresis loss. This also is considered a baseline characterization which would reveal any sensitivity of magnetic measurements to the ductile-to-brittle transition and to precipitation hardening.

Micromagnetics: The magnetic microstructure consists of domains separated by Bloch walls. Motion of the domain walls cause micromagnetic-effects which can be measured at different positions on the B(H) curve. These effects reflect atomic scale interactions similar to those that control dislocation mobility. Barkhausen noise, incremental permeability and magnetostriction are all measurements that are associated with domain wall motion.

The feasibility study is an eight-month project initiated in May, 1997. Preliminary results obtained using each of the measurement techniques will be included in the presentation for both A533B and the copper-precipitation-strengthened steels.

EPRI Activities to Address Reactor Pressure Vessel Integrity Issues

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ABSTRACT

The demonstration of reactor pressure vessel (RPV) structural integrity is an essential element in ensuring the continued safe and reliable operation of U.S. nuclear power plants. The Electric Power Research Institute (EPRI), through its domestic and international member utilities, continues to pursue an aggressive research program to develop technologies and capabilities that will address issues associated with reactor pressure vessel integrity. Ongoing research in the EPRI Nuclear Power Group Materials Performance Program covers a broad range of technical areas associated with RPVs. The R&D program is structured as follows: (1) Materials Issues, (2) Embrittlement Management, (3) Thermal Annealing Methods, (4) Pressure & Temperature (P-T) Limit Optimization, (5) Resolution of PTS Issues, and (6) Microstructural Characterization of RPV Steels.

<ins>Materials Issues and Embrittlement Management</ins>. Activities include: (1) Development of material property databases to assist in resolution of issues associated with Nuclear Regulatory Commission Generic Letter 92-01 and to address uncertainties associated with RPV material properties; (2) Development of vessel-specific flaw distribution estimates and information regarding inspection reliability through characterization of material removed from a decommissioned RPV; (3) Application of miniature specimen testing techniques to provide direct measurement of RPV material fracture toughness, (4) Application of American Society of Testing and Materials (ASTM) Master Curve fracture toughness methodology to RPV integrity characterization for both unirradiated and irradiated materials.

<ins>Reactor Vessel Thermal Annealing Methods</ins>. Activities include: (1) Engineering feasibility demonstration of thermal annealing, and (2) Development of an irradiation-anneal-reirradiation database to evaluate material property behavior following a thermal anneal in order to optimize the annealing process.

<ins>RPV P-T Limit Optimization</ins>. The principal activity involves development of a software tool to calculate RPV heatup/cooldown curves utilizing present guidance provided in Section XI, Appendix G of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

Resolution of PTS Issues.

This work focuses on reevaluation of the fracture mechanics modeling assumptions and acceptance criteria for the pressurized thermal shock (PTS) analyses. The primary goals include development of an alternate approach to evaluate the risk of RPV failure due to PTS events, and providing means by which utilities evaluate corrective measures effectively (such as flux reduction, plant modifications, changes in plant operating procedures, etc.). A plant-specific application of the methodology is also envisioned.

Irradiation-Induced Changes in RPV Steels.

EPRI developed a 5-year program to identify and study the mechanisms that cause radiation embrittlement. The goal of the program is to develop a quantitative link between microstructure and mechanical properties in order to significantly improve predictive capabilities. The program is structured to examine the effects of chemistry, flux, fluence and product form (plate, weld, and forging) on irradiation damage of typical RPV steels. A series of reports will be published that document the irradiation-induced changes and embrittlement behavior of the steels in this program.

Aspects of the Boiling Water Reactor Strainer Debris Blockage Study

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Between September 1993 and May 1997 the Office of Regulatory Research has sponsored three distinct studies that included several tests and experiments as part of the United States Nuclear Regulatory Commission's (NRC's) reevaluation of the debris blockage of emergency core cooling system (ECCS) suction strainers in boiling water reactors. The three studies were: Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris, Experimental Investigation of Head Loss and Sedimentation Characteristics of Reflective Metallic Insulation Debris, and the Drywell Debris Transport Study. The recent studies focused on debris blockages or fouling of strainers. The NRC had previously evaluated debris blockages in the early 1980s as part of unresolved safety issue (USI) A-43, "Containment Emergency Sump Performance." Both, the recent studies and the USI A-43 Studies, evaluated the impact of the accumulation of debris (loss-of-coolant accident (LOCA) generated and/or operational) on suction strainers. The accumulated debris will increase the flow resistance across the strainer thus decreasing the available net positive suction head (NPSH) to the ECCS pumps drawing suction through the suction strainers. So it is important to be able to estimate the head loss across a fouled suction strainer to determine whether the postulated debris loads during a LOCA will cause head losses that will reduce the available NPSH below the NPSH required for pump operation. Also, it is important to be able to estimate the amount of debris that may accumulate on a strainer following a LOCA.

To estimate the amount of debris that would reach the strainer and its impact on head loss the NRC sponsored many experiments and tests:

1. fibrous and particulate debris head loss tests,
2. particulate filtration tests,
3. fibrous and particulate debris settling tests,
4. metallic debris generation test,
5. metallic head loss tests,
6. metallic settling tests,
7. fibrous debris water erosion tests,
8. "small scale" fibrous debris transport experiments, and
9. "large scale" fibrous debris transport experiments.

The first two tests listed above were instrumental in the development of the NUREG/CR-6224 head loss correlation, one of the more significant accomplishments during the recent studies, to predict the head losses across a fouled suction strainer. The correlation was developed by Science and Engineering Associates, Inc.

LOCA GENERATED DEBRIS TRANSPORT IN A BWR DRYWELL

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The objectives of this research were to: a) to estimate drywell transport fractions associated with the transport of LOCA generated fibrous insulation debris and b) to identify important LOCA thermal-hydraulics phenomena (or mechanisms) and plant features that control drywell transport. For the purpose of this drywell debris transport study (DDTS), the drywell transport factor is defined as the fraction of the volume of fibrous insulation contained in the zone of influence (ZOI)[1] that is transported to the suppression pool as a result of steam and water flows that occur during blowdown and long-term ECCS recirculation phase. The focus of DDTS is the transport of fibrous debris by double ended guillotine break in main steam line and recirculation line in the mid region of a BWR drywell. This paper describes the DDTS and presents significant findings.

Figure 1 illustrates transport pathways considered in the study. The overall problem was decomposed into several coupled phenomena, and three phenomena (or mechanisms) were determined to be controlling and were studied experimentally: 1) capture of debris on various structures (e.g., gratings) and enclosures (e.g., reactor cavity) while they are transported by steam flow during blowdown, 2) erosion and washdown of debris captured during blowdown by water flow induced by containment sprays or ECCS recirculation flow, and 3) transport of debris in the pools formed on the drywell pool. The experimental program was divided into the following experiments: 1) separate effects experiments to study inertial debris capture on drywell structure during blowdown, 2) integrated effects experiments to study jet driven transport of insulation debris over drywell structures assembled to the desired congestion levels, and 3) separate effects experiments to study washdown of small pieces of debris subjected to spray flow and erosion of large pieces of debris when subjected to break recirculation flow.

The results of the experiments demonstrated that:

- Debris generated by break jet impingement on fiberglass insulation blankets can be broadly characterized into three size categories: small, medium, large. Transport is strongly controlled by debris size and structural composition, and wetness on drywell structures.
- Wet gratings are the major drywell structures that can capture small and medium debris. Capture efficiency of other structures is relatively small.
- Wet or dry gratings have nearly 100% capture-efficiency to remove large pieces. Erosion of these large pieces by steam flow at velocities up to 180 ft/s is negligible.
- Wet vent plates have a potential to remove up to 20% of small debris and 50% of medium and large debris.
- Majority of the small pieces deposited on various drywell structures will be washed down by containment sprays and/or break overflow. Erosion of large pieces by break overflow is time dependent. Debris produced by erosion resemble small pieces and have near-neutral buoyancy.

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[1] Zone of Influence is the region surrounding the break where impingement pressures are sufficiently large to inflict damage on the insulation blankets.

- Transport in the water pools formed on the drywell floor are strongly dependent on the pool dynamics and the residual turbulence levels. Nearly 100% of the small and medium pieces will be transported to the suppression pool if the pool transport occurs as a result of break break overflow. A potential exists for gravitational settling of medium and small pieces if the drywell pool flow is governed by spray flow [Reference 1].

These experimental results were used in conjunction with engineering computations to estimate transport fractions for a variety of LOCA accident progression scenarios for Mark I, II and III BWR containments. For each scenario an upper bound and central estimate transport fractions were calculated. It is unlikely that transport fraction associated with upper bound would be exceeded. The central estimate is judged to represent a more realistic level of transport. However, central estimates are scenario specific and their application is plant-specific. Table 1 provides these estimates for all three containments. These transport fractions are to be applied to the insulation contained in the entire of ZOI which is conservatively defined for the break consideration.

Table 1. Estimates of Drywell Transport Fractions for Three Generic BWRs [2].

	Estimate for MSL Break		Estimate for RL Break	
	Central Estimate	Upper Bound	Central Estimate	Upper Bound
Mark I	0.15	0.31	0.23	0.39
Mark II	0.20	0.31	0.24	0.39
Mark III	0.16	0.29	0.20	0.39

References:

1. P. Murthy and M. Padmanabhan, "ECCS Strainer Model Study Transport and Entrainment Studies in a Laboratory Flume," 31-94/M216F, Alden Research Laboratory, Inc., 1994.

[2] BWROG Debris Generation Fractions of 0.22, 0.38, and 0.40 for small, large and canvassed pieces, respectively.

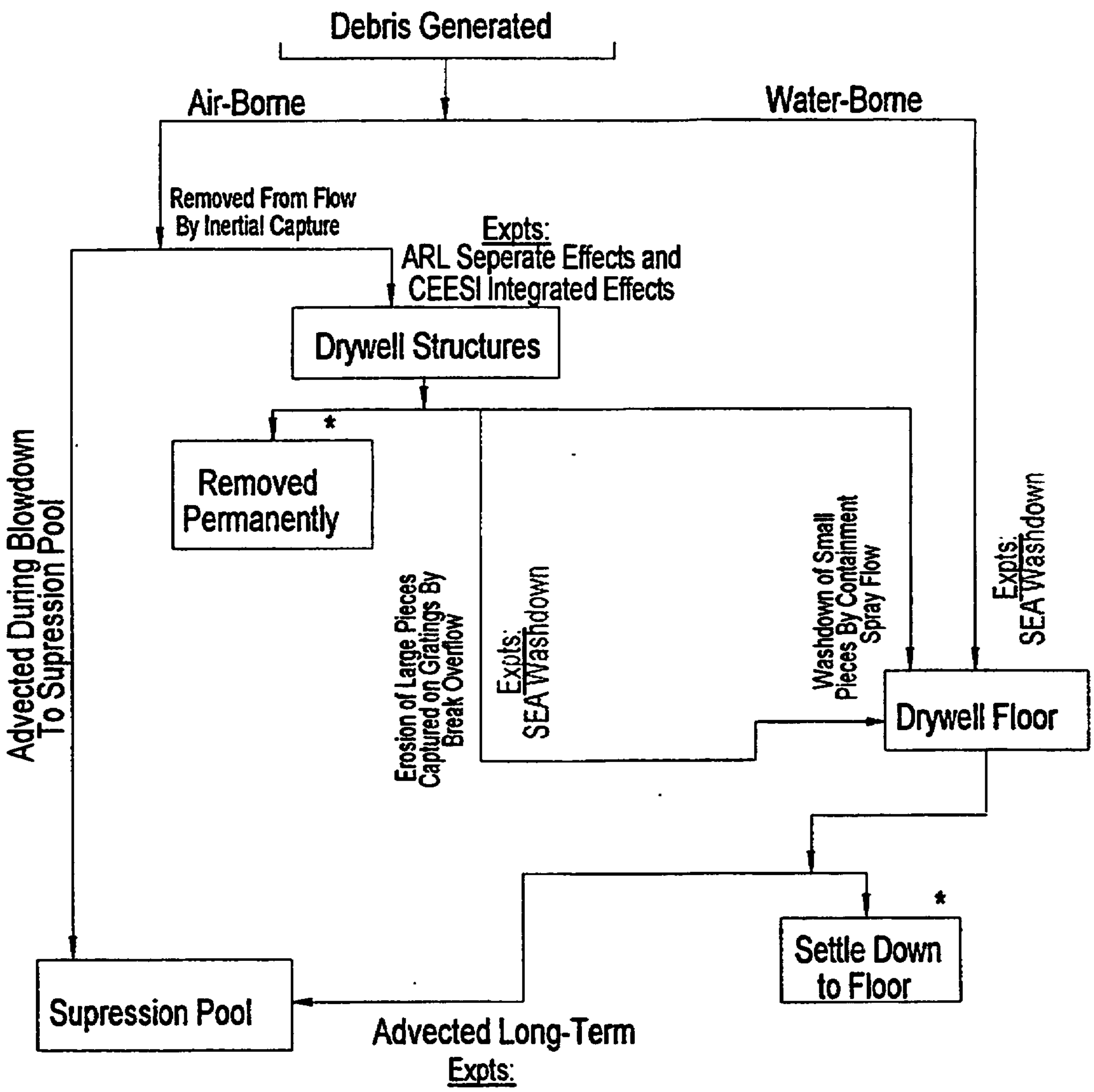

Figure 1. Pathways for Transport of LOCA Generated Fibrous Insulation Debris.

Regulatory Status of the ECCS Suction Strainer/Sump Screen Blockage Issue

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This generic issue deals with the potential loss of emergency core cooling system (ECCS) net positive suction head (NPSH) due to fouling of ECCS suction strainers in boiling-water reactors (BWRs) or sump screens in pressurized-water reactors (PWRs) by debris which is generated by a loss-of-coolant accident (LOCA).

The staff has completed its resolution of this issue for BWRs with the issuance of NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris," dated May 6, 1996, and Regulatory Guide 1.82, Revision 2, "Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident," dated May 1996. BWR licensees are currently implementing their final resolution of the problem. NRC Bulletin 96-03 gave BWR licensees three options to resolve the issue: 1) install large capacity passive strainers, 2) install a self-cleaning strainer, or 3) install a backflush system. The bulletin requested BWR licensees to implement their final resolution by their first refueling outage commencing after January 1, 1997. All BWR licensees have opted to install large capacity passive strainers.

Several licensing issues have arisen during the implementation of NRC Bulletin 96-03. Licensees have been attempting a variety of licensing basis changes under 10 CFR 50.59. These actions by utilities have raised concerns in the staff over the appropriateness of licensee actions conducted under 10 CFR 50.59.

In resolving this issue for BWRs, the staff has learned about several new phenomena which were not considered during the original resolution of Unresolved Safety Issue A-43 (USI A-43). For instance, the potential deleterious effect of particulate debris combined with fibrous debris on suction strainer headloss. As a result, the staff has recently commenced a study of the issue for PWRs based on the current level of knowledge. The staff has begun working with the PWR Owners Groups and NEI to obtain plant detailed information which the staff will need for its study. The purpose of the study is to determine if there are concerns based on the lessons learned from the BWRs which require further action for PWRs.

Generic Safety Issue (GSI) 171 - Engineered Safety Feature (ESF) Failure From a LOOP Subsequent to LOCA: Assessment of Plant Vulnerability and CDF Contributions*

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INTRODUCTION

Generic Safety Issue 171 (GSI-171), Engineered Safety Features (ESF) Failure from a Loss Of Offsite Power (LOOP) subsequent to a Loss Of Coolant Accident (LOCA), primarily addresses an accident in which a LOCA is followed by a LOOP (hereafter called LOCA/LOOP). It was later broadened to include LOOP followed by a LOCA (hereafter called LOOP/LOCA). This issue is concerned with nuclear power plants' (NPPs') ability to respond to a LOOP subsequent to a LOCA and vice-versa, since one of them occurring as a consequence of the other and delayed by few seconds or longer may result in unique conditions not analyzed previously. Several incidents that have occurred in operating nuclear power plants, and also anomalies noted while testing a plant response's to LOCA and LOOP have raised questions about the ability to respond to such accidents. This issue was initially identified by the United States Nuclear Regulatory Commission's (NRC's) Office of Nuclear Reactor Regulation (NRR) in NRC Information Notice (IN) 93-17, "Safety System response to Loss of Coolant and Loss of Offsite Power" issued March 8,1993. A review of the notice by NRC's Committee to Review Generic Requirements noted that "...the staff is considering the need for further generic action to determine if all power reactor licensees should be required to demonstrate the capability to withstand the LOCA/delayed LOOP sequence of concern..." (Letter from E.L. Jordan to D.F. Stenger and R.E. Helfrich, April 12,1994). A prioritization analysis was carried out by the NRC staff and a HIGH priority ranking was given to GSI-171.

OBJECTIVES

To address the issues and concerns raised as part of GSI-171, this paper quantitatively analyzes the accident sequences based on the following tasks:

1) Analyses of LOCA/LOOP and LOOP/LOCA accident sequences considering the loading sequence in response to LOCA and LOOP and the plant's electrical distribution system, along with applicable protection features,
2) Review of Individual Plant Examination (IPE) submittals to identify the extent to which the GSI-171 concerns are addressed,
3) Development of a detailed model for quantifying the Core-Damage Frequency (CDF) contribution of a LOCA/LOOP accident,
4) Development of estimates of LOCA/LOOP frequency based on past LOOP events and estimates of parameters representing the specific conditions during the progression of the accident, using a combination of operating experience data, modeling, engineering analyses and judgment, and
5) Quantification of the CDF contribution of a LOCA/LOOP accident for different plant groups

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

based on a plant's design characteristics and sensitivity analyses of specific plant-vulnerabilities, assumptions in modeling, and variability in the estimated parameters.

The evaluation was carried out for a pressurized- and a boiling-water reactor (PWR and a BWR plant).

RESULTS AND CONCLUSIONS

LOCA/LOOP Accidents

The results of the evaluation show that the CDF contribution of a LOCA/LOOP accident can be a dominant contributor to the total Level 1 internal event CDF for plants with certain vulnerabilities. A comparison of a PWR with BWR shows that BWRs are less vulnerable than PWRs to the accident in terms of the estimated CDF impact; however, the design characteristics that contribute to the CDF are similar.

- The CDF contribution of a LOCA with a consequential LOOP varies approximately two orders of magnitude (PWR: 1.2 E-04 to 2.8E-06/yr; BWR: 3.1E-05 to 6.1E-07/yr) depending on the design characteristics.
- Plants where block-loading to EDG following a LOOP takes place because load-shedding is not implemented, and block-loading to the offsite power is used are expected to have a high CDF contribution; plants where there is sequential loading to offsite power and the EDG, along with load-shedding, appear better equipped to handle this accident and are expected to have a low CDF contribution.
- Some plants may have specific vulnerabilities. Examples include operation with switchyard undervoltage that may increase the probability of a delayed LOOP and overloading of pumps, specific design of load sequencers making lockup highly likely, and the settings in anti-pump circuits so increasing the likelihood of lockout of circuit breakers of safety loads. Such vulnerabilities further increase the CDF contributions of LOCA/LOOP accidents.

LOOP/LOCA Accidents

In a LOOP/LOCA, during the transient subsequent to the LOOP, the pressure in the reactor coolant system (RCS) may reach the set point for the Power-Operated Relief Valves (PORVs) or Safety Relief Valves (SRVs) to open and these may subsequently fail to reclose, leading to a LOCA. IPEs generally model the LOOP/LOCA scenario, but may not address the GSI-171 issues. This study did not quantify the CDF contribution of a LOOP/LOCA considering the GSI-171 concerns, but Licensee Event Reports (LERs) were reviewed to develop estimates of the probabilities of PORVs and SRVs opening subsequent to a LOOP. These estimates are lower than the values used in the IPEs and PRAs reviewed in this study, and consequently, the LOOP/LOCA frequency is expected to be lower than that used in IPEs.

REFERENCES

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Environmentally Assisted Degradation of LWR Components
Session Overview

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In the area of environmentally assisted cracking (EAC) of piping and vessel steels, much information has been collected on materials used in light water reactors (LWR), but some critical measurements have not been made, and there is a serious lack of mechanistic understanding. This understanding is needed to permit NRC to use the available information to address issues such as residual life, inspection of internals, and mitigation of EAC. For example, collection of crack growth data on nickel alloys in simulated reactor environments has begun, but there is as yet insufficient mechanistic understanding to satisfy regulatory needs. Work on analyzing, benchmarking, and validating existing industrial crack growth models has been undertaken, but there is not yet general agreement on a general approach to crack growth predictions, and this causes significant problems in several areas. In the area of irradiation-assisted stress corrosion cracking (IASCC) the lack of basic mechanistic understanding means that, despite an extensive data base, it is still not possible to predict with any confidence whether a particular alloy in a particular application will experience IASCC, though general trends can be predicted.

Historically, the study of EAC phenomena which might degrade safety systems has been focused primarily on boiling water reactor (BWR) primary system piping and reactor vessel internals. EAC of BWR recirculation piping was significant enough to warrant either replacements or substantial repairs of piping in many plants during the middle 1980's. More recently, EAC in reactor internals has raised concerns with regard to maintaining adequate safety margins. When EAC is detected, repairs are often possible and have been implemented in many cases. Future research activities are focused on improvements in predictive capabilities by better understanding the fundamental mechanisms of EAC.

In order to maintain a sound technical base in the area of environmental degradation for timely rulemaking and related decisions in support of NRC regulatory/licensing/inspection activities, the NRC has an ongoing research program at Argonne National Laboratory which addresses several aspects of EAC, and a commitment to the Electric Power Research Institute (EPRI) Cooperative IASCC Research (CIR) program as well as less formal involvement with another IGSCC/IASCC international collaboration and with various American Society of Mechanical Engineers (ASME)-based international cooperative activities. One of the presentations in this session will describe the CIR program in more detail. Two other speakers will discuss the specific environmentally assisted degradation issues specific to BWRs and pressurized water reactors (PWR) in more detail. And finally, another talk will be given on NRC funded research in LWR materials and evaluation of EAC computer codes.

Fundamental Understanding and Life Prediction of Stress Corrosion Cracking in BWRs and Energy Systems

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SUMMARY

Many cases of environmentally assisted cracking have been reported in various energy-related industries involving various subcomponents of boilers, steam turbines, piping, pressure vessels, pressurizers, steam generators, deaerators, etc. The common element in these cracking incidents is exposure in high temperature water of various materials such as austenitic stainless steels, nickel base alloys, turbine steels, low alloy and carbon steels, and their weld metals. Because of the large economic and potential safety consequences, there is a strong driving force to derive design and life prediction codes that account for the relevant multitude of material, stress, and environmental contributions.

A detailed conceptual understanding of crack advance and the crack tip system is an essential precursor to predicting and controlling environmental cracking. This conceptual framework must, of course, be confirmed by critical experiments, then the essential fundamental processes quantified to create, ideally, a deterministic model for life prediction. A fundamental framework is acknowledged as essential, because the variables that affect environmental cracking are so numerous and inter-dependent that factorial (or other designs for) experiments are hopelessly expansive. Once a solid conceptual understanding is established, its use to conceive and evaluate mitigation approaches is very powerful, and its extension to related materials and environments straightforward.

This paper presents an approach for design and lifetime evaluation of environmental cracking based on experimental and fundamental modeling of the underlying processes operative in crack advance. In detailing this approach and its development and quantification for energy (hot water) systems, the requirements for a life prediction methodology will be highlighted and the shortcomings of the existing design and lifetime evaluation codes reviewed. Most examples will focus on BWR components involving stainless steels, low alloy steels, nickel base alloys, and irradiation assisted stress corrosion cracking, although some examples will also be discussed for low alloy steel and Alloy 600 in pressurized water reactors (PWRs) and turbine steels in steam turbines. From its inception in research oriented around BWR issues, this approach has been shown to provides a secure basis for extension to other hot water systems because of their many common characteristics.

Environmentally Assisted Cracking Issues in Pressurized Water Reactors

Warren H. Bamford

This paper will present an overview of environmental cracking experience in Pressurized Water Reactors over the past ten years. These cracking experiences fall into two distinct areas: fatigue crack growth in ferritic and austenitic steels, and PWSCC in Alloy 600 materials. In both these areas significant research has been underway for some time, and the results of this research have been used to solve operating plant issues. Examples will be discussed for each area.

The key applications for environmental fatigue crack growth have been in the areas of low flow stratification, and in evaluation of indications found during in-service inspection, using the rules of ASME Code Section XI. A significant breakthrough in the methodology for characterizing environmental crack growth has just been made, and the results will be discussed.

In the area of stress corrosion cracking in Alloy 600, a significant amount of research work has been completed, and more is underway. The research results to date indicate that the crack growth rate model used in the original safety assessments of the reactor vessel head penetrations in 1992 is still valid, and may be a bit too conservative. An example will be discussed in this area as well.

Current Research on Environmentally Assisted Cracking in Light Water Reactor Environments

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The USNRC research program on Environmentally Assisted Cracking of Light Water Reactor Materials is currently focused on three main tasks: cracking of reactor internals and core shrouds, fatigue initiation in pressure vessel and piping steels, and environmentally assisted crack growth in high–nickel alloys. A small additional effort is focused on the assessment and comparison of proposed models for environmentally assisted crack growth with available experimental data.

Failures of reactor–core internal components in both BWRs and PWRs have occurred after accumulation of relatively high fluence (>5 x 10^{20} n·cm^{-2}, E >1 MeV). This type of degradation is commonly referred to as irradiation–assisted stress corrosion cracking (IASCC). As part of our research effort in this area, stainless steels (SS) with a wide variety of compositions were irradiated in the Halden Boiling Water Reactor and are currently being examined. Preliminary results from these studies indicate a high degree of susceptibility to IASCC of high-purity (HP) Type 316 SS, similar to that observed for HP heats of Type 304 SS. High levels of oxygen and sulfur in SS increase SCC susceptibility even in the nonirradiated state.

Recently, much attention has focused on cracking in core shroud weldments. These components are subject to relatively low fluence and would not usually be considered susceptible to IASCC. Most cases of core shroud cracking have been attributed to classical intergranular stress corrosion cracking (IGSCC) of thermally sensitized SS. However, extensive cracking has been observed in a core shroud fabricated from Type 304L SS, an alloy in which sensitization would not be expected to occur and in which the absence of grain boundary carbides has been confirmed. Weldments from core shrouds that have experienced cracking, as well as shielded-metal-arc (SMA) test welds are being examined to investigate other features that may contribute to cracking. Significant contamination by oxygen, sulfur, and fluorine was observed in the heat-affected zone (HAZ) of the SMA welds. In contrast, contamination in gas-tungsten-arc welds is insignificant. Microhardness profiles in the fusion and heat-affected zones of an SMA weld showed a region of maximum hardness in the HAZ. The local hardening characteristics, together with contamination by oxygen, fluorine, and sulfur, may promote susceptibility of core shroud welds to SCC.

The nature of the cracking experienced by core shrouds is also strongly influenced by residual stresses in the weldments. Under a subcontract, Battelle Columbus Laboratories has calculated weld residual stresses and the associated stress intensity factors for BWR core shroud welds. For cylinder-to-cylinder welds, the axial residual stresses show a "thick-shell" distribution. In the middle of the wall, the stresses are compressive, whereas the stresses are tensile at the inner and outer surfaces. The corresponding stress intensity factors, K, for weldments containing flaws suggest that cracks growing from either the inner or outer surfaces would tend to stop at midwall. For a surface crack that extends over only a portion of the circumference, the results suggest that cracks will tend to increase in length in the angular direction much more rapidly than they will grow throughwall.

Appendix I to Section III of the ASME Boiler and Pressure Vessel Code specifies fatigue design curves. The effects of reactor coolant environments are not explicitly addressed by these design curves, although test data indicate a significant decrease in fatigue life when applied strain range, service temperature, dissolved-oxygen (DO) in the water, and sulfur content of the steel are above a minimum threshold level, and the loading strain rate is below a threshold value. Recent results show that the decrease in fatigue life of carbon and low-alloy steels is caused primarily by the effects of environment on the growth of short cracks. Relative to air, crack growth rates (CGRs) in water with high DO are nearly two orders of magnitude higher for crack sizes <100 µm, and are about one order of magnitude higher for crack sizes >100 µm. In high–DO water, the surface cracks grow from the start as tensile cracks normal to the stress axis, whereas in air or low–DO water the cracks initiate as shear cracks ≈45° to the stress axis. Tests on Types 316NG and 304 SS in LWR coolant environments also indicate a significant decrease in fatigue life in water relative to that in air. However, for these materials, unlike carbon and low-alloy steels, environmental effects are more pronounced in low–DO than in high–DO water.

Besides its widespread use for steam generator tubing, Alloy 600 is used for a variety of structural elements in reactor systems. Cracking has been observed in a number of these components. The effects of temperature, load ratio, and stress intensity on environmentally assisted cracking of Alloys 600 and 690 in simulated BWR and PWR water are being examined. Corrosion–fatigue experiments were conducted on compact-tension specimens of a low-carbon content (0.03 wt.%) heat of Alloy 600 in HP oxygenated water to investigate the effects of load ratio, stress intensity, and heat treatment on CGRs. The specimens were fabricated from material that had been solution-annealed at 1025 and 1115°C for 2 h and heat treated at 600°C for 24 h after solution annealing. Solution annealing produced ASTM grain sizes of ≈1.0-1.5 and 0.2, respectively, which correspond to large average grain diameters of ≈210-250 and 340 µm. The heat treatment conditions do not appear to have a significant effect on the CGRs. Other heats of material with differing carbon contents and heat-treatment conditions may produce a wider variation in the results.

"Best-fit" correlations for the CGRs in water that contained ≈300 ppb DO versus ΔK were obtained for each specimen and for the combined data for all four specimens. At a ΔK of ≈2 $MPa \cdot m^{1/2}$, the rate in water is higher by a factor of ≈10 than that in air. At this ΔK, CGRs for specimens from other heats of Alloy 600 in HP water that contained <5 ppb DO were higher by a factor of ≈2 than that in air.

Cooperative IASCC Research (CIR) Program

J. Lawrence Nelson, EPRI

Irradiation assisted stress corrosion cracking (IASCC) describes intergranular environmental cracking of material exposed to ionizing radiation. The term IASCC is generally applied to environmental cracking that has been accelerated by radiation whether it acts alone or in combination to change water chemistry, material microchemistry, material microstructure, etc. Laboratory and field data show that intergranular stress corrosion cracking of stainless steels and nickel-base alloys can result from long-term exposure to high energy neutron radiation that exists in the core of light water reactors. To date, IASCC in reactor internals has been discovered during routine inspections and thus has not been a cause of unplanned shutdowns. However, concerns for IASCC are increasing.

The implications of IASCC are significant, both in terms of repair and outage costs as well as the potential for cracking in components that may be extremely difficult to repair or replace. Significant advancements have been made in the understanding of IASCC. However, it is clear that major unknowns persist and must be understood and quantified before the life of a reactor component at risk from IASCC can be predicted or significantly extended.

Although individual organizations are continuing to effectively address IASCC, it became apparent that a more direct form of cooperation would be more timely and efficient in addressing the technical issues. Thus in 1995 EPRI formed the Cooperative IASCC Research (CIR) Program. This is a cooperative, jointly funded effort with participants from eight countries providing financial support and technical oversight.

The efforts of the CIR Program are directed at the highest priority questions in the areas of material susceptibility, water chemistry and material stress. Major research areas of the Program are:
1) evaluation of IASCC mechanisms, 2) development of methodology for predicting IASCC, and 3) quantification of irradiation effects on metallurgy, mechanics and electrochemistry. Studies to evaluate various IASCC mechanisms include work to better understand the possible roles of radiation-induced segregation (RIS), radiation microstructure, bulk and localized deformation effects, overall effects on strength and ductility, hydrogen and helium effects, and others. Experiments are being conducted to isolate individual effects and determine the relative importance of each in the overall IASCC mechanism. Screening tests will be followed by detailed testing to identify the contribution of each effect over a range of conditions.

The development of a methodology for predicting IASCC will focus on the dominant elements identified from the mechanistic work described above. The development process would identify variables that are measurable on a component and devise a practical predictive methodology to start. Identification and planning of experiments needed to enhance development of the methodology will be conducted. This experimental planning includes ongoing critical reviews of available data with emphasis on filling gaps in the existing data, as necessary.

Research investigating irradiation effects on material properties is examining defect microstructural development, RIS including grain boundary orientation effects, radiation hardening, and the influence of bulk material composition and temperature. Efforts in the area of irradiation effects on crack-tip mechanics will evaluate the influence of defect type, defect density and dislocation loop size on deformation characteristics. Other mechanics-related areas to be examined include irradiation creep, mechanical cracking and the role of stress/strain during irradiation.

Identification of possible countermeasures to IASCC will be undertaken when sufficient mechanistic understanding of IASCC is obtained and when predictive capabilities are sufficiently developed to provide conclusive direction for work on potential countermeasures. The work in this area is expected to include identification of: potential remedial actions for existing components; new materials; stress reduction techniques; and environmental modifications.

The paper to be presented will describe the completed and ongoing work being sponsored by the CIR Program in the above areas.

Recent SCDAP/RELAP5 Code Applications and Improvements[1]

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Introduction

This paper describes recent applications of the SCDAP/RELAP5/MOD3.1 code, and development and assessment activities associated with the release of SCDAP/RELAP5/MOD3.2. The major focus of recent MOD3.1 applications has been in support of Nuclear Regulatory Commission (NRC) generic assessment of steam generator tube integrity. The subsequent development and assessment activities associated with MOD3.2 have focused on late-phase modeling improvements, which represent the greatest uncertainty in current modeling capabilities.

SCDAP/RELAP5/MOD3.1 Applications

Although recent applications of SCDAP/RELAP5/MOD3.1 include evaluation of the potential for temperature-induced steam generator tube ruptures (SGTRs) in several different operating pressurized water reactors (PWRs), this discussion is limited to results from the Surry PWR analyses. These analyses were particularly challenging because of complex natural circulation flows that can develop as the accident progresses. These flows are important because they can transfer energy from the core to other parts of the reactor coolant system (RCS). The associated heat up of RCS structures can result in pressure boundary failures; with notable vulnerabilities in the pressurizer surge line, hot leg nozzles, and steam generator (SG) tubes. The potential for a steam generator tube rupture (SGTR) is of particular concern because fission products could be released to the environment through such a failure.

A station blackout sequence was assumed to be the accident initiator in all calculations. The specific sequence considered included an immediate loss of ac power and the loss of all feedwater. An additional assumption of sequence progression without recovery or operator actions allowed development of natural circulation flows that could threaten the integrity of the SG tubes (as well as other vulnerable RCS pressure boundaries). Modeling of natural circulation flows was based on data from Westinghouse scaled experiments. All RCS pressure boundaries, including the SG tubes, were assumed to be defect free. In all calculations, the atmospheric dump valve attached to the pressurizer loop SG secondary was assumed to fail in the full open position on its first challenge, which increased the severity of SG tube conditions by depressurizing the SG secondary.

Calculations completed for Surry indicate that creep rupture of the surge line and/or hot leg will occur prior to any SG tube failure. Furthermore, SGTR would not be expected given a surge line or hot leg failure, since the SG tube pressure differential would be reduced to zero as the RCS depressurizes. Consequently, the potential for a SGTR in the Surry PWR is low for conditions that have been considered.

1 Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under DOE Idaho Field Office Contract DE-AC07-94ID13223.

SCDAP/RELAP5/MOD3.2 Improvements and Assessment

The development of SCDAP/RELAP5/MOD3.2 began in the spring of 1994, and contains a number of added code capabilities and modeling improvements since the last version of the code, SCDAP/RELAP5/MOD3.1, was released. These improvements include the completion of several late-phase models identified in the NRC independent review group report issued in January 1993. Completion of these latest models is a major step toward completion of the SCDAP/RELAP5 code resolution plan, developed by the NRC in response to the January 1993 report.

Specific modeling changes in the latest code version include improvements in: (a) molten pool formation and growth, including transient natural circulation heat transfer, (b) in-core molten pool thermal-mechanical crust failure criteria, (c) the melting and relocation of upper plenum structures, (d) interactions between the in-core melt and shroud, (e) BWR control blade/channel box enhancements, (f) ex-vessel CHF heat transfer correlations, and (g) lower plenum debris behavior. The BWR control blade model developed by Oak Ridge National Laboratory has been linked to SCDAP/RELAP5 late phase models to allow relocating control blade materials to participate in the formation and growth of in-core or lower plenum molten pools and debris beds. The MATPRO material properties library was also updated to include material interactions not previously considered. To eliminate abrupt transitions between core damage states and provide more realistic predictions of accident progression phenomena, a transition smoothing methodology was implemented that results in the calculation of a gradual transition from an intact core geometry through different core damage states. Finally, two changes were made in SCDAP/RELAP5 to provide consistency between SCDAP and RELAP5 calculation methods. The first was to update the SCDAP heat transfer correlation package to be consistent with that used in the current RELAP5 code version, and the second was to implement the same implicit coupling of convective heat transfer and hydrodynamics in SCDAP that is currently used in RELAP5.

Assessment results indicate the latest modeling changes have improved both the code to data predictions and code robustness. Changes in the code numerics have improved code run times and computational stability. Comparisons with experimental results show that MOD3.2 gives better predictions of the rate of core heat up and total hydrogen production than observed for MOD3.1. Finally, the implementation of the core-damage transition smoothing has produced a more realistic transition between core damage states.

Conclusions

SCDAP/RELAP5 SGTR calculations have confirmed the importance of natural circulation flows in determining the potential for SGTR under severe accident conditions and, in Surry, have shown that surge line and/or hot leg failure always occurred prior to SGTR, effectively eliminating the potential for SGTR under the conditions considered.

Recent improvements made to the SCDAP/RELAP5/MOD3.2 late phase models have enhanced code predictive capabilities and improved overall code performance. These improvements have been particularly significant in modeling molten pool behavior, and in the transitions from intact core geometry to debris blockages and molten pools.

Overview of MELCOR 1.8.4: Modeling Advances and Assessments

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The MELCOR code development project passed an important milestone in 1997 with the release of MELCOR version 1.8.4. The new version contains many new modeling features as well as improvements to existing models. New model additions to the MELCOR code include a model for predicting enhanced depletion rates for hygroscopic aerosols and a model for predicting the chemisorption of Cesium to the surfaces of piping. Improvements to existing models include: upgrading the core module (COR) to handle flow redistribution resulting from the formation of core blockages, improving the thermalhydraulics (CVH) coupling with COR to handle flow reversal situations, and upgrading the fission product scrubbing model to incorporate the SPARC90 code. Significant upgrading of the COR package core degradation modeling was also included in the new code release version. These changes include a feature to ensure oxidizing molten zircaloy is retained in place behind the outer oxide shell until attaining a user-specified (2400 K default) temperature, and allowing the outer cladding oxide shell to preserve fuel rod integrity until an oxide failure temperature is attained (2800 K default). These two features were implemented because previous versions of MELCOR tended to predict premature fuel collapse following melting of the metallic Zr cladding, resulting in low peak fuel temperatures , hydrogen production and volatile fission product releases in certain calculations.

Throughout the development of, and in preparation for the release of MELCOR 1.8.4, a number of assessment analyses were performed, focused on demonstrating new and improved capabilities in the code. The following assessments are a partial list. The SPARC90 implementation was evaluated against both the stand-alone code and against pool scrubbing experiments performed at Battelle Columbus Laboratory and the ACE experiments, demonstrating the performance of the upgraded pool scrubbing features. The hygroscopic aerosol behavior model was applied both to the ISP-37 VANAM M3 experiment, and to the AHMED experiments. These assessments emphasize the increased aerosol depletion rates associated with water condensation on hygroscopic fission product aerosol. The CORA-13 and the Phebus FPT-0, FPT-1, B9+ tests were used to evaluate modeling improvements in MELCOR's treatments of core degradation, hydrogen generation and fission product release, and demonstrate much improved predicted behavior for these important quantities. Selected results from these assessments are presented in this overview.

While the new MELCOR release provides many improvements over the previous version, post-MELCOR 1.8.4 development activities continue, with focus on additional new models as well as further upgrades to existing models. New models are currently planned for treating iodine behavior in containment sumps, pools, and atmosphere, and for implementing reflood models and the attendant effects on accident progression. Further improvements and additions to the COR package are also planned. These include implementation of enhanced clad failure models to treat clad ballooning and eutectic interaction with grid spacers, and expansion of the COR package to allow for improved representation of UO_2-Zr eutectic behavior, improved melt relocation treatment, greater detail in describing aspects of BWR core degradation (fuel channel, bypass and lower plenum), and more flexibility in modeling "other structures" in the core such as core plate structures (supporting) and PWR control elements (non-supporting). In future years, new models addressing fission product release under HPME conditions, and fission product aerosol behavior in the containment are also under consideration.

DYNAMIC BENCHMARKING PROGRAM FOR THE MAAP4 CODE

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ABSTRACT

Computer simulation of nuclear power plant response can be a full-scope control room simulator, an engineering simulator to represent the general behavior of the plant under normal and abnormal conditions, or the modeling of the plant response to conditions that would eventually lead to core damage. In any of these, the underlying foundation for their use in analyzing situations, training of vendor/utility personnel, etc. is how well they represent what has been known from industrial experience, large integral experiments and separate effects tests. Typically, simulation codes are benchmarked with some of these; the level of agreement necessary being dependent upon the ultimate use of the simulation tool. However, these analytical models are computer codes, and as a result, the capabilities are continually enhanced, errors are corrected, new situations are imposed on the code that are outside of the original design basis, etc. Consequently, there is a continual need to assure that the benchmarks with important transients are preserved as the computer code evolves. Retention of this benchmarking capability is essential to develop trust in the computer code.

Given the evolving world of computer codes, how is this retention of benchmarking capabilities accomplished? For the MAAP4 codes this capability is accomplished through a "dynamic benchmarking" feature embedded in the source code. In particular, a set of dynamic benchmarks are included in the source code and these are exercised every time the archive codes are upgraded and distributed to the MAAP users. Three different types of dynamic benchmarks are used: plant transients, large integral experiments, and separate effects tests.

Each of these is performed in a different manner. The first is accomplished by developing a parameter file for the plant modeled and an input deck to describe the sequence; i.e. the entire MAAP4 code is exercised. The pertinent plant data is included in the source code and the computer output includes a plot of the MAAP calculation and the plant data.

For the large integral experiments, a major part, but not all of the MAAP code is needed. These use an experiment specific benchmark routine that includes all of the information and boundary conditions for performing the calculation, as well as the information of which parts of MAAP are unnecessary and can be "bypassed".

Lastly, the separate effects tests only require a few MAAP routines. These are exercised through their own specific benchmark routine that includes the experiment specific information and boundary conditions. This benchmark routine calls the appropriate MAAP routines from the source code, performs the calculations, including integration where necessary and provide the comparison between the MAAP calculation and the experimental observations.

CONTAIN 2.0 CODE RELEASE AND THE TRANSITION TO LICENSING*

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SUMMARY

CONTAIN is a best-estimate computer code developed by Sandia National Laboratories under US Nuclear Regulatory Commission (USNRC) sponsorship to provide integrated analysis of containment phenomena. CONTAIN provides the analyst with the capability to predict nuclear reactor containment loads, radiological source terms, and associated phenomena under accident conditions. The principal purpose of CONTAIN is to provide the USNRC with predictive containment analysis capabilities and to serve as a tool to provide technical information in support of regulatory decisions. The recently released CONTAIN 2.0 code version[1] represents a significant advance in CONTAIN modeling capabilities over the last major code release (CONTAIN 1.12). The new modeling capabilities include a new dynamic condensate film flow model for heat transfer structures, a new pool tracking model for modeling liquid transport phenomena associated with sumps and suppression pools, improvements in heat and mass transfer boundary layer modeling with respect to heat sinks, improvements in the ability of the code to treat stable stratifications, a new mass and energy conservation tracking scheme, and an improved equation of state for steam. The principal motivation for many of the recent model improvements has been to allow CONTAIN to model the special features in advanced light water reactor designs. As a result, CONTAIN has been used successfully to model the Westinghouse AP600 containment, for several different types of loss-of-coolant-accidents (LOCAs), and to model many of the Westinghouse Large Scale Test (LST) 1/8-scale experiments.

In addition to the AP600 work, the USNRC is currently engaged in an effort to qualify CONTAIN for more general use in licensing, with the intent of supplementing or possibly replacing traditional licensing codes such as CONTEMPT[2] and COMPARE.[3] CONTAIN represents enhanced modeling capability and reflects the more current status in our understanding of containment phenomena. To qualify the CONTAIN code for licensing applications, a number of studies utilizing CONTAIN 2.0 are in progress. These studies are intended to (1) provide comparisons to previous design-basis-accident (DBA) licensing calculations performed with CONTEMPT and COMPARE and (2) establish a methodology for use of CONTAIN in a manner consistent with the philosophy of conservatism noted in the USNRC's Standard Review Plan for containment analysis.

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At present, CONTAIN 2.0 has been used to provide comparisons to licensing calculations for the San Onofre plant, a large dry pressurized water reactor, and the Grand Gulf plant, a Mark III boiling water reactor (BWR). A series of validation calculations involving DBA experiments has also been completed; these include the early General Electric BWR tests, tests utilizing the Carolinas Virginia Tube Reactor (CVTR), and two experiments, ISP-16 and ISP-23, in the German HDR facility. Other validation calculations involving the NUPEC M-8-1 test and the HDR E11.2 and E11.4 experiments were also done to support the DBA analyses. Calculations of separate effects tests and comparisons to other codes have also been performed to validate the CONTAIN approach to heat and mass transfer modeling under both natural and forced convective conditions.

A number of results from the CONTAIN code qualification effort are presented in this paper to illustrate the code capabilities. The differences between CONTAIN and CONTEMPT modeling assumptions are discussed for San Onofre main-steam-line-break and LOCA scenarios. Validation of the CONTAIN approach to heat and mass transfer modeling is also discussed. CONTAIN calculations of NUPEC M-8-1, CVTR Test #3, and ISP-23 are presented to illustrate (1) the ability of CONTAIN to model non-uniform gas density and/or temperature distributions and (2) the relationship between such gas distributions and containment loads.

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Status of VICTORIA: Peer Review and Recent Applications*

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VICTORIA is a mechanistic computer code designed to analyze fission product behavior within the reactor coolant system (RCS) during a severe reactor accident. It provides detailed predictions of the release of radioactive and nonradioactive materials from the reactor core and transport and deposition of these materials within the RCS. These predictions account for the chemical and aerosol processes that affect radionuclide behavior. Coupling of detailed chemistry and aerosol models is a unique feature of VICTORIA; it allows exploration of issues involving deposition and revaporization that cannot be resolved with other codes.

VICTORIA underwent an independent peer review for the US Nuclear Regulatory Commission (NRC), which was completed in April, 1997. The purpose of the peer review was to assess the VICTORIA code and documentation against a set of design objectives and targeted applications and to make recommendations on how the code could be improved. The results of the peer review are viewed as a confirmation of the overall adequacy of the VICTORIA code; however, a number of recommendations were put forward for model improvements. These recommendations were categorized as findings and concerns. A plan for how these recommendations are being addressed will be presented. In particular, modifications that are being made in preparation for the next release version, VICTORIA 2.0, will be discussed.

The VICTORIA code has recently been applied to the analysis of an induced steam generator tube rupture (ISGTR) sequence and to the Phebus FPT-1 and FPT-4 experiments. Each of these applications will be briefly discussed and summarized.

Analyses of an ISGTR sequence, using SCDAP/RELAP5-provided thermal-hydraulic data, was conducted in support of recent NRC activity on steam generator tube integrity. The sequence begins with a station blackout followed shortly by a stuck open

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atmospheric dump valve on one steam generator secondary. Although surge line and hot leg nozzle failures by creep rupture were predicted to occur prior to creep rupture of a steam generator tube, these earlier failures were ignored for the purposes of this calculation in order to investigate the consequences of an ISGTR. The analysis was continued to the point of hot leg nozzle melting, after which fission products would be released into containment instead of to the environment. The primary purpose for the VICTORIA analysis was to get a best estimate of fission product release to the environment during this hypothetical by-pass sequence. Results of this analysis are presented and compared with earlier MELCOR and MAAP predictions for an ISGTR sequence.

The Phebus FP test series is a set of integral, in-pile experiments that are being conducted in France at Cadarache. Two tests, FPT-0 and FPT-1, have been performed to date. The primary difference between FPT-0 and FPT-1 is that FPT-0 was a shakedown test conducted with trace irradiated fuel while FPT-1 used spent fuel from the BR-3 reactor.

Results of the FPT-1 test have recently become available and posttest analyses using VICTORIA were conducted. The purpose of VICTORIA analyses of FPT-1 is primarily one of validation. Points of comparison are releases from the fuel bundle and the deposition pattern in a system of piping, including a 9 m long steam generator tube cooled from 700 to 150°C, leading to a small model containment.

The next planned test is FPT-4, which will be quite different than any of the other Phebus FP tests because the fuel configuration is one of a rubble bed. The test design calls for a set of filters upstream of the piping system; experimental data will focus on fission product releases from the rubble bed, especially those of the less-volatile elements, and on the speciation of the released elements. Pretest analyses are being conducted using VICTORIA to estimate the magnitude and timing of releases. Of particular importance is the predicted release of uranium because the filters must be designed to accommodate this release without plugging.

Human Reliability Assessment and Human Performance Evaluation: Research and Analysis Activities at the U.S. NRC

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The NRC mission "is to ensure adequate protection of the public health and safety, the common defense and security, and the environment in the use of nuclear materials in the United States..." The more specific mission of the human performance and human reliability activities at the NRC is to ensure effective risk-informed and performance-based regulation and oversight of safety-critical personnel performance in the design, operation, maintenance, decontamination, and decommissioning of nuclear reactor sites and other NRC-regulated facilities, as well as the transport, storage, and use of nuclear materials. Consistent with this mission, the Human Performance and Human Reliability Implementation Plan was developed based on the vision that human performance should not contribute to undue risk associated with the use of nuclear materials.

To implement this vision, the NRC has adopted a three-part strategy: 1) to take a proactive approach to identifying human performance and reliability issues important to public health and safety; 2) to increase knowledge of the causes and consequences of degraded human performance and reliability; and, 3) to implement the appropriate regulatory response to such issues. The mission discussed above will be accomplished through regulatory activities associated with research and analysis and with inspection and review. Six programs have been identified. The six programs include:

<u>Development of Technical Basis</u>

Program 1. Develop the technical basis to support human performance evaluation and human reliability analysis, including the technical basis and guidance on management and organizational influences on human performance and facility risk.

Program 2. Develop and update a model of human performance and human reliability.

Program 3. Foster national and international dialogue and cooperative efforts on human performance evaluation and human reliability analysis methods and data.

Operating Event Analyses

Program 4. Conduct operating events analyses and database development to support human performance evaluation and human reliability analysis.

Review and Inspection

Program 5. Provide support to human performance and human reliability inspection and review activities of nuclear reactors.

Program 6. Provide support to human performance and human reliability inspection and review activities for nuclear materials usage.

Each of these programs will be discussed, with special emphasis on research and analysis activities. The plan will guide the agency's efforts to understand, predict, detect, prevent or mitigate the consequences of human errors that are important to the safety of NRC-regulated facilities and licensees.

AN EXPERIMENTAL INVESTIGATION OF THE EFFECTS OF ALARM PROCESSING AND DISPLAY ON OPERATOR PERFORMANCE

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This paper describes a research program sponsored by the U.S. Nuclear Regulatory Commission to address the human factors engineering (HFE) aspects of nuclear power plant alarm systems. The overall objective of the program is to develop HFE review guidance for advanced alarm systems. As part of this program, guidance has been developed based on a broad base of technical and research literature. In the course of guidance development, aspects of alarm system design for which the technical basis was insufficient to support complete guidance development were identified. The primary purpose of the research reported in this paper was to evaluate the effects of three of these alarm system design characteristics on operator performance in order to contribute to the understanding of potential safety issues and to provide data to support the development of design review guidance in these areas. Three alarm system design characteristics studied were (1) alarm processing (degree of alarm reduction), (2) alarm availability (dynamic prioritization and suppression), and (3) alarm display (a dedicated tile format, a mixed tile and message list format, and a format in which alarm information is integrated into the process displays). A secondary purpose was to provide confirmatory evidence of selected alarm system guidance developed in an earlier phase of the project. The alarm characteristics were combined into eight separate experimental conditions. Six, two-person crews of professional nuclear power plant operators participated in the study. Following training, each crew completed 16 test trials which consisted of two trials in each of the eight experimental conditions (one with a low-complexity scenario and one with a high-complexity scenario). Measures of process performance, operator task performance, situation awareness, and workload were obtained. In addition, operator opinions and evaluations of the alarm processing and display conditions were collected. No deficient performance was observed in any of the experimental conditions, providing confirmatory support for many design review guidelines. The operators identified numerous strengths and weaknesses associated with individual alarm design characteristics.

ADDRESSING THE HUMAN FACTORS ISSUES ASSOCIATED WITH CONTROL ROOM MODIFICATIONS

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Advanced human-system interface (HSI) technology is being integrated into existing nuclear plants as part of plant modifications and upgrades. The result of this trend is that hybrid HSIs are created, i.e., HSIs containing a mixture of conventional (analog) and advanced (digital) technology. The purpose of the present research is to define the potential effects of hybrid HSIs on personnel performance and plant safety and to develop human factors guidance for safety reviews of them where necessary. In support of this objective, human factors topics associated with hybrid HSIs were identified. A human performance topic is an aspect of hybrid HSIs, such as a design or implementation feature, for which human performance concerns were identified. The topics were then evaluated for their potential significance to plant safety. Twelve topics were identified as *potentially* safety significant issues, i.e., their human performance concerns have the potential to compromise plant safety. The issues were then prioritized and a subset was selected for design review guidance development. This paper will provide an overview of the issues and will discuss the research currently underway to address them.

A Pilot Application of the ATHEANA Model to a Nuclear Power Plant*

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Summary

A demonstration of A Technique for Human Error Analysis (ATHEANA) was conducted at a pressurized water reactor (PWR) nuclear power plant. This was the first application of the new method. The main goals were to assess the usability of the method as documented, determine its efficacy in identifying and quantifying important human failure events (particularly, errors of commission) in the context of a probabilistic risk assessment (PRA), and identify ways to improve the method and its documentation.

A human reliability analysis (HRA) team was formed, consisting of an expert in PRA with some background in HRA (not ATHEANA) and three personnel from the nuclear power plant. Personnel from the plant included the plant's PRA expert, an operator training instructor, and a licensed senior reactor operator. In addition, plant operating crews participated in simulator runs.

The demonstration was conducted over a 13-week period and was observed by members of Nuclear Regulatory Commission's ATHEANA development team, who also served as consultants to the HRA team when necessary. The results of the demonstration, including human failure events and their estimated failure probabilities, and the users' evaluation of the ATHEANA methodology are discussed.

Also addressed is how simulator exercises were used in the ATHEANA demonstration project to promote the discovery of potential error-forcing contexts as well as possible errors of commission. Included are the purpose of the exercises, identification and preparation of the simulator runs, preparation of the operators involved in the exercises, and the results of the exercises.

*This work was supported by the U.S. Nuclear Regulatory Commission and was performed at Sandia National Laboratories, a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the U.S. Department of Energy under Contract DE-AC04-94AL85000.

Radionuclide Transport in the Environment: A Generic Research Program

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With reductions in budget for high-level waste and low-level waste management, the NRC Office of Nuclear Regulatory Research has terminated research programs in both these areas. However, the Commission has recognized a need to be able to assess doses to the public from releases of radioactive material to the environment. Such releases can occur at any point in the fuel cycle. Current analytical techniques for conducting these analyses generally employ highly simplified models and conservative assumptions to make predictions that are assumed to be worst case estimates of potential doses. In many cases, primarily where the amount of materials released are very small or the environmental systems themselves are simple, e.g. homogeneous and isotropic, such bounding calculations are sufficient for regulatory purposes. When amounts of released radionuclides are large, environmental conditions are complex, and times of concern approach centuries to millennia, conservative approaches and simple models may produce predictions with such large uncertainties that the proper regulatory decisions are not clearly indicated.

The Waste Management Branch, using work formerly conducted for high-level waste and low-level waste as a point of departure, has developed a generic research program to study issues associated with making more realistic dose calculations for expected releases of radioactive materials to the environment. This program is built around a generalized performance assessment process which has been broken down into five areas: Source Term, Engineered Barrier Performance, Infiltration/Flow/Transport, Critical Group/Pathway Analysis, Performance Assessment. Specific research projects are being planned in each of these areas to develop a technical basis for changes to Commission's analytical tools for addressing these problems. The objective of this program is to develop data and models that can be used to improve current performance assessment models and reduce uncertainties in the central estimates and statistical deviation from those central estimates of doses to the public.

Regulatory Framework for Financial Aspects of Decommissioning Facilities

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This paper discusses the proposed amendments to the Commission's regulations regarding appropriate relief and insurance coverage for various spent fuel configurations during permanent plant shutdown. This is part of the NRC effort to eliminate or reduce regulatory burdens for power reactor facilities that are permanently shutdown and in the process of decommissioning.

Stockpiling of Potassium Iodide for Use by the General Public After a Severe Accident

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The Nuclear Regulatory Commission (NRC) received a petition for rulemaking which requested that the NRC amend its regulations to include a requirement that emergency planning protective actions include the prophylactic use of potassium iodide as well as the currently required sheltering and evacuation. The request would amend one of the 16 planning standards that nuclear power plant licensees are required to meet in order to assure that the option of using potassium iodide for the general public is included in emergency plans.

The current Federal guidance to State and local governments on the distribution of KI was promulgated in 1985 by FEMA in its capacity as Chair of the Federal Radiological Preparedness Coordinating Committee (FRPCC). The FRPCC (membership from 15 Federal agencies) was established to coordinate all Federal responsibilities for assisting State and local governments in emergency planning and preparedness for peacetime radiological emergencies.

The 1985 Federal policy recommends the stockpiling or distribution of KI during emergencies for emergency workers and institutionalized persons, but does not recommend pre-distribution or stockpiling for the general public. It recognizes, however, that options on the distribution and use of KI rest with the States. Hence, the 1985 policy statement permits State and local governments, within the limits of their authority, to take measures beyond those recommended or required nationally.

After several years of debate the FRPCC plans to publish a revised Federal Policy statement on the distribution for KI in late 1997. The highlights of this policy statement are:

1. KI should be stockpiled and distributed to emergency workers and institutionalized persons during radiological emergencies. In developing the range of public protective actions for severe accidents at commercial nuclear facilities, the best technical information indicates that evacuation and in-place sheltering provide adequate protection for the general public. However, the State (or in some cases, the local Government) is ultimately responsible for the protection of its citizens . Therefore, the decision for local stockpiling and use of KI as a protective measure for the general public is left to the discretion of the State or, in some cases, the local government.

2. The Federal Government will establish funding for the purchase of a supply of KI for use by the general public. The availability of KI as a protective measure for the general public supplements other options for public officials responsible for protective actions decisions. A few States have indeed included KI as a protective action for the general public. The FRPCC does not want to usurp the State prerogative to incorporate the use of KI as a protective measure for the general public. Therefore, to ensure that States have available to them the option to use KI if they so elect, the Federal Government will be prepared to provide funding for the purchase of a supply of KI.

3. A stockpile of KI is being established by the Federal Government. The Federal Government is required to prepare for a wider range of radiological emergencies. To that end, and as an added assurance for radiological emergencies in which the location and timing of an emergency are unpredictable and for which, unlike licensed nuclear power plants, there is little planning possible, a stockpile of KI is being established by the Federal Government. This Federal stockpile will be available to any State for any type of radiological emergency at any time.

4. Those States or local governments which opt to include KI for the general population will be responsible for the maintenance, distribution, and any subsequent costs associated with this program.

5. The incorporation of a program for KI stockpiling, distribution, and use by any State or local government into the emergency plans will not be subject to Federal evaluation. This is based on the recognition that the use of KI by the State for the general public is a supplemental protective measure, and that the existing emergency planning and preparedness guidance for nuclear power plants are effective and adequate to protect the public health and safety.

Implementation of the New Decommissioning Standard

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The U.S. Nuclear Regulatory Commission published the final rule on radiological criteria for license termination on July 21, 1997. This final rule amends 10 CFR 20, Subpart E, and establishes criteria for the remediation of contaminated sites or facilities that will allow their release for future use with or without restrictions. Guidance is being developed in several areas to support implementation of the rule. These areas include site surveys, institutional controls, ALARA, public participation, and dose assessment.

As part of the work associated with implementing the final rule, a decision methodology has been developed to support implementation of the dose assessment requirements in the new Subpart E. The decision process supports assessment of the entire range of dose modeling options from which a licensee may chose, from changing a single parameter to changing multiple parameters and modifying pathways or models.

Generic exposure scenarios and pathways have been defined based on the NUREG/CR-5512 methodology and can be used without further analysis or justification by licensees who are applying the default scenarios and parameters using the DandD software. The default screening scenarios and pathways provide the licensee with a simple method to demonstrate compliance using little or no site-specific information. The generic models and default parameters are intended to estimate the upper range of the dose that the average member of the critical group could receive. The default parameters were developed probabilistically to control the regulatory risk associated with releasing a site based on source term data alone.

For licensees with more complex decommissioning situations, the decision process supports the modification of model parameters to allow site specific factors to be taken into account while still using the default models. This allows a licensee to use site-specific values in place of some or all of the default parameters. Thus, the dose estimates are more realistic, but may still be conservative for a particular site. The site specific data is used to support modifying or eliminating a particular scenario or pathway, or to demonstrate that a parameter or group of parameters can be better represented by site specific values. Alternative exposure scenarios may be appropriate based on site-specific factors that affect the likelihood and extent of potential future exposure to residual radioactivity.

An Overview of NRC Risk-Informed, Performance-Based Initiatives

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The NRC's research program in probabilistic risk analysis consists of a set of closely-related elements, from basic research to regulatory applications. The objectives of this research program are linked to specific agency regulatory functions, and are as follows:

- (1) To develop, demonstrate and improve methods for assessing the risks of nuclear power plant operations which will facilitate their use in applying the Commission's goal of improved regulatory effectiveness and risk-informed regulatory decision-making, and (2) to develop and promulgate appropriate guidance documents for use by both licensees and staff in uniformly applying risk assessment methods to support agency-wide decision-making, to support evaluation a licensee's requests for license amendments, and to support staff assessments of the significance of abnormal operating events.

- (1) To develop and demonstrate risk assessment methods applicable to non-reactor facilities and operations licensed for the production, processing, or utilization of radioactive materials in industrial, medical, and academic applications, (2) to develop appropriate guidance documents for the use of such methods in risk-informed regulatory applications, and (3) to provide as-needed support for NMSS in the application of risk assessment technology in its regulatory functions.

- (1) To review licensees' IPE and IPEEE submittals to determine if they meet the intent of NRC's Generic Letter 88-20 (IPE) and Supplement 4 to Generic Letter 88-20 (IPEEE), and (2) to analyze information from the review of licensees' IPE and IPEEE submittals to provide generic perspectives and insights from these programs and to determine if regulatory follow-up actions are appropriate.

The meeting presentation and full paper will provide an overview of major programs supporting each of these objectives. Other papers in the session will provide additional detail on specific, key programs.

Development of Risk-Informed Regulatory Guidance to Industry and NRC Staff

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The Nuclear Regulatory Commission has issued for public comment drafts of four regulatory guides (RGs), three Standard Review Plan (SRPs) sections, and a NUREG document. These issuances follow publication of the Commission's August 16, 1995 (60 FR 42622) Policy Statement on the Use of PRA Methods in Nuclear Regulatory Activities. The NRC has developed draft guidance for power reactor licensees on acceptable methods for using probabilistic risk assessment (PRA) information and insights in support of plant-specific applications to change the current licensing basis (CLB). The use of such PRA information and guidance is voluntary on the part of licensees.

The RGs and SRPs are intended to help implement the Commission's August 1995 Policy Statement and to provide an acceptable approach for power reactor licensees to prepare and submit and NRC staff to review applications for proposed plant-specific changes to the CLB that utilize risk information. Currently, draft RGs/SRPs have been developed and issued for comment in the areas of general guidance, Inservice Testing (IST) and Technical Specifications (TS). A draft RG for Graded Quality Assurance (GQA) has also been developed and issued for comment. No SRP has been developed for GQA, since the NRC staff will utilize its inspection process in the GQA area. In addition, the NRC has prepared draft NUREG-1602, "Use of PRA in Risk-Informed Applications," to provide reference information for licensees and NRC staff and it has also been issued for comment. A draft RG and SRP for Inservice Inspection has also been prepared and is now under review by the Commission. With approval from the Commission, the staff will also issue these two documents for review by the public.

The general RG/SRP have been developed to provide an overall framework and guidance that is applicable to any proposed CLB change where risk insights are used to support the change. The application-specific RGs/SRPs (i.e., IST, TS, GQA) build upon and supplement the general guidance for proposed CLB changes in their respective technical areas. Each application-specific RG/SRP references the general RG/SRP, states that the general guidance is applicable and provides additional guidance specific to the technical area being addressed.

The guidance provided in these documents is designed to encourage licensees to use risk information by defining an acceptable framework for the use of risk information on a plant-specific basis, and by promoting consistency in PRA applications. It is expected that the long-term use of risk information in plant-specific licensing actions will result in improved safety by focusing attention on the more risk significant aspects of plant design and operation.

In conjunction with developing these RGs and SRPs, the staff has also been working with several licensees on pilot applications of risk informed regulation in the technical areas listed above. The knowledge gained to date in interacting with licensees on these pilot applications has been used to help define the content and guidance contained in these RGs/SRPs. Additional interactions are expected over the next several months as work on these pilot applications continues and licensees and other interested persons have an opportunity to review the draft RGs/SRPs. The results of these additional interactions will be factored into the final RGs/SRPs, which are scheduled to issued by the end of December 1997.

Component Unavailability Versus Inservice Tests (IST) Interval: Evaluations of Component Aging Effects With Applications to Check Valves

by

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Methods are presented for calculating component unavailabilities when Inservice Test (IST) intervals are changed and when component aging is explicitly included. The methods extend usual approaches for calculating unavailability and risk effects of changing IST intervals which utilize Probabilistic Risk Assessment (PRA) methods that do not explicitly include component aging. Different IST characteristics are handled including ISTs which are followed by corrective maintenances which completely renew or partially renew the component. ISTs which have no followed renewal of the component are also handled. Any downtime associated with the IST, including the test downtime and the following maintenance downtime, is included in the unavailability evaluations. A range of component aging behaviors are studied including both linear and nonlinear aging behaviors. Based upon evaluations completed to date, pooled failure data on check valves show relatively small aging (e.g., less than 7% per year). However, data from some plant systems could be evidence for larger aging rates occurring in time periods less than 5 years. The methods are utilized in this paper to carry out a range of sensitivity evaluations to evaluate aging effects for different possible applications. Based on the sensitivity evaluations, summary tables are constructed showing now optimal IST interval ranges for check valves can vary relative to different aging behaviors which might exist. The evaluations are also used to identify IST intervals for check valves which are robust to component aging effects. General insights on aging effects are also extracted. These sensitivity studies and extracted results provide useful information which can be supplemented or be updated with plant specific information. The models and results can also be input to PRAs to determine associated risk implications.

In the paper, the aging of a component is characterized by its relative aging rate which is the rate at which the baseline component failure rate is increasing per year. The baseline component failure rates used in this work are similar to those used in plant PRA evaluations and may be somewhat more conservative than specific plant data might predict. However, they allow for better assessments on how IST intervals and aging could impact PRA results. General guidelines are given for determining the level of the aging rate for a component based on the characteristics of the component's operating conditions. Because a component can experience different aging behaviors in different circumstances, formulas and evaluations are given for both linear and nonlinear aging behaviors. The sensitivity studies which are carried out for check valves indicate the types of evaluations which can be performed when component failure rates and aging rates are not precisely known. The aging rate values which are used, especially the lower values, are compatible with present data. For certain extended IST intervals, even the low component aging rate values result in significant unavailability increases as compared to the calculated unavailabilities which assume no aging.

AN UPDATE OF PRELIMINARY PERSPECTIVES GAINED FROM INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) SUBMITTAL REVIEWS

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SUMMARY

As a result of the U.S. Nuclear Regulatory Commission (USNRC) initiated Individual Plant Examination of External Events (IPEEE) program, virtually every operating commercial nuclear power reactor in the United States has performed an assessment of severe accident risk due to external events.

To date, the USNRC staff has received 60 IPEEE submittals and will receive an additional 14 by mid 1998. Currently, 49 IPEEE submittals are under various stages of review. The goal of the review is to ascertain whether the licensee's IPEEE process is capable of identifying external events-induced severe accident vulnerabilities and cost-effective safety improvements to either eliminate or reduce the impact of these vulnerabilities. The review does not, however, attempt to validate or verify the results of the licensee's IPEEE. The primary objective of this paper is to provide an update on the preliminary perspectives and insights gained from the IPEEE process.

Thirty of the 49 plant reviews initiated to date have been performed as "Step 1" reviews. This implies that the review is limited to a submittal-only review, supplemented by a single round of requests for additional information (RAIs) to the licensee. Of these 30 submittals, six have been recommended for some level of further review. This will be a "Step 2" review and would typically involve a second round of follow-up RAIs. Depending on the nature of the open items, the "Step 2" reviews may also involve the review of supporting (second tier) documentation and in some cases a site visit.

Because of the NRC's budget constraint and the experience gained in the IPE submittal reviews, an alternate process of screening review has recently been implemented. The objective of the screening review is to identify, based on a somewhat more limited scope review, those licensee submittals that have clearly met the intent of the IPEEE process. Experience to date indicates that the majority of these screening reviews will require some limited scope RAIs before a final judgement on meeting the intent of the process can be made. These RAIs will focus on key aspects of the analysis that have not been fully documented and/or apparent mistakes or oversights with the potential to fundamentally impact the licensee's results. Currently, there are 19 submittals for which the screening reviews have been initiated.

The seismic assessment is one of the significant aspects of an IPEEE submittal. In the seismic area, one of four methods has been employed in virtually all of the submittals, namely, (1) seismic PRA, (2) NRC's seismic margins method, (3) EPRI's seismic margins method, and (4) a hybrid seismic PRA method that employed surrogate element. Many of the insights gained by licensees in this area have resulted from the seismic walkdowns performed for each plant. To date, no licensee (except Haddam

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Neck) has reported seismic vulnerabilities; however, virtually every submittal has identified seismic outliers or housekeeping issues. A number of plants have implemented plant modifications, including changes in plant procedures, as a result of the IPEEE analyses. This paper provides a discussion of the major IPEEE seismic review insights and perspectives, including both methodology and plant perspectives, and also discusses the types of plant modifications that have been reported.

The analysis of internal fires is the second major aspect of an IPEEE submittal. For the fire assessments, one of three methods has been employed in these submittals; namely, (1) the EPRI Fire Induced Vulnerability Evaluation (FIVE) method, (2) fire PRA method, or (3) a hybrid method that combines FIVE-based screening with detailed PRA-based quantification of unscreened scenarios. In general, it appears also that licensees have expended considerable effort in these assessments. Most licensees have also included extensive plant fire walkdowns as a part of the fire assessment, and typically at least one walkdown has been performed together with the seismic analysis team to address seismic/fire interaction issues. In many ways, the results, in terms of identifying major contributors to fire risk, obtained from the various methods employed have been quite consistent, especially in the context of FIVE versus PRA screening results. To date, only one licensee, Quad Cities, has reported a potential fire vulnerability. While no other plants have identified fire vulnerabilities, some licensees have implemented plant modifications or procedural changes as a result of the IPEEE analysis. This paper includes a discussion of these modifications, as well as a discussion of the major perspectives gained to date through the IPEEE review process.

In the area of high winds, flood, and other external events (HFOs), the importance of a given contributor and level of analysis have varied widely from plant to plant as would be expected. Many contributors are assessed using simplistic screening methods, while others may be analyzed in considerable detail. This paper includes a discussion of the most significant findings and perspectives that have developed for HFO events as a result of the IPEEE reviews performed to date.

RESEARCH NEEDS IN FIRE RISK ASSESSMENT

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SUMMARY

The objective of this paper is to discuss a number of areas where improvements are needed in fire risk analysis methods and data to: a) provide a better understanding of the risk contribution due to fires in nuclear power plants, and b) support improved decision making regarding nuclear power plant fire protection. It is expected that the issues presented in this paper will be integrated with previously identified fire protection issues (some of which are based on conditional risk considerations) as the agency develops a fire research program.

Since being prompted by the Browns Ferry fire of 1975, a number of nuclear power plant fire risk analyses have shown that fires can be significant contributors to plant risk. The most important scenarios in these analyses tend to involve the occurrence of relatively infrequent fires whose location and severity are such that critical sets of plant equipment are likely to be damaged by such a fire, if it occurs. These general conclusions regarding the potential magnitude and character of nuclear power plant fire risk appear to be consistent with empirical evidence, where serious fire-induced challenges to reactor core cooling are not common events but have occurred.

While there is little argument about the potential importance of fires, the magnitude of the fire risk and the specific measures needed to efficiently manage this risk are not as clear when considering individual plants. This uncertainty is due to uncertainties in the current state of knowledge concerning the initiation, growth, suppression, and plant impacts of fire-induced nuclear power plant accident scenarios. These latter uncertainties are reflected by the variability in methods and data used by current fire risk assessments (which contribute significantly to variations in predicted fire risk magnitudes and profiles), and by the ongoing dialog between the NRC and industry regarding the usefulness of current fire risk assessment tools in supporting proposed plant changes and the development of a risk-informed, performance-based rule for nuclear power plant fire protection.

Improvements in the staff's ability to thoroughly understand and accurately evaluate nuclear power plant fire risk require efforts in three areas: fundamental research on material properties and scenario phenomenology, the development of methods and tools to apply the results of fundamental research, and the application of these methods and tools to actual plants. It is anticipated that improvements sufficient to ensure good decision making also require efforts in these areas, although probably to a lesser degree. Clearly, the amount of effort required depends on the set of decisions to be made.

As part of its Fire Protection Task Action Plan (dated October 31, 1996), the staff developed a list of 12 potential safety issues recommended for further study. This list, since expanded to 13 issues following the staff's review of the Quad Cities Individual Plant Examination of External Events (IPEEE) study (see Table 1), provides a useful starting point for identifying research needs. However, this list is not specifically aimed at the issue of improving fire risk assessment (note that the issues address concerns at greatly varying levels of detail), nor does it reflect the most recent lessons learned from the IPEEE program.

In order to further define some of the broader issues listed in Table 1 and to ensure that all issues of concern to fire risk assessment are addressed, an approach based on the fire risk assessment analytical process is used. This approach considers the general areas of analysis covered in the assessment (e.g., fire initiation, fire growth and component damage, plant response) and the detailed tasks performed in each area. Based on experience with previous fire risk assessments and with recent IPEEE studies, the results of recent plant inspections, and knowledge of current developments in the fire science and engineering community, a list of issues where research is needed to significantly reduce uncertainties and biases in the results of current assessments is developed. Of course, many of these issues are associated with uncertainties in material properties and fire phenomenology (e.g., the generation, transport, and deposition of smoke in nuclear power plant fires). Others, however, concern impacts on hardware (e.g., the likelihood and consequences of electrical hot shorts) and humans (e.g., confusion due to spurious indications).

The details of the approach used to identify research issues and a discussion of the research issues themselves are presented in the full paper.

Table 1 - Supplemented List of Fire Protection Issues Identified in the Fire Protection Task Action Plan

Fire Impact on Reactor Safety
Availability of Safe Shutdown Equipment
Hot Shorts Resulting in Spurious Operations or Component Damage
Control Room/Cable Spreading Room Interaction with Remote Shutdown Capability
Smoke Effects on Personnel/Equipment
Explosive Electrical Faults
Compensatory Measures for Fire Protection Deficiencies
Seismic Fire Interactions
Fires During Non-Power Operations
Broken/Leaking Flammable Gas Lines
Reliability of Fire Barriers
Equipment Protection from Fire Suppression System Actuation
Fire Detection Methods

PROBABILISTIC SAFETY ANALYSIS OF NUCLEAR MATERIALS

Office of Nuclear Regulatory Research, USNRC

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As the NRC moves toward a risk-informed approach to regulation, probabilistic risk analysis tools are being put to use across the entire spectrum of the Agency's mission. Although most PRAs are studies of power reactor safety, the same techniques can potentially be applied to nuclear materials safety and associated issues. Two programs to develop such techniques currently exist within the Office of Nuclear Regulatory Research: dry cask storage and generally-licensed devices.

Dry cask storage is an alternative storage method for spent reactor fuel. After several years of storage in a spent fuel pool, the decay heat from a spent fuel assembly is sufficiently low that the assembly can be cooled by natural convection in air. However, the inventory of radioisotopes is still significant and the radiation field surrounding the assembly still quite hazardous. Dry casks are designed to hold such a fuel assembly, provide adequate shielding, and provide adequate cooling by means of natural convection. The casks are sufficiently robust to be stored on-site but out of doors, and thus are an attractive possibility for relieving the overcrowded state of spent fuel pools, at least in the short term. Nevertheless, there is inevitably some public risk associated with such a large inventory of radioactive material, and it is intended to evaluate this risk in a quantitative fashion.

Generally-licensed devices are devices containing radioactive material, but which are so robust and well-shielded that the manufacturer rather than the user is licensed by the Agency. The devices of interest in the current study are sealed sources used in industrial thickness gauges, density gauges, and similar gauges utilized in industrial process control. Such devices are extremely robust and are very safe even in a severe industrial environment over a period of many years.. However, there have been incidents where companies have ceased operation due to bankruptcy, the plant equipment has been sold for scrap, and these sealed sources have been smelted in scrap furnaces, resulting in contamination of the entire scrap mill. Most scrap yards now screen incoming shipments for radioactive material, but this screening has not always been successful. There is some sentiment even within the steel industry for more regulation of such devices, and the current study, if successful, will provide the basis for risk-informed regulatory decision making.

HIGH BURNUP FUEL RESEARCH
R. Meyer, Chair

INTRODUCTION

For the past three years, we have had sessions on high-burnup fuel, and those sessions have been weighted heavily with presentations on reactivity accidents. Interest in this subject was generated by a test in late 1993 in the CABRI test reactor in France, and results from that test were given their first major presentation here in 1994. Related tests from the NSRR test reactor in Japan and the IGR test reactor in a Russian program were also presented in that first high-burnup session in 1994.

While we try to highlight research results from NRC programs and from programs in which we cooperate officially (like those just mentioned), last year we also covered an operational problem. That was the problem of incomplete control rod insertion being experienced in some power plants with high-burnup fuel. This year, however, we have a short session and will again restrict the presentations to research results.

The reactivity accidents are still of interest, and a special issue of the journal, Nuclear Safety, has just been published giving major summaries of such work performed to date. That journal issue is an outgrowth of presentations made here at WRSM, and the summary information will not be repeated. Instead, we will hear about some recent results not included in the journal and we will concentrate on other studies being performed to understand test results of this kind.

In particular, there is a significant shift going on in some of these programs to measure fundamental materials properties and to try to relate them to the tests that simulate the accident conditions. The first three papers will address these mechanical properties.

The first paper is of special interest because it presents, for the first time, mechanical properties in fresh and irradiated (high burnup) cladding measured for temperatures and strain rates that are representative of postulated accident conditions. The cladding in this case is Russian Zr-1%Nb cladding from a VVER power reactor. This work will be of interest to those of us with PWRs because niobium alloys are being introduced in PWRs to take advantage of their favorable characteristics at high burnup.

Subsequent papers will discuss mechanical properties testing, a recently released code, and new test results on simulated reactivity accidents.

The final paper will also be of special interest because two of the recent tests to be described have made the news this year. One was a test with MOX fuel that failed a little more dramatically than expected. And the other was a test with heavily oxidized cladding that did not fail as definitively as expected. This presentation will be the first major public presentation of these results by the French experimenters who performed the work.

Development of Data Base with Mechanical Properties of Un- and Pre-irradiated VVER Cladding

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Research program to study mechanical properties of VVER Zr-1%Nb cladding was performed in 1997. Experimental part of the program includes two directions: the measurements of main tensile mechanical properties (yield stress, ultimate strength, total and uniform elongation) and measurements of cladding failure parameters under ballooning conditions. Analytical part of the program contains analysis and summarization of obtained data (including previously published) and the development of the corresponding models of material properties for such computer libraries as MATPRO. Main focus of investigations is directed to cladding characteristics of VVER commercial fuel element with high burnup level.

The current program is the next stage of investigations involved in the analysis of behavior of high burnup fuel rods under RIA conditions. Tests of high burnup fuel rods in IGR, CABRI, NSRR reactors have demonstrated that different mechanisms were responsible for fuel rods failure in PWR and VVER cases. It was noted that the initial level of the cladding ductility can be one of the factors determining the above differences. That's why the corresponding data base characterizing mechanical behavior of irradiated cladding is meaningful for the analysis. Measurements of mechanical properties were performed on ring samples manufactured from the cladding of VVER commercial fuel element irradiated up to burnup of about 50 MWd/kg U. Tensile tests were carried out in the temperature range of 20-950°C under two strain rates - 1.0×10-3 and 0.5 1/s. Similar cladding tubes were used to determine failure parameters during burst tests. Results of these investigations and their analyses are presented in the paper.

Modified Ring Stretch Tensile Testing of Zr-1Nb Cladding,*

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Introduction

As part of a round robin effort between the USNRC, IPSN in France, and the Russian Research Centre-Kurchatov Institute, Argonne National Laboratory conducted nine ring stretch tensile tests on unirradiated samples of Zr-1Nb cladding used in Russian VVER reactors. We used the "modified" ring stretch test discussed by Arsene and Bai;** slightly different geometries were used by the Russian and French researchers. The tests were conducted to determine the circumferential tensile properties of unirradiated fuel cladding, and the results will be used to develop a methodology for conducting similar tests on irradiated cladding segments. This summary discusses the results and analysis of the tests conducted by ANL.

Experimental

In the modified ring stretch test, three components are placed inside a ring cut from the cladding tube: two inserts upon which the tensile pulling force is applied, and a dumbbell-shaped central spacer to minimize specimen bending during the test. The diameter of these components is very close to the inside diameter of the specimen. All components were made of 17-4 PH stainless steel, which was hardened at 482°C for 1 h in argon gas. The surfaces of the inserts and spacer were coated with Molykote G to minimize friction between the components and the specimen. The ring specimens were machined to provide a narrowed gauge section in the circumferential direction. The gauge length of the original samples tested was 1.70 mm, the cross-sectional area was 2.88 mm^2, and the aspect ratio (gauge length to width) was close to 1. The tests were conducted at two temperatures, 25 and 400°C, and two strain rates, 1 and 0.001 s^{-1}, on an Instron Model 1125 tensile machine.

Results

Nine tests were conducted, and the results are summarized in Table 1. Yield strength and ultimate tensile strength decreased and elongation increased when the temperature was increased to 400°C. Reduction in strength was independent of strain rate, while there may have been a small strain rate effect in the elongations, as noted by the decrease in total elongations at 25°C when the strain rate was increased.

Discussion

The strength values were consistent with those reported by the French and Russian researchers; however, the ANL uniform and total elongation values were significantly higher. Elimination of the bending moment by the central spacer probably resulted in the higher uniform elongations in our specimens. However, most of the total elongation difference is believed to be due to the shorter gauge length of our specimens.

To further evaluate the results, finite-element analysis (FEA) was used to model the specimens and the stresses and strains that would develop under the test conditions. FEA determined that the original specimen design was not optimal because the wide gauge section led to an inhomogeneous

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

** S. Arsene and J. Bai, "A New Approach to Measuring Transverse Properties of Structural Tubing by a Ring Test," J. Testing Evaluation, **24** [6], Nov. 1996, pp. 386-391.

plastic strain distribution. As a result, the predicted failure path extended diagonally from the gauge section center toward the fillet junction. Diagonal failures were indeed seen on optical examination of the specimens. Such a long failure path is consistent with the high total elongations found. The analysis has led to an improved design of the "modified" ring stretch specimens, which is being used in additional tests of the VVER cladding. The results will be presented in the paper. Comprison of these results with those from the earlier tests should clarify the disparities found during the round robin exercise.

Table 1. Results of Modified Ring Tensile Tests on VVER Cladding

Test No.	Temperature (°C)	Strain Rate (s^{-1})	0.2% Yield Stress (MPa)	Ultimate Tensile Strength (MPa)	Elongation (%)	
					Uniform	Total
6	25	0.001	334	402	13.8	92.4
7	25	0.001	325	375	11.2	82.3
9	400	0.001	175	197	15.4	101.8
15	25	1	360	420	7.4	71.3
16	25	1	289	409	8.7	71.8
17	25	1	311	387	12.6	56.7
18	400	1	155	203	11.5	96.9
19	400	1	125	174	24.5	71.2
20	400	1	129	192	15.1	84.2

Conclusions

1. The 0.2% yield strength and ultimate tensile strength were determined for unirradiated VVER cladding specimens at two different temperatures and strain rates. Yield strength decreased from an average of 324 to 146 MPa as the test temperature increased from 25 to 400°C. The ultimate tensile strength decreased from 399 to 192 MPa over the same temperature increase. No strain rate effect was found. Strength values are consistent with those found by the Russian research team at Kuchatov and the French research team at CEA.

2. Uniform elongation and total elongation were also determined. The uniform elongation increased from 10.7 to 16.6% as the test temperature increased from 25 to 400°C. The total elongation values are believed to be high because of the specimen design. Finite-element analysis of the specimen design indicated that the plastic strain was inhomogeneous across the gauge section, and the high elongation values would be consistent with the long failure path predicted by the analysis.

3. Using finite-element analysis, we optimized the ring specimen dimensions. Additional tests are being conducted to clarify the differences in elongation found during the round robin exercise.

THE INFLUENCE OF STRAIN RATE AND HYDROGEN ON THE PLANE-STRAIN DUCTILITY OF ZIRCALOY CLADDING

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The ejection of a control rod or its drop can cause a reactivity-initiated accident (RIA) in a light water reactor. The resulting energy deposition within the fuel causes a very rapid increase in fuel and cladding temperature resulting in fuel expansion, pellet-cladding interaction, and potentially cladding failure. Under these conditions, the intrinsic ductility of the cladding is limited by in-service corrosion, hydriding, the dynamic loading imposed by the RIA, and the level of radiation damage sustained after high burnup. Predicting the cladding ductility during a RIA event is further complicated by the likelihood that the cladding is subjected to a range of deformation paths; among these, the transverse plane-strain deformation path is particularly severe in limiting ductility. Under this combination of conditions, the cladding ductility can be quite different from that obtained in conventional tests, and the use of conventional failure criteria for predicting the performance of cladding during a RIA becomes questionable.

The purpose of this research program is to investigate the deformation and fracture of unirradiated but hydrided Zircaloy 4 cladding subjected to following RIA-like loading conditions: transverse (hoop) extension of the cladding under near plane-strain conditions at both room temperatures and at 300° C as well as under both quasi-static and dynamic loading conditions. The results of this study are listed below:

- Both experiment and finite element analysis have been used to design a new straightforward, transverse tension test (see figure 1) to determine cladding failure under plane-strain conditions. In this procedure, we define ductility primarily in terms of the local extensional "limit" strain at the onset of localized necking. This failure strain parameter does not include the strains within a necked region are not included and the failure strain does not depend on specimen gauge length, as do conventional ductility measurements.

- The constitutive stress-strain-strain/rate response has been determined for the cold-worked and stress-relieved Zircaloy 4 cladding deformed at room temperature and at 300°C with the stress axis in the transverse/hoop direction. These data are needed for both our finite element analysis and to assess the resistance of the cladding to failure due to a localized necking instability.

- The failure response of a ring specimen, under transverse uniaxial tension, is contrasted to failure under a transverse plane-strain condition. The failure paths between these two stress states differ; spceifically, failure of the ring specimen is dominated by slip across the specimen width, while through-thickness deformation (as in cladding failure) causes fracture of the plane-strain test.

- The effect of strain rate (10^{-3}/s vs. 10^{2}/s) is to decrease the plane-strain ductility as measured by the limit strain as well as local fracture strain. The magnitude of the ductility decrease as a function of temperature and hydride content is presented.

- The presence of a thickness imperfection (groove of known depth, such as introduced by non-uniform oxidation or by premature failure of a heavily hydrided layer of cladding near its outer surface) is predicted to decrease cladding ductility. Experimental results of specimens containing grooves of controlled depths show failure strain decreases with increasing groove depths in a manner that agrees with our theoretical predictions.

- The influence of hydrogen on the plane-strain cladding failure is examined for samples with a uniform hydride distribution and with hydrides near the cladding surface. The effect of hydrogen embrittlement is presented for both quasi-static and dynamic strain rates, and for room temperature and 300^{o} C conditions.

The results indicate that in order to predict cladding ductility under RIA conditions, factors such as strain path, strain rate, hydrides, and oxidation need to be considered, especially for high burnup fuels.

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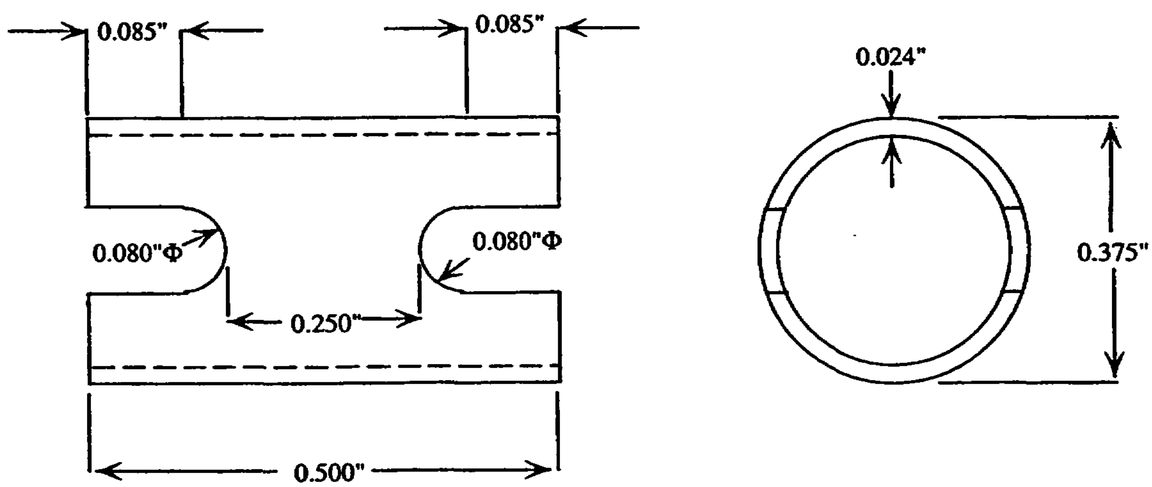

Figure 1: Schematic representation of transverse plane-strain cladding specimen.

DEVELOPMENT AND VERIFICATION OF NRC's SINGLE-ROD FUEL PERFORMANCE CODES FRAPCON-3 AND FRAPTRAN

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SUMMARY

The FRAPCON and FRAP-T codes, developed in the 1970s and early 1980s, are used by the U.S. Nuclear Regulatory Commission (NRC) to predict fuel performance during steady-state and transient power conditions, respectively. Both codes are now being updated by Pacific Northwest National Laboratory to improve their predictive capabilities at high burnup levels. The updates to fuel property and behavior models are focusing on providing best estimate predictions under steady-state and fast transient power conditions up to extended fuel burnups (> 55 GWd/MTU). Both codes will be assessed against a data base independent of the data base used for code benchmarking, and an estimate of code predictive uncertainties will be made based on comparisons to the benchmark and independent data bases.

The FRAPCON-3 code is an updated version of FRAPCON-2 that will be used by the NRC to audit vendor fuel performance codes with an emphasis on thermal, fission gas release, and rod internal pressure analyses. A code assessment of FRAPCON-3 has been recently concluded along with a peer review process that concentrated on those areas where the code will be applied for assessing licensing analyses, i.e., thermal and fission gas release. The code benchmarking data base includes thermal, fission gas release, internal rod void volumes, and cladding corrosion data. The code has also been assessed against an independent thermal and fission gas release data base. Both the benchmarking and independent data bases include rods with steady-state power operation up to rod-average burnups of 70 to 80 GWd/MTU and rods that have experienced power ramping on the order of hours-to-days with rod-average burnups up to 63 Gwd/MTU. The results of these assessments demonstrate that FRAPCON-3 provides a state-of-the-art prediction of thermal, fission gas release, and rod internal pressures up to extended burnups. The code and code documentation including the model description document, code manual with input instructions, and code integral assessment (References 1, 2 and 3) have just been released by the NRC. A comparison of FRAPCON-3 and FRAPCON-2 predictions of fuel melting and end-of-life rod internal pressures for typical commercial light water reactor fuel rods demonstrates that FRAPCON-3 predicts higher temperatures and pressures at high burnups than FRAPCON-2.

The last version of FRAP-T, FRAP-T6, was developed and issued by the Idaho National Engineering and Environmental Laboratory in 1981 with an update in 1983 (Reference 4). Since that time there have been few modifications, with the result that the code does not adequately predict high burnup fuel behavior. During FY-1997, work was begun to update FRAP-T6 to a new version called FRAPTRAN. Modifications to the code are being grouped into three general areas. First, general coding improvements to address known errors, ensure consistency across the code, and to delete undesirable or no longer needed coding and models. Second, model updates to existing models to account for data, knowledge, etc., gained since FRAPT-T6 was released (e.g., radial power distribution, contact conductance, and burnup dependent material properties like fuel thermal conductivity). And third, new model additions to extend the applicability of FRAPTRAN (e.g., modeling to account for fission gas release during fast transients).

The FRAPTRAN modifications accomplished during FY-1997 have included the following: 1) removal of the sensitivity analysis and licensing application coding, 2) replacing the fuel thermal conductivity model with the FRAPCON-3 model that includes burnup, gadolinia and mixed oxide dependencies, 3) updating the contact conductance model to parallel the changes in FRAPCON-3, 4) updating Zircaloy cladding mechanical properties models, and 5) reinstating a restart link between FRAPCON-3 and FRAPTRAN. Modeling of the fission gas release during fast transients will be completed in FY-1998. Comparisons of FRAPTRAN against steady-state data, i.e., centerline temperature data from beginning-of-life power ascensions have shown improvements in the predictive behavior relative to FRAP-T6, and good comparison to FRAPCON-3. FRAPTRAN will be assessed against experimental data sets such as those used for FRAP-T6 (Reference 5) and more recent experiments such as those conducted at Cabri and NSRR (Reference 6).

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THE STATUS OF THE RIA TEST PROGRAM IN THE NSRR

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Summary

To provide a data base for the regulatory guide of light water reactors, behavior of reactor fuels during off-normal and postulated accident conditions such as reactivity-initiated accident (RIA) is being studied in the Nuclear Safety Research Reactor (NSRR) program of the Japan Atomic Energy Research Institute (JAERI). The burnup limits of fuel operating in commercial power-producing LWRs in Japan have been increased from 39 MWd/kgU to 48 MWd/kgU for PWRs and to 50 MWd/kgU for BWRs, and further increase of the limits to 55 MWd/kgU is in consideration. In this condition, a series of experiments with high burnup fuel rods is being performed by using pulse irradiation capability of the NSRR. Recent results obtained from the NSRR power burst experiments with irradiated PWR and BWR fuels are described and discussed in this paper.

During the last Japanese fiscal year, we have add six experiment with irradiated UO2 fuels, consisting of HBO-7, TK-1, FK-1, FK-2, JMH-4 and JMH-5 experiments. The HBO-7 test was performed with 49 MWd/kgU PWR fuel, and fuel enthalpy reached about 80 cal/g at maximum. Although micro-cracks were observed in the outer region of the post-test cladding, the fuel rod survived. Fission gas release in the experiment was only about 8%, which is considerably lower than those observed in HBO-2 through HBO-4.

The TK-1 test is the first NSRR experiment with PWR fuel with low tin (1.3%Sn) cladding. Since the test fuel rod of the TK-1 was sampled from a fuel rod irradiated in only 2 cycles, fuel burnup remains 38 MWd/kgU. This relatively low burnup gives the higher fuel enthalpy during the transient, i.e. 125 cal/g. The fuel did not fail in the experiment. However, significant swelling of fuel rod was observed. The diameter increase of the cladding is about 10% in average over fuel active region, and 25% at maximum. The cladding surface temperature reached about 600 deg C.

The FK-1 and FK-2 are experiments with 45 MWd/kgU BWR fuels. Fuel enthalpy in these experiments reached 112 cal/g and 60 cal/g, respectively. Fuel failure did not occurred in the both experiments. Fuel deformation due to pellet/cladding mechanical interaction (PCMI) was observed.

The JMH-4 and JMH-5 experiments were performed 20% enriched short fuel rod irradiated up to 30 MWd/kgU in the JMTR. The JMH-4 was performed with low energy deposition level, since this test was only for burnup and enthalpy calibration. On the other hand, peak fuel enthalpy in the JMH-5 is about 210 cal/g. Fuel failure occurred, and pressure spikes and mechanical energy generation were recorded. In the JMH-5, water column movement sensor was installed. Mechanical energy conversion ratio in the experiment was about 0.1%.

The paper describes future NSRR test matrix, and a design of a test capsule for high temperature and high pressure conditions also.

THE STATUS OF THE CABRI - REP-Na TEST PROGRAMME PRESENT UNDERSTANDING AND STILL PENDING QUESTIONS

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Since 1993 the CABRI REP-Na programme is regularly presented at the WRSM. This programme has been defined in the frame of the study of the consequences of reactivity initiated accidents such as control rod ejection, in view of the future burn-up increase in the French PWRs. Its main objectives are to investigate the behaviour of high burn-up UO_2 fuel rods and to establish an experimental basis for irradiated MOX fuel rods.

Motivated by NSRR test results with spent fuel which evidenced early rod failures, the CABRI REP-Na programme is realized in the sodium loop of the CABRI facility. Sodium cooling only allows the study of the first phase of the transient during the quasi-adiabatic heat-up of the fuel without a significant clad temperature increase. In parallel, with the CABRI tests, separate effect programmes are undertaken: the PROMETRA programme for clad mechanical properties and the PATRICIA programme for clad-fluid heat transfer determination under fast power transients. Lastly, the SCANAIR code is being developed based on the knowledge for extrapolation to the reactor conditions.

The present paper will give an overview of the recent results of 1997, the actual under standing as well as the identified future needs.

The first five tests with UO_2 fuel have been already presented [1] and have shown the possible occurrence of early rod failure by hydride assisted PCMI in case of high clad corrosion level with spalling while the residual strains of the unfailed rods confirmed the PCMI loading. Fuel fragmentation, large fission release and transient oxide spallation were evidenced.

Three tests with MOX fuel rods have been performed.

The REP-Na 7 using a 4-cycles MOX fuel rod (55 GWd/t) with clad corrosion thickness of 50 μ without evidence of oxide spalling nor hydride "blister" led to rod failure when the mean fuel enthalpy reached 120 cal/g under a slow power pulse (40 ms half width, total injected energy of 175 cal/ g at 1.2 s). The failure resulted in strong flow ejection and high pressure peaks and to fuel ejection into the channel.

The axial location of the first failure at peak power level tends to eliminate the possible effect of the heterogeneity of the MOX fuel with $UPuO_2$ agglomerates.

On the other hand, in opposition to REP-Na 1 failure, the correct state of the cladding with regard to the presence of hydride blisters tends to suggest a contribution of the fuel transient swelling linked to the high fission gas retention in the $UPuO_2$ agglomerates.

Moreover, comparing the REP-Na 6 MOX rod behaviour (47 GWd/t) with similar power pulse leading to a maximum fuel enthalpy of 145 cal/g without failure would suggest the influence of burn-up on the clad loading; in addition, based on the PROMETRA results, the evolution of the stresses and strains as calculated by the SCANAIR code show a risk of failure so that a small difference in the clad loading due to different fuel swelling in both tests might result in failure. Such points have to be more precisely analyzed together with post-test examinations.

The third test of the MOX fuel series, REP-Na 9, using a 2-cycles rod (28 GWd/t, low clad corrosion level ~ 10 μ) did not lead to failure although the energy injection (228 cal/g at 1.2 s) resulted in maximum mean fuel enthalpy of 200 cal/g.

At the present time, all the post-test examinations of REP-Na 7 and REP-Na 9 are still to be done.

In a complement to the first high burn-up UO_2 tests, an additional experiment, REP-Na 8 (60 Gwd/t with corrosion level of 130 μ), has been performed. Its aim is to investigate the influence of the presence of some spallation of the clad oxide layer, on the risk of rod failure under a power pulse typical of reactor case (60 ms half width). It is to be compared to REP-Na 4 (unfailed rod, no oxide spalling, 80 μ ZrO_2) with regard to the state of the oxide layer and with REP-Na 1 with regard to the power transient.

From a first look on the very recent results, the REP-Na 8 test does not show evidence of failure although blisters have been identified on the spalled zone of the rod.

Concerning the PATRICIA programme, now operational with electrically heated rods and Inconel cladding, preliminary results have been obtained in steady state and under transient conditions (PWR and NSRR thermal-hydraulic conditions) and detailed analysis will be available in the near future.

RIA transient simulations under reactor conditions have been made with the SCANAIR code on a 5-cycles PWR rod similar to the REP-Na 4 test rod. High sensitivity to the heat transfer model has been evidenced leading to the occurrence of boiling crisis (DNB) for a mean fuel enthalpy from 60 up to 95 cal/g depending on the heat transfer modeling.

It is shown that the transient spallation, as observed in several REP-Na tests, might accelerate the DNB onset. The further evolution with clad temperature increase and possibility of sustained PCMI or clad creep under internal pressure increases with high fission gas release is a still pending question and calls for global experiment.

A future test programme in a pressurized water loop in the CABRI facility should address the following aspects:

- complete the data base for MOX fuel and very high burn-up fuel corresponding to future managements,
- complement the present CABRI REP-Na programme and reduce the existent uncertainties on the main phenomena,
- obtain the necessary elements in order to quantify the available margins with regard to fuel dispersion and related consequences after failure.

This programme, which should be performed under international cooperation together with the nuclear industry, is under discussion.

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CODE ASSESSMENT IN AP600 BDBA SPACE USING TH-PRA INTEGRATION METHOD

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The thermal-hydraulic (TH) code RELAP5/MOD3 that is used for AP600 analysis has been assessed in the design basis accident (DBA) space and is considered adequate. For the purposes of exploring design margins and providing data to assess the code's capability in addressing scenarios that are in the beyond design basis accident (BDBA) space, some selected accident scenarios in the BDBA space have been tested in the experimental facilities and TH code calculations have been performed for some of the tested accident scenarios. But the scenarios in the BDBA space have not been consistently and systematically explored in its completeness.

The primary purpose of this research is to provide reasonable assurance that the calculated AP600 response from the TH computer code is adequate and can be used as a basis for the BDBA evaluation.

In order to fulfill this purpose, a systematic approach is needed to first identify the classes of accident scenarios in the BDBA space that have a reasonable frequency of occurrence (i.e., probabilistically significant) and have the potential to lead to significant thermal hydraulic concerns in which the code is required to be further assessed and validated. This approach begins with an accident scenario screening process using an integrated TH and probabilistic risk assessment (PRA) method developed in this research. The product of this TH-PRA integration method is the identified BDBA scenarios that are probabilistically significant and thermal hydraulically important.

Once these scenarios have been identified, a decision will be made to determine whether sensitivity calculations will be needed for those BDBA scenarios. Based on the results of the sensitivity calculation, a decision will be made on whether additional testing would be necessary because there might be phenomena or scenarios identified in these sensitivity calculations that may require experimental confirmation. With the results from the sensitivity calculations and/or testing, a decision on whether imposing additional requirements on the code will be made.

Finally, the code with the additional BDBA requirements is evaluated and its adequacy in the BDBA space is determined. If the adequacy is judged to be satisfactory, the objective of the research is met. Otherwise, either additional requirements are imposed on the code or the TH-PRA screening process is revisited, if warranted.

This report summarizes the first part of this research - the TH-PRA integration method in identifying the BDBA scenarios. The method presented here shows how a deterministic analysis and a probabilistic analysis can be linked together to systematically identify dynamic system behavior. This method can be applied in many other areas, e.g., reactor safety issues and vessel pressurized thermal shock, to identify system/structure behavior involving physical phenomena and human interactions.

RELAP5 Code Assessment Using AP-600 Test Facility Data

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The USNRC has sponsored experiments at both the ROSA-AP600 and APEX facilities. Test data from these facilities has been used to establish the adequacy of the RELAP5 computer code to predict the AP600 system response for small break LOCA and other non LOCA accident scenarios. The TRAC-P code is used to simulate large break LOCA scenarios.

The version of the RELAP5 code used for AP-600 analysis is version 3.2.1.2. A number of features and modifications were added to this version specifically to address the unique requirements of the AP-600. These features include:

- Critical Flow. The addition of the Henry Fauske choked flow model, to eliminate the artificial choking of ADS-4 at low pressure, which would result in intermittent delivery of IRWST injection.
- Void Oscillations. An error in the CHF (Critical Heat Flux) low flow interpolation was corrected. Inconsistencies between the RELAP5 flow regime map and the interfacial friction model were corrected. Oscillations that have been caused by the use of an "ad hoc" liquid interfacial heat transfer coefficient have been corrected.
- Low Pressure Rod Bundle Void Fraction. Smoothing the transition between the co-current and counter-current flow regimes has been implemented.
- Flow Circulation. Artificial flow circulation in the core and lower plenum has been eliminated by not using the momentum flux terms for these regions of the reactor. This is being done until the incorrect differencing and donoring in the momentum flux terms is corrected.
- Time Step. Invokes a true Courant time step limitation based on over- extraction of the donored phase.

SCIENTECH and the University of Maryland performed a number of post test calculations for both ROSA-AP600 and APEX runs using RELAP5 version 3.2.1.2. In a number of cases, the results were compared to previous calculations performed with version 3.1.4 which did not have the above code features. The results indicate a major improvement in the RELAP5/MOD3.2.1.2 calculated core level and break flow compared with RELAP5/MOD3.1.4 and the experimental data. Additionally, the RELAP5/MOD3.2.1.2 calculations have shown an increased robustness during the simulations. In many cases very long transients were run to completion without termination from a property or code failure. In one area, condensation and flow of vapor in the IRWST tank, the new code version did not perform as well as the earlier version.

Comparison of the calculation results to test data indicate that, while such key results as the core collapsed liquid level are predicted quite well, there remains a need for further improvement in the fine structure of the predictions. This is particularly evident at low pressure and low temperature conditions. Areas identified as needing further improvement in code predictive capability include modeling of thermal stratification, mass error, natural circulation flow, and condensation and escape of vapor in a liquid pool. The needs for improvement are demonstrated by specific responses from ROSA-AP600 tests AP-CL-09, AP-AD-01 and AP-BO-01 and APEX test NRC-10.

One feature of the AP-600 analysis that is also addressed is the delicate balance of forces which govern the evolution of the later phases of transients during which the passive safety systems are operating to maintain core cooling. Under these circumstances, slight differences in initial conditions and modeling assumptions can result in significant differences in predicted event timing. Examples are given of the types of differences in initial and boundary conditions which can result in significant differences in event timing. These include the initial temperature of liquid in the pressure balance line connected to the core makeup tanks (CMTs) and the manner of modeling CMT heat structures. Overall performance of the safety systems, however, does not appear to be affected by the differences in timing.

An Overview of the Test Results from the NRC AP600 Research Program at OSU

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Abstract - The Advanced Plant Experiment (APEX) at Oregon State University has been used to study the performance of the passive safety systems of the Westinghouse AP600. Thus far, thirty integral system tests have been performed on behalf of the U.S. Nuclear Regulatory Commission. This includes two parametric series; NRC-13 "return to saturation oscillations" and NRC-25 "no reserve tests," which consist of multiple experiments. This paper briefly summarizes the key tests performed in the OSU APEX test facility.

1.0 APEX Description

The APEX facility is an integral system which simulates the primary loop, the passive safety systems, the lower containment compartments and the safety actuation logic of the AP600. It is one-fourth height, operates at a reduced pressure and at one-half time scale. Figure 1 presents a schematic of the APEX primary system.

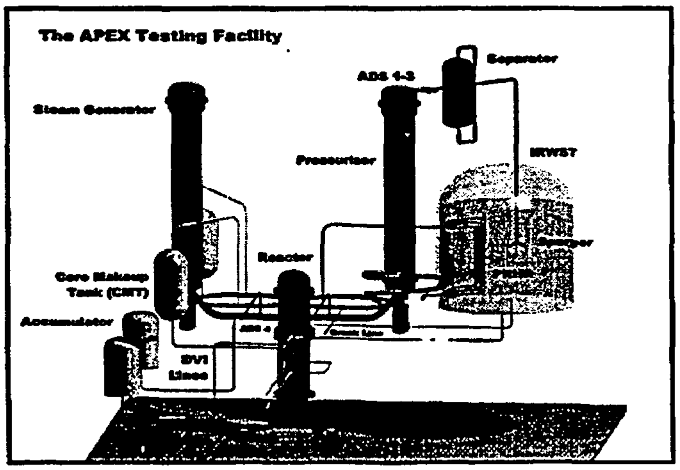

Figure 1. Schematic of the APEX Primary System

The facility is capable of simulating large and small break loss-of-coolant-accidents, station blackouts, and inadvertent ADS actuation scenarios. In addition, simulations can be carried out into long-term core cooling.

2.0 Summary of Key Tests

A wide range of AP600 operating conditions have been studied in the APEX facility. These range from integral system design basis accident scenarios to separate effect, beyond design basis accident scenarios. The following tests represents an effort to study some key AP600 phenomena.

2.1 Return to Saturation Oscillations

The NRC-13 series of tests (NRC-5013, 6113 and 6013) is a parametric study to investigate the flow, pressure and reactor vessel liquid level oscillations that occur during IRWST injection subsequent to the core outlet reaching saturated conditions. These tests were performed under steady-state conditions in which the IRWST liquid level and the core power were incrementally varied. These tests provided significant insights into the oscillation behavior and were useful in developing a predictive model.

2.2 Primary System Thermal Stratification

Thermal stratification in the cold legs was observed for a variety of small break LOCA tests. The data from these tests has been used to determine the conditions for the onset of thermal stratification in the primary loops.

2.3 Counter-current Flow Limitation (CCFL) in the PZR Surge Line

Matrix test NRC-10 examined the effects on system depressurization and core liquid inventory caused by a significant reduction in the ADS-4 flow area. During the test, the liquid in the pressurizer did not immediately drain to the reactor vessel because of the high vapor flow rates through the surge line. The onset of this counter-current flow limitation is of significant interest because of the complex geometry of the surge line.

2.4 No Reserve Tests

The NRC-25 series of tests (NRC-6025, 6125, 6225, 6325, 6425, 7525, 7625, and 7725) represents a parametric study to determine the minimum liquid reserves required to prevent a temperature excursion in the core. The tests are initiated with a failure of all passive safety systems with the exception of portions of the ADS-4 system and the IRWST injection system. The parameters varied were ADS-4 flow area, core power and initial system pressure. The tests were initiated by an ADS-4 blowdown with the reactor vessel liquid level at the bottom of the hot legs. Power was held constant throughout the test. During the system blowdown, ADS-4 liquid entrainment, core boiling and flashing removed reactor vessel liquid inventory. The only means of refilling the reactor vessel was through IRWST injection. However, IRWST injection could not begin until the system pressure dropped below the gravity head pressure of the IRWST. Unlike previous tests, the NRC-25 series was performed at full ADS 4 actuation pressure and time scale. The objective of these tests was to develop a map which identifies the conditions for core uncovering.

2.5 Nitrogen Transport in the Primary

Tests are presently being performed to determine how nitrogen from the accumulators is transported through the primary system during the course of a LOCA. These experiments implement a radioactive gas, Argon-41, to serve as a tracer to track the nitrogen. This phenomenon is important to understanding the effects of non-condensible gases in the primary system.

3.0 CONCLUSIONS

In conclusion, thirty NRC matrix tests have been successfully performed in the APEX test facility. Two of these matrix tests are series which include parametric experiments to study oscillation behavior in the primary system and the approach to core uncovering. The experiments have given significant insight into the following phenomena:

- Return to saturation oscillations
- Primary system thermal stratification
- Counter-current flooding limitations in the PZR surge line
- Conditions for core uncovering
- Transport of nitrogen in the primary system

ROSA-AP600 TEST RESULTS FOR BEYOND-DESIGN BASIS ACCIDENT SCENARIOS

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Westinghouse Electric Corporation has submitted their Advanced Passive 600 MWe (AP600) nuclear power plant design to the U. S. Nuclear Regulatory Commission (NRC) for design certification. This new design features passive safety systems for mitigating accidents and operational transients. The passive safety systems rely either on gravity-driven flow or stored capacity (for example, pressure-driven accumulator injection) to ensure effective operation.

The Japan Atomic Energy Research Institute (JAERI) and NRC have entered a cooperative research agreement to perform a series of full-pressure integral testing on safety response of the Westinghouse AP600 design, using JAERI's ROSA-V Large Scale Test Facility (LSTF). The LSTF was originally built to simulate a Westinghouse-type four-loop 1100 MWe PWR. It has been modified by adding scaled components particular to the AP600 design, including two Core Makeup Tanks (CMTs), a Passive Residual Heat Removal (PRHR) System, an In-Containment Refueling Water Storage Tank (IRWST) and a four-stage Automatic Depressurization System (ADS). In addition, the existing accumulators were modified to provide scaled injection capacity and pressure-flow characteristics. The modified LSTF is a 1/30 volumetrically-scaled, full-height, full pressure model of the AP600.

During the Phase I Program 14 tests were conducted in 1994-1995 to study the transient behavior during simulations of small-break LOCAs, steam generator tube ruptures (SGTR), a main steam line break, a station blackout, and an inadvertent actuation of the ADS. These tests were designed to identify and study the most important phenomena and phenomena interactions for the transients indicated above. In particular, boundary conditions and phenomena that potentially could slow the primary system depressurization and thus prevent uninterrupted, timely IRWST injection from maintaining a cool core were examined. Further, the behavior of the passive safety systems were studied to assure proper guidance is given to plant operators. Throughout the Phase 1 Program no core heatups were experienced and the AP600 safety systems, as reflected in the modified LSTF, appeared to function as designed even though issues were identified for resolution. Some of these issues were studied in the Phase II Program.

The Phase II Program was designed to study key scenarios that belong to the beyond-design-basis accident (BDBA) domain. Six tests, completed by the end of May, 1997, were included in Phase II. These tests were centered on multiple equipment failure together with variations of the test conditions used in Phase I. The Phase II test matrix, together with the selected equipment failures, are listed in Table 1.

Major findings from the Phase II tests are:

- AP-CL-10 and AP-CL-12 show the PRHR system, with no assistance from CMT recirculation, is still capable of removing in excess of all the core decay energy while providing adequate subcooled liquid to the core. Approximately half the core was in a subcooled state in both experiments prior to ADS actuation.
- Complementary to the AP-CL-10 data, the AP-CL-12 data showed: (a) The presence of two accumulators is not sufficient to prevent a core heatup from occurring following the opening of two

Table 1: Summary of Phase II Experiments

Experiment Identifier	Date	Scenario	Failures	Intact Equipment & Comments
AP-CL-10	December 17, 1996	1-in P-Loop CL SBLOCA	Both CMTs; ACC-C; ADS1234C	PRHR; ACC-P; ADS-4P; IRWST; ADS-4P opened 30 min after S-signal.
AP-CL-11	January 28, 1997	2-in. P-Loop CL SBLOCA	CMT-P; ADS124; PRHR; ACC-C; 1 of 2 ADS3	CMT-C; 1 of 2 ADS3; 1 NRHR Pump; ACC-P; IRWST; NRHR pump injected into CMT discharge line downstream of isolation valve. NRHR shutoff head = 1.2 MPa.
AP-SG-02	February 25, 1997	5-tube SGTR; 200% break	Both CMTs; ADS123	Both ACCs; PRHR; IRWST; ADS4 opened manually 1 hr after S-signal. SGTR simulation in C-Loop between SG inlet plenum and tube sheet.
AP-DV-02	March 18, 1997	200% DVI line break; C-Loop	PRHR; Both CMTs; 1 of 2 ADS4P; ADS123	Both ACCs; ADS4C; IRWST; 1 of 2 ADS-4P-- ADS4 opened 30 min after S-signal.
AP-FW-01	April 9, 1997	Loss of Feedwater	PRHR; 1 of 2 ADS3; ADS12; 1 of 2 ADS4P	1 of 2 ADS3; ADS4C; IRWST; Both CMTs; Both ACCs; 1 of 2 ADS4P with delayed trip.
AP-CL-12	May 15, 1997	1-in. P-Loop CL SBLOCA	Both CMTs; 1 of 2 ADS4C; ADS123	Both ACCs; PRHR; ADS4P opened 30 min after S-signal; 1 of 2 ADS4C opened when core temperature reached 460 K; IRWST; sister experiment to AP-CL-10.

where ACC = accumulator; ADS4C = C-Loop ADS Stage 4 valve; ADS4P = P-Loop ADS Stage 4 valve; ADSxyz = ADS Stages x, y, and z; CL= cold leg; DVI=direct vessel injection; NRHR = Normal Residual Heat Removal; SG = steam generator.

ADS Stage 4 valves—similar to that observed in the AP-CL-10 experiment when only one accumulator was available. (b) The addition of a third ADS Stage 4 valve, even if opened after core heatup has begun, is sufficient to restrict the PCT in the LSTF core to less than 910 K.

- One NRHR pump was sufficient to provide long term cooling instead of IRWST injection. During the AP-CL-11 test the core was completely covered and subcooled over more than 60% of its length (and with the degree and extent of subcooling increasing with time) when the experiment was terminated. The IRWST did not inject because the pressurizer level remained high following activation of ADS 3.
- The additional failure assumption for the Phase II SGTR experiment (AP-SG-02), in contrast to the Phase I SGTR experiment (AP-SG-01), of no ADS Stages 1, 2 and 3 valves led to core uncovery before IRWST injection in the latter stage of the transient, though measured temperature excursions did not exceed 150 K (PCT = 530 K). Once IRWST injection began recovery was assured.
- Condensation-induced water hammer was observed in the intact DVI line during the AP-DV-02 experiment when the accumulator injection flow decreased and allowed the formation of a stratified two-phase flow condition in the intact DVI line.
- The transient progression of the loss-of-feedwater experiment (AP-FW-01) was typical (steam generator dryout by secondary mass loss, primary pressurization, primary mass loss due to primary valve actuation). Once ADS4C was actuated, due to low CMT level, recovery proceeded normally and resulted in IRWST injection. Actuation of ADS4P was unnecessary for recovery. No core uncovery was experienced in this experiment.

SENSITIVITY OF BWR STABILITY CALCULATIONS TO NUMERICAL INTEGRATION TECHNIQUES

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Computer simulations have shown that stability calculations in boiling water reactors (BWRs) are very sensitive to a number of input parameters, modeling assumptions, and numerical integration techniques. Following the 1988 LaSalle instability event, a significant industry-wide effort was invested in identifying these sensitivities. One major conclusion from these studies was that existing time-domain codes could best predict BWR stability by using explicit methods for the energy equation with a Courant number as close to unity as possible. This paper presents a series of sensitivity studies using simplified models, which allow us to determine the effect that different numerical integration techniques have on the results of stability calculations. The present study appears to indicate that, even though using explicit integration with Courant number of one is adequate for existing codes, higher-order solution techniques can result in significant improvements of not only accuracy, but also computing time.

The problems associated with the numerical solution of oscillatory-type systems can be illustrated by solving the following ordinary differential equation (ODE)

$$\frac{d^2x}{dt^2} + 2\,\sigma\,\frac{dx}{dt} + \left(\sigma^2+\omega^2\right)x = \epsilon \tag{1}$$

For this particular simulation, we have used $\sigma = 0.1$ and $\omega = \pi$, which results in an oscillation of frequency 0.5 Hz and DR = 0.82. Following the conventional numerical solution approach, we can convert Eq (1) into two first order ODE's and solve them numerically using different integration algorithms and time step. The results are shown in Fig. 1 and indicate that first order integration methods (Euler's method) either overestimate (explicit) or underestimate (implicit) the decay ratio significantly even for reasonable time steps. Second order integration methods are very accurate even for large integration time steps. All methods converge to the analytic solution as the time step is reduced. It can be shown that the percent decay ratio error for first-order integration methods (e.g., Euler) is approximately equal to the integration time in milliseconds. Most time-domain system codes use integration times of the order of 1 to 10 ms; thus, the decay ratio error caused by first-order integration errors should be less than 10% for these codes.

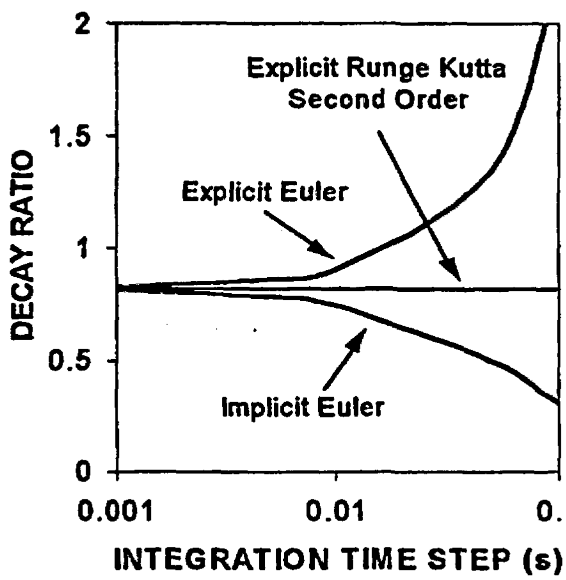

Figure 1. Decay ratio errors induced by first-order numerical integration on Eq. (1)

* Managed by Lockheed Martin Energy Research Corp. For the U.S. Department of Energy under contract DE-AC05-96OR22464

Even though, the partial differential equations (PDEs) that represent BWR stability are significantly more complex than Eq. (1), the qualitative behavior of the solution is similar and one should expect similar behavior. A more complex model of BWR dynamics can be represented by the following equations

$$\frac{\partial \alpha(z,t)}{\partial t} + V \frac{\partial \alpha(z,t)}{\partial z} = Q(t) \tag{2}$$

$$\frac{dQ(t)}{dt} = -\frac{K}{H} \int_0^H \alpha(z,t)\, dz \tag{3}$$

where Eq (2) models the convection of the density wave, α, which propagates at constant velocity, V, and Eq (3) models the power feedback, Q, which is proportional to the core-average density through a density reactivity coefficient, K. This model, although simple, captures most of the numerical problems associated with solving the BWR stability problem in time and space.

It is well known that the convection equation (2) can be solved exactly by using an explicit first order method with Courant number of one. However, when Eq. 23) is coupled to Eq (3), that is not the case and the situation represented in Fig 2 arises. We observe that the decay ratio error is not dependent on Courant number, but on actual time step; thus, the fact that better results are obtained with larger number of nodes is not because of nodalization errors, but because it forces the integration algorithm to take smaller steps. Most stability calculations with time-domain system codes are performed with at least 24 axial nodes and the average Courant number in the core is from 0.2 to 0.5. The integration time step is smaller than a Courant number of one would require because it is limited by either the smallest node or the node with the fastest velocity. Thus, decay ratio errors induced by first-order explicit integration inaccuracies are expected to be in the order of 10%, which is adequate for stability calculations (all NRC-licensed stability codes are assumed accurate to within 20%). The example shown in Fig. 2 also indicates that space nodalization errors are not as important as time-integration errors. Thus, a logical improvement for next-generation codes would be to use higher-order time integration algorithms and larger spatial nodes (preferably of unequal size to maintain a more-constant Courant number over the core). This would allow for significantly longer time steps (up to one order of magnitude) and shorter execution times.

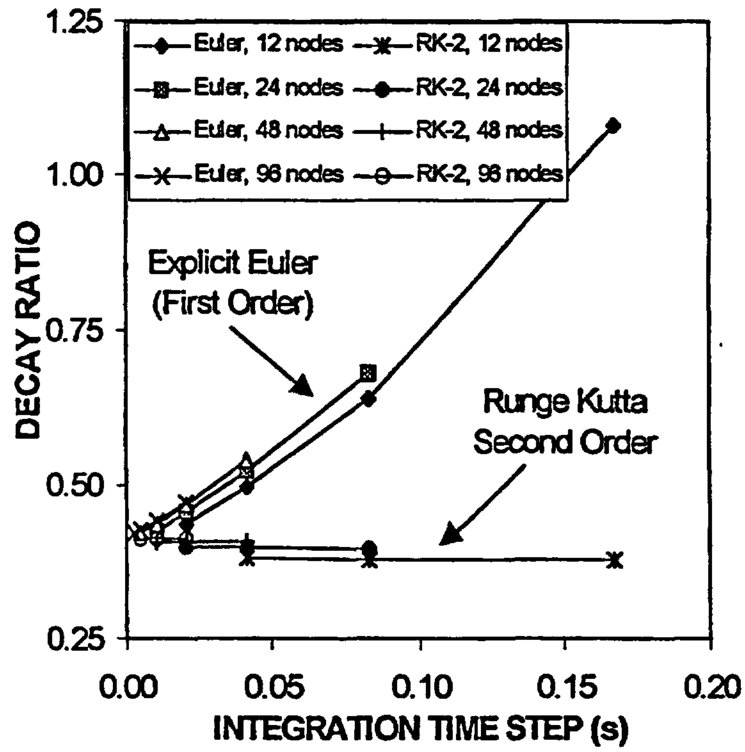

Figure 2. Calculated decay ratio errors for Eqs (3) and (4) for different integration schemes

The PANDA Tests for the SBWR

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The PANDA facility at the Paul Scherrer Institute (PSI) in Switzerland has been used to examine the long-term LOCA response of the containment and in particular of the Passive Containment Cooling System (PCCS) of the General Electric, Simplified Boiling Water Reactor (SBWR).

In the first series of tests discussed here the modular PANDA facility was configured to represent the SBWR containment compartments and PCCS condensers. The SBWR Drywell (DW) and Wetwell (WW) were each represented by *two* vessels with large interconnections; this arrangement allowed dissymetries and three-dimensional effects to develop. The WW has interconnections in both the gas and liquid spaces. The scaling was 1:25 for volumes, power and horizontal areas, whereas relevant vertical heights were maintained 1:1. Process fluids, pressures and temperatures were prototypical. More than 600 sensors were installed for measuring process variables. Decay heat was simulated by controlled electrical heaters in the reactor pressure vessel.

Steady-state PCCS condenser performance tests were conducted in PANDA at the beginning of 1995, as counterpart tests to the PANTHERS tests in Italy. Extensive facility characterization tests were completed in July 1995; these were needed for the accurate description of the facility in computations. The series of ten transient system behavior tests reported here were completed at the end of 1995.

The PANDA transient tests examined system response during the long-term containment cooling period. Their objectives were to demonstrate that:

- The containment performance is similar at a larger-scale to that previously demonstrated with the GIRAFFE tests.
- Any non-uniform distributions in the containment do not create significant adverse effects.
- There are no adverse effects associated with multi-unit PCCS operation and interactions with other reactor systems

The tests also extend the database for qualification of the TRACG code.

Uniform and asymmetric Drywell conditions can be created in PANDA by varying the fraction of steam flow that is injected into each of the interconnected DW vessels representing the SBWR Drywell and by modifying the number of condenser units directly connected to each DW vessel. The large scale of the facility allows mixing and stratification effects to be observed at nearly-prototypical scales. Thus, the test objectives specifically included the study of such mixing of steam and noncondensible gases in the DW and in the WW.

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All ten transient system tests investigated the long-term cooling phase of the SBWR following a Main Steam Line Break. The tests can be divided into three groups: (1) base-case and repeatability tests, (2) asymmetric tests and (3) tests investigating PCCS start-up, parallel operation of the PCCS and of the isolation condenser (IC), bypass leakage between DW and WW, and, "early start."

The initial conditions for the basic tests were the SBWR best-estimate thermodynamic conditions at one hour from the beginning of the LOCA. Steam was symmetrically discharged to both DW vessels. General repeatability tests were performed and the effects of interconnection and sequential refill of the condenser pools were investigated. The tests provided good insights into the operation of the system.

The asymmetric series of tests investigated the influence of asymmetries in the facility configuration on PCCS performance. These tests featured variations in the number of PCCS condensers in operation and of their connection to the DW vessels, as well as the effects of asymmetric steam flow to the DWs. They have thus established the "envelope" response of the containment under a very wide range of possible mixing conditions.

To demonstrate the proper start-up of the PCCS and the purge of the air from the DW to the WW, a PCCS-startup test was performed with practically pure air filling initially the DWs and PCCS condensers. Tests also demonstrated parallel IC and PCCS operation and the effects of leakage flow from the DW directly to the WW air space. The last test started from initial conditions earlier in the transient and included the phase when cold Gravity-Driven Cooling System (GDCS) water is discharged into the reactor pressure vessel thus depressurizing the DWs and causing vacuum breaker openings; this "early start" test established the link between the initial conditions calculated for the other tests and the actual history of the earlier period of the transient.

The tests demonstrated a favorable overall containment system behavior due to the "robust" performance exhibited by the PCCS. The PCCS units were able to perform their function under all conditions tested. No adverse effects due to parallel operation of systems or units were observed. Good understanding of the effects of mixing in the large vessels was obtained.

Current USNRC Review Guidance for Digital Instrumentation and Control Systems

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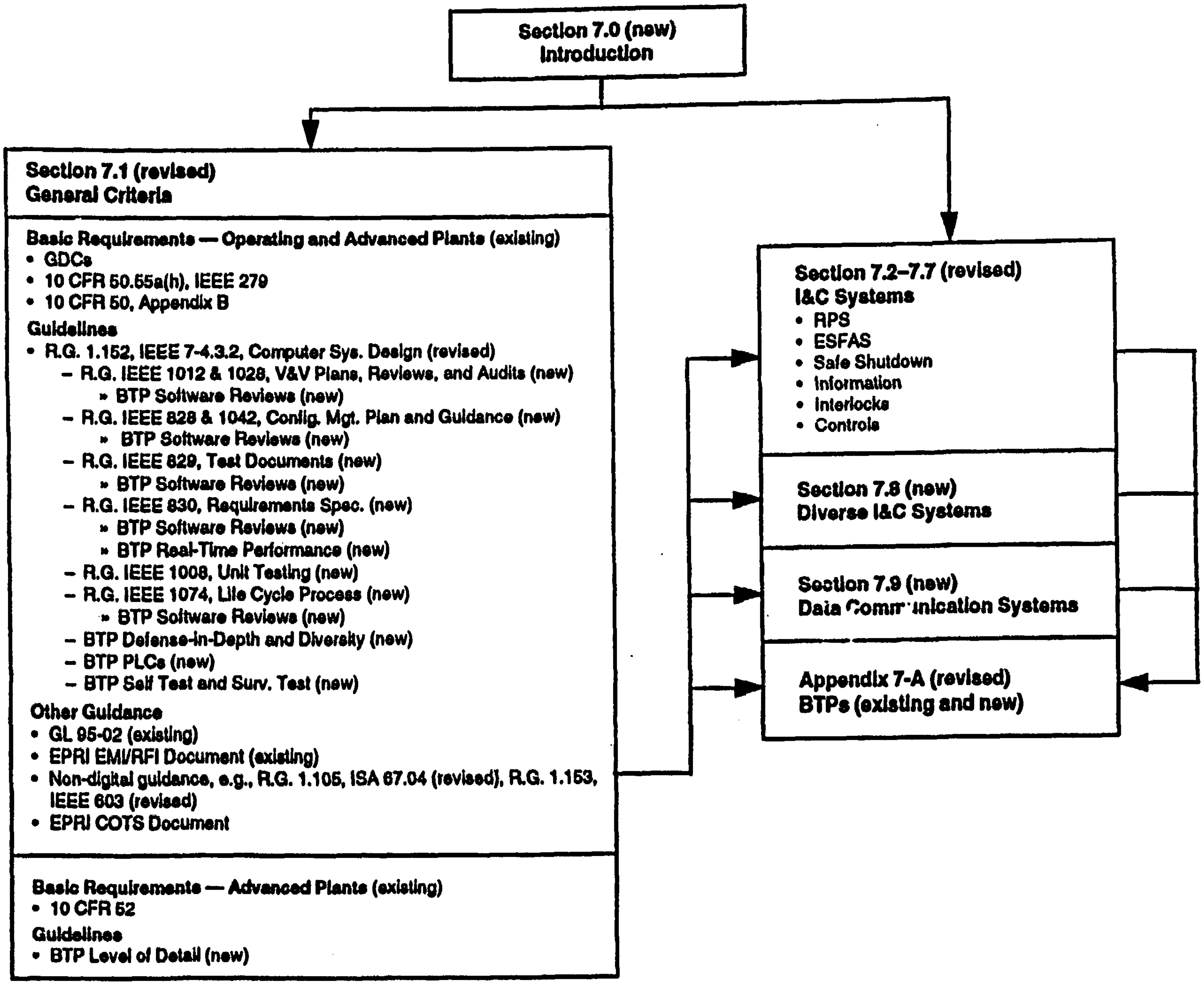
Abstract

USNRC review guidance for digital instrumentation and control systems has been recently published in final form consisting of a revision to Section 7 of the Standard Review Plan (NUREG-0800). This revision will be described with emphasis on key changes, additional Branch Technical Positions (BTPs), and six new Regulatory Guides developed by the Office of Research covering review of software life-cycle processes. As recommended by the National Research Council of the National Academy of Sciences in their recent report, "Digital Instrumentation and Control Systems in Nuclear Power Plants: Safety and Reliability Issues," additional research is necessary to further improve the technical basis for review and regulation of digital systems used in safety systems of nuclear power plants. These issues will be described briefly, and some of our initial thinking about needed regulatory research to address these issues will be presented.


Section 7.0 (new)
Introduction
Section 7.1 (revised)
General Criteria
Basic Requirements — Operating and Advanced Plants (existing)
• GDCs
• 10 CFR 50.55a(h), IEEE 279
• 10 CFR 50, Appendix B
Guidelines
• R.G. 1.152, IEEE 7-4.3.2, Computer Sys. Design (revised)
– R.G. IEEE 1012 & 1028, V&V Plans, Reviews, and Audits (new)
» BTP Software Reviews (new)
– R.G. IEEE 828 & 1042, Config. Mgt. Plan and Guidance (new)
» BTP Software Reviews (new)
– R.G. IEEE 829, Test Documents (new)
» BTP Software Reviews (new)
– R.G. IEEE 830, Requirements Spec. (new)
» BTP Software Reviews (new)
» BTP Real-Time Performance (new)
– R.G. IEEE 1008, Unit Testing (new)
– R.G. IEEE 1074, Life Cycle Process (new)
» BTP Software Reviews (new)
– BTP Defense-in-Depth and Diversity (new)
– BTP PLCs (new)
– BTP Self Test and Surv. Test (new)
Other Guidance
• GL 95-02 (existing)
• EPRI EMI/RFI Document (existing)
• Non-digital guidance, e.g., R.G. 1.105, ISA 67.04 (revised), R.G. 1.153, IEEE 603 (revised)
• EPRI COTS Document
Basic Requirements — Advanced Plants (existing)
• 10 CFR 52
Guidelines
• BTP Level of Detail (new)
Section 7.2–7.7 (revised)
I&C Systems
• RPS
• ESFAS
• Safe Shutdown
• Information
• Interlocks
• Controls
Section 7.8 (new)
Diverse I&C Systems
Section 7.9 (new)
Data Communication Systems
Appendix 7-A (revised)
BTPs (existing and new)

Safety Critical Digital System Architectures

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This paper provides an overview of technologies used in the design and analysis of safety-critical digital systems. The term digital system is used in this paper to imply an embedded, real-time control system containing both hardware and software. The paper begins by surveying applications which require safety critical digital systems. Examples include railway signaling and switching, aircraft flight control, medical electronics, nuclear reactors, automobile braking, and process control. The applications are used to identify common problems, design methodologies, architectures, modeling methods, and evaluation techniques which are employed in the creation of safety critical systems. The technologies identified are presented and illustrated with the example applications. The objective is to describe techniques and approaches which are sufficiently generic to be used in multiple applications.

The paper concludes with an identification of unsolved problems in the field of safety critical systems. The weaknesses of existing approaches are described along with the problematic assumptions which are often used to allow such systems to be designed and analyzed. Example problems include: (1) the lack of effective fault models; (2) ineffective techniques to identify and handle common mode faults; (3) the lack of computer-aided design tools to support the development of safety-critical systems; (4) the impact of the use of commercial off the shelf (COTS) hardware and software; (5) the artificial separation of hardware and software rather than consideration of the complete hardware/software system; and (6) the inability to consider the impact of human effects on the system design and the human interaction with an operational system. The objective of the paper is to propose a research and development agenda for safety critical digital systems.

In summary, this paper provides three primary results: (1) a survey of a variety of applications which require safety critical systems; (2) a summary of problems, approaches, and results which are applicable to multiple applications; and (3) an identification of open issues needing understanding of safety critical digital systems and the problems which remain to be solved.

CURRENT RESEARCH RESULTS ON THE TECHNICAL BASIS FOR ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED DIGITAL I&C SYSTEMS*

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Distributed computer systems and other advanced digital technologies are being used to upgrade instrumentation and control (I&C) systems in nuclear power plants. One concern with using such systems in nuclear power plants, particularly for safety-related applications, is their potential sensitivities to nuclear power plant environments and the possibility of common cause effects resulting from environmental stress. To properly address the qualification needs for safety-related digital systems, it is useful to more fully understand the response of these technologies to likely environmental stress and to better characterize the environments in which they operate. For example, there is the lack of information on potential vulnerability of the I&C equipment under smoke exposure. As another example, until recently little was known about the prevailing ambient electromagnetic interference (EMI) and radio-frequency interference (RFI) environment in nuclear power plants. This lack of information made it difficult to establish electromagnetic operating envelopes for safety-related I&C systems.

These uncertainties prompted the U. S. Nuclear Regulatory Commission (NRC) to initiate the confirmatory research program, *Qualification of Advanced Instrumentation and Controls (I&C) Systems*, at three U. S. Department of Energy national laboratories—Oak Ridge National Laboratory (ORNL), Sandia National Laboratories (SNL), and Brookhaven National Laboratory (BNL). The objective of this research program is to provide the technical basis for environmental qualification of microprocessor-based safety equipment in nuclear power plants. The research approach involves evaluating existing military and industrial guidance, identifying the most significant environmental stressors, investigating the likely failure modes for digital technologies under varying levels of environmental stress, and recommending appropriate methods for qualifying safety-related digital equipment. This paper presents a summary of the results to date for the different projects performed under the research program. The program is divided into five different projects, three of which are being performed by ORNL and one each is being performed by the other national laboratories.

In the first project, BNL conducted a screening study to identify environmental stressors for advanced I&C systems in a nuclear power plant which can be potentially risk-significant, and compared the hardware unavailability of such a system with that of its analog counterpart. The risk screening was based on estimated risk-sensitivities of the stressors, (i.e., changes in plant risk caused by the stressors), and are quantified by estimating their effects on the occurrences of I&C failure and the consequent increase in risk in terms of relative core damage frequency (CDF). The stressors evaluated for risk effects were temperature, humidity, vibration, EMI from lightning, and smoke. The results for the example plant used in the study indicate that humidity, EMI from lightning, and smoke can be potentially risk-significant. The risk-significance of EMI

*Research sponsored by the Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission The opinions and viewpoints expressed herein are those of the authors and do not necessarily reflect the criteria, requirements, and guidelines of the U S. NRC.

from lightning and smoke, however, are sensitive to detection periods for equipment failure following the events. The results also showed that the effects of some stressors, such as humidity, can be sensitive to the location of the equipment. For the levels of the stressors analyzed, risk effects from temperature in digital I&C equipment locations, and from assumed levels of vibrations, appear to be less significant.

In the second project, which was performed by SNL, smoke exposure tests were conducted on digital circuit components and circuit boards to determine failure mechanisms and the effect of different packaging techniques. The component tests focused on short-term effects such as circuit bridging in typical components and the factors that can influence how much the smoke will affect them. These factors include the component technology and packaging, physical board protection, and environmental conditions such as the amount of smoke, temperature of burn, and humidity level. The likelihood of circuit bridging was tested by measuring leakage currents and converting those currents to resistance. The study found hermetically-sealed ceramic packages to be more resistant to smoke than plastic packages. Coating the boards with an acrylic spray provided some protection against circuit bridging. The smoke generation factors that affected the resistance the most were humidity, fuel level, and burn temperature. The use of CO_2 as a fire supressant, the presence of galvanic metal, and the presence of polyvinyl chloride (PVC) did not significantly affect the outcome of these results.

In the third project, ORNL developed recommendations for endorsing design, testing, and installation practices that contribute to establishing electromagnetic compatibility. As part of the confirmatory research, electromagnetic emission profiles at selected nuclear power plants were characterized through long-term plant measurements. These emission profiles were subsequently used to confirm that the recommended test levels based on military (MIL-STD-461 and MIL-STD-462) and industrial (IEEE Std C62.41) standards were appropriately tailored to the nuclear power plant environment (i.e., the recommended envelopes reasonably bound the expected electromagnetic conditions as determined from measured data).

In the fourth project, also performed by ORNL, the potential failure modes and vulnerabilities of distributed computer systems under environmental stress were investigated. An experimental digital safety channel (EDSC) incorporating technologies that are representative of proposed ALWR safety systems was assembled for the tests. Stressors that were investigated included EMI/RFI, temperature, humidity, and smoke exposure. The investigation found communication interfaces to be the most vulnerable elements of the EDSC. The majority of effects resulting from the application of the stressors were communication errors, particularly for serial datalinks. Many of these errors were intermittent timeout errors or corrupted transmissions, indicating failure of a microprocessor to receive data from an associated multiplexer, optical serial link, or network node. Because of similarities in fabrication and packaging technologies, other safety-related digital systems are likely to be vulnerable to similar upsets. As was experienced with the EDSC, intermittent component upsets will typically impede communication, either on the board level (e.g., during bus transfers of data), or on the subsystem level (e.g., during serial or network data transfers). Thus, qualification testing should confirm the response of any digital interfaces to environmental stress that the safety-related system is likely to experience under both normal and accident conditions.

Under the fifth project, which is being performed by ORNL, the results and insights obtained from all the above studies are being used to develop the technical basis for environmental qualification of digital I&C equipment for nuclear power plants. In particular, the technical basis is addressing issues including (1) recommended standards for environmental qualification of safety-related digital I&C systems; (2) determination of the appropriate way in which to address smoke in an environmental qualification program; and (3) resolution of the need for any accelerated aging for equipment to be located in an environment that is normally considered as benign.

Recent Results of an Experimental Study on the Impact of Smoke on Digital Equipment*

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Nuclear power plants are replacing analog instrumentation and control (I&C) equipment with digital I&C equipment; however, there is concern about how these new control systems will be affected by abnormal or severe environments such as smoke from an electrical fire. In 1994 the USNRC initiated a program at Sandia National Laboratories (SNL) to determine the impact of smoke on advanced instrumentation and on safety. The program is providing information on the reliability of digital I&C systems in a smoke environment. The failure modes of digital I&C due to smoke have been identified as (1) corrosion of metal contacts and circuit traces, (2) circuit bridging, and (3) increase of contact resistance. Smoke is expected to cause immediate failures by circuit bridging and an increase in contact resistance, but corrosion is expected to cause long-term failures.

To date SNL has completed three sets of tests. The first set was performed in conjunction with Oak Ridge National Laboratory and their NRC-sponsored program on advanced I&C qualification. Two digital-based systems were exposed to smoke from cable fires using various plausible scenarios and monitored during the exposure. The results showed that smoke from a cable insulation fire may cause transmission errors or interruptions in digital systems. Because these interruptions were intermittent, we believe that they were caused by circuit bridging since corrosion and increased contact resistance should be longer term rather than intermittent effects.

The second set of tests was carried out to determine the effect of circuit bridging on component packages because this was the most likely cause of failure in the first set of tests. Leakage currents were measured between biased pins on empty component packages. Some of the component packages were protected with a conformal coating of acrylic or a chassis body equipped with a cooling fan. The results of the tests showed that smoke immediately reduces insulation resistance, but that the resistance may recover if the smoke is vented. The acrylic coating helped reduce circuit bridging, but the cooling fan on the chassis body increased the amount of soot deposited on the components and thus increased circuit bridging.

The third set of tests was conducted to study the effects of three possible failure modes on a functional circuit board: (1) circuit bridging, (2) corrosion of contacts, and (3) induction of stray capacitance. Stray capacitance can be induced by adding conductive surfaces near high-frequency circuits, and hence is related to circuit bridging. These failure modes were studied on functional boards containing circuits sensitive to these failure modes. The components on the boards were those commonly used in modern electrical circuits. The boards contained high-voltage (to study circuit bridging), high-current (to study corrosion), high-frequency (to study stray capacitance), and high-speed digital circuits. Circuit performance was measured continuously on bare and conformally coated boards during the smoke exposure and for 24 hours after the start of the exposure. The boards were also subjected to a range of smoke levels to try to determine failure thresholds.

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* Research sponsored by the Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission. The opinions and viewpoints expressed herein are those of the authors and do not necessarily reflect the criteria, requirements, and guidelines of the USNRC.
† Sandia is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the United States Department of Energy under contract DE-AC04-94AL85000.

Conformal coatings can substantially increase the smoke tolerance of circuits. In general, there are five types of coatings: polyurethane, epoxy, silicone, acrylic, and parylene. One coating from each of the five types was selected to protect the functional boards. The smoke exposure tests showed that all the circuits performed much better with a conformal coating, although there were minor differences in the performance of the different coatings.

Of the three failure modes studied in the functional board tests, circuit bridging occurred the most often and was the most severe effect. The high-voltage circuit was most affected by smoke (100% change in measured values). Its high impedance (50 MΩ) was shorted during the exposure, but in some cases recovered after the smoke was vented. This shorting occurred for even small concentrations, such as 3 g of fuel per cubic meter of air. Other circuits that were sensitive to bridging, such as the coupled transmission lines, were also temporarily affected during the smoke exposure. The high-speed digital circuit, which could be affected by circuit bridging, only failed at the highest smoke level (200 g/m^3) and in only some of the tests.

The main component in the high-speed digital circuit was an advanced transistor-to-transistor logic chip (FAST). To determine a failure threshold for logic chips from different chip technologies, a nonsmoke experiment was performed in which the circuit bridging of smoke was simulated with a variable resistor. This experiment showed that a FAST chip is tolerant to relatively high leakage currents and hence is more tolerant to smoke than other chip technologies. This experiment also showed that technologies with a high output current are more tolerant of smoke. Because standard complementary metal-oxide semiconductor (CMOS) chips have lower output current drive than FAST chips, they are more sensitive to smoke. These tests suggest that CMOS chips would have been affected at lower smoke levels than FAST chips.

The other two failure modes, corrosion and induced stray capacitance, caused little change in the function of the circuits. The smoke permanently increased resistance of the high-current circuit, implying that the contacts were corroded. However, the change was very small (<2%). The high-frequency circuit, which is sensitive to stray capacitance, showed very little change after a smoke exposure in either the short or long term.

The results of the tests suggest that conformal coatings, type of circuit, and the characteristics of chip technologies are major considerations when designing digital circuitry to be used in nuclear power plant safety systems. Of the three smoke-caused failure modes studied using different circuit types, circuit bridging is the most likely, followed by corrosive attack of the solder joints.

Combining Disparate Sources of Information in the Safety Assessment of Software Based Systems.

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Summary

The use of programmable equipment in safety related systems in NPPs, has many clear advantages compared to conventional equipment. But there has also been a certain reluctance to use programmable equipment in safety critical systems. A reason for this has been the complexity of safety assessment and the licensing of the software in these systems.

The OECD Halden Reactor Project (HRP) is an international institution with participation from 19 countries. USNRC is the main participant from the US. A main research topic over the last twenty years has been software dependability. Particular emphasis has been placed on software in safety critical systems.

There are three complementary principles which should be followed to obtain dependable software. The first principle in this respect is fault avoidance through good software engineering and quality assurance throughout the complete life cycle of the software. The second principle is fault detection and removal through a thorough validation and verification activity. A third principle, which could also be considered is fault tolerance, i.e. the system should be designed so that a single failure will not jeopardize safety. HRP has made research activities on methods of relevance for all these principles, as formal software development method, static analysis, testing, software diversity etc. The paper will start with a short survey of these activities, set in the framework of these principles.

The main topic of the paper will, however, be a discussion on how to combine disparate sources of information in the safety assessment of software based systems. The background for this is that HRP has produced a guideline for reviewing and assessing safety critical software in nuclear power plants for the Swedish Nuclear Power Inspectorate (SKI). These Guidelines were applied in the licensing of the exchange of an analog protection system with a functionally equivalent programmable system in the Swedish nuclear power plant Ringhals. HRP took part as consultants in the licensing process, and the experience gained from this work formed a basis for further considerations about the safety assessment process of a system partly based on proprietary software modules.

The approach is to combine all available information about the system into a diagrammatic framework for the safety assessment, and thereby for the final approval of the system. This framework has the form of an 'influence net', i.e. a directed graph where each node represents an aspect in the total

assessment process. The top nodes in the graph represent the basic information sources which are used in the acceptance process. This information is penetrated through the net down to the bottom node which represents the final acceptance of the system.

A safety assessment of a system depends on information on its reliability. It does also, however depend on information on potential risks to plant and environment, and on particular defenses included to prevent that a potential fault in the system may jeopardize safety. For the reliability assessment there are four main sources of information:

1. Information about producer and development process. This includes the producer's pedigree, production methods, standards followed, quality control etc.

2. Information about the programs. This can be obtained from code listing, but also from other written sources, as user manuals, program specifications etc. To get more details one can perform analysis of this information, e.g. to measure complexity, find possible failure modes etc.

3. Information about verification and validation. This includes both static analysis and dynamic testing. Data from all types of testing, i.e. debug testing, unit testing, acceptance testing etc. may be relevant.

4. Information about usage. Even if the system in question may be unique, it will usually contain various standard, off the shell, software modules. Such modules may have a widespread usage, and information about this usage should be utilized for the reliability assessment.

The various factors mentioned above are of rather disparate nature. Some are of qualitative nature, like the producer's reputation, the development quality etc., others are measurable, but not directly connected to reliability estimation, like program size, program complexity etc., whereas others again, like results from statistical testing, can be used directly in a reliability estimate. It is therefore necessary to have a methodology to combine the disparate evidences. An emerging methodology is to use Bayesian Belief Networks (BBN) to combine such evidences. The objective is to show the link between basic information and the confidence one can have in a system.

A BBN is a connected and directed graph, consisting of a set of nodes and a set of edges between them. To each node there are associated a variable which can be in a set of states, and a prior probability is assigned to each state. The edges in the graph represents the influences between the nodes, and should also be assigned numerical values. The problem on how to assign these values will be discussed. One method which have been suggested is to apply methods for using and eliciting expert judgement. A computer program can then compute the confidence in a target node, which in the present case can be reliability or safety.

This application of BBN is still rather premature, and has not yet been applied in real safety cases. There is, however, a certain research activity on this topic, and this will be presented in the paper.

Review Guidelines for Software Written in High Level Programming Language Used in Safety Systems

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This paper describes guidelines developed for the review of software written in high level languages for use in safety systems. These guidelines were developed using a uniform 3-level hierarchical framework consisting of top level, intermediate, and base attributes. The top level attributes of reliability, robustness, traceability, and maintainability were developed in order to define general qualities of software related to safety. Intermediate attributes were then developed to describe the top level attributes in greater detail. At the lowest level are the base attributes which were defined for the intermediate attributes to be sufficiently specific to derive language specific guidelines. The following paragraphs discuss the top level attributes in further detail.

Reliability is the predictable and consistent performance of the software under conditions specified in the design basis. This top level attribute is important to safety because it decreases the likelihood that faults causing unsuccessful operation will be introduced into the source code during implementation. The intermediate reliability attributes for reliability were defined as predictability of memory utilization, predictability of control flow, and predictability of timing. A total of 18 base attributes relating to static and dynamic memory allocation, control flow constructs, subroutine interfaces, and variable declaration and initialization were defined.

Robustness is the capability of the safety system software to operate in an acceptable manner under abnormal conditions or events. This top level attribute is important to safety because it enhances the capability of the software to handle exception conditions, recover from internal failures, and prevent propagation of errors arising from unusual circumstances. The intermediate attributes for robustness were determined to be diversity, exception handling, and I/O checking. Seven base attributes were defined for control of software diversity, controlled use of exception handling and input and output checking..

Traceability relates to the feasibility of reviewing and identifying the source code and library component origin and development processes, i.e., that the delivered code can be shown to be the product of a disciplined implementation process. Traceability also includes being able to associate source code with higher level design documents. This top level attribute is important to safety because it facilitates verification and validation, and other aspects of software quality assurance. Two intermediate attributes were defined relating to the use of built-in functions and compiled libraries for which source code is not available. The third level of traceability is identical to the second level, i.e., because the intermediate attributes are sufficiently specific to define language-specific guidelines, they are also served as base attributes.

Maintainability is the means by which the source code reduces the likelihood that faults will be introduced during changes made after delivery. This top level attribute is important to safety because it decreases the likelihood of unsuccessful operation resulting from faults during adaptive, corrective, or perfective software maintenance. The intermediate attributes of maintainability were defined to include readability, functional cohesiveness, abstraction, portability, and malleability. A total of 13 base attributes were defined relating to variable names, clarity of programming, comments, global variables, interface complexity, and non-standard (vendor specific) features.

These attributes were then used to develop specific guidelines for the following languages: Ada83 and Ada95, C and C++, International Electrotechnical Commission (IEC) Standard 1131-3 Ladder Logic, Sequential Function Charts, Structured Text, and Function Block Diagrams, Pascal, and PL/M.

For Ada 83 and Ada 95, the guidelines encourage use of strong typing and exception handling features, but strongly discourage the use of tasking. Certain pragmas such as unchecked deallocation or suppression of run-time constraint checking are also strongly discouraged. The Ada 95 guidelines were based on the those for the earlier version of Ada with additional consideration of object oriented features.

Guidelines for C and C++ were developed together because of the close relationship between the two languages. In addition, programs written in C++ are also likely to contain C code as well. For C, the guidelines address the problems in memory allocation and deallocation, pointers, control flow, and software interfaces. For C++, the guidelines address additional issues associated with multiple inheritance, late binding, and large class libraries.

Guidelines for the Programmable Logic Controller (PLC) ladder logic language considered both the language as defined in the IEC 1131-3, and the significant variation among manufacturers. Where possible, the guidelines were made generic. The guidelines emphasize the need for organization of control flow and internal data storage structure, the importance of timing considerations, and the importance of proper variable naming to enhance readability.

Guidelines for SFCs recognized the difference between the programming paradigm for that language and those of other languages. IEC 1131-3 Sequential Function Charts (SFCs) are intended as a way to organize the control flow of lower level software modules written in other languages defined by the IEC 1131-3 standard (including Ladder Logic). The guidelines emphasize the proper use of SFCs given their intended purpose and orientation.

Guidelines for the IEC 1131-3 Structured Text (ST) and Function Block Diagram (FBD) languages assume strict conformance to the standard because there is less variation among vendor offerings. ST is a text-based language similar to Pascal whereas FBD languages use a graphical representation. Both languages are closely associated with PLCs, computers specialized for real-time industrial control. This specialization results in unique I/O capabilities but limited information processing features. The guidelines emphasize the proper use of these languages given their intended purpose and orientation.

For Pascal, the guidelines address not only the ANSI standard, which is fairly limited, but also, the most popular extensions. These extensions are important because they are more widely used in real- time and near real-time systems than is the standard language. The guidelines emphasize the avoidance of dynamic memory allocation, care in the use of pointers, and software interfaces.

PL/M was extensively used in microprocessor control applications but is now no longer being supported by its corporate progenitor. The guidelines are for PL/M are similar to those of C and Pascal. However, a specific concern for PL/M in safety systems is the erosion of the technical base and the importance of planning for the continued presence of personnel and tools so that systems can be maintained throughout their anticipated life.

NRC CODE CONSOLIDATION PROGRAM

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I. Introduction

The NRC currently relies on four different thermal-hydraulic system analysis codes to audit vendor or licensee analyses of new or existing designs, to establish and revise regulatory requirements, to study operating events, and to anticipate problems of potential significance. NRC plans to consolidate all functionality embodied in the suite of codes into one code. Since each of the four codes serves a particular function, the codes will be maintained for a period during the consolidation process, so that user needs are accommodated. These versions will not be static, but will be aggressively maintained and improved throughout the effort. Concurrently, NRC will work to develop innovative code numerics and physical models that, when completed, will be merged with the consolidated code, either during the consolidation program or upon its conclusion. This paper outlines the consolidation program and elaborates on the specific tasks involved and their coordination.

II. Background

The NRC currently maintains 4 distinct thermal-hydraulic codes including: TRAC-P, TRAC-B, RELAP5 and RAMONA. These codes have similar but not identical capabilities. For pressurized water reactors (PWRs), the RELAP5 code is primarily used for simulation of small break loss-of-coolant accidents (LOCAs) and plant transient analyses and provides a one-dimensional representation of the flow field. RELAP5 lacks the models required for large break LOCA analyses, such as a 3-D vessel component and has not been assessed against data for this particular application. TRAC-P has the capability to model multi-dimensional flows and is therefore utilized for large break LOCA analyses. Generally, RELAP5 was developed as a fast running, more simplistic code for long term transients, whereas TRAC-P provides a more detailed description of the flow field in faster transients. More recently, this distinct separation of functionality has eroded and the present capabilities of the two codes overlap. However, the codes often model the same phenomena with different constitutive packages and also employ different numerical schemes. The reactor physics capabilities of both codes are limited to point kinetics.

For boiling water reactors (BWRs), the situation is comparable. The RAMONA code treats the flow field as one-dimensional but incorporates a three-dimensional (3D) kinetics package. A 3D representation of the flow field is provided by the TRAC-B code, but the neutronics models are limited to either point or 1D kinetics. TRAC-B stemmed from the TRAC-P code and was developed in parallel specifically for the BWR. It incorporates BWR specific models, such as the jet pump, and also utilizes a different constitutive package and numerical scheme than TRAC-P. The development of both TRAC codes has proceeded independently.

The NRC thermal-hydraulic codes were developed in the 1970s and therefore do not take advantage of today's abundant supply of inexpensive, fast memory. In addition, older programming languages did not readily provide a means for dynamic memory allocation. As a result, elegant programming styles (such as "bit packing" and "container arrays") were invoked to overcome these limitations. Unfortunately, these techniques produced cryptic coding and compromised readability, maintainability and portability. Presently, a great deal of effort is vested in deciphering these codes in order to improve the numerical techniques or physical models.

User friendliness is associated with the difficulty one encounters when using the code. This hardship arises from the laborious task of input deck preparation and the equally daunting task of interpreting the output. Since using the code is a complex task, different users will generate different results. This notion is deemed the user effect. One method employed to minimize the user effect and to generally facilitate the use of the code is a graphical users interface (GUI). Traditionally, users have relied on command line input through the terminal as the primary means of interaction with the thermal-hydraulic codes, and therefore each code would benefit by the development of a GUI.

Since each code requires modernization, an improved user interface and would benefit from upgrades in physical models and numerics, NRC plans to consolidate the suite of codes into one with an aim of minimizing the dilution of resources that is dedicated to development and maintenance of the four codes. As a result, user needs will be accommodated more expediently, since effort will not be distributed amongst the four codes. Additionally, the consolidation will enhance analysis capabilities, as the NRC and the user community can focus its attention on one code thereby developing the collective expertise far more efficiently than is possible when four codes are utilized. Input deck construction would not be duplicative, as all transients for a plant design would be performed with one code instead of two.

III. Conclusions

It is evident that the existence of the four separate codes dilutes the NRC resources, as effort must be expended in quadruplicate to support code improvements, assessment, analysis and maintenance. To alleviate this problem, NRC plans to consolidate the codes into a single state of the art code. The consolidated code will incorporate all of the capabilities embodied in the separate codes, so that the user needs will not be compromised. TRAC-P will serve as the basis for the consolidation process, since it contains a 3D hydraulic vessel component, utilizes a network solution procedure, which enhances extendibility (the ability to add components), and is more modular (the database structure and first steps in the solution procedure are component specific). To facilitate this effort and to support long term code improvements, TRAC-P will first be modernized, employing Fortran 90 as the coding language and restructured into a more modular design. Incorporation of the currently available code capabilities will then ensue.

The NRC codes will be maintained for a period during the consolidation process, so that user needs are accommodated. These versions will not be static, but will be aggressively maintained and improved throughout the effort. Concurrently, NRC will work to develop innovative code numerics and physical models that, when completed, will be merged with the consolidated code, either during the consolidation program or upon its conclusion. Additionally, a GUI will be developed to aid in the use of the code and to provide an input deck translator so that previously constructed input decks will not become obsolete.

Once the consolidation is completed, NRC will support a transition period during which the code will be further tested and NRC will provide training for use of the code. NRC will not discontinue support of any code until all capabilities are recovered, their functionality well assessed and users are familiarized with the use of the consolidated code. In summary, the NRC plan consists of both recovering all capabilities of the current suite of codes and making substantial improvements.

NRC Generic Graphical User Interface Development for RELAP5

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The NRC is sponsoring the development of a generic tool to assist in the construction and evaluation of nuclear safety analysis models for thermal-hydraulic analysis codes, with the first application being RELAP5. This code is referred to as SNAP, for Symbolic Nuclear Analysis Program. SNAP is being developed to meet the special requirements of the nuclear safety analyst, whose task is to evaluate complex system models composed of many components.

Creating, verifying, and modifying system models are difficult tasks—but ones that the safety analysis specialist must perform frequently. Typically, models must be coaxed through several restarts to reach the desired problem end time. Users must keep numerous files, to plot and analyze the results. To help the user build, modify, submit, monitor, inspect, analyze, and visualize the results, SNAP contains four tools:

1) A preprocessor to aid in developing and modifying input models.
2) A runtime processor to help with job submittal, initialization, restart and status monitoring.
3) A postprocessor for visual presentation of code results.
4) An interface to a database management system for organizing, searching and storing project, plant, and input model information.

The SNAP preprocessor provides a Graphical User Interface — a GUI — which allows the user to import existing RELAP5 input decks and view the structure. The SNAP interface makes interrelationships (flow paths, heat transfer paths, reactivity feedbacks, and controller loops) between components clear both at a logical and a physical level. An input model can then be modified by manipulating the symbols and icons which represent the components. Input dialogs used to change component parameters are descriptive and indicate where input is needed. SNAP provides visualization and error checking tools that help to locate and correct errors in models earlier in the analysis process. Links to the RELAP5 manuals are also provided on-line.

The SNAP runtime module helps with model submission and plotting of results as a calculation progresses. SNAP also has a postprocessor module which presents the calculations from the simulation model in a visually integrated manner that reflect the way the systems and components interact and work together.

In the future, SNAP will also provide expert guides to assist the user through complex modeling tasks such as problem nodalization, initialization, as well as pump, valve, heat structure, and break component setup. The expert assistants (Wizards) are being developed using the knowledge base of experienced engineers, the user guidelines, and the developmental assessment results.

Intelligent and Intuitive User Interface Approach

There have been several previous GUIs developed for safety analysis codes. Unlike some of these earlier codes, SNAP does more than just turn an existing input deck into a series of graphical dialogs. SNAP dialogs and diagrams typically provide a descriptive text and a units field for each variable. Icons have a recognizable shape and scalable proportion, so that they can represent components as in the familiar "nodalization diagram" format. These icons behave like real components, e.g., users can connect these icons only if RELAP5 allows these components to connect.

Generic Basis For Code-to-Code Model Translation

A unique feature of the SNAP is its capability to translate from one simulation code to another. SNAP stores both generic and code specific model information. Generic information includes geometric and performance data for physical pieces of equipment, such as pumps, pipes, elbows, reducers, tees, valves, tanks, etc. Code specific information is used to supplement the generic information for input data items specific to each code, such as model options and flags. SNAP generic component information is translated into the code specific format at "export" time. In the future, users will be encouraged to enter physical information, and let SNAP translate this generic information to code specific information as much as possible.

An advantage to this design is that whenever a new code is added to SNAP, only translators to and from the new code to the generic mode are needed. Without the "generic" intermediate step, each new code would require the addition of translators to and from each existing code.

SNAP Development Environment

SNAP was developed using an object-oriented design approach. SNAP is written in the C++ language, and uses several freely distributable toolkits, including the standard template library (STL), and wxWindows. The wxWindows toolkit is platform independent, with classes that support a UNIX/Motif and Windows NT implementations of SNAP.

SNAP saves and retrieves stored data from a relational database using Object Database Connectivity (ODBC) drivers, so that most database management systems which support the Structured Query Language (SQL) standard, such as ORACLE, SYBASE, etc., should be acceptable. Current plans call for SNAP to run on all common UNIX platforms. SNAP will be ported to Windows PCs in the future.

Summary

The NRC is sponsoring the development of a generic graphical user interface for simulation codes such as RELAP5. Currently, the details of model construction and submission can be an overwhelmingly complex task, even for experienced analysts. This new code, referred to as SNAP, frees the analyst from these burdens and limitations, so that they can use the increased computational power of the newer generations of computers to greater advantage. Besides ease-of-use, models built or modified using SNAP will be less error prone and have greater fidelity to the user guidelines and RELAP5 developmental assessment basis. The SNAP runtime and postprocessor is an advanced tool for results visualization.

THREE-DIMENSIONAL SPATIAL KINETICS FOR COUPLED THERMAL-HYDRAULIC/NEUTRONICS SYSTEMS ANALYSIS CODES

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I. Introduction

A three-dimensional spatial kinetics capability within a thermal-hydraulic systems analysis code provides an analysis tool that more correctly describes the core physics and is consistent with analysis capabilities used in current and near-term licensing submittals of most nuclear utilities. In both the U.S. and abroad, there has been considerable progress in recent years in the development of coupled thermal-hydraulics/spatial kinetics systems codes. This work has been motivated primarily by the prospects of improved nuclear power economics and reactor safety. Recent initiatives to reduce plant operating costs at U.S. nuclear utilities include further extensions of the fuel cycle length and plans to increase the plant power rating. Last year GPU nuclear used TRAC-PF1 coupled with three-dimensional spatial kinetics to analyze the Main Steam Line Break transient in support of their uprate analysis of the Three Mile Island plant [Ivanov, 1996]. The analysis with point kinetics predicted a return to power during the transient, whereas there was no return to power predicted when the same analysis was performed with three-dimensional spatial kinetics. This provides strong evidence that the analysis of certain reactor safety events with a coupled code can provide increased safety margin through reductions in the conservatisms incurred with lower dimensional kinetics calculations [Diamond, 1996]. Furthmore, some transients, such as BWR flow stabilities, can only be addressed with multi-dimensional spatial kinetics. This presentation will address some of the important issues related to the development of a spatial kinetics capability in a coupled systems code.

II. General Issues

The current state of systems thermal-hydraulic and core neutronics codes represents a long period of code development in the respective areas. These codes generally use methods and structures optimally adapted for its specific purpose. Inevitably, the integration of thermal-hydraulics and spatial kinetics codes will require reconciliation of some important differences. These include the integration strategy, the time synchronization method, and certain other issues that impact that effectiveness of spatial kinetics in a systems code.

Perhaps the most basic issue regarding the implementation of spatial kinetics into a thermal-hydraulics systems code is the integration method which determines whether the core simulator should have thermal-hydraulics separate from the systems code. In an "external" integration the core simulator provides its own TH and the systems and core simulator codes are coupled at the inlet and outlet plenums. Conversely, in an "internal" integration the systems code solves the temperature and fluid fields for both the core and ex-core. There have been implementations of spatial kinetics in systems code using both methods. Perhaps the greatest disadvantage of the external integration is that generally, the thermal-hydraulic models in systems codes are more complete and detailed than those of core simulators. Some limitations on phenomena such as counter current flow or high void content could possibly be encountered in the core simulator and not the systems code.

Another fundamental issue in the coupling of thermal-hydraulics and neutronics is the time synchronization of the field calculations. The neutronics and thermal/hydraulics equations generally have very different time constants and the time advancement of the integrated solution requires careful consideration, since it can have a considerable impact on the overall computational efficiency as well as the solution fidelity of the coupled code. The simplest approach would be to advance the thermal hydraulic and neutronic solutions with the same time step. This could be done explicitly, implicitly, or semi-implicitly. In most current implementations, the TH and neutronic solutions are advanced explicitly with the same time step size. The disadvantage of an explicit approach is that small step sizes can be necessary to insure the adequate fidelity. An implicit or semi-implicit approach would increase the maximum time-step size allowed for certain transients, however, some type of iteration at each time step is generally required which implies additional computational cost. Alternately, the thermal hydraulics and neutronics solutions could be advanced with different time step sizes. There has been little research in this area and some further work is warranted on time synchronization in general and specifically on methods that will increase the "implicitness" of the coupled field solution.

Among the issues related to specific spatial kinetics methods, the most important topic is the treatment of new fuel types with existing advanced nodal methods. During the past few years there has been some concern about limitations of the homogenization techniques and the transverse intergration method which have been the foundation of the current generation of advanced nodal methods. Recent applications such as the use of mixed oxide (MOX) fuel and the use of standard UO_2 fuel at high burnups in LWRs has created conditions where severe flux gradients can occur between a fuel assembly with standard uranium fuel and one with high concentrations of plutonium. These conditions can potentially introduce significant error in the homogenization methods and in the quadratic approximation used to characterize the transverse leakage in the node. This is an area that should be followed closely and warrants additional investigation.

The choice of benchmarks is important for code validation and verification. Numerous computational and experimental benchmarks are available to provide sufficient breadth and depth to the assessment of both the spatial kinetics methods and the coupled reactor systems / spatial kinetics capability. Perhaps one of the most worthwhile for the coupled code is the recently proposed OECD/NEA Steam Line Break benchmark developed by Penn State University [Barratta, 1996]. This problem will include three-dimensional core modeling for both neutronics and thermal-hydraulics and will be a valuable PWR systems transient since the main steam line break event is often a limiting event for recent utility initiatives such as cycle length extension and plant uprate. A comparable benchmark problem for the Boiling Water Reactor would be the OECD BWR Stability benchmark which employs measured data from the Ringhals 1 reactor [Lefvert, 1996].

III. Conclusions

There are strong economic and safety incentives for pursuing the integration of full physics, three-dimensional spatial kinetics in reactor systems codes. While the coupled field calculation is currently computationally feasible, important issues remain concerning the method of integration and the most efficient manner of synchronizing the thermal-hydraulic and neutronics calculations. Furthermore there are some basic issues in spatial kinetics methods that must be addressed in light of the recent use of mixed-oxide and high burnup fuels. Once these issues are addressed the nuclear industry should realize considerable economic and safety benefits from reactor analysis with coupled field codes.

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Interfacial Area Measurement and Transport Equation

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Abstract

The importance of the interfacial area in two-phase flow formulation is explained. The use of the interfacial area transport equation to model the dynamic structure changes in two-phase flow is proposed. This transport equation requires several constitutive relations to model the fluid particle coalescence and disintegration. The local interfacial area and interfacial velocity measurements are reviewed. The use of these measurements for the development of the interfacial constitutive relations is discussed.

Two-phase flow is characterized by the existence of the interface between phases and discontinuities of properties associated with them. Various transfer mechanisms between the mixture and wall as well as between phases strongly depend on the structure of the interface. The basic structure of flow can be characterized by two fundamental geometrical parameters. These are the void fraction and interfacial area concentration. The void fraction expresses the phase distribution whereas the interfacial area describes available area for the interfacial transfer of mass, momentum and energy. Therefore, an accurate knowledge of these parameters is necessary for any two-phase flow analysis. The two-fluid model is formulated by considering each phase separately in terms of two sets of conservation equations which govern the balance of mass, momentum and energy of each phase. These balance equations represent the macroscopic fields of each phase and are obtained from proper averaging methods. Since the macroscopic fields of each phase are not independent of the other phase, the phase interaction terms which couple the transport of mass, momentum and energy of each phase appear in the field equations. In the present state of the arts, the closure relations for these interfacial terms are the weakest link in the two-fluid model. The difficulties arise due to the complicated transfer mechanisms at the interfaces coupled with the motion and geometry of the interfaces.. Furthermore, the closure relations should be modeled by macroscopic variables based on proper averaging.

In general, the interfacial transfer terms are given as a product of the interfacial area concentration a_i and driving force. The area concentration defined as the interfacial area per unit volume of the mixture characterizes the first order geometrical effects; therefore, it must be related to the internal flow-pattern of the two-phase flow field. On the other hand, the driving forces for the interfacial transport characterized the local transport mechanisms such as the turbulence, molecular transport properties and driving potentials. In two-phase flow systems, the void fraction and interfacial area concentration are two of the most important geometrical parameters. The interfacial area concentration should be specified by a closure relations, or by a transport equation. The above formulation indicates that the knowledge of the interfacial area concentration and the interfacial structure classified as the flow regimes are indispensable in the two-fluid model. Various transfer mechanisms between phases depend on the two-phase flow interfacial structures. The geometrical effects of interfacial structure can be modeled in a macroscopic field by the interfacial area and void fraction. In order to take into account the effect of entrance, developing flow, coalescence and disintegration, and wall nucleation source, an introduction of the interfacial area transport equation is recommended. The development of the source and sink terms in the transport equation heavily depends on understanding the mechanisms of particle coalescence and disintegration as well as accurate experimental data for the changes in the interfacial area in two-phase flow. The paper discusses the local measurement methods of interfacial area and the development of the area transport equation.

Aspects of Reflood Heat Transfer Modeling

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With the approval of the Appendix K rule revisions, vendors are starting to utilize "Best-Estimate" safety analysis thermal-hydraulic methods to perform large-break LOCA analysis to evaluate the allowable core thermal limits. Even with the application of best-estimate methods, the large-break LOCA still is the most limiting transient and results in establishing the maximum allowable fuel rod linear power level (kw/ft). Typically what has occurred is that as Best-Estimate analysis methods have identified peak linear heat rate margin; this margin has been used by the utility or the vendor for power up- ratings, longer fuel cycles, low leakage core loadings and advanced fuel designs to improve the economics of the nuclear power plant. All of these economic improvements result in the need for higher operating kw/ft values. This is true for both BWR and PWR designs. When the best-estimate methods are applied with the higher linear heat rates, the resulting calculated peak cladding temperatures are nearly the same as those previously calculated using the original Appendix K requirements. However, the difference is that the allowable linear heat rate, (kw/ft) is now higher.

The best-estimate calculations indicate that for nearly all PWR designs the peak cladding temperatures are reached during the dispersed flow film portion of the reflood transient at low pressures, typically one to three bars. A similar situation also occurs in the hot channel for the more modern BWR designs (BWR5 and 6), as well. The flow pattern in the BWR hot channel is co-current upflow during reflood similar to a PWR.

Dispersed flow film boiling also dominates the downflow period of the PWR blowdown transient as well as the reflood transient. Similar heat transfer mechanisms are present for the blowdown downflow period as well as the reflood period. The primary difference is that the vapor convection term is more dominate for the blowdown situation as compared to the reflood phase, and the vapor has less super heat.

The single largest uncertainty in predicting the dispersed flow film boiling heat transfer in reflooding rod bundles is the liquid entrainment at the top of the transition region just above the quench front. In this region, the steam generation from the quenching of the fuel rods results in a very large vapor velocities which entrain and shear liquid filaments into droplets which are then swept into the upper regions of the rod bundle. The entrained droplets provide cooling by several different mechanisms in the upper regions of the rod bundle where the resulting peak clad temperatures are calculated.

The parameters of interest for the Transition Region include:

- flow regime characteristics, with estimates of the interfacial area
- quench front velocity and transition region liquid and vapor velocities
- entrained drop size, number density, and velocity at the top of the transition region
- vapor temperature
- void fraction
- flow quality
- convective enhancement of spacer grids
- wall temperature
- wall heat flux
- vapor convection to rod surfaces
- droplet shattering effects caused by spacer grids

The parameters of interest for the dispersed flow film boiling region above the transition region include:

- vapor superheat temperature and velocity
- droplet shattering effects of spacer grids
- drop size, velocity, and number density
- spacer grid convective enhancement
- void fraction
- radiation heat transfer to droplets, vapor, and colder surfaces
- spacer grid rewetting
- convective enhancement caused by entrained drops
- vapor convection to rod surfaces
- wall temperatures

Also, since the different mechanism are of comparable magnitude, improving one particular model is difficult since very little data is available to isolate its particular contribution to the total wall heat flux. Therefore, compensating errors can result as the code's predictive capabilities are improved.

The different heat transfer mechanisms which are present in the high temperature portions of the rod bundle where the peak cladding temperatures are calculated represent a combination of a "two-step and three-step " dispersed flow film boiling process. A "two-step" dispersed flow film boiling process consists of heat transfer from the wall to the vapor flow by convection as well as by radiation. There is also wall-to-wall radiation heat transfer and wall to entrained droplet heat transfer. The vapor is the heat sink and quickly reaches superheated conditions as it receives energy from the wall. The second step of the "two-step" process is the heat transfer between the superheated vapor and the entrained droplets. The interfacial heat transfer between the drops and the vapor result in a lower vapor temperature which is the fluid heat sink for the wall heat transfer. The "two-step" film boiling process becomes a "three-step" process as the wall temperature decreases such that there can be intermittent direct (or near direct) droplet-wall contact heat transfer. It is believed that the direct wall contact heat transfer component occurs within and just above the transition region which results in improved heat transfer. The improved heat transfer in this region can be seen from the FLECHT-SEASET test data.

The ability of a best-estimate computer code to accurately predict the integrated effects of the individual phenomena for the dispersed flow film boiling region is a challenge. The uncertainty in the individual models is large and the integrated effects of the uncertainties will accumulated as the calculation progresses upward along the heated channel. The error or uncertainty accumulation is one reason that the code predictions of temperatures above the mid-plane of the FLECHT-SEASET rod bundle or the Japanese Cylindrical Core Test Facility have always been worst than predictions lower in the bundle, closer to the transition region. Also, code calculations for lower flooding rates which are reflected in a slower quench front velocity and a longer transient time, also show poorer predictions relative to the test data.

This paper will describe the different modeling methods and techniques used for reflood heat transfer for PWRs and BWRs. Also, modeling needs for the advanced best-estimate computer codes will be discussed.

BORON MIXING EXPERIMENTS FOR CFD AND SYSTEM CODE ASSESSMENT

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The University of Maryland at College Park (UMCP) 2x4 Loop has been involved in a three year study of boron-dilution related reactivity transients. The experimental program was devised to investigate:

- the formation of a boron-dilute volume of coolant in the primary system – flow from a non-borated interfacing system; boiling-condensing mode coolant accumulation in the steam generator
- the transport of the boron-dilute volume – resumption of natural convection; actuation of reactor coolant pumps
- the mixing of the boron-dilute volume along its path to the core – diffusion; turbulent flow; geometric discontinuity induced

The concomitant review of literature on the subject revealed limitations of current computer codes in simulating phenomena associated with boron-dilute volume reactivity transients. This paper proposes a series of tests that can assess code performance in such transients. These tests are in the context of the current UMCP boron mixing experimental program. The tests employ the integral test facility configuration of the Loop, but are proposed with more controlled initial conditions then would be expected in a prototype. This compromise preserves the test realism, while providing an assessment problem that can be more clearly evaluated.

Proposed Loop Tests

Three tests are proposed: one open test and two closed (blind) tests. The open test consists of a single Once Through Steam Generator (OTSG) configuration (one hot leg, one or two cold legs). The system is solid, and the simulated boron-dilute slug is injected from an external tank. Injection is done with the actual reactor coolant pump.

The single loop is well suited for an open test, in that it is relatively easy to instrument and the boundary conditions can be well defined. The simplest single loop configuration utilizes a single cold leg. Using two cold legs would give some indication of downcomer flow bypass.

The two blind tests both use a dual OTSG configuration (two hot legs, four cold legs). This fully integral configuration allows a detailed investigation of such phenomena as flow bypass, slug mixing, and slug transport (axially and circumferentially) in the downcomer region.

The first blind test will be injection into a solid system, simulating slug formation from an interfacing system leak. The system is filled and conditioned (heated and mixed) prior to test initiation. The slug is injected from an external supply into the bottom of the steam generator. The pumps are used to initiate the slug motion

The second blind test will be injection into a partially voided system, simulating injection during small break LOCA recovery. The system is initially filled to approximately cold leg elevation level. The boron-dilute slug will then be introduced into the bottom of a steam generator (simulating slug formation and HPI refill), and the boron-dilute slug is injected from the steam generator by the reactor coolant pumps, once the level has recovered such that the pumps can be restarted.

Instrumentation

To fully characterize the slug behavior in the Loop, the following instrumentation will be used:

- transport – flow measurement in the cold legs and downcomer
- mixing – temperature measurement in cold legs and downcomer; if possible, conductivity probes at the downcomer inlet and along the core plate
- slug level and volume – DP measurement
- heat losses – temperature measurement in the core and downcomer

For both the open and closed tests, sufficient repetitions would be made in order to provide a quantification of uncertainty in the instrument measurements.

Code Features Tested – Required Data

Both the boron-dilute transport in a solid system (open test and the first blind test) and transport in a partially voided system (second blind test) can be modeled with system or CFD codes A major requirement of both types of codes is well-defined boundary and initial conditions. For system codes the transport and mixing through the entire system will be provided by thermocouple readings along the path of the slug. For CFD codes, a coarse slug flow field in the downcomer will be provided through temperature and, if possible, velocity measurements.

Flow Visualization

In addition to the integral tests in the Loop, a separate flow visualization study will be performed in a transparent replica of the downcomer and lower plenum. These studies will provide detailed measurements for comparison to the code results. Through the use of Laser Induced Fluorescence (LIF) techniques or dye tracer studies, a quantitative measure of the slug evolution can be provided as it exits the active cold leg, proceeds through the downcomer, and enters the reactor core. Laser Doppler Velocimetry (LDV) measurements of the circumferential and axial velocity can be made to provide localized measurements in the downcomer.

Results and Findings of the Seismic Analysis Research Program Relative to 1995 ASME Design Rules

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This paper describes the U.S. Nuclear Regulatory Commission (NRC) Seismic Analysis of Piping research program for reviewing the technical bases for the ASME Code seismic design criteria for nuclear plant piping as revised by the Code 1994 Addenda. This program was established in 1993 and is being conducted by the Energy Technology Engineering Center for the NRC.

The paper describes reviews and assessments performed as part of the program, including reviews of previous evaluations performed in support of the revised design criteria and independent assessments of margins demonstrated in the previous joint Electric Power Research Institute (EPRI)/NRC Piping and Fitting Dynamic Reliability (PFDR) program.

In addition, this paper presents an analytical study of frequency effects on seismic margins for nuclear power plant piping systems. The study is based on results of seismic testing of piping components conducted as part of a previous joint EPRI/USNRC program. Analysis techniques involving nonlinear modeling and identification of the test data are described, and analytical fatigue margins are presented. It is observed that the margins provided by current. ASME Code provisions are low for stiff piping systems. Furthermore, the results suggest the possibility of large deformation response for highly flexible yielding systems considering the presence of P-Δ effects. A ductility based margin is presented as a means of describing large-deformation induced failure modes which are not being fully addressed by the fatigue based margin.

Finally, a technical basis for establishing appropriate seismic margins in nuclear plant piping is presented. The establishment of the margins was based on considerations of current seismic PRA evaluation methods. Margins were developed such that piping will not control the plant High-Confidence, Low-Probability-of-Failure (HLCLPF) seismic capacity, and consequently the current seismic PRA methodology can be maintained.

Overall findings from the research project are summarized along with the pertinent peer review group comments (As a part of this project, a peer review group consisting of experts in piping, structural dynamics, seismic engineering, and materials engineering has been constituted to review progress and provide independent comments)

Preliminary Results of the Steel Containment Vessel Model Test

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The Nuclear Power Engineering Corporation (NUPEC) of Japan and the U.S. Nuclear Regulatory Commission (NRC) have been co-sponsoring and jointly funding a Cooperative Containment Research Program at Sandia National Laboratories. The purpose of the program is to investigate the response of representative models of nuclear containment structures to pressure loading beyond the design basis accident and to compare analytical predictions with measured behavior. This is accomplished by conducting static, pneumatic over pressurization tests of scale models at ambient temperature. One of the tests was a test of a mixed scaled model with 1:10 in geometry and 1:4 in shell thickness of a steel containment vessel (SCV), representing an improved boiling water reactor (BWR) Mark II containment.

This paper describes the preliminary results of the high pressure test of the SCV model. The preliminary post-test measurement data for the gap between the SCV model and the contact structure and the preliminary comparison of test data with pretest analysis predictions are also included.

The SCV model test was intended to accomplish the following specific test objectives:

1) to provide experimental data for checking the predictive capabilities of analytical methods to represent some aspects of the static internal pressure response of a steel containment,

    a) beyond the elastic range, without consideration of contact with a surrounding shield structure or thermal effects, and

    b) after contact with a surrounding shield structure,

2) to investigate the failure mode of the SCV model, and

3) to provide experimental data useful for the evaluation of actual steel containments.

To meet these objectives, the high pressure test was conducted using a monotonic pressure rise. The high pressure test of the SCV model was conducted on December 11-12, 1996 at Sandia National Laboratories. After approximately sixteen and a half hours of continuous monotonic increase in pressure achieved by pumping nitrogen gas into the SCV model, the test was terminated when the pressure in the model dropped very rapidly, even with the pressurization system operating at its maximum flow rate of 1300 standard cubic feet per minute. The cause of failure of the SCV model was a tear resulting in leakage; the failure mode was not catastrophic. The maximum internal pressure achieved during the test was 4.66 MPa, which is 5.97 times the design pressure.

Seismic Tests
of a Prestressed Concrete Containment Vessel

Part 1 - Test Program

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Since 1980 a series of seismic proving tests of nuclear power facilities have been carried out by the Nuclear Power Engineering Corporation (NUPEC), using the large-scale, high-performance shaking table at the Tadotsu Engineering Laboratory. The tests are sponsored by the Ministry of International Trade and Industry (MITI) of Japan.

As a part of the overall program, NUPEC has started seismic proving tests of a prestressed concrete containment vessel (PCCV) used for PWR type nuclear power plants (NPPs) and a reinforced concrete containment vessel (RCCV) used for ABWR-type NPPs since 1992.

The objectives of the tests are to prove the structural and functional integrity and to grasp the seismic margin of CCVs subjected to severe earthquakes. These shaking tests are the first large-scale ones for CCVs with liner installed on the inner surface of the concrete cylinder.

This report focuses on the PCCV test and describes the modeling concept, model geometry, test procedures, pretest analysis and so on.

Due to the objectives and constructability of linear system and cylinder wall, the harmonized scale was adopted to meet all the requirements.

The shaking test of the PCCV was completed this June, and the processing of test data and its evaluation is under the way.

Seismic Test of a Prestressed Concrete Containment Vessel

Part 2 - Analytical Investigations

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In a cooperative program between the United States Nuclear Regulatory Commission (USNRC) and Nuclear Power Corporation (NUPEC) of Japan, USNRC is conducting pre and post-test analyses of the prestressed concrete containment vessel (PCCV) model being tested by NUPEC at the Tadotsu Engineering Laboratory. Pretest analyses have been conducted for the PCCV model using a highly sophisticated concrete constitutive model in a general three-dimensional finite element time history analysis procedure. The analytical modeling accounted for structural damage mechanisms such as tensile and shear cracking in the concrete, concrete crushing, and steel yielding. The mode of failure of the PCCV model exhibited unique features which reflect the particular design of the model. The lessons learned from both the test results and the analytical studies will contribute significantly to our understanding of the behavior of concrete containment structures subjected to earthquake motions. Insights derived from the studies are presented.

Guidelines for Probabilistic Seismic Hazard Assessments and a Trial Application

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Under the sponsorship of the Nuclear Regulatory Commission, the Department of Energy (DOE), and with contribution by the Electric Power Research Institute, a panel of scientists was convened to perform a study of probabilistic seismic hazard assessment (PSHA) methodologies. The panel was named the Senior Seismic Hazard Analysis Committee (SSHAC); its guidelines were published as NUREG/CR-6372, which is popularly referred to as the SSHAC report.

The SSHAC was tasked with developing an improved methodology that would be useable for regulatory applications for about the next decade for both regional and site-specific analyses. In evaluating existing methodologies and general principles, they found that most of the problems in past PSHA applications were caused by flawed expert elicitation and procedural aspects of the methodologies used. Therefore, the SSHAC concentrated on procedural guidance for PSHA and on rigorous treatment of uncertainties. Where necessary, the SSHAC also provided guidance for the subjects of seismic source characterization and ground motion estimation.

Their overall conclusion is that there are important pitfalls in using experts effectively, and that the key task is technical integration. Depending on technical complexities and regulatory significance, the study is led by either a Technical Integrator (TI) or a Technical Facilitator/Integrator (TFI) who is responsible for the results of the PSHA. The TI is commonly used for less complex tasks, such as a site-specific study for a bridge or other project. The TFI is employed for more complex regional studies or for investigations related to a critical facility, such as a nuclear power plant. The TFI would commonly consist of two or three individuals with the requisite range of experience in earth sciences and expert elicitation. The TFI evaluates a range of hypotheses and models presented by the experts, and arrives at a representation of the knowledge of the group and of the scientific community at large. The expert elicitation depends heavily on group interaction and structured workshops where available facts are presented. The aim of the TFI process is to develop as much of a consensus as possible; however, where that goal is not reached and where there may be "outlier" opinions, it is up to the TFI to formulate the most consistent result, including behavioral aggregation involving qualitative judgment.

With respect to uncertainties in seismic hazard assessment, the SSHAC adopted a rigorous treatment based on a distinction between epistemic and aleatory uncertainties. Epistemic uncertainties are based on a lack of scientific understanding that may be reduced in the future. Aleatory or "random" uncertainties cannot be reduced for all practical purposes.
These terms were chosen to avoid multiple meanings associated with words such as "uncertainty" for epistemic. Further characteristics of the SSHAC methodology involve careful documentation

of the PSHA process and of the data and models used. Also required is adequate peer review in both the TI and TFI processes, including technical and process peer review. In the course of their work, the SSHAC held several workshops that served to refine the guidelines and prove their efficiency.

Two of the most significant aspects of the new guidelines provided by SSHAC are the TFI concept and a departure from relying on inflexible aggregation schemes, such as a priori equal weights. The guidelines were reviewed by a committee of the National Academy of Sciences (NAS) and given generally positive comments. The review committee, in particular, agreed with and further emphasized the principle of not relying on mechanical aggregation schemes.

An approximately two year study by Lawrence Livermore National Laboratory (LLNL) is nearing completion. The study had the goal of testing and implementing the SSHAC guidelines for the specific case of the southeastern United States and of two nuclear plant sites in that region, namely Vogtle and Watts Bar. Workshops and expert elicitations were held in accordance with SSHAC principles, with emphasis on seismic source characterization, because ground motion elicitations were already tested during the SSHAC workshops. This project has shown that the TFI procedures can lead to an unusual degree of agreement among experts through thorough discussion of the available data, and through interaction between the experts. Together with the focusing effect of the TFI, this leads to narrower margins of variation without any coercion. For the southeastern U.S. this lead to an integrated map of source zones that incorporated the opinions of all the experts involved, even though they began with fairly different source zone maps. This is in stark contrast to the previous situation, where each expert produced a series of map interpretations, leading to a large number of source zone maps, most of which were totally different from each other.

The process used for the southeastern U.S. source map eliminated several variations in source zones, because different experts were able to agree on a compromise solution that would not significantly change the final hazard. In some cases, such as near the Watts Bar plant, where a change in zone boundaries can change the site hazard substantially, differing opinions were incorporated by using three versions of a source zone boundary. Each zone boundary variant was assigned a probability, thus incorporating the range of expert opinions. The only difficulty that has remained in this and other projects using the SSHAC guidelines is the treatment of uncertainties. Many of the experts still have problems following a rigorous distinction between various epistemic and aleatory uncertainties.

The NAS review panel pointed out that the SSHAC guidelines should be applicable to a wider range of subjects than just seismic hazard. Indeed, the DOE has already used the guidelines for an investigation of volcanic hazards at the Yucca Mountain waste repository site. The DOE intends to apply them to other subjects as well, such as characterization of hydrology at that site. Overall, it is generally acknowledged that the guidelines represent a substantial advance in PSHA, and that they are useful for a variety of applications that depend on expert opinions.

Technical Basis for Concrete Anchorage Criteria

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Since the issuance of Inspection and Enforcement Bulletin (IEB) 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," and the resolution of Unresolved Safety Issue A-46, "Seismic Qualification of Equipment in Operating Nuclear Power Plants," a significant amount of anchorage test data have been published. Data has been generated in the U.S. by researchers, anchor manufacturers, architectural & engineering firms, and regulatory bodies to answer regulatory inquires and /or resulting from the introduction of new anchorage systems.

Experience has shown that the design or installation of anchorage for equipment can be a weak link in the seismic performance of equipment. Thus to assure that equipment (motors, pumps, electrical cabinets, etc.) is seismically adequate for the safe shutdown of a nuclear plant during an earthquake, seismic verification of equipment anchorage is necessary. There is also a need for anchorage criteria that is consistent and that provides adequate regulatory guidance. Discussed in this paper is the technical basis for anchorage criteria that can be used for operating reactors, license renewal efforts and advanced reactor designs.

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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

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10. SUPPLEMENTARY NOTES

C. Bonsby, NRC Project Manager; Transactions prepared by Brookhaven National Laboratory

11. ABSTRACT *(200 words or less)*

This report contains summaries of papers on reactor safety research to be presented at the 25th Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel in Bethesda, Maryland, October 20-22, 1997. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, U.S. NRC. Summaries of invited papers concerning nuclear safety issues from U.S. government laboratories, the electric utilities, the nuclear industry, and from foreign governments and industry are also included. The summaries have been compiled in one report to provide a basis for meaningful discussion of information exchanged during the course of the meeting, and are given in order of their presentation in each session.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	
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TWENTY-FIFTH WATER REACTOR SAFETY INFORMATION MEETING

SESSION SCHEDULE

Monday 8:30 am	**PLENARY SESSION** *(Congressional Ballroom)*	
	Grand Ballroom B&C	***Grand Ballroom D&E***
10:00 am to Noon	1 PRESSURE VESSEL RESEARCH E. Hackett	2 BWR STRAINER BLOCKAGE AND OTHER GSIs A. Serkiz
1:30 pm to 5:00 pm	3 ENVIRONMENTALLY ASSISTED DEGRADATION OF LWR COMPONENTS A. Lund	4 UPDATE ON SEVERE ACCIDENT CODE IMPROVEMENTS AND APPLICATIONS C Tinkler
Tuesday 8:30 - 9:45 am	**PLENARY SESSION** *(Grand Ballrooms B-C-D-E)* **Risk Informed Regulation**	
10:00 am to Noon	5 HUMAN RELIABILITY ANALYSIS & HUMAN PERFORMANCE EVALUATION A. Ramey-Smith	6 TECHNICAL ISSUES RELATED TO RULEMAKINGS J. Murphy
1:30 pm to 5:00 pm	7 RISK-INFORMED, PERFORMANCE-BASED INITIATIVES M Cunningham	8 HIGH BURN-UP FUEL RESEARCH R Meyer
Wednesday 8:30 am to Noon	9 THERMAL HYDRAULIC RESEARCH & CODES I J. Uhle	10 DIGITAL INSTRUMENTATION AND CONTROL J Calvert
1:30 pm to 5:00 pm	11 THERMAL HYDRAULIC RESEARCH & CODES II J. Uhle	12 STRUCTURAL PERFORMANCE A Murphy

All Lunches and Monday's Reception are in the Congressional Ballroom
Breaks are in the "Bethesda Room"

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