NUREG/CP-0165

Transactions of the Twenty-Sixth Water Reactor Safety Information Meeting

To Be Held at
Bethesda Marriott Hotel
Bethesda, Maryland
October 26–28, 1998

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 26th Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel in Bethesda, Maryland, October 26-28, 1998. 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.

An asterisk [*] in place of a page number in the Contents indicates summary not submitted in time for inclusion in this report.

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

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Evaluation of Margins in the ASME Rules for Defining the P-T Curve for an RPV

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The pressure-temperature (P-T) curve defines the upper bound to the permissible operating envelope for a reactor pressure vessel (RPV) during the normal start-up and cool-down transients. As irradiation-induced embrittlement of the RPV material accumulates during the operating life of the RPV the P-T curve must be recalculated. The result of this recalculation is a progressive tightening of the P-T operating envelope. In recent years a number of electric utility companies with nuclear plants have reported that the available P-T operating envelope has become so restricted that operation of the reactor during the heat-up and cool-down transients has become very difficult. This development has prompted an evaluation of the inherent margins in the current ASME P-T curve rules to determine if they can be modified so as to increase the available P-T operating envelope while assuring the retention of adequate safety factors. Rules for defining the P-T curve are given in appendix G to section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Those rules require the use of a 1/4-thickness flaw and consider stress intensity factor (SIF) only at the deepest point of the flaw. Acceptable SIFs are conservatively defined using a lower bound to the dynamic fracture toughness database (KIR). Loading conditions considered in the ASME P-T curve rules are limited to pressure loading and loads from transient-induced through-the-wall temperature gradients. A safety factor (SF) of 2.0 is applied to the SIF generated by pressure loading. This SF is intended to compensate for loading conditions not included in the P-T curve analysis rules. Loading conditions in this category include residual stresses in the structural welds, stresses produced by clad/base-material differential thermal expansion, and stresses produced by pressure on the crack face. Assessment of margins in the current ASME P-T curve rules is in progress in the NRC-funded Heavy Section Steel Technology (HSST) program at Oak Ridge National Laboratory. Results from this assessment are expressed as the ratio of the median estimate allowable pressure (PM) divided by the allowable pressure determined using the current ASME Code rules (PCODE). The margin assessments are based on a median fit to the EPRI KIc data base and incorporate results from research conducted within the HSST program. Research results included in the margin assessment include the influence of local brittle zones on the slow-loading fracture toughness of RPV steels (KIc), use of experimentally determined residual stress values for the RPV structural welds and the cladding, and use of the FAVOR computer program to search the portion of the periphery crack front located in ferritic material and identify the most limiting SIF to the KIc fracture toughness ratio. The SF of 2 on pressure loading is eliminated in the calculation of PM because all known factors, which could influence the governing SIF ratio, are included in the PM analysis model. A margin assessment (PM/PCODE) was generated for a reference semi-elliptical surface flaw with a 6:1 aspect ratio and a depth of 1.0 inch.

TECHNICAL BASIS FOR REVISED P-T LIMIT CURVES IN SECTION XI

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Westinghouse Electric

The startup and shutdown process for an operating nuclear plant is controlled by pressure-temperature limit curves, which are developed based on fracture mechanics analysis. These limits are developed in Appendix G of Section XI, and incorporate four specific safety margins:

1. Large flaw, ¼ thickness
2. Safety factor = 2 on pressure stress
3. Lower bound fracture toughness
4. Upper bound adjusted reference temperature (RT_{NDT})

There are two lower bound fracture toughness curves available in Section XI, K_{IA}, which is a lower bound on all static, dynamic and arrest fracture toughness, and K_{IC}, which is a lower bound on static fracture toughness only. The only change involved in this action is to change the fracture toughness curve used for development of P-T limit curves from K_{IA} to K_{IC}. The other margins involved with the process remain unchanged. There are a number of reasons why the limiting toughness in the Appendix G pressure-temperature limits should be changed from K_{IA} to K_{IC}.

Use of K_{IC} is More Technically Correct

The heat-up and cool-down process is a very slow one, with the fastest rate allowed being 100° per hour. The rate of change of pressure and temperature is often constant, so the stress is essentially constant in this case. This is clearly a static process, but it should be emphasized that all operating transients (levels A, B, C and D) correspond to static loadings, with regard to fracture toughness.

The only time when dynamic loading can occur and where the dynamic/arrest toughness K_{IA} should be used for the reactor pressure vessel is when a crack is running. This might happen during a PTS transient event. Therefore, use of the static toughness K_{IC} lower bound toughness would be more technically correct for development of P-T limit curves.

Use of Historically Large Margin No Longer Necessary

In 1974, when the Appendix G methodology was first codified, the use of K_{IA} (K_{IR} in the terminology of the time) to provide additional margin was thought to be necessary to cover the uncertainties and a number of postulated but unquantified effects. Almost 25 years later, significantly more is known about these uncertainties and effects. For example, the flaw size distribution has been measured for prototypical vessel welds, the fracture toughness database has increased by more than an order of magnitude, and both K_{IA} and K_{IC} remain lower bound curves[1], as with the original database[2]. Therefore, additional margin to account for these uncertainties and effects is no longer necessary. This is especially true when the unneeded margin may in fact reduce overall plant safety.

Overall Plant Safety is Improved

The primary reason for making this change is to reduce the excess conservatism in the current Appendix G approach that could in fact reduce overall plant safety. By opening up the operating window relative to the pump seal requirements, the chances of damaging the seals and initiating a small LOCA, a potential pressurized thermal shock (PTS) initiator, are reduced. Moreover, excessive shielding to provide an acceptable operating window with the current requirements can result in higher fuel peaking and less margin to fuel damage during an accident condition.

Reactor Vessel Fracture Likelihood is Very Low

It has long been known that the P-T limit curve methodology is very conservative[3,4]. Changing the reference toughness to K_{IC} will maintain a very high margin. The full paper will show a series of P-T curves developed for the same plant, but with different assumptions concerning flaw size, safety margin and fracture toughness.

Summary and Conclusions

Technology developed over the last 25 years has provided a strong basis for revising the ASME Section XI pressure-temperature limit curve methodology. The safety margin which exists with the revised methodology is very large, whether considered deterministically or from the standpoint of risk.

Changing the methodology will result in an increase in the safety of operating plants, as the likelihood of pump seal failures and/or fuel problems will decrease.

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Fabrication Flaws in Reactor Pressure Vessels

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Pacific Northwest National Laboratory (PNNL) under contract to NRC has performed nondestructive and destructive examinations of welds taken from reactor pressure vessels, which were manufactured for cancelled nuclear power plants. One such vessel is the Pressure Vessel User Research Facility (PVRUF) vessel, which was located at the Oak Ridge National Laboratory (ORNL). The objective has been to determine the numbers, locations and sizes of flaws in the vessel welds, and to develop empirical estimates of fabrication flaw rates for use in fracture mechanics structural assessments. Simulations were performed with RR-PRODIGAL to predict the numbers and sizes of flaws in the welds. The objective of these calculations was 1) to provide a basis for generalizing the observed flaw rates and for estimating rates for flaws having depths significantly greater than those observed, and 2) to validate the computer modeling code by comparing predicted and observed flaw distributions.

The PVRUF vessel was assembled by Combustion Engineering in the late 1970's and early 80's for a nuclear power plant that was not completed. The vessel has a diameter of 4.39-meter (173-inch), a height of approximately 13.34-meter (525-inch), and is made out of A533B steel. The wall thickness varies from one region to the next, but within 25-cm (10-inch) of the beltline welds it is 22-cm (8.6-inch) thick.

Examination of the welds were performed to determine the density and size distributions for the fabrication flaws within the welds. The very sensitive SAFT-UT (Synthetic Aperture Focusing Technique for Ultrasonic Examination) was used to detect and characterize flaws. Data were obtained from three types of examinations:

SAFT-UT Examinations Performed Onsite - PNNL staff moved the SAFT-UT system to ORNL, trained ORNL staff in operations. ORNL staff examined 100 percent of the beltline region welds with access from the vessel inner surface and 50% of the intermediate to upper shell course. The data collected at ORNL were then sent to PNNL for detailed analyses to identify all zones that may contain larger fabrication flaws.

SAFT-UT Examinations Performed at PNNL - Sections of welds that contained all potentially large flaws were remove from the PVRUF vessel and shipped to PNNL for detailed examinations. With access from the various sectioned surfaces of the segments it was possible to achieve an enhanced level of sensitivity and resolution. In a number of cases some flaws which had been conservatively characterized by the preliminary examinations to be a single large flaw were more accurately determined to be several closely spaced smaller flaws.

Destructive and Radiographic Examinations - Small cubes containing the larger indications were cut from the weld segments. The sizes and locations of the defects within the cubes were confirmed by radiography. The final step was to section some of the cubes to obtain metallographic confirmation of selected defects.

The total number of flaws was about 2500 with most of these flaws having through-wall dimensions of less than 3-mm. All of the larger flaws were confirmed and characterized on the basis of the detailed SAFT-UT, radiographic examination, and destructive examinations. The sizes of the flaws in the material which remained at Oak Ridge were reestablished by using sizing rules adjusted on the basis of experience gained from the enhanced SAFT-UT, RT and destructive examinations.
Figure 1 shows observed flaw size distributions from PVRUF vessel examinations. The plot includes flaws in the original weld metal along with some relatively large and significant flaws, which were detected in regions of repair welding.

The measured data show a much larger number of flaws of very small size than predicted by the RR-PRODIGAL simulations. This was not a significant concern, because these flaws are too small to be important to structural integrity, and were also below the size domain addressed by the RR-PRODIGAL model (i.e. flaw depths much less than the weld bead dimension are excluded). The observed and predicted flaw depth distributions come into relatively good agreement when the flaw depths approach 5-mm (roughly the radial dimension of the weld bead).

Differences between the two rates are believed to be within the level of accuracy associated with uncertainties in the predictive model and to be consistent with the expected random vessel-to-vessel variations in flaw occurrence rates. This agreement enhances the value of the PVRUF data because the model calculations are based on an extensive body of knowledge regarding welding flaws and the factors that contribute to their occurrence. The fact that flaw rates for the PVRUF welds agree with expected trends as predicted by the RR-PRODIGAL code indicates that the PVRUF vessel is representative of a larger population of vessels relative to flaw occurrence rates. The agreement between the model and data also enhances the level of confidence in the ability of the RR-PRODIGAL model to predict the numbers and dimensions of large through wall sizes of flaws in vessel welds.

Figure 1 also compares the PVRUF data with estimates provided by the Marshall flaw distribution. An assumed flaw density of 400 flaws per cubic meter in combination with the "with inspections" version of the conditional size distribution provides a reasonable approximation to the PVRUF data over the depth range of 5-20 mm.

Future work will measure flaw occurrence rates for welds removed from other unused reactor vessels with a focus on BWR vessels (River Bend, Shoreham and Hope Creek).

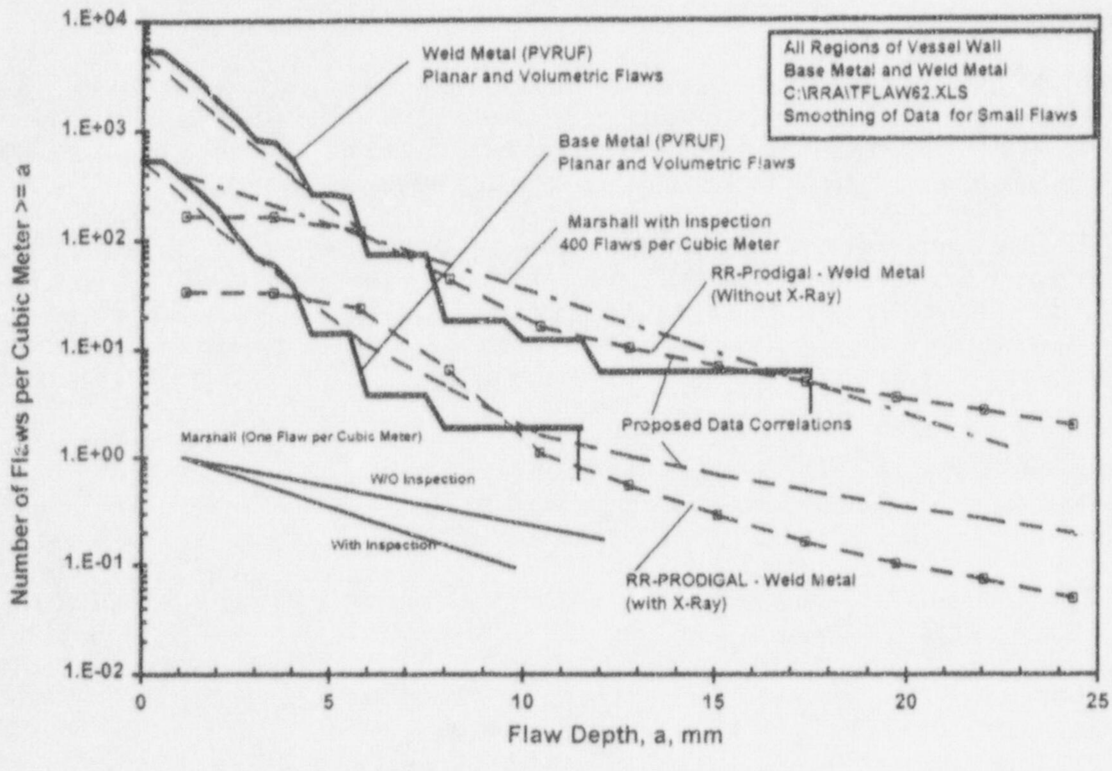

Figure 1 Observed Versus Predicted Flaws for Welds of PVRUF Vessel

NDE of Shoreham Reactor Pressure Vessel for Flaw Distribution Development

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Abstract

A joint program between EPRI and the US Nuclear Regulatory Commission (NRC) is underway to determine the distribution (size and number density) of fabrication defects in reactor pressure vessel (RPV) welds. A series of nondestructive examinations (ultrasonic - UT) are being performed on material removed from the decommissioned, unirradiated, Shoreham RPV. EPRI-sponsored UT inspections are being performed at the EPRI Nondestructive Evaluation (NDE) Center. Results obtained from the ultrasonic examinations will be used to determine appropriate areas for further evaluation through destructive sectioning and metallographic analysis. The destructive sectioning will be performed to characterize the defect type and determine the defect distribution for the material evaluated. In addition, it is anticipated that results from the NDE and destructive analyses will provide information regarding ultrasonic inspection performance and reliability for consideration in RPV operating criteria. The inspection activities being performed at the EPRI NDE Center will be discussed.

Revised Source Term Rebaselining for Operating Reactors

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The NRC's reactor site criteria, 10 CFR Part 100, require that a fission product release into containment be postulated and that offsite radiological consequences be evaluated against the guideline dose values given in Part 100. Other Commission regulations, in 10 CFR Part 50, GDC 19, address regulatory requirements on accident radiological doses for the control room. The evaluation of the release of fission products into containment (called "source term") is used for judging the acceptability of both the plant site and the effectiveness of engineered safety features. The original source term, which was based on releases from a severely damaged core, was published in 1962 by the U.S. Atomic Energy Commission in Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactors." Since that time there have been significant advances in our understanding of the timing, magnitude and chemical forms of the fission product release from severe reactor accidents. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," was published in February 1995, and reflects that extensive research and experience culminating in the development of a new or revised source term. The development of the revised source term was originally intended for initial application to advanced reactors though it was recognized that current reactors may want to utilize the revised source term in licensing actions. The impetus for operating reactors to adopt the revised source term is that through its more realistic characterization of the source term, plants may modify existing restrictive plant features, (e.g., component actuation times, leakage control systems).

The objective of the re-baselining effort was to develop a better understanding of the impacts of implementing the revised source term (NUREG-1465) at operating reactors. The major areas examined were:

- i) effect on calculation of individual offsite and control room dose
- ii) effect on calculation of dose for equipment qualification
- iii) effect of potential plant modifications, including assessment of changes to plant risk

It has been recognized since the development of the revised source term that changes in the prescription of the source term from that originally described in TID-14844 would influence the major areas of dose analysis and could prompt plant, technical specification and procedure modifications. The most significant changes in the source term are the treatment of the fission product release as a time dependent process and the release of radio-iodine primarily as an aerosol. In the revised source term the fission product release is assumed to occur over roughly two hours as opposed to the TID source term which assumed the release of the entire source term occurs instantaneously at the beginning of the accident. In addition, in the revised source term, 95% of the radio-iodine is assumed to be released as an aerosol, CsI, with the remaining 5% as a combination of inorganic and organic vapors. This is in contrast to the original TID source term which prescribed the opposite ratio, 95% of the iodine as vapor and 5% as aerosol. Therefore, plant systems originally provided to mitigate an instantaneous source term by very rapid actuation would not be required to perform under such stringent requirements with the revised source term. Likewise, systems needed to remove iodine vapors are less important under conditions where iodine is an aerosol. As part of rebaselining these issues and a number of other more subtle differences between the source terms, as well as the impact of improved modeling of fission product processes, were explored to position the staff for review of operating plants use of the revised source term and rulemaking. Rebaselining was the vehicle for assessment of the likely dominant issues associated with implementation of the revised source term as revealed by analysis of representative plants (Surry, Zion, and Grand Gulf) and as such provides the technical basis for development of regulatory guide criteria.

The overall impact of implementing the revised source term in the majority of cases is to produce lower calculated doses, ranging from a slight reduction up to an order of magnitude decrease, for an individual, whether for the Exclusion Area Boundary, Low Population Zone, or control room. Another finding from the rebaselining activity was that the time dependent release of the new source term, coupled with plant characteristics, can result in a substantial shift in the 2 hour interval associated with the maximum dose. In Grand Gulf LOCA dose analyses, the worst two hour interval for the total effective dose equivalent (TEDE) of 6.8 rem began at 2.2 hrs. By comparison the calculated dose for the first two hours was 2.0 rem.

An evaluation of the equipment qualification dose using the TID and NUREG-1465 source terms revealed that the containment atmosphere dose using the revised source term was similar. With respect to doses for equipment exposed to containment sump water, the revised source term again produces similar results. However, the revised source term produces somewhat higher doses later in time due to a much higher inventory of cesium.

Analyses were also performed with the MELCOR severe accident code both to evaluate the extent of safety margins maintained with the revised source term calculations and to evaluate related thermal hydraulic issues. The MELCOR analyses indicated that the DBA dose calculations still have substantial margin (a factor of 2 or greater) even though the DBA dose with NUREG-1465 may be well below the earlier TID analysis. Finally, an evaluation of potential plant changes was undertaken with the objective of assessing the impact on the DBA dose calculation and the impacts on plant risk. The general conclusion from these studies was that indeed many of the types of changes proposed could be made and the DBA dose would remain within acceptance limits, though the plant specific changes will need to be reviewed.

Having concluded the rebaselining initiative, the NRC did not identify any issues that would prevent implementation of the revised source term at operating reactors. Further, the rebaselining activities have provided a technical basis to support rulemaking and changes to associated regulatory guides. The NRC is therefore proceeding to commence rulemaking. The NRC is also proceeding with the evaluation of implementation of the revised source term for the reactors that have submitted applications.

Recent MELCOR and VICTORIA Fission Product Research at the NRC[1]

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The MELCOR and VICTORIA severe accident analysis codes, which were developed at Sandia National Laboratories for the US Nuclear Regulatory Commission, are designed to estimate fission product releases during nuclear reactor accidents in light water reactors. MELCOR is an integrated plant-assessment code that models the key phenomena in adequate detail for risk-assessment purposes. VICTORIA is a more specialized fission-product code that provides detailed modeling of chemical reactions and aerosol processes under the high-temperature conditions encountered in the reactor coolant system during a severe reactor accident. This presentation focuses on recent enhancements and assessments of the two codes in the area of fission product chemistry modeling.

Recently, a model for iodine chemistry in aqueous pools in the containment building has been incorporated into the MELCOR code. The model calculates dissolution of iodine into the pool and releases of organic and inorganic iodine vapors from the pool into the containment atmosphere. The main purpose of this model is to evaluate the effect of long-term revolatilization of dissolved iodine. Inputs to the model include dose rate in the pool, the amount of chloride-containing polymer, such as Hypalon, and the amount of buffering agents in the containment. Model predictions are compared against the RTF experiments conducted by AECL, specifically International Standard Problem 41.

1This work was supported by the United States Nuclear Regulatory Commission and was performed at Sandia National Laboratories, which is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the United States Department of Energy under Contract DE-AC04-94AL85000.

Improvements to VICTORIA's chemical reactions models were implemented as a result of recommendations from a peer review of VICTORIA that was completed last year. Specifically, an option is now included to model aerosols and deposited fission products as three condensed phases in addition to the original option of a single condensed phase. The three-condensed-phase model results in somewhat higher predicted fission product volatilities than does the single-condensed-phase model. Modeling of UO_2 thermochemistry was also improved, and results in better prediction of vaporization of uranium from fuel, which can react with released fission products to affect their volatility. This model also improves the prediction of fission product release rates from fuel.

Finally, recent comparisons of MELCOR and VICTORIA with International Standard Problem 40 (STORM) data are presented. These comparisons focus on predicted thermophoretic deposition, which is the dominant deposition mechanism. Sensitivity studies were performed with the codes to examine experimental and modeling uncertainties.

THE PHEBUS F-P INTERNATIONAL RESEARCH PROGRAM ON SEVERE ACCIDENTS: STATUS AND MAIN FINDINGS

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ABSTRACT

The international Phebus F.P. program[1] was launched in the late 80s to investigate key phenomena involved in LWR severe accident: core melt progression up to, and including, the late phase, the subsequent release of fission products and of structural materials, their transport in the cooling system and their deposition in the containment with a special emphasis on iodine volatility. The experiments are carried out in the Phebus facility located at Cadarache, France, by the French Institut de Protection et de Sûreté Nucléaire. In contrast to past experimental programs performed in this area, Phebus F.P. uses a significant amount of prototypical fuel (10 kg) and a scaled reactor circuit and containment model, under realistic thermal-hydraulics and physico-chemical conditions.

Two tests were successfully carried out in December 1993 and July 1996, using respectively trace irradiated fuel (FPT-0) and medium burn-up fuel irradiated in the Belgium BR3 reactor (FPT-1). The experimental conditions for both tests were rather similar, simulating a cold leg break under a low pressure, steam rich, environment. They have both reached a state of bundle degradation never achieved so far in any previous in-pile experiment, with respectively 50% and 30% of fuel relocated and the formation of molten pools. Subsequently, large fractions of the volatile fission product inventory, but also significant amounts of structural materials and a few percents of the low-volatile fission product inventory have been released and transported to the containment model.

Only a few data on the fission product chemistry were obtained from the first test, owing mainly to the very low amount of fission products present in this first test. Nevertheless, specific instruments, as thermal gradient tube, or combinations of indirect measurements, enable to infer certain conclusions with respect to their volatility. Instrumentation and test procedures were considerably improved for FPT-1. Data evaluation of the second test, in particular post-test analyses of samples, is still in progress at the time of writing. However, from the results already available, a consistent set of observations can be derived from both tests on bundle degradation and fission product behaviour, with indications on burn-up effects.

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[1] M. SCHWARZ, P. von der HARDT, The Severe Accident Research Programme PHEBUS F.P. : First Results and Future Tests, 23rd Water Reactor Safety Information Meeting, p 239-261 - Bethesda, Maryland - October 23-25, 1995

Certain phenomena, observed in both experiments, as the low temperature at which core melt progresses or the presence of gaseous iodine in the cold leg, could not be reconstituted by standard code calculations and point at weaknesses in the computer models. A wide-word cooperative effort is in progress to interpret these results and improve code modelling, with the help of specific separate effects tests in the following areas: high temperature core material interactions, radioiodine chemistry, aerosol physics, containment thermal-hydraulics and mass transfers.

In the meantime, the next two tests of the Program are being actively prepared. FPT-4 will use a rubble bed of irradiated fuel to investigate the release of low volatile fission products and actinides at high temperatures up to the molten pool configuration. Pre-test analyses have shown the rather large uncertainty which exists in the predicted mass of fuel which can volatilize during the test. The use of filters in the in-pile test package will allow to use the experimental circuits a few months later to perform FPT-2, a repetition of FPT-1 but under steam poor conditions. Containment conditions will be also somewhat different, as the sump water will be alkaline and hotter than the atmosphere during the 4-days chemistry phase. Possible poisoning of coupons made of catalytical materials, provided by different laboratories, will also be studied during FPT-2.

Discussions are ongoing within the Phebus F.P. expert groups to define the last two tests of the matrix: FPT-3 would investigate the effect of boron carbide control rods on both core degradation and fission product chemistry , whereas air ingress conditions would be studied in FPT-5.

On the Re-entrainment of Aerosols from a Boiling Pool after a Severe Core Melt Accident

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Keywords
aerosol re-entrainment, containment venting, entrainment modeling

Abstract

One major objective of the REVENT program is the experimental determination of the aerosols re-entrained from the pool boiling during controlled filtered venting of the containment after a severe core melt accident. Experimental studies are carried out in a scaled-down model containment (factor 1:20). A simplified model based on integral balance equations is used to describe the breakup of bubbles, and furthermore, to estimate the amount of aerosols entrained from the water pool. The model indicates that aerosol entrainment depends largely on the size of the bubbles in the pool as well as on the gas velocity above the pool surface. These findings motivate experimental studies of the bubble size and the velocity field inside the model containment.

Introduction

In western nuclear power plants several layers of protection are provided ensuring its proper operation. However, probabilistic studies show that the global frequency of a severe core melt accident is 10^{-5}/year per plant. In the case of such an accident the containment provides only a limited protection against releases of fission products. Controlled filtered venting of the containment as an additional safety concept is one possibility to avoid containment failure by over-pressure. The release of active and non-active substances into the containment atmosphere and their transport inside the containment are parts of our research effort. Aerosol release and aerosol behaviour during several accident scenarios were investigated in a large number of separate effect studies and integral experiments. But, most of the previous research programs do not consider the concept of containment venting, which will lead to severe changes in the containment atmosphere. Containment venting may cause flashing of the water pool at the bottom of the containment or, at least, will cause boiling of the pool. Thus previously deposited aerosols will be re-entrained out of the water into the open venting system. Releases, which were otherwise small, may increase by orders of magnitude due to this effect.

Objectives

The major objective of the REVENT program is to quantify the amount of aerosols which are separated in the filter systems during containment venting (Müller 1997). Both water soluble (Na_2SO_4, CsI, KI) and non soluble (SiC) model substances are used as fission product simulants during the pilot scale experiments. The amount of aerosols entrained from the containment sump has been estimated by an integral model. The

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objective of this paper is to present a brief summary of the modeling approach and to discuss the influence of such parameters as the size of the bubbles in the pool and the fluid velocity in the containment atmosphere.

Experimental

The REVENT facility allows to measure the content of aerosol released from a boiling pool surface at quasi steady state conditions (Müller 1997). The decay heat is simulated with an electrical heating plate working at a constant power of 20 kW. The plate with a diameter of 55 cm is mounted at the bottom of the model containment (Volume 5 m^3). In the sump saltwater (soluble substances) or solid particles (non soluble) suspended in water is boiling above the heating plate. Fig. 1 illustrates relevant transport phenomena inside the containment atmos-phere. The concentration of the model substance in the sump is c_{BP}, the concentration in the condensate is c_C. For the presented experiments the containment atmosphere was saturated and did not contain non condensible gases. The mass flow through the condenser $\dot{m}_C$ was adjusted so that the pressure inside the containment was constant (4 bar). To obtain quasi steady sate conditions $\dot{m}_C$ was recirculated into the sump. Experiments were conducted under venting and steady state conditions. In fig. 2 the measured values of c_C are shown for the different model substances. The condensate concentration is found to increase for increasing c_{BP}.

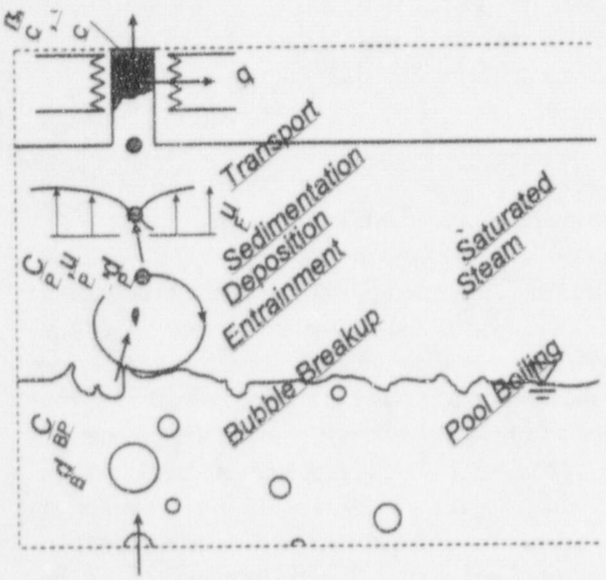

Fig.1 Schematic of the transport processes inside the model containment.

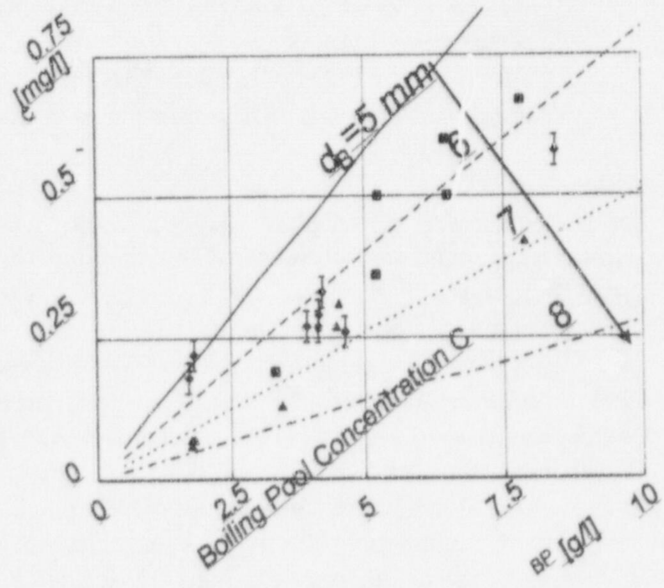

Fig.2 Experimental results and model predictions for the condensate Concentration C_C as a function of C_{BP}.

Model

Balance equations for mass and momentum were used to study the importance of the different transport processes leading to the measured values of c_C (Müller *et al.* 1997). Simplifying assumptions are: (a) all bubbles in the pool have the same size d_B, (b) the velocity above the sump $\mathbf{u}_F$ has no horizontal component, (c) the vertical component is equal at all locations above the sump and shows no time dependence. (d) Wall deposition of aerosols is not considered. The values of c_C obtained by this approach are shown in fig. 2 as well. Consistent with the experimental findings c_C increases for increasing c_{BP}. In addition, the bubble diameter d_B appears to affect the condensate concentration: smaller bubbles cause c_C to increase. Comparing the experimental findings and the model predictions the average bubble diameter was estimated to be in the order of 5 mm.

Discussion

The comparison between experimental data for c_C and the results predicted from the model shows that the size distribution of the bubbles in the pool is of importance to understand the breakup mechanism. Measurements of the diameter distribution of d_B are therefore being performed at the REVENT facility. To verify assumptions (b) and (c) the velocity field in the containment atmosphere is measured by means of a laser sheet technique. From present experimental findings there are indications that these assumptions should not be used if non condensible gases are present in the containment atmosphere.

Financial support provided by the Swiss Federal Nuclear Safety Inspectorate (HSK) is gratefully acknowledged.

[1] Müller, M. (1997) *Re-entrainment von Aerosolen aus einem Containmentsumpf während der gefilterten Druckentlastung nach einem schweren Kernschmelzunfall*; Diss. ETH Nr. 12138; Zürich

[2] Müller, M., Rudolf von Rohr, Ph. (1997) REVET Program: modeling of aerosol re-entrainment from boiling pool during controlled filtered venting after a severe core melt accident, *J. Aerosol Sc.* **28**, Suppl.1, pp. 711-712

The Linear-Nonthreshold Dose-Response Hypothesis: A Critical Reevaluation.

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Although it has been customary, for radiation protection purposes, to assume that the risks of mutagenic and carcinogenic effects of ionizing radiation increase as linear-nonthreshold functions of the dose, the existing data do not exclude the possibility that thresholds for such effects may exist in the low dose domain. Therefore, it is generally acknowledged that the linear-nonthreshold theory (LNT) needs to be reevaluated periodically in the light of advancing knowledge. Such a reevaluation of the LNT is presently being conducted by Scientific Committee 1-6 of the National Council on Radiation Protection and Measurements (NCRP). The conclusions of this evaluation are yet to be formulated, but its methodology and present status form the basis of the following report.

The reevaluation in question has been guided by the Committee's charge specifically to review the weight of scientific evidence for or against the LNT and not to concern itself with policy ramifications. In order to provide the breadth of expertise needed for the task, the members of the Committee were drawn from each of the various relevant disciplines. Also, in an effort to ensure that no pertinent data were overlooked, the Committee has sought supplementary input from the scientific community at large, through solicitations published in the open literature and through a formal workshop convened expressly for the purpose.

In evaluating the arguments for and against the LNT, the Committee has endeavored to consider all relevant lines of evidence, epidemiological, experimental, and theoretical. Information seemingly at variance with the LNT includes: 1) evidence that irradiation may elicit adaptive responses which can enhance the capacity of exposed cells or organisms to withstand further exposure, and 2) that the net effects of low-level irradiation have appeared under certain conditions to be actually beneficial ("radiation hormesis"). Information in support of the LNT, on the other hand, includes evidence that: 1) lesions in DNA (including double-strand breaks, the repair of which is error-prone) appear to increase in frequency as linear-nonthreshold functions of the dose; 2) DNA lesions that remain unrepaired or are misrepaired may give rise to mutations and/or chromosome aberrations, the frequency of which appears to increase as linear-nonthreshold functions of the dose; 3) although carcinogenesis is a multistage process, a single mutation or chromosome aberration may suffice to increase the likelihood of neoplastic transformation in a suitably susceptible cell; and 4) neoplasms of some, but not all, types appear to increase in frequency as linear-nonthreshold functions of the dose in susceptible laboratory animals and human populations.

These and other lines of evidence are still in the process of being evaluated, and although the schedule for completion of the task is somewhat uncertain, the Committee hopes that a full report of its evaluation will be ready for submission to the NCRP by the Fall of 1998. In any case, given the conflicting lines of evidence and gaps in existing knowledge, the report is likely to conclude that any assessment of the risks attributable to low-level irradiation is fraught with uncertainty .

Utilizing A Decision Framework With DandD Models and Parameter Analyses For License Termination

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Summary

A decision methodology has been developed by Sandia National Laboratories for the Nuclear Regulatory Commission (NRC) to support implementation of the dose assessment requirements in 10 CFR Part 20 Subpart E. This decision process provides a logical and consistent framework that supports licensee planning of decommissioning activities and NRC review of license termination requests. The decision framework includes the entire range of dose modeling options a licensee may utilize, from NRC prescribed screening to complex site-specific models.

The Decontamination and Decommissioning (DandD) software package, developed by Sandia National Laboratories for the Nuclear Regulatory Commission (NRC), provides a user-friendly analytical tool to address the technical dose criteria contained in NRC's Radiological Criteria for License Termination rule (10 CFR Part 20 Subpart E; NRC, 1997). Specifically, DandD implements the NRC's screening methodology, allowing licensees to convert residual radioactivity contamination levels at their site to annual dose. DandD is consistent with both 10 CFR Part 20 and the corresponding implementation guidance currently under development by NRC. NRC's screening methodology employs prudently conservative scenarios, fate and transport models, and default parameter values that have been developed to support decisions to release certain sites given only information about the level of contamination. Therefore, a licensee has the option of specifying only the level of contamination and running the code with the default parameter values, or, in the case where site specific information is available, altering the appropriate parameter values then calculating dose. DandD implements models for four different scenarios: residential, building occupancy, building renovation, or drinking water. The screening methodology only applies to the residential and building occupancy exposure scenarios. The screening methodology is an integral part of the larger decision framework, allowing and encouraging licensees to optimize decisions on choice of alternative actions at their site, including collection of additional data and information.

For the simplest level of analysis, the user is required to provide a minimum amount of site-specific information. In general, only information about source concentration is required for screening. This level of analysis is automated in DandD, and therefore provides certain licensees with a simple and cost-effective method to demonstrate compliance using a minimum amount of information. This level of analysis implements the generic scenarios and models from NUREG/CR-5512, Volume 1, and uses deterministic values for all model parameters that have also been defined to be prudently conservative through a systematic process of assessing the variability of each parameter across all sites and then defining default values that produce generic dose estimates that are unlikely to be exceeded at any real site.

If a licensee has site-specific information for certain parameters, they may choose to replace the default parameter values with alternative values, and employ the default transport and exposure models. This level of analysis, which can easily be conducted with DandD, was previously referred to as Level 2 screening. Licensees are not required to conduct the "Level 1" screening calculations prior to proposing changes to parameter values if they have such information to do so.

The default parameter values for the NUREG/CR-5512 models (which are implemented in DandD) are based on probability distributions representing the variability across all sites in the country. As a consequence, the licensee would likely need little supporting information to defend significant changes to the parameter values. For example, the probability distribution used in defining the default values for the depth to groundwater for the NUREG/CR-5512 residential scenario models is based on the variety of possible hydrogeologic settings. Many sites should be able to defend a greater depth to groundwater than the default value. This approach of moving away from the "prudently conservative" values used in the NUREG/CR-5512 modeling could be used by all sites until the point that further reduction in simulated dose would require model changes. This would require the licensee to step away from using DandD. At that point, new models and parameter values would have to be developed and defended by the licensee. Model changes should lead to less conservative models and lower doses with each iteration, because the NUREG/CR-5512 models are designed to be inherently conservative, however the conservatism of presently-used models has yet to be fully evaluated or quantified.

USING MARSSIM FINAL STATUS SURVEY TECHNIQUES AT A LWR SITE

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An important step in the development of the survey methodology contained in the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) and NUREG-1505 involved performing actual field testing of these new methods. The first of the field tests was performed at a LWR site.

The areas selected for the survey contained low-levels of residual radioactivity from site operations that would be classified as Class 1 or Class 2.. To be an effective test, the level of "residual activity" was selected to be near the guideline, since an area with either no residual activity, or a heavily contaminated area would be easy to identify using almost any reasonable procedure.

The field test was conducted on both interior and exterior survey units. The radionuclides of concern were ^{60}Co and ^{137}Cs. The following example activity and concentration release criteria were used, based on a conservative, screening level approach:

Surface activity
^{60}Co 1260 dpm/100 cm^2
^{137}Cs 1120 dpm/100 cm^2

Soil concentration
^{60}Co 1.18 pCi/g
^{137}Cs 0.83 pCi/g

Global fallout levels of ^{137}Cs in the region were expected to be in the range of zero (highly disturbed) to 0.5 pCi/g (excess fallout from runoff) with a standard deviation on the order of 0.2.

The exterior Class 2 survey unit was 3000 m^2 in size. Soil samples were collected using a 62 cm^2 diameter core cutter to pull a 15 cm deep section of soil. These samples were then analyzed for ^{60}Co and ^{137}Cs by Ge spectrometry.

The interior Class 1 survey unit was 61 m^2 in size. A GM probe and scaler unit were used for the surface measurements. Since these measurements are not radionuclide specific, a reference area was found in nearby building constructed of similar material. Background in the reference area was estimated to be about 2240 dpm/100 cm^2 with a standard deviation of about 400 dpm/100 cm^2.

This paper gives the survey design, data analysis procedures, and results that were used to successfully demonstrate the use of MARSSIM and NUREG-1505 for conducting final status surveys at a LWR site.

NRC Perspective on Selected Materials Integrity Issues: Session Overview

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Predicting and maintaining the physical integrity of components in service can present a challenge in many arenas in which the USNRC has regulatory authority. This physical integrity can be challenged by aging effects, irradiation effects, fabrication defects, and repair defects, as well as other in-service demands. Ensuring that the components maintain physical integrity is important for maintaining safety and reliability of systems in nuclear power plants, as well as in the safe transportation, storage, and disposal of spent nuclear fuel. In response to the concerns regarding integrity of selected components in service, the Office of Nuclear Regulatory Research (RES) has funded work to evaluate weld cracking and provide a flaw size screening criteria for spent fuel dry storage cask welds. In addition, RES is initiating cooperative research programs to investigate the feasibility of welded repairs in highly irradiated stainless steels and evaluate the integrity of spent fuel and spent fuel casks that have been stored up to 18 years in dry storage.

Investigating the Feasibility of Welded Repairs of Highly Irradiated Stainless Steels in Boiling Water Reactors

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A. Louise Lund, US Nuclear Regulatory Commission
Robert C. Thomas, Electric Power Research Institute
Robin L. Dyle, Inservice Engineering

As reactors age, the availability of repair technology is of more interest. Use of welded repairs within the reactor vessel is desirable due to their durability, strength and broad applicability. Welded repair methods allow refurbishment of components without the installation of costly and cumbersome mechanical devices and the associated design modification considerations. Use of welded repairs however presents its own set of unique issues to be resolved. One major issue is the formation of cracks when welding highly irradiated stainless steels. This cracking occurs when helium, a product of the transmutation of boron and nickel in the presence of neutrons, is present in base materials during welding.

The US Nuclear Regulatory Commission and Electric Power Research Institute/Boiling Water Reactor Vessel and Internals Project (EPRI/BWRVIP) are currently involved in a cooperative effort to evaluate the feasibility of welding on highly irradiated stainless steels in the boiling water reactors. The program will attempt to determine: locations for which repair is feasible, the level of helium that eliminates welding as a repair option, welding techniques that can be employed, and the range of helium concentrations for which each technique is applicable.

A near term benefit is to find a method for determining (and eventually predicting) the helium concentration in locations within the vessel that require repair. An additional benefit of this program will be the opportunity to corroborate predictive neutron flux model results with actual field material samples.

Cooperative Research on LWR Spent Nuclear Fuel in Dry Storage

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Abstract

Since 1986 a large quantity of spent-fuel has been placed in dry cask storage in Independent Spent Fuel Storage Installations (ISFSIs) at commercial nuclear power stations. NRC license extensions are needed to continue storage beyond 20 years. Information on the long-term integrity of spent-fuel and dry storage casks under dry storage conditions is not currently available to support cask license extensions by the NRC.

The Idaho National Engineering and Environmental Laboratory (INEEL) has been involved in a dry storage test and demonstration program since 1985. Prototypes of several commercial dry storage casks were procured or constructed on-site, and multiple tests and evaluations of the viability of dry cask storage of spent-fuel were conducted. In 1985 and 1986, three casks were tested by placing Westinghouse PWR spent-fuel assemblies in the GNS Castor V/21, Westinghouse MC-10, and Transnuclear TN-24P metal casks. Fuel rods from 48 assemblies were subsequently consolidated into 24 canisters, which were tested in the TN-24P. In 1989, 17 of those 24 canisters were used to test the Pacific Sierra Nuclear VSC-17 concrete cask. Monitoring of cask temperatures was initially performed. Continued routine monitoring consists of visual surveillance of the casks, monitoring of the gas pressure inside the casks, and monitoring of radiation fields around the casks.

Preparations to determine how the spent-fuel and dry cask internals have behaved under extended storage conditions at the INEEL are now being made by Lockheed Martin Idaho Technologies Company, with funding from the NRC, EPRI, and DOE-RW. This research effort will provide enhanced cask monitoring, and will allow for inspection and performance of selected material tests of the spent-fuel and cask internals in the GNS Castor V/21 and the Pacific Sierra Nuclear VSC-17 casks. This program will start in FY-1999 and is planned to continue through at least FY-2001.

Weld Cracking in Spent Fuel Dry Storage Casks

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Due to a limited storage capacity in spent fuel pools, some nuclear power plants are temporarily storing spent fuel on site in specially designed dry storage casks. One type of these dry storage casks has experienced weld cracking problems during closure welding. This particular cask design incorporates 2 lids which are both welded to the outer shell; the inner shield lid provides shielding from the spent fuel while the outer structural lid is used for structural integrity, additional shielding, and redundant sealing of the confinement system. NDE has found cracks up to 18 in. long in these lid to vessel welds. The cracks are attributed to undocumented weld repairs, moisture, hydrogen induced cracking, and improper fit-up during welding.

To prevent cracking in future casks several changes have been proposed: a 200°F preheat and postheat; use of low hydrogen electrodes; sequenced welding; and use of low sulfur, calcium-treated, vacuum-degassed steel in the construction of future casks. A delay time before inspection enhances the probability that any cracks which could develop after welding will be discovered. In addition to the current dye penetrant (PT), visual examination (VT), and helium leak check, a new ultrasonic examination (UT) will be required to provide reasonable assurance of the integrity of the weld. Flaws discovered during UT which do not meet a given screening criteria must be either repaired or reexamined using Linear Elastic Fracture Mechanics (LEFM) or Elastic Plastic Fracture Mechanics (EPFM). The analytical and experimental work used to determine this screening criteria is described in a companion paper.

This paper will describe the cracks found in these welds and explain the approach used to resolve this problem.

Determination of Flaw Size Screening Criteria for Spent Fuel Dry Storage Cask Welds

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A companion paper described weld cracking problems experienced in a particular dry cask storage system. Ultrasonic examination (UT) of the outer structural weld of this system's sealed canister will be performed to determine if any indicated flaws are acceptable or require repair and/or additional analysis.-

Allowable flaw sizes were calculated using ASME Section XI, IWB-3600 and Appendix A considering a horizontal drop accident occurring at a given temperature as the limiting operating condition. The weld membrane stress due to horizontal drop accident conditions, as reported in the safety analysis report, is 43.3 ksi, and the residual membrane stress due to welding of the outer structural lid was conservatively assumed to be equal to the material (SA516 steel) yield stress (38 ksi) SA516- Grade 70 Charpy V-notch impact energy and fracture toughness specimens were tested by both the licensees using the cask and an NRC contractor. These measured toughnesses were then adjusted to obtain a conservative lower-bound value and to account for dynamic loading rate effects. These potential membrane stresses and adjusted toughness values were input into an analytical model to determine the appropriate flaw size screening criteria at various operating temperatures.

UT inspection of a mockup of the canister closure weld confirmed the feasibility of performing the inspection under field conditions and showed that the UT method was sensitive and reliable in detecting the screening flaw sizes. This paper will present the flaw screening methodology, details of fracture toughness testing of the cask materials, and an overview of the UT inspection procedure.

Issues in the Design of Human-Systems Interfaces to Digital Systems

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The U.S. Nuclear Regulatory Commission's (NRC) human factors engineering (HFE) design review guidance is described in Chapter 18 of the Standard Review Plan (NUREG -0800), Human Factors Engineering Program Review Model (NUREG-0711, and the Human-System Interface (HSI) Design Review Guideline (NUREG-0700, Rev. 1). While the NUREGs -0800 and -0711 mainly address the process aspects of HFE considerations, the latter addresses the detailed implementation of an HSI design.

In the development of NUREG-0700, Rev 1, several topics were identified as "gaps" because there was an insufficient technical basis upon which to develop guidance. One gap was hybrid HSIs; i.e., HSIs that result from the combination of digital and tradition HSI technologies. New demands may be imposed on personnel for the operation and maintenance of these systems. These demands may result from many factors including: characteristics of the new technologies, characteristics of the mixture of new and traditional technologies, the process by which the hybrid HSI is developed and implemented, and the way in which personnel are prepared to use the hybrid HSI.

The NRC is currently sponsoring research to (1) better define the effects of hybrid HSIs on personnel performance and plant safety; and (2) develop HFE guidance to support safety reviews in the event that a review of plant modifications involving a safety-significant aspect of HSIs is necessary. This guidance will be integrated into existing regulatory guidance documents and will be used to provide the NRC staff with the technical basis to help ensure that the modifications or HSI designs do not compromise safety.

HSI technology changes and their potential effects on personnel performance were identified based upon published literature, interviews with designers and subject matter experts, and plant visits. These results were reported at last year's WRISM and elsewhere (O'Hara, Stubler, & Higgins, 1996). The topics were evaluated with respect to their potential safety significance (Stubler, Higgins, and O'Hara, 1996). One topic found to be potentially significant to safety and selected for the development of HFE guidance was Design Analysis, Evaluation, and Implementation of Hybrid HSIs.

This topic addresses analyses and evaluations conducted during the design of upgrades and the way upgrades are introduced into the HSI and incorporated into plant operating practices. Important considerations included the effects upon personnel of temporary and changing HSI configurations, which result from the installation of HSI upgrades. Additional considerations include training and personnel acceptance of HSI changes. Thus, the topic addresses the life cycle of an HSI upgrade from initial planning through design, evaluation, and installation.

With regard to its application to hybrid HSIs in the context of plant modifications, the existing guidance is also limited in an additional way. While the guidance provides for tailoring of the review methods and criteria to the unique circumstances of an individual review, no guidance is available to assist in the identification of the process elements and criteria that are necessary. The extent of plant modifications can range significantly, e.g., for a replacement "in-kind" of a single HSI component to an extensive control room modification from analog to digital technology. Thus, when and where to apply that guidance that is available needs to be addressed.

The objective of the phase of the research that is reported in this paper was to develop human factors review guidance addressing the process by which hybrid human-system interfaces are developed, implemented, and integrated into plant operations. To support this objective, several tasks were performed including:

- Development of a technical basis using human performance research and analyses that are relevant to upgrades,

- Development of HFE review guidelines in a format that is consistent with existing NRC review guidance, and

- Identification of remaining issues for which research results were insufficient to support the development of NRC review guidance.

The status of each will be briefly addressed below.

Technical information related to system development and modification was reviewed. This information was used to provide the technical basis upon which design review guidelines were developed. The NUREG-0700 guidance development methodology was used to convert this technical basis into technically valid review guidance. A characterization framework for describing key characteristics of hybrid HSIs that are important to human factors engineering (HFE) reviews was developed. Since the guidance addressed the design and implementation process, rather than the detailed design implementation, the guidance was organized according to the ten design process elements identified in NUREG-0800 and -0711, such as operating experience review, function analysis, task analysis, verification and validation.

In the course of the guidance development process, several human performance issues associated with upgrades were identified that could not be addressed with the available technical basis. They represent topics for which further research is necessary. These issues include:

- The Role of HSI Consistency as Applied to Traditional and Digital HSIs
- The Effects of HSI Design on Crew Coordination and Cooperation
- The Role of Training in HSI Skills
- The Effects of the Installation Process for HSI Upgrades upon Personnel Performance
- Personnel Acceptance of Upgrades.

In conclusion, design review guidance addressing the design, evaluation, and implementation considerations of HSI upgrades has been developed. This guidance complements the design review guidance that was already developed in other phases of the project to address the characteristics associated with specific technologies such as soft controls, advanced information systems, computer-based procedures, and digital system maintenance. The guidance was peer reviewed and is expected to be published at the end of 1998.

Reliability Modeling of Safety-Critical Digital Systems

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Summary

Safety-critical systems are employed in a wide range of applications including nuclear reactors, aircraft flight control systems, medical electronics, railroad control systems, and automobile braking controllers. The availability of low-cost, high-performance microprocessors has led to their proliferation in a variety of applications, many of which are safety-critical in nature. As the use of microprocessors has increased so has the complexity of safety-critical systems. The increasing complexity has compounded the problem of designing the system and assessing its reliability and safety. Modern safety-critical systems include sensors, analog hardware, digital hardware (including microprocessors), software, actuation devices, and human operators integrated into a complete functional unit to control or monitor some dynamic system. The complexity of the complete system makes it impossible to identify, characterize, enumerate, and evaluate all the possible ways that the system's components may fail. The result is an increasingly difficult task of estimating how reliable and safe the system will be once it is placed into operation.

This paper describes an approach for modeling and assessing the reliability and safety of complex systems used in modern safety-critical applications. The modeling and assessment process consists of several steps that include, for example: (1) establishment of quantitative and qualitative reliability and safety measures; (2) establishment of assumptions and models to be used in the assessment process; (3) partitioning of the system into manageable subsystems; (4) development of an overall system model; (5) identification of crucial parameters through sensitivity analysis of the system model; (6) estimation of parameters through a combination of analytical, simulation, and statistical models; and (7) estimation of the quantitative and qualitative reliability and safety measure for the system. This paper describes each phase of the process and illustrates each phase with simple examples.

The complete process has been applied to a commercial safety-critical application. The application is an embedded real-time controller that has been installed and in operation at more than 100 locations throughout the world for more than 10 years. The system is microprocessor based and includes approximately 30,000 lines of assembly language software. Approximately 80% of the software are diagnostics that are included to detect faults and initiate a safe shutdown in the event of faults that cannot be tolerated. The application is fail-safe in that the system is designed to shut down to a safe state in the event that it is unable to continue correct operation. The system has real-time constraints that require it to respond to input changes or fault conditions within 250 milliseconds, or the system is considered failed in an unsafe manner. The results of the modeling and assessment process have been used in the licensing of this system by a public utility commission.

This paper describes each step of the assessment process as it has been applied to our application. The critical quantitative measure in this case is the Mean Time Between Hazardous Events (MTBHE). The MTBHE requirement was 10^5 years for the complete system. The overall system reliability and safety model was developed using Petri nets. Parameter estimation was performed through the use of a hierarchy of simulation models. The simulation models allow the real software to be executed on a model of the hardware so faults can be inserted in the complete hardware/software system. The input and output

modules were simulated at the physical device (transistor, resistor, capacitor, and gate) level, while the microprocessor was simulated at the instruction set architecture level. The combination of analytical and simulation approaches allowed the effects of millions of faults to be evaluated. This paper shows the values obtained for each parameter estimate and the value obtained for the MTBHE of the complete system.

Finally, this paper summarizes research issues that must be addressed to support the future enhancement of reliability and safety assessment of hardware/software systems. First, techniques are needed to support modeling systems at higher levels of abstraction. Complexity is a problem for both design and analysis, and higher levels of modeling abstraction are needed to accommodate increasing complexity. Second, fault and error models are needed to represent corruption that can occur at the higher levels of modeling abstraction. The higher levels of complexity and the increasing use of commercial off the shelf hardware and software make it impossible to model all system components at the physical device level. Similarly, corruption cannot continue to be modeled at low levels of abstraction. Third, improved simulation techniques are needed to support a hybrid analysis of systems. The term hybrid is used here to imply analysis approaches that use analytical models as well as simulation models. Simulation models are needed to gain a better understanding of how the integrated hardware/software system functions and performs in the presence of faults. Such simulations are complex and time consuming, and techniques are needed to make them more efficient and quicker to perform. Fourth, the simulation models are used to produce data that provides a means of estimating certain crucial system parameters. Since exhaustive simulation is seldom feasible statistical techniques must be used to estimate parameters using the available data. New statistical modeling approaches are needed to provide more accurate estimates of the important parameters. Fifth, modeling and simulation tools are needed to allow designers and analysts to more easily construct the necessary models and perform the analysis. Sixth, design principles are needed to allow the creation of systems that are easier to model and analyze. For example, it is well known that time-triggered systems are much easier to analyze than event-triggered systems, and this might be one of the fundamental design principles. Design principles might lead to generic architectures and methodologies that are easier to design, model, and analyze but still provide the flexibility for competing companies to add their own unique properties. Finally, experimental research is desperately needed to provide proof of the concepts developed in research projects. Researchers need to become more actively involved in the design, implementation, and testing of safety-critical systems for real-world applications. It is through such experimental work that we will learn if new ideas and approaches are feasible and actually work as proposed.

The Importance of Fault Detection Coverage in Safety Critical Systems

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Fault coverage is the probability that a fault is detected given that the fault occurs. There are numerous techniques available to quantify fault coverage and they can be partitioned in two distinct ways: (1) coverage modeling and (2) coverage estimation. As its name implies, fault coverage modeling is the development of a model for the response of a component to the occurrence of a fault. Coverage estimation, however, treats coverage as a model parameter and analytically derives a coverage estimate. There are three primary model types used in examining coverage [1]:

(1) *axiomatic models*: analytical models used to model structure and the dependability and/or performance behavior of a system [2].

(2) *empirical models*: statistical models used to model complex and detailed descriptions of a system's parameter(s) using data collected from physical models.

(3) *physical models*: prototypes that actually implement the hardware and/or the software of an actual system.

Axiomatic modeling of fault coverage is a behavioral representation of a system's response to faults, which can be categorized as coverage modeling. These models are embedded in the overall system model, and the actual number of coverage models required is a function of the system under test. There have been numerous refinements to the axiomatic fault coverage models and the various models that have been developed are presented in the following sections. These models are categorized into two sections: error handling without time limitation and error handling with time limitations.

The initial iteration of fault coverage models are categorized as models that perform error handling without time limitations. These models ignore any type of interference that could occur during error handling, and typically consist of various forms of Markov and semi-Markov models. In these models, it is assumed that the time spent in states handling errors is negligible with respect to the time spent in states where errors are not present. If the transient lifetime is considered, which in reality is a major concern, then these models have very little applicability in developing accurate fault coverage estimation. In order for coverage models to be robust, consideration must be given to the lifetime of the fault and/or error and such models are categorized as error handling with time limitations. In all of these models, estimates are required for the various parameters required by the models to derive their coverage estimate. Hence, the accuracy of these coverage estimates are a function of the parameters required by the models.

The use of empirical models for fault coverage estimation requires detailed statistical analysis that must address four important questions [3]:

(1) How can the fault coverage value be accurately estimated?

(2) How can any error in the estimate be quantified?

(3) How are fault samples selected?

(4) How can accurate estimates for fault coverage be obtained in a reasonable time?

Empirical modeling relaxes many of the assumptions and restrictions present in axiomatic models. Parameter estimation requires that the system fault space be sampled in some fashion to provide a representative sample of the entire fault set. Once this sampled fault space is obtained, then a physical model is experimentally subjected to this fault set. Using the data collected from this experiment, statistical analysis is performed to estimate the parameters. It is shown in [4] that this technique can be used for predicting the system's expected fault coverage. There are numerous sampling strategies available, including techniques that attempt to reduce the variance of the estimate. This type of sampling is referred to as a variance reduction technique (VRT).

The purpose of VRTs is to increase the accuracy of the parameter estimate so that the required number of sample points can be further reduced. VRTs exploit some attribute of the system to increase the accuracy of the parameter estimate(s). Importance sampling, multi-stage sampling, stratified sampling and regression analysis are all examples of VRTs [5], [6]. The estimates that are derived using variance techniques require an a prior knowledge of the system's fault data. If the assumed distribution differs from the actual distribution for the fault data, then the accuracy of the coverage estimate is questionable.

It is the purpose of this paper to categorize existing fault coverage models and estimators as either axiomatic or empirical, and to summarize their methodology. Through this classification and summarization process, the evolution of fault coverage models and estimators is provided, which demonstrates the refinements of these techniques.

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Safety Criteria and Certification of Programmable Industrial Control Systems Based on IEC 61508 and UL 1998

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ABSTRACT

Manufacturers are using more and more innovative technologies to create components, products and systems with enhanced capabilities and improved performance. When the technologies involve programmable or computerized components, benefits such as reduced parts costs and the ability to rapidly incorporate new features have been realized. With the increased reliance on microprocessors and software to deliver this enhanced functionality, product designs are using more complex controls than their traditional electro-mechanical counterparts. These modern systems need to be designed, constructed and tested according to a set of safety requirements, arrived at through technical experience and consensus. Two important safety standards have been developed, one in the US and one in Europe, to address the safety-related aspects of these systems. In the United States, UL 1998, "Standard for Safety for Software in Programmable Components," has been developed to address non-networked embedded software used in safety-related applications. In the European Community, the International Electrotechnical Commission has developed IEC 61508, "Functional safety: safety-related systems." We describe some of the similarities and differences between these two safety standards and describe the process that UL's Programmable Systems Certification Services (PSCS) uses to assess compliance with these requirements.

Putting Principles into Practise:
The Formal Development of a Theorem Prover

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Formal methods for software specification, verification, and development have for many years been a focal point of research at the OECD Halden Reactor Project. A major accomplishment has been the establishment of a complete methodology for the practical application of algebraic specification in formal software development. The methodology is supported by the HRP Prover, an automatic theorem prover developed at the Halden Project to facilitate exploration of animation and theorem proving techniques in formal software development. The applicability of the methodology has previously been demonstrated in a case study on the development of a reactor safety system. More recent achievements include a uniform approach to the translation of Petri nets into algebraic specification, thus combining graphical and textual notations.

The paper presents results from a project involving the use of this methodology in the development of a new version of the HRP Prover. In spite of the large number of tools available that support formal methods, very few of these have been developed in accordance to the same principles. The novelty of the reported project relates to the fact that the methodology used in the development of the tool is identical to the methodology supported by the tool. In particular, the formal development of the HRP Prover appears to be unmatched from earlier developments of theorem provers in its extensive use of algebraic specifications.

The project has delivered a tool that facilitates formal development of modular, well-structured programs by the use of automatic program generation from specification or design. As a consequence, functional requirements can be realised directly, while improving the traceability of requirements from the program code. This also simplifies maintenance and further development, since new program code can be constructed directly from changed or added requirements. One example of a modular extension relates to the automatic transformation between classes of specifications. Two related classes that capture the concept of state are so-called state-based and transition-based specifications. The paper demonstrates how the automatic transformation between these two classes facilitates a more efficient approach to specification and design, especially when the development involves the integration of graphical and textual notations.

The reported results from the development project are of relevance to the formal development of a wide range of language-oriented tools, involving aspects like analysis, transformation, and code generation. In particular, the approach employed appears to facilitate combination of complementary specification notations. This is exemplified in the paper by the integration of Petri nets and algebraic specifications. The paper aims at giving proper attention to the advantages as well as the limitations of the approach, and in particular to the transferability of the results in other contexts, involving other specification languages.

Reevaluation of Regulatory Guidance on Modal Combination Methods for Seismic Response Spectrum Analysis

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Regulatory Guide 1.92 "Combining Modal Responses and Spatial Components in Seismic Response Analysis" was last revised in 1976. The purpose of this project was to re-evaluate the current regulatory guidance for combining modal responses in Response Spectrum Analysis, evaluate recent technical developments, and recommend revisions to the regulatory guidance. In addition, Standard Review Plan (SRP), Section 3.7.2 was also reviewed to identify related sections which may need to be revised. The objectives were addressed through a literature review of past studies, supplemented by analysis of a piping system model previously utilized in NUREG/CR-5627, "Alternate Modal Combination Methods in Response Spectrum Analysis".

The project evaluated (1) methods for separation of the in-phase and out-of-phase modal response components; (2) methods for combination of the out-of-phase modal response components; (3) the contribution of "missing mass"; and (4) combination of the three elements of response to produce the total response. Numerical results from Response Spectrum Analysis were compared to corresponding time history analysis results to assess the accuracy of the various combination methods tested.

For separation of in-phase from out-of-phase modal response components, the methods proposed by Lindley-Yow, Hadjian and Gupta were evaluated. For combination of the out-of-phase modal response components, the Square Root of the Sum of the Squares (SRSS), NRC Grouping, NRC Ten Percent, NRC-Double Sum Combination (DSC), Rosenblueth's DSC, and Der Kiureghian's Complete Quadratic Combination CQC methods were evaluated. For treatment of the "missing mass" contribution, the method of Kennedy was evaluated. Response Spectrum Analyses were conducted by combining elements of the above methods to construct complete Response Spectrum Analysis solutions.

Baseline time history solutions, utilizing both Mode Superposition and Direct Integration methods were also generated. Excellent correlation was obtained between the two time history methods by including the missing mass contribution to the mode superposition solution. Missing mass proved to be significant even though the mode superposition solution included 31 modes, up to 70 Hz.

Based on the literature review and the numerical results generated in this project, it was concluded that Rosenblueth's DSC and Der Kiureghian's CQC methods for combining out-of-phase modal response components, coupled with the Lindley-Yow or Gupta method for separation of in-phase from out-of-phase modal response components, and inclusion of Kennedy's missing mass contribution for modes with frequencies above the Zero Period Acceleration (ZPA) frequency of the response spectrum constitute the best methodologies currently available for Response Spectrum Analysis.

The NRC Grouping, Ten Percent and DSC methods for combining out-of-phase modal response components produced more conservative but less accurate results; removal of these methods from the Regulatory Guide should be considered, because absolute summation of closely spaced modal responses has no documented technical basis. SRSS remains applicable in the absence of closely spaced nodes. Separation of modal responses into in-phase and out-of-phase components for modes with frequencies below the ZPA frequency of the response spectrum, coupled with the missing mass approach to account for higher frequency modes, produced more accurate results than commonly applied past methods.

At the request of the Project Peer Review Panel, an extensive investigation of individual modal contributions to both the response spectrum and time history solutions was conducted for selected support reactions, in order to explain the significant differences in prediction in the original analysis. Once the phenomenon was understood, several variations on the initial modal combination methods were defined and additional response spectrum analyses were conducted. Improvement in accuracy was achieved. As part of this additional analytical effort, a methodology was defined to utilize the SDOF oscillator responses from the response spectrum generation analysis to establish the frequency above which modal responses are in-phase with the input time history. This technique should be applicable to all response spectra.

During the course of the project, static analysis results for mass times ZPA were generated utilizing system mass distributions developed by both the dynamic analysis option and the static analysis option of the computer code for the same piping model. Significant differences in solution were noted, which are attributed to differences in the treatment of mass. In the dynamic analysis, mass is apportioned to node points, while in the static analysis, mass is treated as distributed along the element length. While this situation may not exist in all piping analysis codes, it is important to verify that consistent static analysis results are obtained between dynamic and static mass distributions. Lack of consistency is evidence that the element breakup in the piping model is not sufficiently refined.

Post Test Analysis of a PCCV Model Subjected to Beyond-Design-Basis Earthquake Simulations

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ABSTRACT

A scaled model Prestressed Concrete Containment Vessel (PCCV) was tested up to ultimate failure under simulated earthquake loadings at the Nuclear Power Engineering Corporation's (NUPEC) Tadotsu Engineering Laboratory in Japan. The mixed-scale model was first subjected to a series of design-level earthquakes, and then the magnitudes of the earthquakes were increased in several stages until the cylinder walls catastrophically failed in shear. Under sponsorship from the U.S. Nuclear Regulatory Commission (USNRC), state-of-the-art, non-linear dynamic finite element analyses were completed for the PCCV structure. Both pre- and post-test analyses were performed. This paper summarizes post-test analyses that correspond to the larger-than-design-basis seismic tests. Under these "severe" conditions, the concrete cracks, spalls, and crushes while the reinforcing bars and liner undergo extensive plastic deformation during the repeated dynamic load cycling. The analytical results are compared to the measured structural response of the scaled-model structure. The principal objective of this work has been to evaluate how well state-of-the-art analyses can predict the structural response and eventual failure of a prestressed concrete structure under seismic loadings that are considerably larger than the structure was designed for.

Steel Containment Vessel Model Test: Results and Posttest Analysis[1]

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Abstract

Two static, pneumatic overpressurization tests of scale models of nuclear containment structures at ambient temperature are being conducted by Sandia National Laboratories for the Nuclear Power Engineering Corporation of Japan and the U. S. Nuclear Regulatory Commission. The joint research program consists of testing two models: a steel containment vessel (SCV) model and a prestressed concrete containment vessel (PCCV) model.

This paper summarizes the conduct of test of the SCV model, which is a mixed-scaled model (1:10 in geometry and 1:4 in thickness) of an Improved Boiling Water Reactor (BWR) Mark II containment, and posttest activities. A concentric steel shell, identified as the Contact Structure, was installed over the SCV model prior to the test to represent some of the structural characteristics of the reactor shield building in the actual plant. The SCV model and Contact Structure were instrumented with strain gages and displacement transducers prior to the overpressurization test, which was conducted on December 11-12, 1996 at Sandia National Laboratories. The test was terminated when a large tear developed adjacent to the equipment hatch reinforcement plate and pressure could not be maintained.

The test data are compared with the pretest analytical predictions by the sponsoring organizations and others who participated in a blind pretest prediction effort. Posttest analysis efforts focused on resolving inconsistencies between the predicted and measured free-field strains and local strain concentrations near the equipment hatch. Posttest metallurgical evaluations on specimens removed from the SCV model were also performed and the results are discussed.

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NEW SEISMIC DESIGN SPECTRA FOR NUCLEAR POWER PLANTS

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In 1996 the Nuclear Regulatory Commission (NRC) amended its regulations to update the criteria used in decisions regarding nuclear power plant siting, including geologic, seismic, and earthquake engineering considerations for future applications; USNRC (1996). As a follow-on to the revised siting regulations, it is necessary to develop state-of-the-art recommendations on the design ground motions commensurate with seismological knowledge and engineering needs. The current design spectra in Regulatory Guide 1.60 (USNRC 1973) were based on limited, principally western United States earthquake strong-motion records, available at that time. Since 1996 the NRC has funded a project to develop up-to-date seismic design spectra for the US. The work combines empirical and analytical approaches, supplementing data where they are sparse using theoretical methods to develop the recommended spectra for a range of earthquake magnitudes and distances. Soil conditions necessarily involve site-specific parameters, and we demonstrate and recommend procedures to account for local soil effects on earthquake motions. A Review Panel consisting of Carl Stepp (Chair), David Boore, Allin Cornell, I.M. Idriss, and Robert P. Kennedy review the work and offer guidance on procedures. The prime contractor is Risk Engineering, Inc., with Pacific Engineering and Analysis developing databases, spectral shapes, and site response procedures.

Databases for the western US are available in the form of strong motion accelerograms for moment magnitudes M in the range 5.0 to 7.6 and source-to-site distances R of 1 to 100 km. Rock conditions in California are generally soft, with shear wave velocities of 700-1500 m/s. Central and eastern US (CEUS) strong motions are sparse for North America. Thus it is necessary to augment the CEUS empirical motions with analytically derived spectral shapes. This analysis uses a point- or finite-source representation of the earthquake rupture, attenuates both body and surface waves, accounts for near-surface attenuation of high frequencies, and assumes that ground motion is a band-limited, white noise process. Calibration of the model with available records confirms the underlying assumptions and provides estimates of the model parameters. One outstanding issue, however, is whether the seismic energy at the source has a "single-corner" or "double-corner" spectrum; this is the focus of independent research, and the current project will include each model as an alternative. Rock conditions in the CEUS are generally hard, with shear wave velocities exceeding 1500 m/s.

The databases of strong motion records and empirical attenuation relations form the basis for recommended spectral shapes on rock for defined M and R bins, augmented as necessary by analytically derived shapes. In the application of these spectral shapes for design, the M-R combination is defined by the dominant earthquake as determined from a probabilistic seismic hazard analysis (PSHA). Examples of procedures for defining the dominant earthquake are described in McGuire (1995) and USNRC (1997).

In addition to the spectral *shape*, the overall level of the spectrum must be specified. This choice may be made from a PSHA by defining a target annual frequency of exceedence for the spectrum. Alternatively the level could be defined using an acceptable annual frequency of failure at the component level and convoluting the seismic hazard results with component fragility curves to relate component performance to seismic hazard. Results using test sites in the CEUS and on the west coast, and using typical component fragility characteristics, indicate that the annual frequency of component failure is about 15 to 45 times less than the annual frequency of exceedence of the design spectrum, using realistic design procedures. This means that, for example if the median frequency of exceedence of a site's design spectrum is $1x10^{-4}$, the median component frequency of failure is about $3x10^{-6}$. The ultimate choice of a recommended spectral level must be made with a combination of analysis to determine acceptable failure frequencies, calibration to accepted existing design procedures, and judgment.

Recommending spectral shapes for soil sites requires additional procedures. One straightforward method is to conduct the PSHA with site-specific soil attenuation equations, to obtain seismic hazard curves and uniform hazard spectra (UHS) for the soil surface. However, as recommended in NRC (1997), it is often more practical to conduct the PSHA for rock outcrop conditions and later translate these to soil surface motions, because various facilities may sit on different soils, or detailed site-specific data may not be available early in the project. In this case a site's rock UHS at a target annual frequency of exceedence can be translated to a soil UHS at the same or similar frequency of exceedence, accounting for uncertainties in the soil properties. Procedures to accomplish this are demonstrated in the project.

In addition to recommended spectral shapes, the project will archive a database of strong motion records for the recommended **M** and R bins, for both rock and soil conditions. These will be empirical records for bins where data are abundant, augmented by artificial motions derived to have the correct frequency content for bins where data are sparse or non-existent.

A final set of recommendations concerns criteria to match artificial motions to recommended spectral shapes and levels. Such motions might be used for input to detailed dynamic analyses of building response, for example. The recommended procedures for developing artificial motions concentrate on matching response spectral amplitudes at multiple frequencies and dampings, and put less emphasis on matching power spectral density functions.

This NRC-sponsored project will offer a number of recommendations on choosing spectral shapes, selecting design levels, and generating time histories of motion for the design of nuclear facilities. The objective is to achieve consistent design levels across the country for a range of seismic environments and site conditions. Procedures developed in this project to define ground motion for a risk-consistent, performance-based design are an integral part of the recommendations. A second objective is to make the procedures easy-to-understand and technically justified, so that they will be readily accepted. There is a need to strike a balance between the engineering conservatism required to achieve the safe design, seismological knowledge, and preservation of important earthquake ground motion characteristics, such that realistic responses are considered. The results from this research will also provide tools for the seismic design of non-reactor facilities and will influence the design of non-nuclear facilities.

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DAMAGE MECHANICS-BASED ASSESSMENT OF TIME-DEPENDENT STRUCTURAL DETERIORATION AND RELIABILITY

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ABSTRACT

Introduction

Steel containments and liners in nuclear power plants (NPPs) may be exposed to aggressive service and environmental effects over a 40-year service life. Among the mechanisms having the potential to cause steel pressure boundary structures to deteriorate in service are corrosion, elevated temperature creep, low-cycle fatigue, and load-induced inelastic deformation. While corrosion is reasonably well-understood, the same cannot be said about the underlying mechanisms giving rise to creep. fatigue, or inelastic deformation damage. Moreover, the initial stages of such damage often occur without perceptible manifestation. By the time that damage reaches a detectable stage, a significant fraction of the remaining service life or residual strength (or margin of safety) may already have been exhausted.

Condition assessment of a containment metallic pressure boundary should provide quantitative evidence that structural performance will continue to meet or exceed a minimum standard of acceptability in the foreseeable future. Quantitative evaluation of the effects of damage accumulation on time-dependent structural behavior is difficult. Many methods for modeling structural deterioration require a measurable flaw to be applicable. Most are empirical or semi-empirical in nature. Finally, structural damage growth is an intrinsically random process.

Technical Approach

Time-dependent structural reliability analysis provides the framework for integrating information on material and structural degradation and damage accumulation, service and environmental factors and nondestructive evaluation technology. Research in progress, supported by Oak Ridge National Laboratory and the US Nuclear Regulatory Commission, is aimed at: identifying mathematical models to evaluate structural degration and damage

accumulation; recommending statistically-based sampling plans for nondestructive evaluation; and assessing the probability that structural capacity will not degrade to an unacceptable level during a future service period (Naus, et al, 1996; Ellingwood, et al, 1996; Oland and Naus, 1998).

A recent phase of this research has explored the use of the relatively new field of continuum damage mechanics (CDM) as a tool for evaluating damage accumulation in steel pressure boundary structures (Bhattacharya and Ellingwood, 1998). CDM deals with the aggregate effects of micro-structural defects, expressed in terms of quantities that are observable at the structural level, e.g., changes in the elastic modulus or stiffness. It is particularly well-suited for analyzing damage that may occur over an extended period of time without visible manifestation. The governing damage laws are derived from the fundamental principles of thermodynamics, and can be extended naturally into the stochastic domain. CDM has the potential to reduce the level of empiricism associated with other approaches to modeling structural damage accumulation.

Results

Time-dependent reliability analyses were performed for steel elements and steel pressure boundary components subjected to corrosion, ductile damage from sustained load, elevated temperature creep, and low-cycle fatigue. The CDM approach was validated for limit states involving ductile damage, creep and low-cycle fatigue. It was found that corrosion has the most significant impact on time-dependent reliability of steel components, the other mechanisms having a more localized effect. The estimated conditional failure rates for structural components increase in a nonlinear fashion with time. Neglecting this nonlinear behavior may lead to an erroneous appraisal of time-dependent margins of safety.

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High Frequency Imaging of Thickness Degradation in Steel Containment Vessels and Liners

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In the U.S., some nuclear power plants are subject to moisture-induced, thickness degradation of embedded steel containments several inches to a foot below the interface where the steel enters concrete, see Figure 1. This is an area which cannot be inspected by traditional Ultrasonic Thickness testing, due to the inaccessibility imposed by the concrete. The overall objective of this activity is to demonstrate the feasibility of using high frequency bistatic acoustic imaging techniques for the detection, quantification, localization and mapping of thickness reductions in the metallic pressure boundary of nuclear power plant containments.

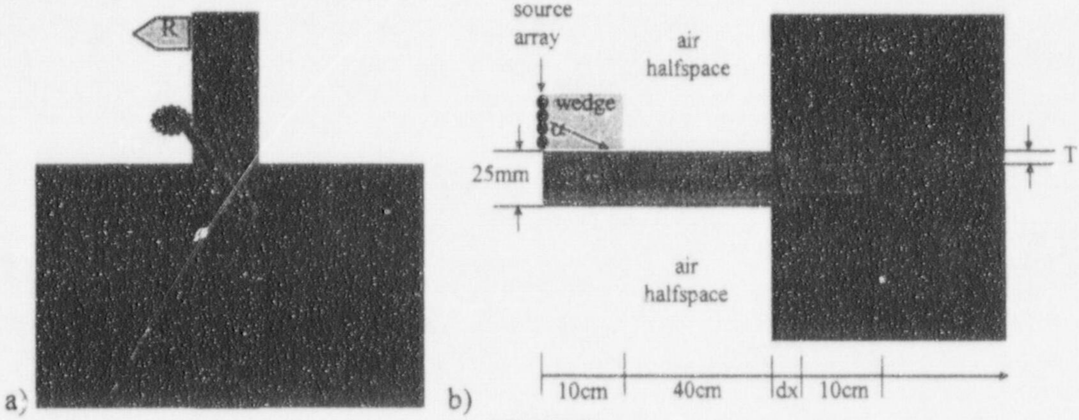


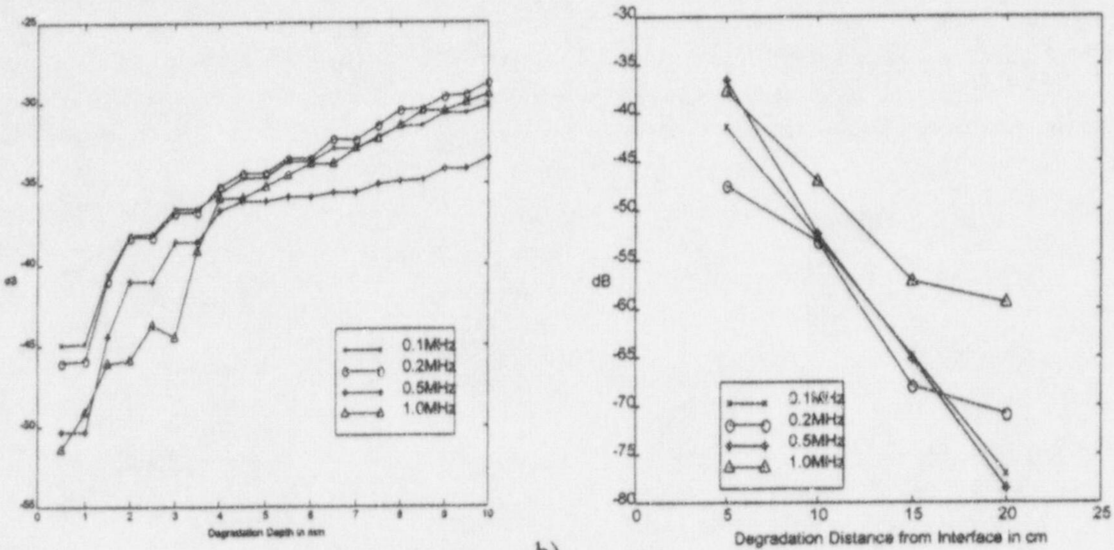
FIGURE 1

The Range-Dependent version of the OASES code (RD-OASES) from MIT was used to calculate the acoustic field in this stratified elastic scenario. The problem is broken up into vertical sectors, each characterized by an arbitrary number of fluid or solid elastic horizontal layers. The field within each sector is calculated for a single frequency via an exact, full-elastic wavenumber integration. The field in each sector is propagated to the next by a virtual array of point sources. A single scatter model is used, i.e. the field is calculated once forward through the sectors and then once backward. Thus, the received field due to an arbitrary source array can be calculated at arbitrary depth and range positions.

A 2-D model of the embedded steel containment scenario is shown in Figure 1b. A nominal 1" thick steel layer, which represents the containment, is surrounded by air halfspaces left of the midpoint and by concrete halfspaces right of the midpoint. Degradation of the steel is represented by a 2-sided 10cm long notch. Its thickness, T, and distance from the interface, dx, are parameters of the simulation. A nominal 1" long source array in a "wedge" material, modeled as a fluid, represents the ultrasonic sensor. The array is steered down to couple to a 45 degree shear wave in the steel for frequencies of currently available commercial sensors. The model assumes a plane geometry, i.e. all layers extend homogeneously and infinitely out of plane. Normal stress was computed.

In order to determine the dependence of the field on frequency and depth of the degradation, the model of Figure 1b was run parametrically. Frequencies of 0.1, 0.2, 0.5 and 1MHz were run for a two-sided degradation depths ranging from 0.5mm to 10mm incremented by 0.5mm. Figure 2a shows a plot of Signal Level in dB versus degradation depth in mm for a family of frequency curves, where Signal

Level represents the backscattered level seen by sensor located at 0.3m due to unit normal stress input at that position. The curves all have a distinct rise at 2mm. This indicates that a practical system will be able to distinguish between small "surface irregularities" in the 0.5 to 1mm range and a true degradation in the 2-10mm range. The dependence over the 4-10mm range is small, only about 10dB, which implies that careful calibration will have to be done to determine degradation depth accurately. These results are for the sharp-sided notch. Other research suggests that the pitting and shape of the degradation can have a significant effect on the backscatter level. This simulation indicates a "best case" backscatter scenario.



a) b)

FIGURE 2

The location of the degradation from the interface is a significant parameter in the problem. This determines how much concrete a practical system could penetrate. In order to determine the dependence of the field on this penetration depth, the model of Figure 1b was run parametrically. Frequencies of 0.1, 0.2, 0.5 and 1MHz were run for a two sided, 4mm deep degradation located a 5, 10, 15 and 20cm from the interface. Figure 2b shows a plot of Signal level in dB versus Penetration depth in cm for a family of frequency curves. The curves show a very high attenuation of the received signal with distance. The concrete adds 3-4dB of two-way loss per centimeter of concrete penetrated. This will severely limit the penetration capability of any practical system. Since this dependence is much greater than that seen with degradation thickness, this may limit a system's ability to effectively map degradation depths versus range. Calibration of such a system will be difficult because of this high loss. For example, the signal return from a 4mm degradation 7cm away is equivalent to the signal return from a 8mm degradation 5cm away. Other studies, not shown here, indicate that backscatter strength is not very sensitive to frequency or angle of transducer (<5dB) for the sharp-edge degradations examined here. This may not be true for tapered degradations, higher losses should be expected.

Acknowledgements

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The Halden Reactor Irradiation Facilities and Equipment

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The Halden Reactor Project can make available a wide range of versatile test rig designs to be irradiated under HBWR conditions or in light water loop systems with prototypic LWR thermal/hydraulic conditions and water chemistries. Refabrication and re-instrumentation equipment have been developed and are applied to study performance of fuel segments and materials pre-irradiated in commercial reactors. The irradiation programmes are complemented by non-destructive inspections at the reactor site and post-irradiation examinations at the hot cell laboratories of Institutt for energiteknikk at Kjeller, Norway.

An overview is given of test rig designs, in-core instrumentation, experimental techniques in use and interim inspection and PIE capabilities available for the fuel and materials test programme at Halden.

The OECD Halden Reactor Project Fuels Testing Programme - Methods, Results and Plans

W. Wiesenack

OECD Halden Reactor Project

The fuels testing programme conducted in the Halden reactor (HBWR) is aimed at providing data for a mechanistic understanding of phenomena which may affect fuel performance and safety parameters. It is based on more than thirty years of experience and the development of reliable in-core instrumentation, versatile irradiation rigs and loop systems for the simulation of light water reactor conditions.

The fuels performance studies focus on implications of high burnup. The instrumentation typically allows to assess thermal property changes as function of burnup, fission gas release as influenced by power level and operation mode, fuel swelling, and pellet-clad interaction. Relevant burnup levels (> 50 MWd/kgU) are provided through long term irradiation in the HBWR and through utilisation of re-instrumented fuel segments originating from commercial light water reactors. While UO_2 fuels still represent the majority of the test materials, other variants such as mixed oxide and Gd-bearing fuel receive increasing attention.

The thermal behaviour of urania fuel as function of burnup has been investigated with a number of experiments which constitute a data base for the assessment of UO_2 conductivity degradation. The derived modification of UO_2 thermal conductivity is suitable for the explanation of temperatures measured in re-instrumented BWR fuel segments which have been further irradiated in the Halden reactor.

Various aspects of fission gas release are investigated with a number of experiments. The paper provides examples of release behaviour during normal operation as function of burnup and grain size, data to establish the threshold for appreciable release (> 1%) at high burnup, and an assessment of the contribution of gas release from the rim.

Regulations usually require that rod overpressure due to fission gas release does not lead to increased fuel temperatures due to clad lift-off and opening of the fuel-clad gap. The Halden Project is therefore conducting experiments to assess the cladding creep behaviour at different stress levels and to establish the overpressure below which the combination of fuel swelling and cladding creep does not cause increasing fuel temperatures.

Pellet-clad mechanical interaction (PCMI) is manifested with clad elongation measurements, and data originating from re-instrumented high burnup fuel are shown. The measurements provide information on the strain during a power increase, the relaxation behaviour, and the extent of a possible ratcheting effect during consecutive start-ups.

Further investigations as indicated above are foreseen in the current and next programme period from 2000 to 2002. It is envisaged that the behaviour of mixed oxide fuel, Gd-bearing fuel and other variants developed in conjunction with burnup extension programmes will be studied. To this end, fuel segments irradiated in light water reactors and having reached high exposure have been procured. Some of them will undergo a burnup extension in the HBWR to reach burnups not yet achieved in LWRs, while others will be re-instrumented and tested for a shorter duration only.

Achievements and Further Plans for the OECD Halden Reactor Project Materials Programme

T.M. Karlsen

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The materials programme at Halden is aimed at addressing the effects of operating conditions and water chemistry variables on core materials behaviour, particularly as related to zircaloy cladding corrosion and Irradiation Assisted Stress Corrosion Cracking (IASCC), the materials degradation phenomenon which affects the structural integrity of stainless steel and nickel based components. The aim of the experimental work is to improve the understanding of materials ageing processes, to demonstrate the benefits of mitigation measures and to evaluate cladding corrosion behaviour with extended cycle operation and increased burn-ups. The studies are performed in loops which simulate light water reactor environments in terms of thermal-hydraulic, radiation and water chemistry conditions.

In the cladding corrosion studies, several of which have been performed in PWR facilities, the separate effects of parameters such as high lithium concentrations, cladding stress, heat flux and hydriding on oxidation rates in high and low tin cladding materials have been assessed. In an investigation currently in progress, fuelled test segments, prepared from modern cladding materials in the irradiated and un-irradiated condition are being exposed to representative PWR operating conditions. A range of unfuelled coupon specimens are also installed in the facility. The test, which has been designed to enable corrosion behaviour with increasing burn-up and fluence accumulation to be evaluated, is to be run with high heat loads and appreciable subcooled boiling. Plans for future cladding corrosion studies include evaluation of the effects of water chemistry variables on the deposition and re-dissolution of crud on cladding surfaces. The planned investigation may be extended to encompass also the effects of water chemistry on radiation build-up and cobalt deposition on primary circuit materials.

The IASCC experiments have typically been conducted in loop facilities operating under representative BWR conditions. In the series of crack growth studies that have been performed at Halden to date, several objectives have been addressed. In the first investigations, the use of instrumented, wedge-loaded Double Cantilever Beam (DCB) specimens a in-pile crack growth sensors and the benefits of hydrogen water chemistry (HWC) in controlling cracking in both thermally and radiation sensitised stainless steels were demonstrated. In a follow-on qualification study, actively loaded DCB and Compact Tension (CT) specimens were developed for crack growth versus stress intensity experiments. The specimens were equipped with pressurised bellows which enabled on-line control and variation of applied load. Bellows loaded CT specimens containing welded-in inserts of irradiated material (with fluences ranging from 1 to 9×10^{21} n/cm^2) were employed in a recently completed study in which in-pile crack growth rates were measured as a function of applied stress intensity and normal versus hydrogen water chemistry conditions.

Reconstituted CT specimens (prepared from irradiated core component materials and other suitable sources) will continue to be utilised in new investigations where the main objectives will be to produce reliable data on rates of crack growth in high fluence materials exposed to normal and hydrogen water chemistry conditions. In addition, the benefits of noble metal or other coatings in mitigating IASCC will be determined and it is envisaged future crack growth studies will be extended to incorporate exposure in PWR environments.

Achievements and Further Plans for the OECD Halden Reactor Project Man-Machine Systems Programme

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The development efforts in the OECD Halden Reactor Project's Man-Machine Systems (MMS) area were initiated on the basis of the experience gained through the Halden reactor dynamics experiments and the use of in-core instrumentation. Initially, efforts were spent on practical demonstrations of advanced concepts for closed loop control, and core power distributions and plant load-follow control were successfully demonstrated on the Halden reactor. Subsequently, a shift of priority towards human factors based designs, utilising an experimental control room linked to a full scope PWR simulator took place. In the later periods, issues related to upgrading the instrumentation and control systems for the purpose of increased safety and improved operation have been given increased attention. The activities have thus been focused on providing information supporting the design and licensing of upgraded, computer-based control rooms and to demonstrate improvements through system validation experiments carried out in the Projects experimental control room HAMMLAB, as well as in pilot installations in nuclear power plants. A process is soon completed to provide this facility with new full scope simulators for BWR, PWR and VVER reactors, complemented with a Virtual Reality laboratory.

The current research programme comprises four main areas:

- The work on *man-machine interaction research* aims at providing knowledge about the capabilities and limitations of the human operator in a computerised control room environment. Experiments are carried out to assess operator performance in varying plant conditions and in presence of various support systems. Important issues which are addressed include mechanisms affecting human error and methods for its prediction; effects of alarm processing and presentation; effects of level of automation on operator performance as well as studies of operator performance at night.

- The work on *experimental control rooms* investigates how modern technology can be taken into use to improve plant safety and efficiency. Here, experimentai, but still realistic control room prototypes are built and evaluated providing useful results which can be used when upgrading existing control rooms. An important part of the programme is directed at Information Presentation Methods where issues related to navigation, overview displays, and new forms for displays are investigated.

- The activities on *plant surveillance and operations systems* investigates the potentials for improved plant operation through implementation of new methods and systems for plant surveillance. Systems developed at Halden are made in such a manner that they easily can be taken over by the member organisations. Among systems developed at Halden are tools for signal validation, reactor core monitoring and predictive simulators, advanced alarm systems, computerised procedures and systems for fault diagnosis when plant disturbances occur.

- The programme on *enhancement and assessment of system quality* addresses the issues of how to develop high quality software systems, with particular weight on methods which are relevant to safety. The work includes development and investigation of methods and tools for analysing and assessing existing software with respect to various quality aspects. Particular emphasis is placed on use of formal methods in safety critical software development. Furthermore, verification and validation methods, such as program analysis and testing methods, are investigated.

Since the current fleet of commercial nuclear power plants will continue to operate for many more years an emerging high priority issue is how to safely upgrade and modernise the existing control rooms. In this context we at the Halden Project believe that there will be increasing needs for human factors experiments and control room studies and we have accordingly prepared the experimental facility HAMMLAB for such experiments through the upgrade project HAMMLAB-2000. Through experiments and demonstrations in the highly versatile new facility the Halden Project will continue to provide answers to emerging questions in the human factors field.

The full paper will give an overview of current activities and achievements as well as an overview of the future planned Man-Machine Systems activities at the OECD Halden Reactor Project.

Overview and Results from the Human Error Analysis Project 1997-1998. Performance Measures, Complexity profiling, and Initial Results from the first Main Experiment.

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The ongoing Human Error Analysis Project (HEAP) was initiated within the OECD Halden Reactor Project in 1994. Its objectives are to develop a better understanding and explicit model of how and why 'cognitive errors' occur, and to provide design guidance to avoid, or compensate for, cognitive errors.

During 1994-1996, results lead to practical insights concerning, diagnostic strategies and styles, verbal protocol analysis, aspects of operator support systems affecting diagnosis, and success rates for diagnosis and error types. From 1996 the project's scope was extended to consider error prediction, error recovery, method development, and investigation of complexity.

HEAP has investigated human error within complex simulator based experiments using the Cognitive Reliability and Error Analysis Method (CREAM). Theoretical and methodological work has also been carried out in the area of operator error recovery. Preliminary results from both areas will be presented.

Extensive method development has also been carried out within HEAP, to allow the study of operator cognitive activity within realistic situations. New measures for operator performance, situational awareness, workload, and plant performance, have been developed and will be briefly described.

Understanding of what makes a control room situation difficult to handle is also important when studying operator performance with respect to both prediction, and improvement of the human performance. Therefore, HEAP has been investigating complexity of the operator's work situation. From these investigations a definition and measures of complexity, are being developed. A Complexity Profiling Questionnaire has been developed, based on factor analytic results from operators' conception of complexity. Initial validity of a set of identified complexity factors has been shown, by prediction of both crew and plant performance from ratings of the complexity of scenarios.

The full paper will provide an overview of these activities, initial results and findings, and their potential applicability.

Prospects on Combining Software Quality Assurance Techniques

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The effectiveness of software quality assurance largely depends on the success of combining complementary techniques. This is a frequent observation both in research and practise, but there is still a lack of consensus on the usefulness of the various combinations. In many instances, even the effectiveness of a single technique is difficult to measure.

On basis of several research activities at the Halden Project, the paper discusses some of the options available for combining software quality assurance techniques. While the various options reflect a wide variety of techniques, several classes of combinations involve the use of formal methods. In particular, the combination of graphical and textual notations represents a promising approach to making formal specifications comprehensible to a wider group of users. The paper presents research results related to the combination of Petri nets and algebraic specifications, as well as to the development of a graphical front-end to the editing of algebraic specifications.

The practicality of formal methods is further enhanced by a proper combination of formal methods with traditional development techniques. The paper presents results from a research project concerned with using, and measuring the effect of, formal methods in real-life software development.

A third class of combinations concerns the use of complementary formal verification techniques. A distinction is made between theorem proving and model-checking, both of which represent well established approaches to the verification of software specifications. On basis of the advantages and shortcomings of the respective techniques, consideration is given to the possibilities of a combined approach.

Finally, there are several options available for the combination of program testing techniques. While formal verification is concerned with proving correctness of a specification, testing normally focuses on the (possibly symbolic) execution of a specification or a program. The paper discusses how testing can be made more focused if being guided by the PIE-technique, which is a dynamic failure-based technique for performing program sensitivity and testability analysis.

Several of the options for combining software quality assurance techniques relate to the problem of integrating software process- and product quality. While process-oriented quality assurance focuses on the underlying process in the software development, product-oriented quality assurance emphasises the testing and evaluation of the final program. Due to their inherent limitations, neither of the these two classical approaches appears to be fully satisfactory in ensuring adequate software quality. The paper demonstrates how this problem in particular, as well as the general problem of using and measuring the effectiveness of combined techniques, relates to the need to incorporate a variety of factors into quality assurance or assessment. It is demonstrated how these factors, which represent disparate evidences of software quality, can be combined in quality assessment by means of so-called Bayesian networks.

Human Factors Engineering and Control Room Design Using a Virtual Reality based Tool for Design, Test and Training

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The Halden Project has in co-operation with a Swedish utility developed a user-centred approach to control room design. The main new features of this approach are the development of a control room philosophy, the application of Virtual Reality technology in the design process, integration of a special-purpose tool called the Design Documentation System and the use of a process display prototyping tool called Picasso-3.

The control room philosophy is meant to prepare the ground when developing a new control centre concept. It is based on functional and operational requirements. The philosophy identifies the roles of the operators, and all functional requirements put on the control centre including support facilities and infrastructure. The control room philosophy is viewed as a functional requirement specification, which later guides the design, development and selection of control room layout, systems and technologies.

The use of Virtual Reality (VR) technology is introduced to enable the end-users of the control room design (the control room operators) to easily participate and in a straight forward manner specify their preferred control room design. Console instruments can be animated, and interactive simulations of interesting control room scenarios can be performed for evaluation purposes. The VR design tool substitutes the need to build physical mock-up's to test and evaluate a proposed design.

The Design Documentation System (DDS) is integrated with the VR design tool. The purpose is to assist control room operators in keeping to a structured and well documented approach during the design process. The final outcome is a fully documented and archived design process and a complete design specification which can be exported to vendors as CAD files. Support for interactive design validation against guidelines such as NUREG-0700, is also possible.

The optional use of the integrated process display prototyping tool Picasso-3 enables the design team to develop and test layout of mimic displays, interaction dialogues, navigation methods and presentation styles in the new control room. Picasso-3 can be integrated to the Virtual mock-up.

The reason for developing the VR based Design, Test and Training tool (VR DT&T) is the positive experience gained with early end-user involvement in the design process. By taking advantage of existing documentation as well as operational knowledge and experience it is believed that the final design is made more robust and relevant than it otherwise ever would be.

The methodology is currently applied in a large modernisation program at the Swedish Oskarshamn BWR plant. The full paper will describe the approach in more detail, emphasising the steps in the design process, the content of the control room philosophy and finally the practical use of the VR tool.

Full Range Signal Validation of PWR Plant Data
and Fast Transient Classification applied in Alarm Handling
Using Neural-Fuzzy Models

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The surveillance and control of any industrial plant is based on the readings of a set of sensors. Their reliable functioning is essential since the output from the sensors provide the only objective information about the state of the process. The signal validation task is to confirm whether the sensors are functioning properly.

Real-time process signal validation is an application field where the use of fuzzy logic and artificial neural networks can improve the diagnosis of faulty sensors or drift in sensor readings in a robust and reliable way.

The present work describes the transient and steady state on-line validation method of plant process signals using artificial neural nets (ANN) and fuzzy logic pattern recognition. The use of ANN's for signal validation has several advantages. The most important are - it is not necessary to define the physical model of the monitored process - and properly trained ANN's are less sensitive to the measurement noise than the model based techniques.

The signal validation model is based on a set of ANN's each driven by a pattern recognition algorithm. A classifier, based on fuzzy and possibilistic clustering techniques, identifies the incoming signal pattern (a snapshot of process signals) as a member of one particular cluster from a set of clusters which cover the entire operating range represented by the possible combinations of steady state and transient values. Each cluster is associated with one ANN previously trained only with data belonging to this cluster. During the real time operation the classifier provides an automatic switch-over mechanism to allow the best tuned ANN to be used.

This method has been developed at the OECD Halden Reactor Project and tested on simulated scenarios covering the whole range of PWR operational conditions provided by Electricite' De France (EDF) and the Centre D'Etudes De Cadarache (CEA) in France. Both the method and the results from the referenced study will be presented in the full paper.

Events and faults in nuclear power plants can set off transients which subsequently can activate a large number of alarms presented in a rapid sequence to the operators. A robust method to suppress less important alarms has been sought for a long time. Starting from the work outlined above the Halden Project has developed a ANN based system performing a fast classification of the occurring transient, then providing this information as input to an alarm handling system which again applies this information to perform event-driven alarm suppression. The full paper will report on the first phase of this project including a description of the method and the prototype system.

Discussion of Comments from a Peer Review of A Technique for Human Event Analysis (ATHEANA)[1]

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Ann Ramey-Smith, US Nuclear Regulatory Commission
Dennis C. Bley, Buttonwood Consulting, Inc.
Alan M. Kolaczkowski and Susan E. Cooper, Science Applications International Corp.
John Wreathall, John Wreathall & Co.

In May of 1998, a technical basis and implementation guidelines document for A Technique for Human Event Analysis (ATHEANA) was issued as a draft report for public comment (NUREG-1624). In conjunction with the release of the draft NUREG, a peer review of the method, its documentation, and the results of an initial test of the method was held over a two-day period in Seattle, Washington, in June of 1998. Four internationally-known and respected experts in human reliability analysis (HRA) were selected to serve as the peer reviewers and were paid for their services. In addition, approximately 20 other individuals with an interest in HRA and ATHEANA also attended the peer review meeting and were invited to provide comments. The peer review team was asked to comment on any aspect of the method or the report in which improvements could be made and to discuss its strengths and weaknesses. They were asked to focus on two major aspects:

(1) The soundness of the philosophy underlying ATHEANA. Are the basic premises on solid ground and is the conceptual basis adequate?

(2) Is the ATHEANA implementation process adequate given the description of the intended users in the documentation? Assuming the technical basis is adequate, is the guidance for conducting the search and quantification processes and for integrating the results into the PRA adequate, e.g., clear, effective, usable?

The four peer reviewers asked questions and provided oral comments during the peer review meeting. They also provided written comments approximately two weeks after the completion of the meeting.

Detailed comments from the peer reviewers addressed the strengths and weaknesses of many aspects of the methodology including:

- use of terminology and organization of the document,
- use and effectiveness of the underlying psychological models and error taxonomy,
- use of retrospective analysis and the associated database of human-system events,
- treatment of performance shaping factors (PSFs),
- usefulness and adequacy of the search process for identifying potentially important unsafe human actions and their error-forcing contexts (EFCs),
- and the usefulness and adequacy of the ATHEANA quantification process.

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

All of the reviewers thought the ATHEANA method had made significant contributions to the field of PRA/HRA, in particular by addressing the most important open questions and issues in HRA, by attempting to develop an integrated approach, and by developing a framework capable of identifying types of unsafe actions that generally have not been considered using existing methods. The reviewers had many concerns about specific aspects of the methodology and made many recommendations for ways to improve and extend the method, and to make its application more cost effective and useful to PRA in general. Details of the reviewers' comments and the ATHEANA team's responses to specific criticisms will be discussed.

METHODOLOGICAL AND APPLICATIONS ISSUES IN FIRE RISK ASSESSMENT*

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Summary

This paper discusses two of the current fire risk assessment (FRA) research activities being conducted by the Office of Nuclear Regulatory Research (RES) of the U.S. Nuclear Regulatory Commission (NRC). These activities involve: (1) the analysis of the likelihood and failure modes of fire-induced circuit failures and (2) the assessment of model uncertainties in the predictions of available fire models. These topics are of interest because they will support the development and use of risk-informed, performance-based fire protection approaches through improvement of FRA methods and tools and because they illustrate the wide range of methodological and applications issues being addressed by the NRC/RES program.

Tools for Circuit Failure Mode and Likelihood Analysis

An FRA needs to specify both the likelihood of fire-induced equipment failure and the effect of the failure on plant systems. The equipment of most interest to FRA is electrical cables. Depending on the mode of cable failure assumed, the fault effects could include loss of equipment function, false instrument and control indications, or spurious equipment actuations. The latter two effects are typically assumed to result from "hot shorts," i.e., ungrounded short circuits involving powered conductors or between multiple conductors in an instrument or control cable. A number of FRAs have shown hot shorts to be significant direct and indirect risk contributors. Direct contributions typically derive from spurious opening of primary system valves leading to flow diversion or loss of coolant. Indirect contributions typically derive lower success probabilities assumed for post-fire recovery actions that might be made significantly more complicated when the hot short potential is considered.

Current FRAs are generally limited to simplistic hot short analyses. The assumed conditional probability of a single hot short given a fire-induced cable fault is commonly based on reference to a "generic" probability distribution derived subjectively from limited information. The probability of multiple hot shorts is typically obtained by multiplying the single hot short conditional probability a corresponding number of times. This approach does not explicitly address potentially important issues such as the circuit design, the function of the cable, the cable design, the nature of the postulated fault, and the number and characteristics of other cables in the same raceway. Further, the latter procedure ignores dependencies, both aleatory and epistemic. A second concern with current FRAs is that they typically do not address the full set of effects that may result from a fire-induced circuit fault. They usually do not consider potential hot-short effects such as the simulated closure of a switch or erroneous control or instrumentation signals nor do they propagate the electrical faults caused by the cable damage through the affected electrical

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

circuit(s) to determine if additional faults (or even fires) occur. If fault propagation is considered, it is not done probabilistically and is generally limited to an assumption that failure of a power cable might de-energize the associated power bus back to the first in-line circuit protection device (fuse or breaker).

Recognizing that the specific response of a circuit to fire-induced cable damage will vary with circuit design, the focus of the current NRC effort is on the development of input needed to perform a probabilistic systems analysis. The objectives of the work are: a) to develop an improved understanding of the mechanisms linking fire-induced cable damage to potentially risk significant failure modes of power, control, and instrumentation circuits; b) to develop improved methods and data for estimating the conditional probabilities of key circuit faults given damage to one or more cables; and c) to develop sample estimates of the conditional probabilities of key circuit failure modes applicable to currently operating U.S. nuclear power plants. The estimation process will include an identification and quantification of the key uncertainties in the estimates; a) to gain risk insights concerning fire-induced circuit failures, especially those associated with cable hot shorts; and b) to identify areas where additional work needs to be done to improve understanding of the risk associated with fire-induced circuit failures. This work represents the first phase of a detailed study of the issue. The results of this analysis will be used to determine where additional work is needed (possibly including experiments).

Output Model Uncertainty: Framework, Methodology and Tools

Whenever model predictions are used to support decision making, it is desirable that the uncertainties in these predictions be explicitly quantified. Methods for estimating "output parameter uncertainty," i.e., uncertainty in the model output due to uncertainties in the values of model input parameters (often referred to as "parameter uncertainty"), are well known and routinely applied in many situations. On the other hand, there currently is no consensus concerning formal methods for estimating "output model uncertainty," i.e., the additional output uncertainty due to approximations inherent in a given model (often referred to as "model uncertainty").

In FRA applications, simulation codes are available to predict the dynamic behavior of variables that are, in principle, measurable. Furthermore, limited amounts of experimental data useful for estimating output model uncertainty are available. However, estimation is not straightforward because: a) the experiments do not cover all possible situations to which the model will be applied; b) the values of the model parameters needed to simulate the experiments may not be well known; and c) there currently is a controversy as to how model uncertainty (and therefore output model uncertainty) should be precisely defined and quantified. According to one school of thought, the probability that a given model is "valid" or "acceptable" needs to be estimated. According to another school of thought, estimation of the conditional probability that an output variable takes on a value in a given range, given the prediction of a model, is key.

Some simple approaches for quantifying uncertainty in model predictions in the presence of model uncertainty and parameter uncertainty have been proposed. However, these approaches have not been fully tested, nor has their relationship with the different schools of thought mentioned above been investigated.

The objective of the initial NRC work in this area is to develop a theoretically sound, practical methodology for quantifying the uncertainty in model predictions which accounts for data sparseness, model uncertainty, and parameter uncertainty. Subsequent work will, using relevant data, apply the methodology to fire models and will develop software for automating this application.

Incorporating Aging Effects into Probabilistic Risk Assessment — Program Overview

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INTRODUCTION

The operation of complex systems, such as light water nuclear reactors, over long periods of time (e.g., 20-40 years) invites the potential of age-related degradation and a reduction of the strength of passive components in light water reactors. The U.S. Nuclear Regulatory Commission (NRC) sponsored the nuclear power plant aging research (NPAR) program during 1985-1994 to gather information about nuclear power plant aging. This program collected a large body of information, mainly *qualitative*, on plant aging and its potential effects on plant safety. Incorporating this body of knowledge into modern risk assessment techniques such as probabilistic risk assessment (PRA) has been envisioned as an effective and systematic method to assess the impact on plant risk resulting from aging of SSCs (systems, structures, and components). However, this body of knowledge had not yet been formally integrated into modern risk assessment techniques.

A number of age-related degradation mechanisms [e.g., fatigue, irradiation embrittlement, stress corrosion cracking and flow-accelerated corrosion (FAC), not fully accounted for in the original design, have caused failures and raised questions about the safety of older nuclear power plants. To better capture these age-related issues for plant safety, the methods development and analysis presented in this paper utilize PRA techniques and models. Consequently, the process will allow for modeling of aging of passive components in a PRA so that the physical effect of aging on core damage frequency can be estimated.

Traditional PRAs, in general, do not include passive SSCs in its models. Passive SSCs which may be affected by aging are nominally much more reliable than the active components modeled in PRAs. But, these aging effects may make the nominally reliable passive SSCs less reliable as time progresses. A case in point is FAC. A particular FAC problem area is the wall thinning of carbon steel pipe, a passive component. FAC in carbon steel pipe systems is characterized by the simultaneous dissolution of iron from the iron oxide fluid interface and the formation of an iron oxide film at the oxide metal interface. Bulk flow plays a vital role in providing a sink for dissolution.

APPROACH

Aging issues such as IGSCC and FAC are complex, multi-parameter phenomenon, and the susceptibility of a given site or SSC cannot be determined by considering only a few parameters. Even though aging models can predict observed results fairly well in some cases, such calculations are subject to large uncertainty. Since the development of detailed aging models based on reliability physics is still in its infancy, approximate models that modify failure rates directly have been proposed. For example, the linear failure rate model assumes that the total failure rate, $\lambda(t)$, is the sum of random failures, λ_0, and

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[1]MIT collaborator.

aging, αt. This assumption of linearity in time has been questioned; in fact, it is shown in the NUREG/CR-6157 report that the failure mechanisms discussed earlier do not necessarily lead to failure rates that are linear in time.

A general drawback of these approximate aging models is that it essentially represents a parametric approximation, made at a relatively high level, of failure processes and mechanisms that usually have a rather complex physical behavior. Thus, although a particular parametric reliability distribution may adequately fit the available failure data over a specified region, it will nevertheless always constitute a drastic simplification when examined from the point of view of the underlying physical phenomena. The obvious danger under these conditions is that a careless or superficial choice and application of parametric reliability models may obscure the understanding of the role played by important physical processes and may inhibit the management of aging.

The modeling technique used is an extension of modern PRA techniques. Physical aspects of aging mechanisms are incorporated into the PRA model using basic events contained in fault or event trees. As part of this incorporation of physical aging models into the PRA, it may be necessary to augment the existing fault or event trees to encompass the additional aging mechanisms. For example, if sections of piping are susceptible to FAC, the system fault trees containing these sections of piping may need to be modified to account for the pipe segments. Once the pipe segments are incorporated into the PRA, the physical aging model representing FAC could be "tied" to the pipe segment basic events, thereby allowing an analyst or regulator access to physical parameters (e.g., fluid velocity, steam quality, temperatures, pH) that drive the FAC phenomenon. Consequently, having the physical process incorporated directly into the PRA yields risk insights based upon the aging process rather than an abstraction of failure data into a statistical probability parameter.

To accomplish the incorporation of the physical process into the PRA, a stress-strength interference (or reliability physics) model based on physical FAC phenomena is utilized. We developed load and capacity (or stress and strength, respectively) probabilistic distribution using the KWU-Kastner-Riedle (KWU-KR) FAC model. The uncertainties associated with both the parameters and the model of the KWU-KR are assessed. An adjustment factor E is introduced to account for the uncertainty in the model.

Then, we applied the SAPHIRE PRA software with the Surry IPE to model select pipe segments of the Surry nuclear power plant. We calculated impacts on the core damage probability (CDP) due to FAC for a 10-year in-service-inspection period of time. As part of this calculation, sensitivity and uncertainty evaluations were performed.

Preliminary results find that the core damage probability due to FAC for 10 years has an impact (an increase of 31%) when compared with the 10-year failure probability due to the nominal transient initiating event from a loss of main feedwater. However, the CDP due to FAC has a smaller impact (less than 1%) when compared with the 10-year overall nominal CDP. Nonetheless, this investigation leads us to believe that modeling of additional passive SSCs could potentially have a non-negligible impact on overall CDP. Further, nuclear power plants that have a risk profile where feedwater transients are more important to overall risk could see very significant impacts from aging effects such as FAC. In addition, preliminary results indicate that attempts to model aging mechanism using the linear failure rate model are suspect.

ACCIDENT SEQUENCE PRECURSOR PROGRAM
LARGE EARLY RELEASE FREQUENCY MODEL DEVELOPMENT

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SUMMARY

The U.S. Nuclear Regulatory Commission's (NRC) Accident Sequence Precursor (ASP) program was initiated by the Office of Nuclear Regulatory Research (RES) to provide a probabilistic method for reviewing operational experience to determine and assess both known and previously unrecognized vulnerabilities that could lead to core damage accidents. The ASP program is currently implemented by the Office of Analysis and Evaluation of Operational Data (AEOD). Standardized Level 1 plant models developed using the SAPHIRE computer code[1] are used to assess the conditional core damage probability for operational events during full power operation occurring in commercial nuclear power plants (NPP). Because not all accident sequences leading to core damage will result in the same radiological consequences, in 1995 work was begun to study the feasibility of developing simplified Level 2/3 models using SAPHIRE to estimate the radiological consequences associated with the radioactive release to the environment resulting from the Level 1 core damage scenarios. Once developed, these simplified Level 2/3 models were linked to the Level 1 models to provide risk perspectives for operational events. Ten Level 2/3 prototype models were completed to represent the various pressurized water reactor (PWR) and boiling water reactor (BWR) nuclear steam supply systems and containment types.

During the development of the simplified Level 2/3 models, several deficiencies were identified with the prototype models. These deficiencies were primarily the result of the simplification process necessary to develop these models. In addition, in 1997 the NRC began moving towards using large early release frequency (LEAF) as a surrogate for the early fatality quantitative health objectives [refer to Draft Guide (DG)-1061 (future Regulatory Guide (RG)-1.174) *An Approach for Using Probabilistic Risk Assessment in Risk-informed Decisions on Plant Specific Changes to the Current Licensing Basis*]. As outlined in DG-1061, LEAF is defined as the frequency of a radiological release occurring prior to an off-site evacuation and having a large enough source term that may result in early-term fatalities. The focus of the Level 2/3 ASP project was then modified to address LEAF and to correct the deficiencies of the simplified Level 2/3 models. The development of LEAF models is also thought to be a more useful and practical approach for use in event assessment and the AEOD ASP program.

In the spring of 1998, work was begun to develop detailed LEAF models for the ten plant classes established during the Level 2/3 model development work. The LEAF models, as well as the previously developed Level 2/3 models, are based on the accident progression event trees (APETs) developed during the NUREG-1150 analyses.[2] Because these models will be "detailed" rather than "simple", it is envisioned that the initial ten plant models will be capable of being modified to represent any NPP within each plant class given sufficient plant-specific data. Modification of the simplified Level 2/3 models to represent other NPPs could not be readily performed.

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NRC SUPPORT FOR THE KALININ (VVER) PROBABILISTIC RISK ASSESSMENT*

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At the Lisbon Conference on Assistance to the Nuclear Safety Initiative, held in May 1992, it was agreed that special efforts should be undertaken to improve the safety of the nuclear power plants designed and built by the former Soviet Union. As a result, the U.S. Nuclear Regulatory Commission (NRC) and the Federal Nuclear and Radiation Safety Authority of the Russian Federation (GAN) agreed to (among other things) work together to carry out a probabilistic risk assessment (PRA) of a VVER-1000 reactor in the Russian Federation (R.F.). This was a recognition by both NRC and GAN that this technology has had a profound effect on the discipline of nuclear reactor safety in the West and that the technology should be transferred to others so that it can be applied to Soviet-designed plants. Unit 1 at the Kalinin Nuclear Power Station (KNPS) was chosen for the PRA, and the effort was carried out under the auspices of GAN with the assistance of several other Russian organizations.**

The NRC provided financial support for the PRA with funds from the Agency for International Development and technical support primarily through Brookhaven National Laboratory (BNL) and its subcontractors. The latter support was carried out through workshops, by documenting the methodology for doing a PRA in a set of guides, and through periodic reviews of the technical activity.

The workshops consisted of one 8-week long workshop at BNL at the start of the project, followed by 1-week workshops in Russia approximately every six weeks over a period of one and one-half years. The first workshop took place after the initial plant familiarization and information gathering. It consisted of scheduled seminars to provide training on specific technical issues (e.g., development of event sequence diagrams), independent work by the Russian PRA team with interaction with the U.S. experts as needed, and meetings with the

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

**In addition to GAN, the following organizations were involved: GAN's Scientific and Engineering Centre for Nuclear and Radiation Safety, Kalinin Nuclear Power Station, The Experimental and Design Office Gidropress, Nizhny Novgorod Project Institute Atomenergoproekt, and Rosenergoatom Consortium.

U.S. experts to review work in progress. The followup workshops were on technical subjects that enter into the analysis at later times (e.g., human reliability analysis) and subjects that needed further elucidation (e.g., common cause failure analysis).

The procedure guides complemented the workshops. The first draft of the guides was used for the Kalinin PRA, and a final version [1] is to be published to be of assistance to PRA practitioners in other countries, in particular those with VVER plants. The procedure guides are limited to accidents involving the reactor core and that occur while the plant is operating at full power. Internal initiating events, including internal fires and floods, are considered as well as seismic events. Guidance is provided for a Level 1, 2, and 3 PRA; however, the Level 3 PRA guidance is limited to offsite consequences.

It was assumed that the team carrying out the PRA would be familiar with the set of guides developed by the International Atomic Energy Agency (IAEA) for carrying out a Level 1 PRA for internal events [2]. The IAEA document represented an internationally acceptable approach. The new guides improve on the existing guides by: (1) taking into account recent work in the field, (2) considering special problems that might be specifically present for the VVER experience, and (3) improving upon the guidance already provided. The idea was not to duplicate the existing guidance found in the IAEA document or the material in other guides that have been produced by the NRC [3,4]. For subjects not well documented in the open literature (e.g., the approach taken for human reliability analysis), detailed guidance was given; for tasks where a firm understanding was already well established and documentation freely available (e.g., system modeling), minimal guidance and appropriate references were provided.

These efforts have led to the completion of the PRA for internal events. Ongoing work is being done to add results for internal fires and floods and seismic events. It is our belief that the type of assistance provided by the NRC has been instrumental in assuring a quality product and transferring important technology for use by regulators and operators of Soviet-designed reactors.

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Incorporating Maintenance Effectiveness Into PRA Applications

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Maintenance effectiveness can be defined in different alternative ways:

1. Maintenance effectiveness is the degree to which risk is controlled from component aging and deterioration.
2. Maintenance effectiveness is the degree to which reliability is maintained from component aging and deterioration, while maintenance downtime unavailability is controlled.
3. Maintenance effectiveness is the degree to which component failure rate increases from aging and deterioration are controlled.

These alternative definitions focus on the different levels of maintenance effects. The term "aging and deterioration" is used in the definitions since some people differentiate aging, which they interpret as degradation with component age, from deterioration, which they interpret as being related to time but not necessarily being correlated with component age. If aging is broadly interpreted as including any mechanism that increases the component failure rate with time then the term "deterioration" is redundant and is not needed. This broad definition is utilized for simplicity.

To incorporate maintenance effectiveness into PRA applications, component aging thus needs to be considered. This in turn means that increases in component failure rates with time need to be considered. If component failure rates didn't increase with time then there would be no need for maintenance. Present PRAs assume component failure rates are constant with time. This is adequate to obtain a snapshot of the risk at a given time or averaged over a given time period. For any other applications, present PRAs can be viewed as potentially penalizing maintenances since only maintenance downtime contributions are explicitly modeled.

To evaluate maintenance effectiveness, the benefits of maintenance in controlling aging mechanisms, as well as the maintenance downtime contribution and other possible maintenance deficiencies, need to be incorporated into PRA applications. Various approaches can be used to evaluate maintenance effectiveness. Performance level approaches do not require time-dependent component failure rate models, but instead evaluate maintenance effects on component performance and system performance. These performance level models are useful for determining the balance between maintenance's reliability effects and unavailability effects on system performance and risk performance. The NRC Maintenance Rule requires this balancing, but it generally has not been fully implemented. Performance level approaches are discussed along with their applications.

If time-dependent component failure rate models are considered then they need to be practical. They can be based on available engineering knowledge about the component and do not necessarily require detailed time-dependent failure data. Available failure and maintenance data, even though not detailed, can also be used to obtain ranges for component aging behaviors that are consistent (consonant) with the data. These aging behaviors can be used in PRA applications to determine ranges for maintenance effectiveness. Component failure rate approaches are presented in the paper which are felt to be practical. Their applications are also discussed.

The demonstrations of maintenance effectiveness evaluations that are described in the paper indicate some important findings. First of all, the demonstrations indicate that maintenance often has more effect on risk then testing does. This is a good reason why maintenance effectiveness should be incorporated in PRA applications. The demonstrations also indicate that large failure rate increases can occur between certain types of maintenances, which in turn can cause large CDF increases to occur. These large swings in risk can occur even though no overall time trend in the component failure behavior is realized. The demonstrations furthermore indicate that the performance indicators used for the Maintenance Rule generally perform poorly in identifying increasing risks from ineffective maintenances. These demonstrations are limited but indicate the need to evaluate more informative and more powerful indicators. Specific recommendations are made.

Uses of maintenance effectiveness evaluations in PRA applications are examined in the last part of the paper. Specific measures of maintenance effectiveness are also examined. These examinations indicate the useful implementations that can be carried out in improving maintenance effectiveness, in trading off maintenances with tests, and in extending the scope and benefits of risk-informed applications.

RISK COMPARISON OF PERFORMING SCHEDULED MAINTENANCE AT POWER VS. DURING SHUTDOWN

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This paper presents a comparison of the risk of performing scheduled maintenance at power operation and during various periods of shutdown. The motivation for the study comes from the fact that, for reasons of cost, the nuclear industry is attempting to maximize the time at power and reduce the length of refueling and maintenance outages, and consequently the nuclear industry is interested in increasing the amount of scheduled preventive maintenance performed during power operation. It is therefore of interest to determine the risk of performing preventive maintenance at power operation, and compare this risk to that of performing the maintenance during various times during shutdown and refueling. Risk-informed regulatory decision making requires this; work already performed shows that the risk during some periods of shutdown is not negligible, and hence one cannot estimate the risk change from performing maintenance at power operation instead of at shutdown by assuming that the risk from performing maintenance during shutdown is negligible. Two plants are studied, the Surry pressurized water reactor (PWR) and the Grand Gulf Nuclear Station (GGNS) boiling water reactor (BWR). The work on the PWR was performed at Brookhaven National Laboratory, and that on the BWR at Sandia National Laboratories. Some generalizations to other plants are made. The scope of the studies is limited to conventional internal events (including loss of offsite power, but excluding fires).

Two types of risk measures for scheduled maintenance can be defined. The first type consists of conditional risk measures, and measures the increase in risk from taking a component, or a group of components, out for maintenance given that the plant is in a particular plant operational state (POS). One calculates the risk, given the component (or group of components) is out for maintenance and subtracts the risk, given the component (or group of components) is not in maintenance, but given that one is in the given POS. The second type of risk measure multiplies these risk measures by the duration of the scheduled maintenance performed in a calendar year on the component, or group of components, and gives the increase in the annual risk if the scheduled maintenance is done in that particular operating state. This latter type of risk measure is the most relevant, for making risk-informed decisions concerning what operational state scheduled maintenance should be performed in. Both types of risk measures were calculated in the studies. The studies also assessed the uncertainty in the risk measures, but these will not be presented in the summary. In calculating the uncertainty in the risk measures, it is important to take into account the correlation between the risk in the case where the component is known to be out for maintenance and the case where it is known to not be out for maintenance. The uncertainty in the difference of these risks can be much less than the uncertainty in the base case risk, because of the correlations between the two cases.

Some highlights of the results for Surry will be discussed first. Because it was already known that the risks were very low during refueling, when the water level is high above the reactor flange, these risks were not quantified here. Instead, the focus was on cold shutdown. The results here are affected by the fact that, at Surry, the containment was assumed open during this period. Within the cold shutdown operational state, the risk can vary with the time since shutdown, since the variation in decay heat affects the timing of the accident and the time available for recovery, and since the short-lived radioisotopes decay as the time from reactor trip increases, which affects the early fatality risk. Four time windows were defined: window 1 extends from shutdown to 75 hours from shutdown; window 2 extends from 75 hours to 240 hours; window 3 extends from 240 hours to 32 days; and window 4 corresponds to times in excess of 32 days from shutdown.

The Surry plant, as modeled in the study, consists of two units, each with a dedicated diesel, and a third diesel shared between the two units. Maintenance on the Unit 1 dedicated diesel generator (DG1) has a greater risk impact (on Unit 1) than maintenance on the shared diesel generator (DG2). For DG1, the increase in the yearly core damage frequency (probability of core damage in one calendar year) is 6E-6 per year, for a seven day maintenance outage at power, while in window 4 it is 4E-7 per year, more than an order of magnitude lower than at power. The increase in expected yearly total latent fatalities from 7 days per year maintenance on DG1 at power is 2.2E-4 per year, while during time window 4 the corresponding increase is 6.7E-5 per year. For DG3, the risk impact of maintenance on Unit 1 is less, but one also has to consider the risk impact on Unit 2. Also, since it is unlikely that both units will be shut down for maintenance or refueling at the same time, the most likely risk reduction strategy would be to perform maintenance on DG3 when one unit is in time window 4 of cold shutdown or in the refueling POS, and the other unit is at power. Maintenance on diesel generators at power has a greater risk impact than maintenance on the other components studied.

The turbine-driven auxiliary feedwater pump (TD-AFW) is not used during shutdown; since seven days of maintenance per year on the TD-AFW at power results in an increase in the yearly core damage frequency of 1.6E-6 per year, and an increase in expected total latent cancer fatalities of 5E-5 per year, it may be desirable to perform maintenance on the TD-AFW during cold shutdown.

The study on Grand Gulf considered the following POSs: full-power operation (POS 0), cold shutdown (POS 5), refueling with vessel water level at the steam lines (POS 6), and refueling with the vessel flooded up to the upper containment pool and the refueling transfer tube open (POS 7). For POS 5, only the time windows before the refueling POS were considered. The components considered were the emergency diesel generators (EDGs), the standby service water (SSW) system motor-driven pumps (MDPs), the reactor core isolation cooling (RCIC) system turbine-driven pump (TDP), and the high pressure core spray (HPCS) system MDP. Maintenance on single components and selected pairs of components was considered.

The risk increase measures from performing scheduled maintenance on the single components considered were of the same order of magnitude at power operation and during cold shutdown. During refueling, however, the risk measures are one to two orders of magnitude less. For example, for EDG1, the expected increase in the yearly core damage frequency from a maintenance duration of about 4.4 days per year (.012/yr) is 3.4E-7/yr at power, 2.2E-7/yr in POS 5, 1.9E-9/yr in POS 6 and negligible in POS 7. For the same component, the increase in the expected dose to the population within 50 miles of the plant is 6.1E-4 Sv per year at power, 1.1E-3 Sv per year in POS 5, 7.3E-6 per year in POS 6, and negligible in POS 7. There appears to be no significant risk advantage to performing preventive maintenance activities on the EDGs, the SSW MDPs, or the HPCS MDP during cold shutdown instead of during power operation. There is a risk advantage to performing such maintenance during refueling, but the magnitude of the risk benefit depends on the length of the maintenance outage.

Thermal-Hydraulic Research on Passive Safety Systems for Next-Generation PWRs using ROSA/LSTF

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A thermal-hydraulic research on next generation PWRs has been conducted at JAERI using the ROSA-V Large Scale Test Facility (LSTF). The LSTF was originally built as an integral facility for the investigation of thermal hydraulic behavior occurring during accidents in the conventional Westinghouse type PWRs. The LSTF provides full height and full pressure simulation for the four loop PWR using a volumetrical scaling factor of 1/48. The facility has also been used for the ROSA/AP600 program after the addition of the passive safety components unique to the AP600. The present research on the next generation PWRs focuses not only on the AP600 but also on general thermal-hydraulic phenomena related to passive safety components proposed in literature or by JAERI for the use of next-generation PWRs. Such safety components include a gravity-driven injection system (GDIS), a primary automatic depressurization system (ADS), and a natural circulation core cooling system using a dedicated heat exchanger or steam generators (SG).

Since the above-mentioned passive safety system is generally characterized by a small driving force, a low fluid velocity, and low pressure and temperature conditions, important thermal hydraulic behavior are 1) thermal hydraulic interactions among the safety systems and the other systems (system effects), 2) thermal or two-phase stratification, i.e., multi-dimensional behavior, 3) noncondensable gas behavior including the dissolved gas in the water injected form the safety systems, and 4) natural circulation cooling at low pressure. The experimental simulation using an integral facility is considered as a best tool especially for the investigation of the system effects. The full height simulation is also beneficial for the investigation of low pressure behavior where the static head is comparable to the system pressure.

Tests have been conducted at the LSTF for this research activity, which are classified into two categories: "phenomenon-oriented tests" for the investigation of parameter effects and "design-assessment tests." The ROSA-AP600 program, conducted jointly with USNRC, is an example of the latter category. JAERI is also now negotiating with the industries for a possibility of conducting the design-assessment tests for the Japanese next-generation PWR. The tests in the former category focus on the thermal-hydraulic phenomena related to the passive safety systems which are technically interesting and seem to have impacts on the enhancement of safety, the simplification of a system, and the cost reduction for next generation PWRs.

In this paper, results of several phenomenon-oriented tests focusing on a combined use of a SG secondary-side ADS (SADS) and GDIS will be presented. The combined use of SADS and GDIS is a candidate of safety systems for some next-generation PWR designs including Mitsubishi New PWR-21. The experimental results showed several thermal-hydraulic behaviors typical to these safety systems, including the primary depressurization due to natural circulation cooling, a nonuniform flow behavior among SG U-tubes, an accumulation of the noncondensable gas originally contained in the injected water, liquid holdup in U-tubes due to the countercurrent flow limiting, and long-term passive core cooling with the GDIS injection.

From the assessment of the RELAP5/MOD3 code using the present data, it was found that the inability of the code to predict the U-tube nonuniform flow behavior resulted in overprediction of the primary depressurization rate at a pressure less than 1 MPa, and exaggerated oscillation of the natural circulation flowrate in the primary loop. The mechanism of the nonuniform behavior among U-tubes will also be discussed.

Computational Two-phase Flow Dynamics and Heat Transfer for Analysis of LWR Transients

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SUMMARY

A Computational Fluid Dynamic (CFD) model is developed to simulate the LWR transient such as dryout and reflux two-phase flow and heat transfer with phase change. The CFD model is based on the Navier-Stokes formulation in conjunction with a set of constitutive relationships for water-steam system. The model combines the high-resolution capability of the state-of-the-art CFD methods with a novel approach that allow for the tracking and delineation of the dynamic interfacial water-steam boundary. Reynolds number and Raleigh number are included in the parabolic-type governing equations to identify the flow and heat transfer regimes in water, vapor and two-phase regions. In this method the need for temporal and spatial averaging is completely eliminated and the use of closure relationships for the treatment of the interfacial transport phenomena is minimized. The geometrical void fraction in this formulation is replaced by a dynamic vapor-phase fraction that is defined based on the internal energy in a fine computational grid. The dynamic vapor-phase fraction, which is a parameter with both local and instantaneous value, plays a pivotal role in tracking the liquid-vapor interfacial boundary.

The numerical method used is based on Semi-Implicit Method for Pressure-Linked Equations with Energy-Based Iteration (SIMPLE-EBI). In this method a cyclic series of guess-and-correct operations are used to solve the governing equations. The velocity components are first calculated from the momentum equations using a guessed pressure field. The pressures and velocities are then corrected to satisfy the continuity equation. During each iteration, the water-steam interfacial boundary and the cell average internal energy are determined using energy-based equations.

The turbulent viscosity in the entire system is determined by solving the two supplementary equations (k-ε model). This requires the solution of two additional partial differential equations at each time step. Because of the stiff nonlinear terms in the ε equation, the two-equation turbulence model often causes instability. A decoupled line-by-line TDMA (Tridiagonal Matrix Algorithm) is used to solve the equations sequentially over the entire computational domain. Using the velocity scale obtained from solving the governing equation for k and a time scale obtained from a combination of k and ε, an expression for the turbulent viscosity is obtained.

An important feature of the presented interface-tracking technique is the ability to simulate the full sequence of bubble formation, growth and departure. This is achieved by adding the buoyancy and surface tension terms to the governing equations and also by adding bubble dynamic relationships.

Among many others, the model is used to simulate a case involving bulk boiling and condensation of water in a constant volume container. The results include the evolution of water-steam interface, the formation and development of the liquid film covering the side wall surface, the temperature distribution and the convection flow field. Figures 1 and 2 show the evolutions of temperature distribution and water-steam interfacial boundary unde: micro-gravity condition, respectively.

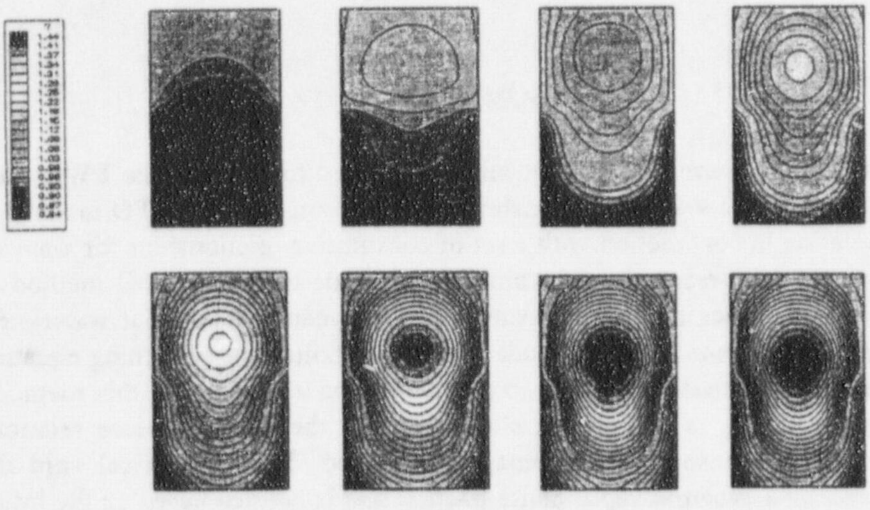

Figure1. Evolution of temperature distribution under micro-gravity condition.

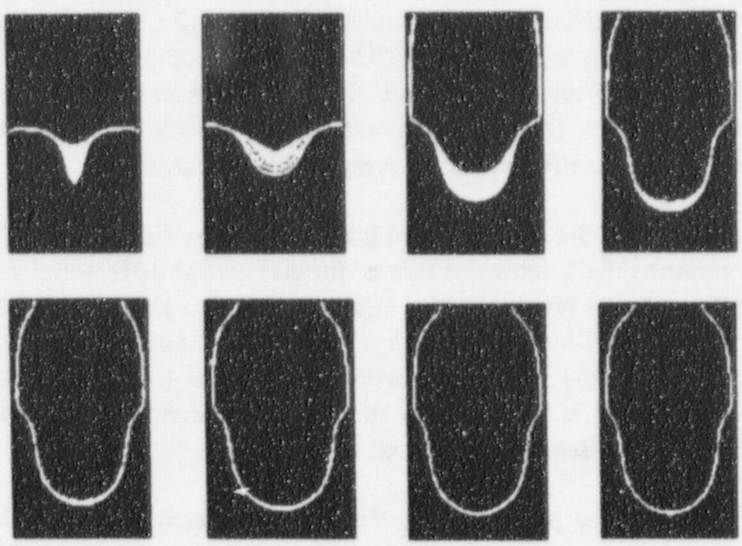

Figure 2. Evolution of water-steam interfacial boundary under micro-gravity condition.

FASTNET : proposal for a ten-year effort in thermal-hydraulic research

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The presentation is the result of a collective effort launched in 1996 by CEA and extended later to its French partners, IPSN, EDF and FRAMATOME, in order to explore new avenues of research in thermal-hydraulics in the next ten years and to propose guidelines.

The context is defined mainly by thermal-hydraulics applied to nuclear engineering. The scope of the study was broaded in 1997 by a work group of representatives of CEA, EDF, FRAMATOME and IPSN. With 57 reactors in operation providing more than 75% of the production of electricity, France will not replace the present generation of plants until 2010. The extension of life span of this generation and the preparation of the next generation are the main elements of the context. R and D in thermal-hydraulics should contribute to a better evaluation of operational and safety margins and to improved competitivity of nuclear plants. Methods are already well established and efficient. Thus a qualitative jump in the description of the physics is required for a significant improvement of an already good situation.

An evaluation of the state-of-the-art and new trends in thermal-hydraulic research was made in its three major activities : physical modeling, numerical techniques and software, experimental techniques and instrumentation. Greater computing capacities provide the opportunity to carry out more local and detailed simulations. Thus the large structures of the flow could in principle be resolved directly while the modeling supported by experimental proofs would be needed for the unresolved small scales. This would lead more universal and versatile numerical tools.

However, this is a very difficult challenge in a discipline characterized by strong non linear interactions especially in two-phase flows. This will require a long-term effort which was proposed along three lines of development :

- Simulation of single-phase turbulent flows, particularly with Large Eddy Simulation.
- Improvement of the averaged two-fluid model of two-phase flows. This would include a transport equation for the interfacial area allowing a better physical description of the transitions between flow regimes and a multifield model allowing a better description of complex two-phase separated flows. These extensions can greatly improve the one-dimensional description widely used in system codes. With additional developments on turbulence in two-phase flows, they can improve the description of three-dimensional two-phase flows.
- Simulation of local phenomena on a microscopic scale in two-phase flows in a box containing few particles or bubbles. This requires the development of techniques for interface tracking. It could

provide information to be confirmed by experiments for the modeling of closure laws of the improved two-fluid model in the short term. In the long term this could prepare the simulation of large structures in two-phase flows.

These developments in modeling will be possible only if improved instrumentation allows access to local values of quantities like interfacial area, phase velocities. The development of this instrumentation achieving these goals is thus a major part of the program. Experiments devoted to generic situations can provide the data base needed for the development or qualification of the new models.

Developments are to be shared with the international community of thermal-hydraulics, as exchanges in the past have been very fruitful for example the meetings organized in 1996 and 1997 by OCDE on Transient thermal-hydraulics and Neutronic code requirements or on Advanced Instrumentation.

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REMIX97: A Computer Program to Predict the Downcomer Fluid Temperature Transients Due to Safety Injection at Low (or Zero) Loop Flow Conditions*

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Rapid cooling of a reactor pressure vessel wall, accompanied by high coolant pressure, is referred to as pressurized thermal shock (PTS). Overcooling events are of safety concern in PWRs because the resulting high state of stress along with irradiation-induced loss of ductility and inner-surface flaws can yield through-the-wall crack propagation and reactor pressure vessel failure.

Many thermal-hydraulic aspects of overcooling events, except the thermal stratification effects, can be analyzed by using system codes, such as TRAC and RELAP. The term "thermal stratification" here means nonuniformity in temperature and density in a direction transverse to the flow path. Such thermal stratification is obtained at low (and zero) loop flow, and it is not represented in system codes currently used to simulate overcooling events with PTS potential.

The results of thermal hydraulic transient analysis performed in support of the NRC PTS study exhibited rather complicated high pressure injection (HPI) and loop flow histories. Periods of high loop flows (perfect mixing and system code results applicable) were interdispersed with periods of flow stagnation (stratification). The predicted flow stagnation was due to loss of coolant inventory and the associated interruption of the natural circulation flow paths by steam bubbles. Much of the experimental and analytical efforts in quantification of stratification and associated cooldown effects, were thus focused only on stagnated loop flow conditions.

The regional mixing model (REMIX code model), which has been utilized in support of the NRC PTS study, is based on a fundamentally-orientated zonal approach which integrates local mixing behavior into an overall system response. The model accounts for countercurrent flow limitation between the cold and hot streams at the cold leg/downcomer junction, and incorporates plume mixing rates which are consistent with data from idealized plume geometries. The regional mixing model and associated computer code REMIX was intended for vertically downward, low Froude number injections ($Fr_{HPI} \sim 1$) of interest to Westinghouse and Combustion Engineering designed reactors.

For very high Froude number HPI injections ($Fr_{HPI} \sim 16$) of interest to Babcox & Wilcox designed reactors, forceful jet impingement on the opposite cold leg boundary result in a significant increase in local mixing and entrainment which is not depicted in the entrainment model incorporated in the REMIX code. However, since the extent of mixing in the regional mixing model is also controlled by the counter-current flow limitation at the cold leg/downcomer junction, the cooldown transient can be calculated on the basis of maximum entrainment as it was done in the NEWMIX code. The computer code NEWMIX is identical to REMIX except that the mixing at the point of HPI location is assumed to occur with the maximum entrainment controlled only by counter-current flow limitation (independent of HPI nozzle orientation). It should be noted that computer codes REMIX and NEWMIX are only applicable when there is no loop flow (complete loop stagnation condition). Specific versions of these codes, REMIX-S and NEWMIX-S have also been developed for applications to experimental simulations involving solute induced buoyancy.

The previous thermal hydraulic analysis of small-break LOCAs of PTS-potential in the Calvert Cliffs plant was revisited in 1988. Using a more recent version of the TRAC code (with improved condensation modeling under low flow and high vapor fraction), it was concluded that the previously envisioned fully stagnated loop flow regime at high primary system pressure was not possible. Furthermore, it was shown that even the very low loop flows are important in moderating the cooldown transients.

A study has been conducted at Brookhaven National Laboratory for the U.S. Nuclear Regulatory Commission (NRC) to provide a number of modeling improvements to the REMIX code that include extending the regional mixing model to include low-loop flow conditions. This improved version of the code combines REMIX, NEWMIX, REMIX-S and NEWMIX-S into a single code REMIX97.

This paper presents an overview of the extended regional mixing model and the associated computer code REMIX97. Comparison of REMIX97 calculated results with thermal mixing experimental data obtained from various integral test facilities are presented and shown to be in excellent agreement.

Containment/ Reactor Coolant System Analysis of a Large Break Loss-of-Coolant Accident in the AP600 Using RELAP5/MOD3

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SUMMARY

Several large break loss-of-coolant accident (LBLOCA) containment calculations were performed for the Westinghouse Advanced Passive 600 MWe Reactor (AP600) using RELAP5/MOD3 Version IG. All of the calculations used a 49 node containment module developed to assess containment effects on AP600 reactor coolant system (RCS) behavior.

The first calculation used mass and energy flows from the RCS as boundary conditions for the containment, which were developed by Westinghouse for the FSAR design basis containment analysis. The RELAP5 model also used outside containment shell heat transfer coefficients calculated by the CONTAIN code. This calculation was used as a benchmark to compare basic containment models (volumes, elevations, and heat structures) against similar calculations using CONTAIN and WGOTHIC. Even though the RELAP5 containment models are not well assessed against experimental data, the calculated containment pressure was comparable to the CONTAIN and WGOTHIC results.

For the second calculation, the 49 node containment module was linked to the 100 node RELAP5 AP600 RCS Long Term Cooling Model (LTCM), and an integrated RCS/containment LBLOCA was performed. In this case the second and highest pressure peak disappears, because:

1) the FSAR mass and energy release is significantly higher than the "best-estimate" release calculated by RELAP5, and

2) the excess energy in the FSAR release goes upward as steam to the upper containment, because of the conservative energy partition assumed by Westinghouse, and based on an accepted and approved methodology for licensing analysis.

A sensitivity study was performed using the integrated model by increasing then decreasing the calculated internal containment shell heat transfer by a factor of three. The results showed that the factor of three reduction still does not result in a high second pressure peak for the integrated model with "best-estimate" release.

A second sensitivity study was done using the integrated model. For this study, the outside containment shell heat transfer was varied. In the first variation, the outside coefficients from the CONTAIN analysis uses only the un-wetted values. This has the effect of assuming that no passive containment cooling system (PCCS) water has been provided to the outside of the containment. In this case the containment

pressure drops until about 5 hours and doesn't rise to the design pressure until after 24 hours. This indicates that a substantial margin exists in the time required to initiate PCCS cooling compared to what would be required using the conservative FSAR energy release. In the second case of this study, "wetted" coefficients were only applied to the upper dome region of the containment. In this case the pressure turns over before the design pressure is reached.

Integrated containment/RCS analyses are helpful in assessing the conservatism of certain design basis FSAR assumptions used for containment analysis.

SAFETY OF NEXT GENERATION REACTORS

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ABSTRACT

A technique was developed to evaluate the applicability of data from small scale facilities for validation of codes for analysis of nuclear safety with emphasis on the next generation of reactors. The technique first divides an accident into phases based on the components that come into play as the accident evolves. Conservation equations, resolved to the component level and their interconnections, are derived for the active components in each phase. The equations are then non-dimensionalized and reference parameters are selected such that the dependent variables and their time derivatives, other than the system response of interest, are of order 1. Order of magnitude analysis is then performed for each equation and then between equations, based on the numerical values of the non-dimensional coefficients for each term, with only the large order terms being retained. The resulting equations then contain terms whose impact on key system responses (e.g., reactor vessel level) are ordered in terms of the magnitude of the non-dimensional groups multiplying the O[1] dependent variables. The reduced set of equations and non-dimensional groups are validated with experimental data where possible. The validation process is meant to demonstrate that the important terms have been retained and enhance confidence in the system of equations used to capture the main processes occurring in each phase.

The methodology was demonstrated by evaluating the applicability of small-scale facility data for next generation reactor SBLOCA. Based on the non-dimensional equations, the dominant non-dimensional groups, and hence the dominant physical mechanisms and their dependence on geometric and operational parameters, were identified for a particular scenario, an AP600 cold leg break, starting from the initiating event through long term cooling. The important parameters entering the groups included elevation differences between the reactor vessel and other components, PRHR heat transfer rates, fluid thermo-physical properties, liquid levels in tanks, and flow resistances in the CMT lines and IRWST lines. It was also shown that, after the

beginning of CMT draining and accumulator injection, the dominant processes do not depend on break size provided they are small. The dominant processes were dependent on plant geometry and the operation of engineered safety features, such as the automatic depressurization system. The same transient events were evaluated for three experimental facilities and the same non-dimensional groups, and hence mechanisms, were shown to be important. It was found that these non-dimensional groups covered the range expected in the AP600, indicating that while there may be some distortions in scaling for a particular facility, between them, the important phenomena were captured and the small-scale facility data appear applicable for SBLOCA in the AP600 system. In more general terms, the methodology appears suitable for assessing scaling of various facilities for other postulated accidents and for other reactor concepts.

Technical Challenges for Life Cycle Management-
Oconee Nuclear Station

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Summary

Life Cycle Management (LCM) is one of the keys to the future, since thinking, planning, and acting within a structured business-minded process is required now more than ever. Plant staff and management must recognize the need for integrated, holistic solutions that identify and address the life limiting issues which face their plants.

The potentially life limiting issues for nuclear power plants (NPPs) come from a variety of sources but can be grouped into five categories of concern: Technical, Regulatory, Environmental, Political, and Economic. These five categories allow the engineer to view issues based on the source of the concern and to measure its impact on their nuclear business. There are many common LCM concerns, but the importance of each will vary from NPP to NPP depending upon relative significance of the source to that particular NPP. This paper stresses the following technical challenges faced by successful life cycle management programs:

- Material aging
- Obsolescence of component technology
- Lack of spare parts

Obsolete component technology can cripple the operation of a NPP. Oconee Nuclear Station is dedicated to upgrading components, as needed. Many large components have been replaced recently, particularly computer-related components. As obsolescence becomes an issue for a particular component or component type, engineering studies are performed to determine feasible replacement options. Management is responsible for authorizing and allocating money to replacement of these components.

Duke Power's Commercial Grade Program is dedicated to managing the spare parts dilemma at its NPPs. Personnel within the Procurement Engineering department are responsible for the program implementation as well as for performing the analysis required to dedicate commercial components for nuclear QA applications.

The preparation of the license renewal submittal for Oconee Nuclear Station led us through a systematic process with respect to material aging. By following the approach laid out by the license renewal rule, we provided ourselves with the knowledge of recognizing what programs and activities we are performing that manage material aging in the plant. For the purpose of extending the license by 20 years, many of those activities may need enhancing. But for the purposes of operating an aging NPP to the end of its current license term, we gained much insight into the kinds of activities required to manage the material aging challenge.

The challenges that face aging NPPs are expansive in nature and require systematic solutions. At Oconee our preparation for the future by investing in license renewal requires that we be prepared to address these challenges with the often drastic and expensive measures required to attack them with full force. Preparing for the future means investing today.

AGING MANAGEMENT, AN INTEGRAL PART OF EDF MAINTENANCE PROGRAM

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Electricité de France is now operating 58 PWR nuclear power plants which produce 75 % of electicity in France. Besides maintaining safety and availability on a routine basis, it is outmost important to protect the investment which is now 50 % amortized. That is the reason why EDF is devoting important ressources to implement aging management concern as an integral part of operation and maintenance programs (for example through appropriate data collection). This leads to specific repair and replacement projects but also to important anticipation efforts, taking in account the high level of standardization of the units.

A particular organisation has been set up to continuously observe and analyze all activities so as to make sure that aging concern is correctly taken in account in strategies and that no decisions are susceptible to impair plant lifetime. This "lifetime program" is paying attention to technical issues associated with main components but is also dealing with issues related to economics and industry situation.

Reports are regularly presented to EDF managers and decisions are taken in relationship with future plant construction program.

PLANT LIFE MANAGEMENT ACTIVITIES OF LWR PLANTS IN JAPAN

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ABSTRACT

Plant life management activities of LWR Plants has started substantially since 1994 in Japan and it's been conducted by the utilities and MITI co-operatively.

In Japan , it's not same as US ,there are no limitations for an operation period or plant life because we have no laws or rules which prescribe a licensed operation period for nuclear power plants. If an annual inspection checked by the regulatory authority ,MITI, is completed without any problems , one year operation renewal is permitted. This cycle can be repeated without limit.

However , we consider it very important to evaluate the long-term integrity of major systems,structures and components of an aged nuclear plant and to ensure the safe ,steady and highly reliable long-term continued operation. So,we've come to recognize the importance of the Plant Life Management Activities.

Japanese Plant Life Management study consists of two phases. Part 1 study started in 1994 and was already conducted and made public in 1996. Part 2 study is now on-going and be scheduled to be completed in near future.

1. PLANT LIFE MANAGEMENT ACTIVITIES

1.1 BASIC CONCEPT

Part1 study was carried out as a feasibility study to obtain the outlook for nuclear plant soundness over the representative components assuming 60 years operation. To confirm the soundness of plant life time , three candidate power plants that had been in service for many years were selected ,those were Fukushima Dai-ichi Nuclear Power Station Unit 1 (BWR), Tsuruga Power Station Unit 1 (BWR), and Mihama Power Station Unit 1 (PWR).

For the technical assessment , seven or nine typical components were chosen for evaluation such as the reactor pressure vessel , the reactor internals, the primary containment vessel and so on. These components were chosen because of their importance in the safe operation and in the event of any degradation of these components during operation, repair or replacement cannot be performed easily.

The purpose of Part 2 study is to improve plant reliability and thus we should review components not only a safety point of view , but from the perspective of avoiding a forced shutdown, considering a wider range of components (consisting of thousands of items) of the three candidate power plants. It's intended to develop measures against aging degradation and establish the methods and period of inspections from this assessment for future implementation.

1.2 THE RESULTS OF PART 1 EVALUATION

Through the technical assessment, it was discovered that some of the components should be provided with adequate measures in operation and inspections to insure proper operation for a 60 year period. For example, in the case of BWR plants, it was discovered that planned measurement against thinning of carbon steel pipe wall thicknesses was necessary. Furthermore , in the case of Reactor Internals , it is necessary to have scheduled inspections to avoid the damage due to IGSCC at which non-low carbon stainless steel is applied as basic material. In the case of PWR plants, it was discovered to be necessary that measures be taken against stress corrosion cracking of parts made of Inconel 600 alloys.

In both BWR and PWR plants, it was also discovered that certain aging degradation phenomena had to be monitored by periodical inspection on the components that were susceptible.

It was confirmed, however, that plant integrity can be preserved basically by the continuity of present maintenance methods.

2. PART 2 EVALUATION METHOD

The evaluation methods of Part 2 study is as follows.

a) Selection of Components for Assessment
Aiming to further improve the reliability of nuclear power plants, the components to be assessed are classified into two categories: "Important Components for Safe Operation" and "Important Components for the Avoidance of a Forced Shutdown".

b) Classification and Selection of Typical Components
The components chosen for assessment as noted previously were divided into groups of similar items, such as, type, construction, operating circumstances (which include operating location, fluid properties, etc.) and materials. Typical components were then selected from among these groups based on importance, operating conditions (pressure, temperature) and other critical criteria.

c) Assessment
The typical components selected under paragraph (b) will be broken down to a part level and technically assessed by considering the phenomena of aging degradation. This technical assessment will verify the preservation of plant integrity and on the appropriateness of current inspection and maintenance programs (additional items may or may not be required), assuming that the plants are to remain in long term service.
The next step will be to apply the results of the assessment of the typical components to other components in each group with consideration to the differences between components.

d) Formulation of Maintenance Plans Based On Long-Term Operation
Plans will be formulated to adopt for an actual power plant the assessment results which were made considering the aging degradation. Specifically, the maintenance activities and the research and development plans will be developed.

3. THE RESULTS OF PART 2 STUDY

Basically confirming that there is no problem in continuing the practice of the existing maintenance program against aging degradation assuming 60 years operation.

Throughout the Part 2 study, 40~50 recommending items were pointed out to improve maintenance methods for long term operation. For example, in the case of foundation bolts ,it is common to every components, it cannot be inspected as it is burried in concrete. So it is necessary to conduct sampling inspections on proper occasions.

Conservative seismic evaluation against aging components shows there is not any problems for the earthquake resistance.

4. CONCLUSION

In Japan, appropriate and adequate inspections and maintenance are performed on the LWR nuclear power plants which have been in operation for over twenty-five years. In addition, these plants are provided with adequate preventive measures based on detailed surveys of troubles experienced with plants in foreign countries, and the latest available technical knowledge. Accordingly, there have been no relations between the plant operating years and the number of abnormalities in Japanese domestic plants.

However , in order to improve plant reliability, it is important to take proper maintenance and management measures against aging degradation on those nuclear power plants that have been in service for a considerable length of time.

Furthermore, through the evaluation of the numerous components in this study, those items that are susceptible to aging degradation can be realized. And it is important to develop rational inspection and maintenance methods against younger power plants with this study.

EPRI NUCLEAR LIFE CYCLE MANAGEMENT TECHNOLOGY PROGRAM: CURRENT ACTIVITIES

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EPRI

SUMMARY

Life Cycle Management (LCM) of nuclear power plants can be viewed as the integration of all those various activities which establish the useful operating life of a nuclear power plant. These activities include engineering, operations, maintenance, economic evaluations, environmental assessment, aging management, license renewal and other activities. Since its inception in the mid-1980's, the EPRI LCM program has focused on providing products to assist nuclear utilities in obtaining maximum benefit from their existing nuclear facilities. This paper is intended to provide an overview of current activities within the EPRI Nuclear Life Cycle Management Technology Program, in the areas of license renewal, aging management, and asset management. Specific projects reviewed include: Calvert Cliffs Life Cycle Management Program, Life Cycle Management Guidelines, LCM Implementation Demonstration, Generic License Renewal Technical Issues Resolution, Concrete Structures Aging Reference Manual (COSTAR), and Valuation and Management of Nuclear Assets. In addition, this paper provides a brief status report on the Joint DOE-EPRI Strategic Research and Development Plan to Optimize U.S. Nuclear Power Plants.

ESTIMATING THE UNCERTAINTY IN REACTIVITY ACCIDENT NEUTRONIC CALCULATIONS[1]

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INTRODUCTION

The design-basis reactivity initiated accidents (RIAs) are the rod drop accident in a boiling water reactor (BWR) and a rod ejection accident (REA) in a pressurized water reactor (PWR). Licensing calculations must show that the fuel response is acceptable for these accidents. Fuel response is judged on the basis of the fuel enthalpy. In the past, these calculations have been done using very conservative methods. However, with the possibility that for high burnup fuel the acceptance criteria may become more stringent, and with new, more rigorous, calculational methods available, it is expected that best-estimate methods will be used in the future. Hence, it becomes important to understand the uncertainties inherent in such best-estimate calculations.

The present study focuses on the calculation of local fuel enthalpy for the PWR rod ejection accident. As had been identified in a previous study [1] of the BWR rod drop accident, the most important parameters controlling the power excursion during an RIA event are the reactivity worth of the control rod, the delayed neutron fraction, and the fuel reactivity feedback as determined by the fuel temperature (Doppler) reactivity coefficient and the specific heat of the fuel. In addition, since many methods do not calculate the fuel enthalpy for a specific fuel rod directly but rather calculate the assembly average fuel enthalpy (or power), the intra-assembly peaking factor used to obtain the fuel rod enthalpy becomes an important parameter.

ANALYSIS

Sensitivity calculations were done with the PARCS code [2] which has a three-dimensional neutron kinetics model. The code is coupled with the RELAP5 system thermal-hydraulics code, but for this application, it was run as a stand-alone code. The calculations were based on initial conditions for hot zero power operation. The sensitivity of total energy to changes in delayed neutron fraction and control rod worth from the three-dimensional calculations generally followed what would be expected based on simple point kinetics. For example, the relative change in total energy per unit relative change in delayed neutron fraction was a function of control rod worth. At high rod worths, the sensitivity was low (approximately unity), but as the rod worth decreased to the point where the excursion was just prompt-

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[1]This work was performed under the auspices of the U.S. Nuclear Regulatory Commission.

[2]Visiting Nuclear Engineer from the Korea Institute of Nuclear Safety.

critical, the sensitivity became very large. Although the sensitivity increases, the magnitude of the energy deposition also decreases as control rod worth decreases to just prompt-critical.

The question of how to obtain the fuel rod power when it is the assembly average power that is calculated is being addressed in two ways. The first way is by comparing results calculated with PARCS (which calculates assembly average powers) with results calculated using the BARS code [3] (which calculates the power in each fuel pin explicitly). A second way will be possible when PARCS has a flux reconstruction model which is expected to be available in the fall of 1998.

The comparison of PARCS and BARS results is not straightforward because the two codes differ in ways other than the explicit representation of the fuel pins. In spite of those differences and the challenges they represent in comparing results from the two codes, the comparison should still be informative. Deficiencies in one or both of the codes may surface which is important since both codes are undergoing validation work. It may be determined that the detail in the heterogeneous method is essential for an accurate determination of the consequences of the REA, or it may turn out that other uncertainties in the calculations are more important.

CONCLUSIONS

Work is ongoing to understand the uncertainties in calculations of the REA. The work to date has focused on calculations which show that the uncertainty in energy deposition becomes less sensitive to the uncertainty in the important core parameters as the energy deposition increases. The energy deposition is determined by the ejected control rod worth, the delayed neutron fraction, and the fuel temperature feedback. It is only when the resulting power excursion is close to (and above) prompt-critical that the sensitivity becomes large. The effect of approximating the pin power using calculations of assembly average power and estimates for the intra-assembly peaking factor are also being investigated. This work is being carried out using results from a code which calculates the power for an assembly and one that calculates the power on a pin-by-pin basis.

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Validation of a Pin-by-Pin Neutron Kinetics Method for LWRs

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Summary

During some reactivity initiated accidents (RIAs) in light water reactor (LWRs), very large deformations in the power distribution over the reactor core can take place. In analysis of these transients it is difficult to evaluate the real uncertainty in calculation of peak fuel enthalpy due to neutronic code because of a lack of adequate experimental data. When large in-assembly power peaking factors occur in LWR, widely used 3D assembly-by-assembly neutronic codes based on the neutron diffusion theory underestimate to a great extent the peak power in fuel rods and a method for pin-by-pin reconstruction of power distribution requires very complicated procedure of validation for transients. The problem may be solved by comparison of calculational results of assembly-by-assembly diffusion code with ones obtained by qualitatively different, advanced code in the framework of more detailed modeling of the reactor core.

Recently in Russian Research Centre "Kurchatov Institute" an NPP dynamic model for VVER and PWR RIA modeling has been developed. The model is based on coupling of the RELAP5/MOD3.2 thermal hydraulic code with the BARS 3D pin-by-pin neutronic code. The BARS code was developed on the basis of the advanced method of heterogeneous reactor theory, which distinguishes by some advantages from diffusion theory using homogenized fuel assemblies.

The neutron database of BARS is calculated by the TRIFON code. TRIFON solves the multigroup neutron transport equation in various reactor cells using the collision probability method taking into account detailed structure of resonant cross sections. The TRIFON code allows to take into account precisely rim-effect on the Doppler coefficient of reactivity.

The main aim of this report is to determine uncertainties in reactivity coefficients and power distribution for LWR pin-by-pin calculation by using the advanced heterogeneous method.
Calculations of the reactivity coefficients in PWR and BWR fuel cells have been done by the TRIFON code with neutron database generated on the basis of ENDF/B-VI library. The calculational results were compared with available in literature Monte Carlo (the MCNP code) benchmark calculations with ENDF/B-V and ENDF/B-VI libraries. 10 PWR cells with fresh fuel of different enrichment (from 0.7 % through 3.9 %) at two temperatures were calculated. On the basis of these calculations the Doppler coefficients were determined. Comparison of the data shows very good agreement for the TRIFON and MCNP-3A, MCNP-4A results. 77 BWR cells with fresh and burnup fuel of different enrichment at varied temperatures of the fuel and the moderator and at different void fraction were considered. The following reactivity coefficients

were calculated: the Doppler one, the void one and the moderator one. The mean square deviation in calculated parameters between the TRIFON and the MCNP results are the following:

- the Doppler coefficient: 4% for the fresh fuel and 7% for the burnup fuel;
- the void coefficient: no more than 4%;
- the moderator coefficient: no more than 3 standard deviations of the Monte Carlo calculation.

To evaluate uncertainties of the BARS code in calculation of pin-by-pin power distribution, 107 ZR-6 benchmark critical assemblies with VVER-1000 type fuel lattice were considered. The BARS calculations were performed with 5 group neutron database generated by the TRIFON code using ENDF/B-VI based library. Calculational results for the multiplication factor (K-eff) and pin-by-pin power distribution were compared with the experimental data. The K-eff value averaged over all assemblies is 0.9961 with the mean square deviation of 0.0021. The mean square deviation in pin-by-pin power distribution calculated by the BARS code is 1.5% for 41 critical configurations where corresponding experiments have been carried out.

To verify the BARS transient model, experimental results of the power dynamic behavior obtained at the pulsed graphite reactor IGR were used. This reactor is intended for reactor fuel tests under RIA conditions. The basic feature of such a transient is the fact that the power surge are generated by a control rod withdrawal and are suppressed by the negative feedback due to reactor core temperature increase. Such experiments modeling the control rod ejection accident in LWR with temperature feedback are unknown. The time dependence of the reactor power was measured by means of a set of in-core detectors and out-core ionization chambers. Comparison of calculated and measured power time distributions gives the excellent agreement of the results.

To illustrate some features of the pin-by-pin LWR transient model of the BARS code, calculational results of the control rod ejection RIA modeling at hot zero power conditions in VVER-1000 are presented. The calculations were carried out by the coupled RELAP5/BARS codes for the initial fuel load of Zaporozskaya NPP Unit 5. Neutronic model consists in 5 energy groups, 6 delayed neutron groups, 5 axial harmonics, and about 75,000 calculational cells including about 51,000 fuel cells. The results demonstrate that detailed power shape after control rod ejection is of a very complex structure.

The results presented in this paper show that the BARS code can be successfully used for LWR RIA modeling and for intercomparison with 3D diffusion neutronic codes.

Oxidation and Quenching Experiments with High Burnup Cladding under LOCA Conditions

Revision of Previous Data and Main Trends of Recent Tests

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Summary

Within the framework of burnup extension request, experimental programs have been undertaken in France, jointly by EDF, IPSN, CEA and FRAMATOME, in order to provide the Safety Authority with relevant data concerning the behaviour of high burnup fuel under accidental conditions. Relative to LOCA, the initial studies were focused on the behaviour of cladding alone, bearing a simulated or actual in-reactor corrosion state representative of high burnup reactor operation. With reference to the current LOCA acceptance criteria, the main issues concern the high temperature oxidation kinetics and the to thermal shock resistance upon quench after significant oxidation of the cladding.

The TAGCIS program was first conducted to investigate the thermal shock behaviour of unirradiated cladding bearing a pre-corrosion state simulating the end-of-life state of high burnup fuel claddings. The main tendancies of the results indicate that :

- The initial oxide scale appears very poorly protective relatively to high temperature oxidation and may be considered fully transparent to oxygen diffusion ;
- Initial oxide offers a very low mechanical resistance to quench loads so that initial corrosion does not induce additional brittleness of the cladding as compared to an as-received cladding of equal metal thickness.

The TAGCIR tests series was performed on actually irradiated cladding samples taken from high burnup rods irradiated over 5 cycles in a commercial EDF PWR and having reached a rod burnup close to 60 GWd/tU. The main trends of the tests results indicate that :

- Irradiated zircaloy exhibits an increased oxidation rate as compared to unirradiated zircaloy ;
- No failure upon quench occured below the acceptance limit ECR=17% relative to the transient oxidation alone. The failure maps have indicated a lower brittleness for irradiated cladding than for unirradiated cladding under two-side oxidation ; the tendancy for one-side oxidation tests was not so clear but appeared first opposite as for two-side oxidation.

In the aftermath, the HYDRAZIR program was undertaken, based on the main TAGCIR outcome that the behaviour of irradiated zircaloy under LOCA conditions is essentially borne by the hydrogen charged in the cladding ; the objectives of this program are to quantify the influence of hydrogen content on oxidation kinetics and quench resistance.

The HYDRAZIR tests series was performed on the previous TAGCIR facilitiy with incorporation of significant experimental improvements. These improvements make questionable the results obtained previously from the TAGCIS and TAGCIR tests series, namely the failure limits relative to oxidation amount. A detailed investigation of the uncertainties on temperature distribution associated to the evolutive experimental geometry of the inductive heating furnace, led us to reconsider the failure limits derived for the two-side oxidation tests series in the TAGCIR experiments. As a result, the failure limit relative to the usual ECR parameter (Equivalent Cladding Reacted) appears shifted towards higher values. Additionally, a further examination of the one-side oxidation tests conditions has revealed that the results of these tests series were not reliable ; they have been consequently excluded from the current data analysis.

The HYDRAZIR program consists of oxidation kinetics experiments and oxidation/quenching experiments on unirradiated hydrogen pre-charged zircaloy-4 specimens.The hydrogen content varies from 500 to 5000 ppm weight, thus ranging from the in-reactor corrosion charging (below 1000 ppm) to the transient charging that may occur, after burst, on the inner side of the cladding under steam starved oxidation conditions. The program, still under progress for complementary tests series, has provided important results relative to the performance under LOCA conditions of pre-hydrided zircaloy. The main tendancies of the results indicate :

- an oxidation kinetics enhancement for 500 ppm H content, around +20% at 1200°C compared to as-received zircaloy kinetics, and increasing with temperature; however, the kinetics enhancement vanishes for 1000 ppm and higher H contents

- a similar (or possibly slightly better) quench resistance to that of as-received cladding for H content below 1000 ppm, but severely degraded at 2000 ppm content and above ; the analysis of these tests results is still preliminary but post-test metallographic examinations have shown important structural changes in the metal layer for high H content cases.

Although the high hydrogen content influence is not a burnup extension issue, it points out the need for a proper taking account of hydrogen effects in LOCA transients evaluation, that may involve alternative embrittlement criteria including hydrogen contribution.

This paper will briefly review the TAGCIR results update and present the main trends of current HYDRAZIR tests results.

Zr-1%Nb (VVER) High Burnup Fuel Tests under Transient and Accident Conditions

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The procedures of fuel development and licensing include the required application of experimental data. Particular attention is paid to the examination of fuel behavior at burnups above 45-50 MWd/kgU. This provides important information to verify analytical codes, to provide the possibility for further burnup increase, and to understand transient and accident operating conditions.

At the Research Institute of Atomic Reactors (RIAR), a special technology was developed for tests, in the MIR research reactor, of VVER refabricated and full-scale fuel rods taken out of standard fuel assemblies after operation at nuclear power plants. The tests are carried out under special conditions, for example, power cycling and power ramping, those under LOCA conditions, as well as high burnup tests. Experimental fuel rods and fuel assembly specimens are provided with detectors to measure parameters of fuel rods and coolant (fuel rod gas pressure, fuel temperature, change in fuel length, neutron flux intensity, coolant temperature and pressure). Change in fuel rod power is provided by removal of the reactor control rods. Changes in reactor power or a special facility is used to provide different rates of the power changes (0.1-100%). The information measuring system allows one to interrogate and simultaneously record all the necessary parameters with a frequency of up to 10 kHz. The experimental facility is located in the water loop channels (8 channels). LOCA tests are carried out in the special loop. Integral experiments with fuel assembly specimens were carried out in the MIR reactor using different values of pressure difference for the coolant and within the fuel rods, maximum temperatures on the cladding during dry out from 700 to 1300°C. Currently the loop is being additionally fitted with special equipment to carry out a number of experiments under maximal design basis accident conditions (break in the principal circulating tube). The preliminary tests of the systems will be carried out at the special facility with "hot" coolant.

Special experiments with specimens of pellets, fuel rods and refabricated fuel rods are carried out at an electrically heated facility in the hot cells. The temperature dependance of radionuclide release from the spent fuel and changes in the structure of the cladding and the fuel pellets are being currently examined. The experiments to examine the spent fuel rod behavior under LOCA conditions with external fuel rod heating have already been started. As a result of the examination, the dependance on the accident history, time before depressurization, and change in fuel rod forms are obtained.

The primary part of the accidents connected with reactor core cooling by means of flooding of the heated fuel by cold water was examined at the electrically heated facility followed by further study of the simulated fuel rod properties.

A number of the experiments simulating reactivity-initiated-accident (RIA) conditions with high burnup fuel were carried out at the impulse reactor as well as the SM reactor. As a result of these experiments, the limits of fuel rod depressurization and failures as well as the mechanism of their faults are obtained.

NSRR Pulse Irradiation Experiments and Tube Burst Tests

Toyoshi FUKETA, Fumihisa NAGASE, Takehiko NAKAMURA,
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Japan Atomic Energy Research Institute

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). 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 and the status of out-of-pile, separate-effect experiments including tube burst tests are described and discussed in this paper.

We have add four experiments with irradiated LWR fuels for the last Japanese fiscal year, consisting of TK-2, TK-3, TK-4 and FK-3 experiments, and have a plan to perform TK-5 and TK-6 experiments at several weeks before the 26th WRSM. The paper presents results from these recent experiments, and discusses PCMI failure including relationship between cladding corrosion and possibility of fuel failure.

In the tests TK-2 and TK-3 with PWR fuel with low tin (1.3%Sn) cladding, the TK-2 resulted in fuel failure and the TK-3 did not. The primary difference between the two tests is the elevations where the fuel specimens were sampled. The test fuel of the TK-2 was sampled from the higher elevation and had 25 μm oxide layer at the periphery of the cladding, while the TK-3 rod had 8 μm oxide layer. In the test TK-2, mechanical forces generation was detected. Thermal-to-mechanical conversion ratio in the experiment is 0.08%, and 8% of fuel pellets were dispersed into coolant water.

The three experiments with 41 to 45 MWd/kgU BWR fuels, FK-1 through FK-3, resulted in no failure, where fuel enthalpy in these experiments reached 70 to 150 cal/g at maximum.

The NSRR experiments with high burnup PWR fuels suggested that expanded fuel pellets mechanically interact mechanically with Zircaloy cladding during the pulse irradiations, and that cladding embrittled by noticeable hydrogen absorption and hydride localization failed at the early stage of the transients. In order to quantify the effects of hydrogen absorption and hydride localization on reduction of cladding integrity and probability of the failure, out-of-pile, separate-effect tests are being conducted at JAERI. A series of burst tests at high pressurization rates on the cladding samples is one of the separate-effect tests. The objective of the burst tests is to simulate the quick mechanical interaction between the pellet and the cladding during pulse irradiation and to investigate the influence of hydrogen concentration, hydride localization, cladding temperature, and pressurization rate on the cladding failure behavior. This paper describes the results of the tests at room temperature with the cladding samples having different hydrogen concentrations and hydride distributions. The cladding surface temperatures at the fuel rod failures were below 370 K in the pulse irradiation test at NSRR. Accordingly, the present study are available to interpret the results of the NSRR experiment.

The results from the burst tests are summarized as;

(1) Hydrided Zircaloy-4 cladding samples exhibiting axially extended failure opening in the present test. The opening is quite similar to those observed in the NSRR experiments for high burnup PWR fuels.

(2) The residual hoop strain was reduced with increase of the hydrogen concentration. Increase of the pressurization rate might slightly reduce the residual hoop strain of the hydrided samples.

(3) Strength and ductility of the cladding were obviously reduced by localization of hydrides at the cladding outer surface, namely the formation of the hydride rim. Low ductility at the hydride rim could have the great influence on the cladding failure at the beginning of the transient in the NSRR experiments.

The paper provides future NSRR test matrix and a plan of out-of-pile, separate-effect tests.

Loads to be Considered in Qualifying Cables for LOCA

Wolfgang Michel and Lueder Warnken
Siemens AG, KWU, Germany

Summary

Qualifying equipment for LOCA-loads many aspects have to be considered:

1. kind of LOCA and coverage of different brake postulates and positions
2. distribution of loads, spatial and in time
3. relevance of loads and ageing factors to be considered
    - generally and
    - based on experience
4. effects of accelerated aging on the results to be achieved, considering synergy
5. transition of the corresponding relevant loads to test conditions, considering the influence of margins
6. experimental demonstration according to test objective (type test or single purpose test)
7. limiting conditions reproducing LOCA-loads in test facilities
8. influence of operation and maintenance on LOCA-resistance
9. experience gained and how to be introduced in the established EQ-procedure.

With few exceptions, the above mentioned aspects have been discussed world wide in the past and many investigations have been made as well. Never the less, from the point of view of a German manufacturer of NPPs as well as a manufacturer of equipment for NPPs some issues seem still neither be formally established to the necessary extend nor sufficiently addressed.

This regards specifically the aspects of the relevance of loads to be considered, if and how they should be applied in the process of EQ and how to introduce the results of experience in the same procedure.

The background of the presentation is not only physically and technically. The practical and commercial aspects have to be considered too. To reduce costs qualifying electrical equipment as far as reasonable achievable without violating the safety objectives cannot be neglected any longer. The average suppliers are no more willing to provide equipment for NPPs significantly different to their series products (provided they are generally able to withstand LOCA-loads).

The presented paper will discuss these issues based on the example of cables to the major extend, but complemented where necessary by some other examples in order to facilitate the understanding.

Finally some suggestions how to proceed in the future will be made and some hints to improve existing standards will be given.

Correlation between physical degradation of artificially aged cables and their dielectric behavior during LOCA

Kjell Spång, Ingemansson Technology AB, Göteborg, Sweden

Consideration of aging is important in qualification of components for nuclear power plants. This is mostly taken care of by subjecting the components to accelerated aging before they are exposed to LOCA conditions. In order to avoid extreme acceleration factors the components may be qualified for a life which is shorter than the expected service life. This qualification is then complemented with an on-going procedure in which the qualified life is controlled and extended.

Two different methods for control and extension of the qualified life, condition monitoring and repeated qualification testing. In condition monitoring certain parameters are used to establish the condition of the component after aging before the component is subjected to LOCA test. If the component passes the LOCA test, it is assumed that the component is qualified for a design basis event as long as the condition parameter value which was measured before the LOCA test is not passed.

Of special interest is then to know if there is a correlation between the electrical behavior of the component during LOCA and the value of the condition parameter. This correlation has been studied for some components, including cables of type Lipalon, Rockbestos and Dätwyler, as part of a research program in Sweden, supported by the Swedish Nuclear Inspectorate and the utilities.

In order to achieve significant results, the artificial aging performed in the study included more severe environmental conditions than would normally be used for qualification of the cables for installation in the Swedish nuclear power plants.

Mechanical degradation due to the artificial thermal and radiation aging of complete cables, mainly representative of cable jacket materials, and of the separately tested leads was determined by indenter measurements. The mechanical degradation of the cable jacket material was also determined by elongation-at-break measurements. The chlorosulphonated polyethylene (CSPE, Hypalon) insulation materials, included in the jacket material of the Rockbestos cable and the jacket and leads of the Lipalon cable, showed very pronounced mechanical degradation as result of the conditioning. The EPDM insulation materials, included in the jacket and leads of the Dätwyler cable, were less affected. The crosslinked polyethylene insulation materials, included in the lead insulation of the Rockbestos cable, were only slightly affected.

The dielectric degradation of the cables was determined by measurement of insulation resistance and capacitance and of dielectric loss factor. The aging resulted in a pronounced degradation in the dielectric behavior, which is shown as significant increase in the dielectric loss factors and, in the most severe cases, in significant reduction of the insulation resistance as well as increase of the capacitance.

The LOCA test subjected the cables to overheated steam for 3 hours at 181°C and 4 BarG, followed by 160°C and 4 BarG for 3 hours and 120°C for 44 hours. The tests included cable samples, which had been thermally aged for various duration at 95, 120 and 142°C, for the Lipalon cables also at 80°C. Cables which had been subjected to high humidity or vibration and thermal aging were included. The cable samples were subjected to ionizing radiation at 10^4Gy/h for 50 hours before LOCA testing (total dose 500 kGy).

The effects of the radiation on the Lipalon and Dätwyler cable jackets were significant. Also the dielectric loss factors were affected for the Lipalon cables. This was the case also for the separately exposed insulated cable leads.

The measurements during LOCA testing included insulation resistance and dielectric loss factor.

The comparison between the indenter values measured after the artificial aging before LOCA and the values of maximum loss factors during LOCA showed a positive correlation for the Lipalon cables. The Rockbestos cables which had been aged to the most severe conditions were in such a poor shape that the dielectric loss factor couldn't be measured during LOCA, but there was a positive correlation between the indenter values before LOCA and the dielectric loss factors after LOCA.

Use of dielectric loss factor as a measure of the degradation before after the aging showed similar correlations to the loss factor measured during LOCA.

The Dätwyler cables showed no consistent relationships between mechanical condition before LOCA testing and the dielectric parameter values during LOCA. There was a very high correlation between the aging time and the indenter values, but the degradation was not severe enough to result in any significant influences on the dielectric behavior during LOCA.

Lipalon cables which were subjected to high humidity at the thermal aging show a significant reduction of insulation resistance during LOCA compared to cables which had been subjected to the same thermal aging in dry atmosphere.

The results also show that the cables which were subjected to intermittent vibration during the thermal aging indicated significantly lower insulation resistance, higher capacitance and higher loss factor at LOCA than the cables which had been subjected to the same thermal aging without vibration. The vibration levels used in the study were in excess of those occurring in normal installations in nuclear power plants.

To summarize, for the cables which were degraded to a degree that they showed significant changes in dielectric behavior during or after LOCA, there was a positive correlation between the mechanical degradation after the aging (in terms of indenter values) and the values of dielectric loss factors during LOCA.

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In Situ Partial Discharge Detection in Power Plant Cables

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Dc test often is used as a diagnostic test in power plants or production facilities cables due to ease and economy of use. However, the continuity of dc testing recently has been questioned due to limited effectiveness and also due to the destructive nature of test. Alternative test methods involving very low-frequency voltage have been developed. However, this technique does not provide information about the extent of degradation nor does it identify the location of defects. Moreover, this technique also may introduce trapped space charges.

Partial discharge (PD) monitoring is perhaps the best predictive tool to check the insulation integrity of cables. It is nondestructive and is a reliable test. It also provides information about the degree of deterioration and about the location of insulation defects. Remaining cable life estimation can be determined with PD testing. PD monitoring rapidly is gaining popularity in replacing hipot testing in cable. Recent PD monitoring equipment advancements allow real-time resolution recording of PD pulses, both in time and frequency domains. Both on-line and off-line PD measurements can be made on cables. Off-line testing normally is conducted by isolating the cable from the system and applying an external voltage to the cable. Voltages higher than the operating voltage of the cable normally are used. On the other hand, on-line PD monitoring technique normally is based on detecting PD produced by the cable-operating voltages. On-line testing is made while the cable is in operation.

Most of the power plants cables are extruded-dielectric cables. Extruded cables are insulated with organic materials such as polyethylene and EPR. The organic insulation can not resist the electron/ion bombardment and related events generally inherent to PD. PD pulses attack the insulation causing failure. Therefore, extruded cable is designed to operate free of any PD. If PD develops, it eventually will lead to a failure. Lead-time depends on the size and type of defect and on the cable type. Smaller defects take several months-to-years to cause a cable failure. However, in larger defects, such lead-time is days and sometime minutes. The majority of large defects are the result of smaller defects enlarged over time by PD. Therefore, the PD technique only is useful if it is capable of detecting PD produced by small defects. These PD are usually within the magnitude of the background noise.

This paper presents on-line PD detection system applicable to power cables used in power plants and production facilities. The system can also be utilized to test control and instrumentation cables. However, for control and instrumentation cables, off-line testing normally is used. The technique is applicable to both shielded and unshielded cables.

The PD system presented in this paper measures PD in both the time and frequency domains. The system is capable of detecting PD in a very noisy environment such as in power plants and production facilities. The combination of both domains is used to distinguish between PD generated in the cable under test and outside interference resulting from background RF noise and discharges generated in adjacent equipment.

Inductive couplers are used to pick up PD from cable under test. Inductive couplers are devices used to convert the magnetic signals resulting from PD to current pulses. They are designed to filter the 60 Hz signals and coupled to the PD higher frequency signals. In general, they detect PD pulses in the frequency range of 200 kHz to 200 MHz. These couplers consist of either high- frequency current-transformers clamped to the cable or high-frequency coils placed in close proximity to the cables. The signal from the inductive sensor then is coupled to a high-frequency preamplifier to increase the sensitivity of the system.

The frequency-domain testing is conducted using a spectrum analyzer. The analyzer is capable of conducting measurements in both full and zero-span modes. In the full span mode, the frequency range can be adjusted to examine signals in narrow-frequency bands as well as wide-frequency bands. The zero-span mode is used to examine single- frequency pulses in a time domain. The sweep time of the zero spans is used to find PD pulses occurring at one or more cycles of the operating voltage.

The time domain measurements normally are made using a pulse phase analyzer. The pulse-phase analyzer is capable of recording PD pulses sorted by their phase angle and magnitude.

Noise rejection also is made through a novel technique called visual differential noise rejection (VDNR). Noise surrounding the equipment under test is picked up by an antenna. The signal from the antenna, noise, and the signal from the PD coupler, noise and PD, are connected to a differential operational amplifier (DOA). The signal from the DOA is displayed on the spectrum analyzer in a wide-frequency band (1 MHz to 200 MHz). Gain control units are placed in front of both the non-inverting input, PD coupler, and the inverting input, antenna, of the DOA. Noise identification is made by changing the gain of the gain control unit of the inverting input. However, noise cancellation is made by adjusting the gain of both inputs of the DOA.

Distinguishing of PD from interference is achieved as follows: First, the spectrum analyzer is used to display the detected signals in respect to their frequencies. Each signal then is examined in the zero-span modes. PD signals will occur at the crest of the operating voltages. On the other hand, noise will have no pattern to follow. Once the PD-frequency range is identified, the signal then is displayed in the time domain using the pulse-phase analyzer. A filter system is used to allow the pulse-phase analyzer to record only pulses within the frequency range of the PD. Interference outside this frequency window will be rejected. The pulse-phase analyzer is fed with a reference ac cycle of the operating voltage. The phase-angle pattern will identify if that PD signal is generated in the equipment under test or in adjacent equipment. The pulse count and magnetite are used to indicate the problem's severity. The phase resolved partial discharge (PRPD) is used to identify the type of defect that produces the PD. The PRPD is a visualization tool obtained when the amplitude and phase of each individual PD pulse is recorded and sorted relative to the test voltage. The location of defects is achieved by comparing the frequencies of the detected partial discharge pulses. Higher frequencies indicate that the PD source is close to where the PD testing was made.

LIMITATIONS OF THE ARRHENIUS METHODOLOGY

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The Arrhenius methodology has been utilized for many years to predict polymer lifetimes in various applications. For nuclear power plant cables, it is often used 1) as a method for preaging cables to their 40-year operating lifetimes prior to DBE testing and 2) as a way to compress the aging times of equipment during LOCA and post-LOCA exposures. Unfortunately, there are numerous potential limitations associated with the Arrhenius approach, many of which can lead to non-Arrhenius behavior. This paper will review two of the most important of these limitations, diffusion-limited oxidation (DLO) and changes in the dominant reactions as the temperature changes. The former is a physical effect that is dependent upon the geometry (thickness) of the sample tested. It can lead to non-Arrhenius behavior for some degradation parameters, but not for others, dependent primarily on whether the parameter probes or is correlated to a surface property or reflects internal material. In many instances, we will show that DLO effects can be safely ignored. Changes in the dominant reactions with temperature, on the other hand, can happen for any material, making extrapolations beyond the experimental temperature range problematic. Unfortunately, when changes in mechanism occur, it invariably results in a reduction in effective Arrhenius activation energy, leading to lower than expected material lifetimes. Thus it is critically important to derive methods for testing the Arrhenius extrapolation assumption. One approach that we have developed involves ultrasensitive oxygen consumption measurements. Results from the application of this approach will be reviewed.

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Insights and Results from EPRI Cable and Component Aging Research Programs

Gary J. Toman, Nutherm International, Inc.
John Hutchinson, Plant Support Engineering, Electric Power Research Institute

Introduction

EPRI has been developing aging management techniques for environmentally qualified components. This paper will discuss the insights and results from EPRI research activities and the additional efforts that still remain to allow implementation of condition monitoring for cables. Management of aging of cables and components may be approached in many ways such as analysis of plant environments, identification and control of hot spots (adverse localized environments), and application of condition monitoring ranging from visual/tactile evaluation through sophisticated physical, chemical, or electrical testing. Attributes of these types of activities will be discussed.

Identification of Localized Adverse Environments

The development of the EPRI Adverse Localized Equipment Environments Guideline will be described. In addition, the evaluation of the effects of design temperatures zone by zone on cables will be described.

Correlation of Natural to Artificial Aging

After 13 years of cable specimen natural exposure in environments up to 61°C, only the neoprene jackets on certain cables are showing signs of limited aging. The results from the natural exposure have been as expected.

From the data developed in the EPRI Natural Versus Artificial Aging Program, the correlation of elongation-at-break with density changes under artificial aging will be described. Then, the density and elongation-at-break results from in-plant aging will be compared to that from artificial aging. The reasons for the difference in results and why the conservatism exists will be discussed.

Condition Monitoring of Cables

<u>Practical Concerns Raised by IAEA Round Robin Testing and In-plant Testing</u>

IAEA sponsored a set of round robin tests to evaluate cable material condition monitoring techniques. New and aged cable specimens were sent to laboratories around the world. The round robin test program will be described, and the diversity of condition monitoring results obtained on identical specimens will be discussed as will the degree of specificity necessary to obtain similar results from lab to lab.

In addition, practical issues identified during in-plant Indenter testing of cables will be discussed.

Use of Sensory Inspection in Cable Aging Management

While visual/tactile inspection may not give the precise degree of aging that has occurred to a cable, it can identify real problems requiring further evaluation and can also show where there is no concern. The technique can provide much information quickly for cable types dominantly used in US. Basic issues and tactics for visual/tactile inspection will be described.

Diagnostics Matrix for Selecting Useful Evaluation Techniques

EPRI has developed a matrix of diagnostic techniques for evaluating cable problems including aging. The matrix leads the user from the problem to a set of evaluation techniques and provides descriptions of the techniques, their limitations, and sources for more detailed information.

Conclusions

The development of useful condition monitoring techniques for cables is progressing. However, practical issues remain for many techniques with regard to field application and repeatability of results.

"Accident Testing of Artificially Aged and Naturally Aged Electric Cables"

Rober. Lofaro and Michael Villaran
Brook aven National Laboratory

Summary

Electric cables are used extensively in nuclear power plants to operate safety equipment, and transmit instrumentation and control signals. However, before they can be used, the cables must first pass a qualification process which involves preaging of the cable to simulate actual service conditions, followed by testing under simulated accident conditions. Recently, the U.S. Nuclear Regulatory Commission has identified issues regarding this qualification process which are being addressed in a focused research effort. Brookhaven National Laboratory is the lead laboratory assisting the NRC in this research.

In the BNL program, three types of low-voltage instrumentation and control cable are being tested in six test sequences. The cable types being studied are 1) cross-linked polyethylene (XLPE) insulation with Neoprene® jacket, 2) ethylene propylene rubber (EPR) with unbonded Hypalon® jacket, and 3) EPR insulation with bonded Hypalon® jacket. Each test sequence includes four groups of cables, which are 1) unaged control specimens, 2) cables artificially aged to match the naturally aged cables in group 3, 3) naturally aged cables from a nuclear facility, and 4) cables artificially aged to simulate 20, 40 or 60 years of service per IEEE Standard 323-1974 guidelines. All four cable groups are exposed to simulated accident conditions as part of the test sequence. Condition monitoring data is obtained periodically throughout the artificial aging and accident testing. To date, two test sequences have been completed.

Results of Test Sequence 1

The first test sequence included cables with XLPE insulation and Neoprene® jackets. Naturally aged cables that are approximately 10 years old were obtained from a decommissioned nuclear power plant, along with matching unused cable of the same type and materials. The artificial aging for the group 2 cables included 2.86 hours of thermal aging at a temperature of 120 °C followed by 0.6 Mrad of radiation at a dose rate of 0.33 Mrad/hr. The group 4 cables were thermally aged for 650 hours at a temperature of 150 °C, after which they were exposed to 25 Mrad of radiation at a dose rate of 0.47 Mrad/hr. The aging parameters were selected based on those used in the original qualification of the cables.

The accident exposure included 150 Mrad of radiation followed by exposure to high temperature/pressure steam and chemical spray. The profile used included a double peak at 174 °C and 113 psig, and a duration of 7 days. During the accident exposures the cables were energized and loaded using a pressure transmitter circuit. A separate circuit was used for each cable with all pressure transmitters monitoring a common pressurized manifold. Circuit performance parameters monitored were circuit current, leakage current and blown fuses.

All cables were initially in excellent condition with no cracking or embrittlement of the jacket or insulation. After artificial aging, the group 2 cables were still in excellent condition and showed little or no signs of aging. The group 4 cables showed significant aging degradation after thermal aging including cracking and embrittlement of the jackets. The insulation was slightly stiffer, however it was not cracked. In general, exposure to the accident radiation produced slight stiffening for the cables.

During the accident exposure no performance anomalies were observed for the cables in groups 1, 2 or 3. However, several anomalies were observed for the group 4 cables, including leakage currents and blown fuses. Post test examination showed that the splices used to attach the cables to the instrument test leads were not adequately installed, which allowed moisture intrusion into the splice area. It was determined that this splice deficiency led to the performance anomalies noted. Therefore, it was concluded that all cable specimens performed acceptably during the accident testing.

Results of Test Sequence 2

The second test sequence included cables with EPR insulation and unbonded Hypalon® jackets. Naturally aged cables that are approximately 25 years old were obtained from a decommissioned nuclear power plant, along with matching unused cable of the same type and materials. The artificial aging for the group 2 cables included 30.3 hours of thermal aging at a temperature of 121 °C followed by 3.25 Mrad of radiation at a dose rate of 0.49 Mrad/hr. The group 4 cables were thermally aged for 84 hours at a temperature of 121 °C, after which they were exposed to 25.5 Mrad of radiation at a dose rate of 0.49 Mrad/hr. The aging parameters were selected based on those used in the original qualification of the cables.

The accident exposure included 150 Mrad of radiation followed by exposure to high temperature/pressure steam and chemical spray. The profile used included a single peak at 171 °C and 60 psig, and a duration of 7 days. During the accident exposures the cables were energized and loaded as described for the first test sequence.

All cables were initially in excellent condition with no cracking or embrittlement of the jacket or insulation. After artificial aging, both the group 2 and group 4 cables were still in excellent condition and showed little or no signs of aging. Exposure to the accident radiation produced slight stiffening of the cables.

During the accident exposure no performance anomalies were observed for any of the cables. Therefore, it was concluded that all cable specimens performed acceptably during the accident testing.

Preliminary Observations

While testing is not complete, several preliminary observations can be made from the results obtained from the first two test sequences. From test sequence 1, the cables in group 4 were severely degraded after artificial aging to simulate 20 years of service. This suggests that the artificial aging requirements to simulate 40 years of service may be conservative.

The CM data obtained indicate that several of the techniques being studied are promising tools for in situ monitoring of cable degradation. These include the indenter, oxidation induction and ac impedance. Visual inspections also appear to be useful for monitoring cable degradation.

Temperature and Dose Monitoring of Surroundings, and Accelerated Aging of Electric Cables in Nuclear Power Plant

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Electric cables for nuclear power plant are essential to safety operation of the plant. Therefore, it is important to estimate precisely degradation of the electric cables during a use of 40 years that is established as a legal life time of nuclear reactor. There are two estimation methods of the cable degradation, one is accelerated aging method, and the other diagnostic one which is required non-destructive and more conventional methods. We study currently methods of the latter, and describe the former here.

The accelerated aging method is composed of following two stages, first, we monitor temperatures and dose rates, which are main factor of cable degradation, on actual surroundings around the installed cables in nuclear power plants, and, second, on the base of these measured condition, various insulators of the cables are tested and evaluated by the accelerated aging of irradiation and thermal to estimate the life time of the cables.

<u>Monitoring methods of temperature and average dose rate around the cables in nuclear facility:</u>

It is difficult to measure dose-rate by survey meters at many point inside of a container vessel of nuclear reactor. Alanine-polystyrene dosimeter which is developed in Japan is useful for these measurement. This dosimeter is a small size(3 mm ϕ ,3cm long), and can be used to measure a low dose rate for a very long term, because there is little attenuation of radicals in or after the irradiation. Radicals in alanine are produced in proportional to irradiation dose. For example, the alanine-polystyrene dosimeters will be put on places where are to be measured a dose rate around the installed cables during ten months or one year that is an interval of maintenance of nuclear reactor. There is little attenuation of doses in the alanine-polystyrene dosimeter when it put on the places under 60℃ for 200 days. There are decay data of dose on this dosimeter until 380 days. Using these decay data, the accumulation doses could be monitored around the cables in nuclear power plant.

There is a temperature measurement system of optical fiber as a sensor, which is used for a long distance. This is based on that the power of back scattering light of Raman light is proportional to temperature of a optical fiber installed, when laser pulse light sends to the fiber placed in various temperature. When optical fiber material has high radiation resistance, such as pure silica core fiber, a distribution of temperature along the optical fiber is obtained over a length of 1km at accuracy of 1℃ per 1m under low dose rate(about 1Gy/h). By using these methods mentioned above, we can obtain more accurate condition about dose rates and temperatures around the installed cables in nuclear power plant.

<u>Accelerated aging method for the cable:</u>

The condition of dose rate and temperature around the installed cables in a container vessel is mainly said as dose rates less than 1Gy/h and as temperatures of ca.50℃ under operation. We must evaluate simultaneous radiation and thermal deterioration of cable insulator and sheath materials for 40 years. The acceleration in dose rate is simply obtained by using higher dose rates. For example, when dose rate is 100 times higher than 1Gy/h, the irradiation is accelerated by 100 times, i.e., the irradiation time is 100 times shorter than that of 1Gy/h, because, in low dose irradiation, active regions produced by γ-ray in polymer insulator do not overlap each other. On the other hand, Arrhenius' plot of deterioration is used in the acceleration of thermal aging, in which we extrapolate higher temperature data(over ca.120℃) to lower temperature ones which are actual conditions(around 50℃). However, as there is a possibility in which degradation behavior in the polymer materials is different between higher temperatures and lower temperatures, it is necessary to obtain proper activation energies for the acceleration of thermal aging . We found that the activation energy of deterioration in the polymer materials changes at a boundary of ca. 100℃; the activation energy is 127kJ/mol(30.5 kcal/mol) above 100℃, and 56.5kJ/mol(13.5kcal/mol) below 100℃ for ethylene-propylene rubber(EPR) used as a cable insulation material. This is mainly caused an ununiformity of oxidation inside of the polymer. By using this changing activation energies, deterioration test period can be shorten to several months instead of 40 years.

In our accelerated aging method, we must set up the degradation condition in which the accelerating values of a dose rate and a temperature to the base condition are coincident. For formulated EPR, we had examined the simultaneous radiation-thermal accelerated test on the condition of 50(50Gy/h,117 ℃), 100(100Gy/h,128 ℃), 300(300Gy/h,143 ℃) and 600 (600Gy/h,152℃) time- acceleration to the base condition of 1Gy/h,50℃. If we adopt the life-time of EPR as time when a elongation at break reaches to 100%, its life-time is estimated 26years by extrapolating the Arrhenius' plot obtained from those accelerated test data. This value agrees well with degradation test data by 70 ℃ for a long time.

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10. SUPPLEMENTARY NOTES

S. Nesmith, NRC Project Manager; Transactions prepared by Brookhaven National Laboratory

11. ABSTRACT *(200 words or less)*

This contains summaries of papers to be presented at the Twenty-Sixth Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 26-28, 1998. 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 and information exchange during the course of the meeting, and are given in the order of their presentation in each session.

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Twenty-Sixth 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 Chair: E. Hackett	2 Severe Accident Research, Fission Product Behavior Chair: J. Schaperow
1:30 pm to 5:00 pm	3 Nuclear Materials Issues and Health Effects Research Chair: C. Trottier	4 Materials Integrity Issues Chair: A.L. Lund
Tuesday 8:30 - 9:45 am	**PLENARY SESSION** ***(Congressional Ballroom)***	
10:00 am to Noon	5 Digitial Instrumentation and Control Chair: J. Calvert	6 Structural Performance Chair: A. Murphy
1:30 pm to 5:00 pm	7 The Halden Program Chair: J. Persensky	8 PRA Methods and Applications Chair: N. Siu
Wednesday 8:30 am to Noon	9 Thermal Hydraulic Research Chair: C. Boyd	10 Plant Aging I Chair: J. Vora
1:30 pm to 5:00 pm	11 High Burn-up Fuel Chair: R. Meyer	12 Plant Aging II Chair: J. Vora

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

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