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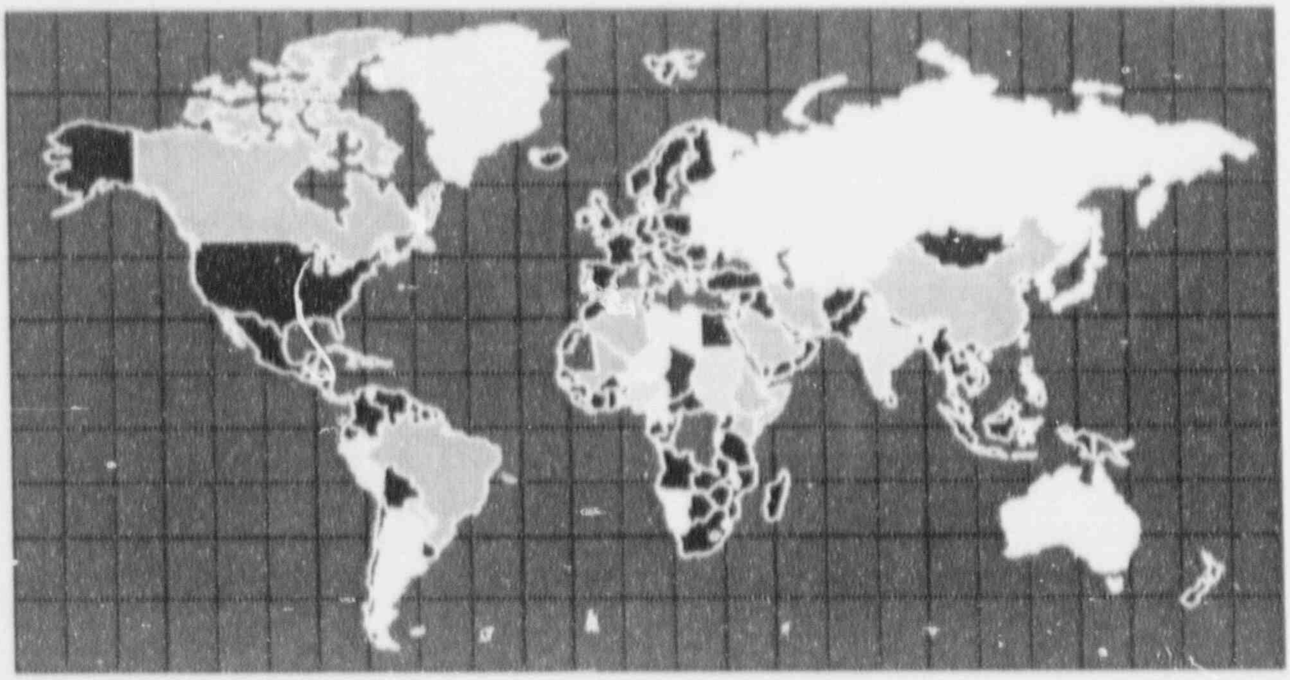
Aging Research Information Conference–Abstracts of Papers



Held at
Holiday Inn Crowne Plaza
Rockville, Maryland
March 24-27, 1992

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U.S. Nuclear Regulatory Commission

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ABSTRACT

This report contains abstracts of papers to be presented at the Aging Research Information Conference held at the Holiday Inn Crowne Plaza in Rockville, Maryland, on March 24-27, 1992. This conference is held to disseminate research results in the area of nuclear power plant aging from programs sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission. The conference will also provide an opportunity for engineers and scientists from around the world to exchange technical information and discuss future international cooperation. The abstracts appear in the order in which they will be presented at the conference, and they are grouped by technical session. The full papers and the agenda for the conference will be published as separate documents.

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NRC Plant Aging Research Program - Overview

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Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

ABSTRACT

Aging degradation in operating nuclear power plants must be managed to prevent safety margins from eroding below the levels provided in plant design bases. "Aging" is universal in nature. No industrial complex including nuclear power plants (NPP) should be considered immune from its effects. For NPP, aging is manageable if its symptoms are recognized and predicted, if it is monitored, and if appropriate steps are taken for timely mitigation of age-related degradation.

The Division of Engineering's (DE) Plant Aging Research program is a multi-branch, multi-disciplined, integrated approach to understanding and managing age-related degradation in safety-related components, systems, and structures (CSS) by the Materials Engineering, the Electrical and Mechanical Engineering, and the Seismic and Structual Engineering Branches.

The hardware oriented engineering research is concerned with degradation of: 1) the components of the primary system pressure boundary (with principal attention paid to the reactor pressure vessel), 2) safety-related electrical and mechanical components and systems, and 3) the civil engineering structures and materials, including containment. Emphasis on the reactor pressure vessel stems from its importance to plant safety. It is a component which is not redundant and for which failure is not acceptable. The pressure vessel integrity research program is the oldest of three areas of aging research providing a high quality information data base to understand and make regulatory decisions involving fracture mechanics; fatigue life, including initiation and propagation; material properties and flaw growth; irradiation effects; neutron dosimetry and surveillance data analysis; non-destructive examination (NDE) techniques; and annealing issues including validation testing. Programs in the electrical and mechanical engineering area and the civil engineering area are similarly developing the needed technical information data bases and guidance for understanding and managing aging of selected CSS in nuclear power plants of all ages and types.

The aging management process central to these efforts, as developed by the DE research program, consists of three key elements: 1) select and prioritize components, systems, and structures for which aging must be managed, 2) identify and understand the relevant aging mechanisms and their effect on the properties or performance of the selected CSS, and 3) take appropriate action to manage degradation through effective inspection, surveillance, condition monitoring, trending, preventive and corrective maintenance, and mitigation to prevent reduction in safety margins.

This paper provides an overview of the aging-related research programs sponsored by the Division of Engineering, Office of Nuclear Regulatory Research.

NUCLEAR POWER PLANT COMMON AGING TERMINOLOGY

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ABSTRACT

The U.S. NRC and the nuclear industry programs and activities related to understanding aging mechanisms, condition monitoring, failure and residual life evaluations, and maintenance have created the need for developing uniform terminology in many of these areas.

A technical committee with nine members from the utility industry and regulatory research established the scope of nuclear power plant common aging terminology and used a systematic technical and lexicographical approach in developing common definitions. The definitions cover 93 terms related to degradation, life cycle, and aging management of systems, structures, and components. Results have been issued by the Electric Power Research Institute (EPRI) for trial use and comment. This paper gives an overview of the effort and solicits constructive feedback.

RECORDKEEPING TECHNOLOGY TO SUPPORT AGING MANAGEMENT

J. S. Dukelow
Pacific Northwest Laboratory

Pacific Northwest Laboratory[a] has investigated the capability of current recordkeeping technology to support aging management. This paper discusses technical issues associated with potential enhancements of nuclear plant records systems from the perspective of the lessons learned about equipment aging degradation mechanisms and associated surveillance and monitoring techniques during the U.S. Nuclear Regulatory Commission's Nuclear Plant Aging Research Program. It considers both the specific types of technical data needed to ensure continued safe operation and the use of new technology to upgrade record systems.

Specific topics discussed in this paper include:

- equipment reliability data needed to support the assessment of the impact of aging on the continued operation of the plant
- operational history data to support the assessment of residual life of mechanical and structural components and piping
- tools for the analysis and trending of equipment reliability data and operational history data
- design and implementation of plant record systems that will provide comprehensive and usable engineering design basis for the plant
- proposed improvements in the data input process for the plant records system
- computerization of plant records systems, including conversion of existing records into machine readable forms.

(a) Pacific Northwest Laboratory is operated for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830. This work was conducted under NRC FIN B2865.

TREATMENT OF AGING IN ROUTINE SURVEILLANCE AND TESTING PRACTICES - EXAMPLES

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Pacific Northwest Laboratory

J. I. Vora
U. S. Nuclear Regulatory Commission

Routine surveillance and testing are performed on components and systems that are important to the safety of nuclear power plants. These surveillances and tests ensure the operability and availability of those components and systems. The results of these activities can be used to detect, monitor, and trend age-related degradation. Further, this information/data could be useful to focus the surveillance and monitoring of those components and systems susceptible to significant age-related degradation, optimize maintenance effectiveness, and maintain the safety envelope as plants advance in age.

Routine surveillance and testing practices (RSTPs) can be effectively used to detect and manage aging. For example, RSTPs serve as the major means of aging detection in dynamic plant equipment such as pumps, valves, breakers, and switches. However, the effects of aging have not always been recognized explicitly in the RSTPs. More specific attention could be paid to 1) the types of aging mechanisms that are active on the components and systems that are important to safety, 2) the most effective methods of mitigation to counter the effects of the subject aging mechanisms, and 3) the RSTPs that can be used to detect, trend, and augment control of age-related degradation.

Pacific Northwest Laboratory[a] under the Nuclear Plant Aging Research (NPAR) Program is evaluating RSTPs for representative components from three perspectives: 1) the extent to which they address age-related degradation; 2) their potential contribution to accelerated aging and service wear (e.g., fast starts of diesel generators, and the loads imposed on the auxiliary feedwater pump while testing in a pumping mode); and 3) their capabilities for detecting the symptoms of aging. These evaluations are performed by 1) reviewing, from an aging perspective, representative components, and 2) recommending improved RSTPs (and, if applicable, the associated bases) to account for aging. Examples of evaluations for the emergency diesel generator system and the service water system follow.

Emergency Diesel Generator System

The emergency diesel generator (EDG) system provides backup electrical power needed by a nuclear power plant in the event of a loss of offsite ac power. The EDG components include the instrument and control, fuel oil, starting, cooling, lubricating oil, intake, exhaust, and generating systems,

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(a) Pacific Northwest Laboratory is operated for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830. Work is conducted under NRC FIN B2865.

and the engine. An NPAR study determined that these components are subject to degradation attributable to corrosion, vibration-induced fatigue, oxidation, thermal stresses, shock, contamination, biofouling, manufacturing and design errors, improper or excessive operation, improper or inadequate maintenance, and adverse environmental conditions. Many, if unmitigated, may enhance with time and age. Each of the component systems of the EDG is subject to its own set of degradation mechanisms, and each mechanism causes varying degrees of degradation. Some RSTPs are effective for detecting aging mechanisms, some are neutral, and some promote degradation. The evaluation concluded with recommendations to improve and supplement the RSTPs.

Service Water System

The three types of service water systems (SWSs) perform vital safety functions in nuclear reactors and are the final link between the reactor and the ultimate heat sink, which is typically a sea, river, lake, or cooling pond. SWSs also provide cooling to safety-related equipment, such as EDGs and emergency core cooling systems. Depending on its design, all or part of a SWS may be exposed to raw or relatively aggressive treated water. Therefore, SWS components are subject to many age-related degradation mechanisms.

Aging in SWSs is a complex subject and is only now being studied in sufficient depth to focus on effective methods of mitigating aging mechanisms. An NPAR study has identified the age-related degradation mechanisms that are active in SWSs and where within the SWS these mechanisms are operative. The study has also summarized a review of the RSTPs to evaluate their effectiveness in detecting age-related degradation mechanisms, and has developed recommendations to improve and supplement the RSTPs. Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment," dated July 18, 1989, contains specific recommendations to address SWS aging. However, there are critical areas within the SWS that the GL does not address, such as the need to verify that the degradation rates of heat exchanger tubes, tubesheets, and waterboxes will remain above the minimum test limits through the next surveillance interval. A similar statement could be made for other components in the SWS. The NPAR recommendations augment those of the GL to form a complete set of RSTPs that focus on the aging management of the SWS.

Degradation Modeling - A Key to Understanding Effects of Aging and Maintenance*

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SUMMARY

Component degradation modeling is the analysis of component degradations for the purpose of developing models of the degradation process and its implications. Degradation modeling can encompass many different areas, from the microscopic modeling of material degradation processes to macroscopic modeling of times of occurrences of degradations. In this paper, we present basic concepts, approaches, applications, and data needs of degradation modeling using times of occurrences of component degradations and failures.

We discuss degradation modeling from the viewpoint of understanding the effects of aging and the role of maintenance in mitigating the aging effects. We argue that degradation modeling is a key to understanding the effects of aging and maintenance and should be the principal focus of aging analysis. Since degradations generally occur before failures, detecting aging trends in degradations allows the aging effects to be corrected before they impact failures. Furthermore, degradations generally occur more frequently than failures, providing a larger data base for analyzing aging effects.

In this paper, we discuss the basic concepts and mathematical development of a simple degradation model where the operational performance of a component is divided into three states - normal operating state, degraded state, and a failure state. Traditional reliability analysis treats only two states of a component - normal operating state and failure state. Using Markovian approaches and renewal theory, we establish relations among the states using rates of degradation and failure occurrences. The relations are used to define the estimates of the effectiveness of maintenance in preventing degradations from becoming failures.

Specific applications of the modeling approaches are performed for "active" components. Both standby and continuously operating components are analyzed. All the applications demonstrate the usefulness and benefits of analyzing

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

component data using degradation modeling approaches. In case of residual heat removal (RHR) system pumps, a standby "active" component, degradation rate shows a "bathtub" curve where a distinct, increasing aging trend is observed at later ages. The pump failure rate does not show any increasing aging trend for the same period demonstrating early indication of aging trends through analyses of component degradations. For air compressors, a continuously operating component, both the degradation and failure rate show aging trends. The failure rate, which is significantly lower than the degradation rate in the first three years, increases significantly faster than the degradation rate and reaches the same value towards the end of the ten year period analyzed. The effectiveness of maintenance in preventing age-related degradations from transforming to failures is also determined which is useful in understanding the effectiveness of existing maintenance activities.

Degradation modeling approaches can have broader applications in aging-risk studies, in defining the effective maintenance practices, and in analyzing component reliability performance. Extensions of degradation modeling approaches to study reliability effects of different maintenance intervals, different maintenance durations, and different maintenance efficiencies also are discussed. This extension will help define the maintenance activities to mitigate aging effects, complement to the evaluation of the effectiveness of existing maintenance practices. We also discuss additional application areas and the data needs of the degradation modeling approaches.

RISK EVALUATIONS OF AGING

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Aging of active and passive components can cause significant risk increases if the aging is not effectively managed. It is therefore important to evaluate the risk effects of aging in order to prioritize aging contributors and to evaluate the effectiveness of aging management programs. A methodology for risk evaluations of aging has been developed as a part of the Nuclear Plant Aging Research (NPAR) Program being conducted by the Office of Research of the U.S. Nuclear Regulatory Commission. The methodology has been reported in NUREG/CR-5510 (1) and in Reference (2).

The methodology which has been developed allows any current probabilistic risk assessment (PRA) to be used to quantify and prioritize aging effects. The PRA does not need to be completely requantified to incorporate aging effects, but instead suitable risk importance coefficients are extracted from the PRA and are combined with aging models for the individual components. To apply the methodology, component aging failure rate data are required, which can be based on engineering knowledge and generic data as well as plant specific data. Useful sensitivity studies can also be carried out by systematically varying the component aging rates to determine the degree of aging control of a test and maintenance program and to identify the most aging-impacting contributors.

The risk evaluations of aging methodology has been peer reviewed and has been applied to support the regulatory analysis for license renewal rulemaking as described in NUREG 1362 (3). In the demonstrations and applications which have been carried out, even when many components are aging, very few significantly impact the core damage frequency. However, if these relatively few, risk-important contributors are not identified and are not effectively maintained, large core damage frequency increases can occur. Furthermore, existing test and maintenance programs may not adequately focus on the risk-important contributors.

This paper describes highlights of a procedures guide which has been recently completed as a further step in transferring the technology to the user. The procedures guide is presently under review and will be published shortly as a NUREG/CR. Specific steps in transforming a standard PRA to an age dependent PRA are described, including specific aging models which can be used. The discussions focus on a Level 1 PRA, or equivalently a probabilistic safety assessment (PSA), which evaluates system unavailabilities and core damage frequency. However, the same procedures can be applied to Level 2 or 3 PRAs which evaluate radioactive releases or public health risks. Procedures for prioritizing aging contributors and for evaluating the risk-effectiveness of aging management programs are described. Demonstrations and insights are also given.

REFERENCES

1. NUREG/CR-5510 "Evaluations of Core Melt Frequency Effects Due to Component Aging and Maintenance", June 1990.

2. W.E. Vesely, "Incorporating Aging Effects into Probabilistic Risk Analysis", Reliability Engineering and System Safety, Vol. 32, No. 3, 1991, pp. 315-337.

3. NUREG-1362, "Regulatory Analysis for Proposed Rule on Nuclear Power Plant License Renewal", July 1990.

A Technique Of Including The Effect Of Aging Of Passive Components In Probabilistic Risk Assessments[a]

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The probabilistic risk assessments (PRAs) being developed at most nuclear power plants to calculate the risk of core damage generally focus on the possible failure of active components. The possible failure of passive components is given little consideration. We are developing methods for selecting risk-significant passive components and including them in PRAs. The methods provide effective ways to prioritize these passive components for inspection, and where inspection reveals aging damage, mitigation or repair can be employed to reduce the likelihood of component failure. We demonstrated a method by selecting a weld in the auxiliary feedwater (AFW) system. The selection of this component was based on expert judgement of the likelihood of failure and on an estimate of the consequence of component failure to plant safety. We then modified and used the PRAISE computer code to perform a probabilistic structural analysis to calculate the probability that crack growth due to aging would cause the weld to fail. The PRAISE code was modified to include the effects of material properties with age and changing stress cycles. The calculation included the effects of mechanical loads and thermal transients considered in the design and the effects of thermal cycling caused by a leaking check valve. We modified an existing PRA (NUREG-1150 plant) to include the possible failure of the AFW weld, and then we used the weld failure probability as input to the modified PRA to calculate the change in plant risk with time. The results showed that if the probability of component failure is high, the effect on plant risk is significant. However, this particular calculation showed little change in plant risk for 48 years of service. The success of this demonstration shows that this method could be applied to nuclear power plants.

The demonstration showed the method is too involved for handling a large number of passive components and therefor simpler methods are needed. A simpler method identified was to screen to limit the number of passive components and for the limited list use simpler methods to estimate the aging failure rate. Once the list of passive components is identified and the aging failure rates are estimated the PRA can be modified to incorporate them.

a. Work sponsored by the U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under DOE Contract No. DE-AC07-76ID01670; Dr. G. H. Weidenhamer, Technical monitor

The screening can be done in several ways. A method identified was to select passive components based on consequence. The consequence of passive component failure can be estimated using the PRA as a guide and a plant specific review. Active components in the PRA can be used as surrogates for the passive components to determine the "importance" or contribution to the risk. Another method is to use this list and further reduce it by eliminating those that do not have an identifiable aging mechanism. Another is to include all passive components that could lead to system malfunction.

In the analysis we performed at the INEL the modified version of the PRAISE code was used to estimate the aging failure rate of one specific weld, but this code is too large to efficiently calculate the failure rate of a large number of passive components in a nuclear power plant. Therefore techniques were identified to estimate the failure frequency on a large number of components. These techniques include data analysis of actual failures, expert solicitation, questionnaires, simple structural probabilistic models, as well as the large models such as PRAISE. Data analysis is the most direct approach. Statistical analysis and aging techniques developed by NPAR can be used to determine the failure probability parameters if a number of failures have been observed--but this data is usually not available on passive component failures. Expert solicitation can be used to estimate the failure frequency and "wear out" portion of the bathtub curve. This technique has been demonstrated by a committee sponsored by the NRC, called Tirgalex, and by Pacific Northwest Laboratories in their work on risk-based inspection, but this technique is only as good as the experts and information available to them. In addition, where it may be easy for an expert to estimate stress or temperature difference it can be difficult for an expert to estimate a probability from this information. Questionnaires have been developed and used to estimate the failure frequency of passive components especially during the design phase of a project. The questionnaire lists the parameters for individual components such as types of materials, numbers and relative size of mechanical and thermal cycles, etc. so that a failure frequency estimation can be made. The basis of the failure frequency estimation is numerous probabilistic structural analysis calculations on similar piping systems. Simple probabilistic structural analysis models are the most exciting development. These computer models allow simple user friendly input in a model that is developed from the larger structural analysis codes. Many calculations can be performed in a short period of time to perform optimization studies, specify inspection frequencies and accuracies necessary to obtain a specific failure probability. Larger models such as PRAISE can be used on very large consequence passive components where more accuracy is needed.

Screened passive components can be included in a PRA and standard risk analysis results can be used to identify "target level" aging failure rates that must be maintained to control risk.

Validation Issues In Aging Risk Evaluations*

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SUMMARY

As nuclear power plants grow older, the aging plant components may be increasingly susceptible to failure with consequent potential for increase in plant risk. Evaluation of plant risk from aging is important not only from a safety perspective but to implement effective and efficient aging management programs to assure reliable plant operation as well. Significant efforts have been made in recent years to develop methodologies to evaluate the risk impact from aging of nuclear power plant components and systems. The aging risk evaluation approaches which have been developed are based on extension of Probabilistic Risk Assessment (PRA) techniques and can be used to predict overall risk to plant from aging and/or to prioritize detailed contributors to risk from aging. In these approaches, the aging effects are incorporated into existing plant PRAs to predict plant risk behavior with age. Different approaches may be used to incorporate the aging effects into the PRA and a number of parametric models are available to describe component or system behavior with age. The prioritization is determined through an evaluation of the risk contribution of individual components and systems. There are, however, uncertainties in these evaluations arising primarily from, (a) sparseness of available component aging data, (b) assumptions in component aging behavior modeling, and (c) assumptions in plant risk models and in risk quantification approaches. These variabilities in modeling assumptions and data can be significant and can affect the applicability of aging risk evaluations.

This paper summarizes the issues involved in the validation of aging risk evaluations for different applications. How these issues are relevant in aging risk estimations, aging prioritizations, and/or other inferences are discussed. The objective of the validation study is to evaluate the sensitivity of the aging risk evaluations to modeling and data variabilities and to identify applications which are robust against such variabilities. As a first step sensitivity studies using a NUREG-1150 plant are currently underway to assess the impact of model and data uncertainties on component prioritizations. The results of this study will help identify prioritization schemes which are robust and meaningful for applications. Recent developments from this work are discussed.

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MAINTENANCE PRACTICES TO MANAGE RISK ASSOCIATED WITH AGING-RELATED SAFETY ISSUES

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As part of the Nuclear Plant Aging Research (NPAR) Program, supported by the Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratory (PNL)[a] has developed a risk-based method that may be applied to establish maintenance practices for operating nuclear power plants. The approach uses existing results of probabilistic risk assessments (PRA) and related studies and relevant industry experience to provide guidance for managing risk associated with component aging. The developed method consists of three major steps: the first step develops a graded approach to maintenance, which identifies and prioritizes risk significant systems, structures, and components (SSCs) to focus maintenance effectiveness; the second step analyzes and characterizes system and component aging degradation; the third step identifies safety significant SSCs and their associated aging failure mechanisms useful to develop an effective aging maintenance program.

This paper provides suggestions that may be considered during the development of maintenance practices intended to focus on the management of risk associated with SSC aging. The suggestions can be applied to a) develop a risk-based graded approach to maintenance, b) assess the impact of the aging process on safety significant SSCs, c) develop a maintenance program that recognizes the aging of safety significant SCCs as a unique risk factor, and d) incorporate a plan for managing this risk area.

A risk-based graded approach to maintenance involves selective and judicious assignment of resources to maintain facilities and equipment based on site-specific risk quantification. Thus, it is essential that a site-specific maintenance plan ensures that the risk associated with aging of safety significant SSCs be included in this risk quantification. Two primary issues that relate directly to potential risk are 1) priority of performing maintenance and 2) the depth and extent of procedures, training, qualification of personnel, procurement control, quality assurance, reporting, and documentation. For nuclear applications, techniques normally employed in formulating a graded approach to maintenance for a specific facility are the PRA, described in NUREG/CR-2300; Reliability Centered Maintenance (RCM), a technique borrowed from the aviation industry; a judicious review of both plant-specific and industry-wide experience, as documented in LERs, bulletins, notices, the Nuclear Plant Reliability Data System (NPRDS), and in-plant maintenance, operations, and quality control records. The recorded results of a plant's self-assessment are also useful in formulating a graded approach. There are, however, advantages and limitations associated with each of these techniques;

(a) Pacific Northwest Laboratory is operated for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830. Work is conducted under NRC FIN B2865.

hence, all three should be considered when formulating a graded approach for a specific site.

Assessing the effect of aging on safety significant SSCs, identified as a result of site-specific risk quantification, requires a thorough understanding of aging mechanisms and the stressors imposed by these mechanisms. It must be understood that equipment items and component parts age and deteriorate over long outage periods as well as during operating periods, due to numerous aging mechanisms. Additionally, many system hardware failures that occur during service are traceable to built-in (latent) manufacturing defects that accelerate specific aging mechanisms. These defects may pass quality inspection during manufacturing but finally become evident during plant operation. In NPAR program studies, PNL has identified twenty-seven component aging mechanisms. A glossary of these issues is included in NUREG/CR-5490, Vol 1. Results from related PNL studies regarding system and component prioritization activities may also be applicable.

In developing an effective maintenance program that minimizes risk resulting from aging of safety significant SSCs, it is useful to review the lessons learned by others who have been confronted with similar issues. Where appropriate, these lessons learned can be used for developing maintenance plans. Feedback mechanisms, which continually improve and refine the program, are vital if the maintenance program is to address changing needs, such as the degradation of plant equipment due to aging.

To establish a broader perspective for managing age-related degradation, PNL analyzed effective maintenance activities used by two commercial industries and two military organizations to manage the aging of systems and components. The four programs considered were: a) the U.S. commercial airline industry; b) the U.S. Air Force B-52 bomber; c) the U.S. Navy Ballistic submarine; and d) the Japanese nuclear power industry. The maintenance programs of these four organizations offered lessons potentially useful for managing aging in commercial nuclear power plants. A summary of the maintenance-related activities to address system and component aging, based on the approach taken by each of these four organizations, is presented in Table 1 of the final report for this study (PNL-7823). PNL also reviewed maintenance practices at various U.S. commercial nuclear power plants.

The suggestions set forth in this paper could be useful in establishing overhaul frequency, developing inservice inspection programs, developing outage plans and establishing control procedures. These suggestions could also be useful in selecting the proper balance among maintenance methods (e.g., preventive maintenance versus corrective maintenance), selecting replacement components, especially where diminishing manufacturing source is an issue, and establishing design modification needs to ensure adequate maintainability with respect to aging issues. Moreover, the approach set forth has the potential for enhancing maintenance effectiveness to manage aging by providing suggestions for focusing maintenance on risk significant SSCs.

Environmentally Assisted Cracking and Fatigue of Reactor Structural Materials in LWR Environments*

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

Fracture-mechanics CGR tests have been performed on composite specimens of A533-Gr B/Inconel-182/Inconel-600 plated with nickel, and on homogeneous specimens of A533-Gr B material plated with chrome and nickel. Conventional unplated specimens have also been tested to provide baseline data. In the tests to date, plated specimens have been more susceptible to environmentally enhanced cracking than unplated specimens. The proposed revised ASME Section XI da/dN versus ΔK curves were used to calculate equivalent da/dt versus K_{max} curves for comparison. Even for the plated specimens, the data are in reasonably good agreement with the predictions for all data from water containing 200 ppb dissolved oxygen.

Irradiated austenitic SSs can become susceptible to SCC. This susceptibility been attributed to radiation-induced segregation (RIS) or depletion of elements such as Si, P, S, Ni, and Cr. High (HP)- and commercial-purity (CP) Type 304 SS specimens were obtained from control blade absorber tubes after irradiation to fluences of up to 2×10^{21} $n \cdot cm^{-2}$ (E > 1 MeV) from two operating BWRs. Microchemical and microstructural changes in the steels were studied by Auger electron spectroscopy. Significant RIS of Si, P, Ni, and an unidentified element or compound associated with an Auger energy peak at 59 eV was observed in the CP material. Except for Ni, such segregation was negligible in the HP material. No evidence of S segregation was observed in either material. However, chromium depletion from grain boundaries was more pronounced in the HP than in the CP material. Slow-strain-rate-test were conducted on the CP and HP materials in air and in simulated BWR water. The HP material showed significant intergranular stress corrosion cracking in water and the strain to failure for comparable fluence levels was lower in the HP material than in the CP material.

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* Work supported by the office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission FIN A22122, Program Manager Dr. J. Muscara.

RISK-BASED INSPECTION FOR MANAGEMENT OF AGING DEGRADATION

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As part of the Nondestructive Evaluation Reliability Program, sponsored by the Nuclear Regulatory Commission (NRC), Pacific Northwest Laboratory (PNL) is developing methods that use risk-based approaches to establish in-service inspection (ISI) plans of nuclear power plant components. The method first uses results of probabilistic risk assessment and failure modes and effects analysis technique to identify and prioritize the most risk-important systems and components for inspection priorities. Once high-priority components have been identified, an approach recommended by the American Society of Mechanical Engineers (ASME) Research Task Force on Risk-Based Inspection Guidelines is used to determine the target (acceptable) risk and failure probability values for individual components. Aging degradation that may impact on the estimated component failure probabilities are addressed. Inspection programs (method, frequency, extent) are then developed to manage these aging mechanisms. Probabilistic structural mechanics techniques will be applied to establish inspection strategies that will ensure that component failure probabilities are maintained at acceptable level. After candidate inspection strategies yielding component failure probabilities less than identified target values have been determined, decision analysis techniques can be used to optimize inspection strategies.

The Surry Nuclear Power Station, Unit 1 was selected for demonstrating the methodology. The specific systems selected for analysis were the reactor pressure vessel, the reactor coolant, the low pressure injection, and the auxiliary feedwater. The results provide a risk-based ranking of components within these systems, and provides a basis for relating the proposed target risk to acceptable failure probability values for individual components. Aging mechanisms (e.g., irradiation damage, embrittlement, erosion/corrosion, etc.) with the potential to impact on target rupture probabilities of components are also discussed.

Because of similarities in objectives, the PNL program is coordinated with an ASME Research Task Force on Risk-Based Inspection Guidelines. The initial task force document has made general recommendations on the application of risk-based methods to ISI, and forms the basis of future proposals to ASME for improved codes and standards. Results of PNL studies are being made available to the ASME group to demonstrate the usefulness of the risk-based methodology.

Improved In-service Inspection Program for Management of Degradation in Steam Generator Tubing

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This paper discusses extensive research supported by the U. S. Nuclear Regulatory Commission to develop information on the failure pressure of degraded steam generator tubing and the reliability and effectiveness of eddy current (ET) in-service inspection (ISI) techniques to detect and size degradation. In 1976, the NRC authorized the Pacific Northwest Laboratory to conduct the Steam Generator Tube Integrity Program to develop the needed information.

A main objective of the program was to establish validated models, based on experimental data, for predicting the failure pressure of service-degraded tubing under normal operating and accident loading conditions. More than 600 specimens of Inconel 600 tubing were mechanically flawed to simulate known or postulated defects present in steam generators. These specimens were burst and collapse tested at steam generator operating temperatures under controlled loading conditions. From the data, constitutive equations were developed relating failure pressure to flaw size and geometry. The constitutive equations were validated by testing tube specimens with artificial flaws created by chemical means and by testing service-degraded tubes taken from the retired-from-service Surry 2A steam generator.

Information on the reliability of ET inspection techniques to detect and size flaws in laboratory and service-degraded tubes was developed throughout all phases of the program. The most extensive and realistic data base was obtained from round robin examinations of the Surry 2A steam generator.

Four round robins were performed to determine the reliability of conventional multi-frequency ET inspections and alternative NDE methods. To validate the in situ NDE results, more than 550 tube segments were removed from the generator. Pitting and wastage were the predominant tube defects found in the specimens examined. Estimates of the probability of detection (POD) and sizing accuracy were obtained by matching the ET inspection results with data from both visual and destructive metallographic analysis of the removed specimens. Results indicated that the POD depended mainly on flaw severity. For pitting/wastage type flaws, the POD increased with wall loss and approached 0.9 for flaws greater than 40% through-wall.

Wide variations in the reported ET depth estimates were observed between specimens with similar wall-loss for an individual team and also within the same specimen for data from different inspection teams. The team-to-team variations for a given specimen appear to result from differences in analysis procedures or the analyst's interpretation of the complex ET signal patterns. Defect morphology and distribution within the corroded region was considered the major cause for the variations between specimens with similar wall-loss. However, dents and surface deposits near the defects also contributed to the sizing variations. In general, ET tended to undersize the pitting/wastage type defects, especially for the severely-degraded specimens. Improved sizing accuracy was noted for one team that employed special frequency mixes to enhance the signal-to-noise ratio by suppression of signals due to denting, copper deposits and support plates. Also, ultrasonic and rotating ET probes were successfully used to augment conventional ET/bobbin-coil data to obtain improved sizing accuracy.

The results of the ET reliability studies were used to develop models of POD and ET sizing error to provide a basis for evaluating and comparing various two-stage sampling plans for steam generator tube inspection. For this study the goal of ISI was to identify all defective tubes which could fail by leak or burst during reactor operation should a main steam line break occur. In this work a defective tube was defined as one with $\geq$ 75% through-wall degradation at the time of ISI. This definition was developed from burst test data which indicated that, on average, tubes with degradation about one inch in axial extent and 85% through-wall would fail under main steam line break loading conditions. A 10% flaw growth rate per operating cycle was assumed to arrive at the definition of a defective tube.

Analytical and Monte Carlo simulation studies were performed which indicated that a 40% systematic sequential sampling plan was almost as effective as 100% inspection, assuming some clustering of degraded tubes. This sampling strategy relies on two key concepts to achieve a high level of effectiveness. First, a relatively large, uniformly distributed initial sample is used to maximize the probability of finding isolated defective tubes, and second, detection of tube degradation of any level triggers second-stage inspection to aid in finding defective tubes which may be in close proximity. In order for this sampling strategy to be effective high inspection reliability for detection and sizing of degradation is needed, even when degradation is $<$ 75% through-wall. As a consequence, performance demonstration qualification criteria have been developed to provide high inspection reliability by establishing appropriate thresholds on POD and flaw sizing performance. The thresholds were selected because of the need for high reliability for identifying defective tubes and good reliability for finding degraded tubes. The POD curve and flaw sizing requirements were also based, in part, on the levels of performance observed in the ET reliability studies.

AGING MANAGEMENT OF MAJOR LWR COMPONENTS[1]

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The Aging Assessment and Mitigation Project has comprehensively evaluated degradation mechanisms affecting the structural integrity of the major light water reactor (LWR) components and has identified several options for managing their aging. The three primary degradation mechanisms acting on the components are embrittlement, fatigue, and corrosion (including stress corrosion cracking). This paper focuses on the management of stress corrosion cracking (SCC) mechanisms: primary water stress corrosion cracking (PWSCC) of pressure boundary components in PWRs, and intergranular stress corrosion cracking (IGSCC) and irradiation-assisted stress corrosion cracking (IASCC) in BWR vessels. Effective aging management of the SCC mechanisms includes evaluation of interactions between design, materials, stressors, and environment; identification and ranking of the susceptible sites; reliable inspection of damage; mitigation of damage, including modifications in water chemistry; and repair and replacement using corrosion-resistant materials.

Primary water stress corrosion cracking has caused both axial and circumferential through-wall cracks in Alloy 600 components constituting the PWR primary pressure boundary. Two major concerns about PWSCC failures are an unisolable pressure boundary leak and corrosion of any carbon or low-alloy steel base metal exposed to leaking borated coolant. PWSCC cracks have been found in tubes on both hot- and cold-leg sides and in tube plugs of recirculating steam generators. Recently, these cracks have also been reported in pressurizer instrument nozzles and heater sleeves and in control rod drive mechanism nozzles. Fabrication records of all Alloy 600 components in PWRs need to be reviewed to estimate residual stresses and to characterize microstructure so that the components can be ranked according to their susceptibility to PWSCC. Components with high residual stresses and no intergranular carbides have high susceptibility if operating temperatures are high, and such components need to be inspected for PWSCC damage. In addition, methods need to be developed for ultrasonic examination for boric acid corrosion of base metal around the failed Alloy 600 nozzles. Alloy 690, a PWSCC resistant material, may be used for replacement of failed Alloy 600 components.

Intergranular stress corrosion cracking has caused through-wall cracking in BWR pressure vessel nozzle welds and is a potential degradation mechanism for vessel interior attachment welds. Two major concerns are a crack initiating in the weld propagating into the vessel base metal and a leak through the primary pressure boundary. Welds most susceptible to IGSCC are those having Alloy 182 weld material; having high residual, applied, and thermal tensile stresses; and having

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[1] Work sponsored by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under DOE Contract No. DE-AC07-76D01570; Dr. G. H. Weidenhamer, Technical Monitor.

a high electrochemical potential. In addition, the IGSCC susceptibility of welds increases as the BWR coolant conductivity increases. The estimated residual tensile stresses at the attachment welds not subject to postweld heat treatment are high (280 to 480 MPa), whereas the stresses at the welds subject to postweld heat treatment are low (140 to 280 MPa). The applied stresses are generally low except at the welds attaching the jet pump riser brace to the vessel. Thermal stresses are due to differential thermal expansion of the weld and base metal. The electrochemical potential of the attachment welds in the upper region of the vessel is high because of the high oxidizing power of the BWR coolant leaving the core, where it is subject to radiolysis. Specialized equipment and techniques are being developed for remote automated inspection of penetration welds, for example, inspection of the incore monitor housing-to-lower head welds. An SCC monitor may be employed to estimate any IGSCC crack initiation or growth in the nozzle and the attachment welds.

Recent SCC test results indicate that the hydrogen water chemistry, which has been found effective in suppressing IGSCC in recirculation piping, has a potential to protect the attachment and nozzle weld materials. However, it does not provide the same level of protection to all weld locations on the vessel, because the electrochemical potential varies. Susceptible welds in the upper region of the vessel, which have a higher electrochemical potential, will require a greater amount of hydrogen injection than the ones in the downcomer region or in the recirculation piping. The greater amount of hydrogen injection has several adverse effects that should be addressed, such as an increase in the steam line radiation fields associated with an increased partitioning of N-16 in the steam phase. Damaged welds may be repaired by replacing Alloy 182 material with the corrosion resistant Alloy 82 material, by applying corrosion resistant cladding to protect susceptible materials from exposure to BWR coolant, and by using clad overlay as a short-term solution. Underwater wet welding and cutting techniques are being developed for the vessel repair.

Irradiation-assisted stress corrosion cracking can occur in the highly irradiated BWR reactor internal components fabricated from stainless steel and nickel base alloys. Several failures of the stainless steel components, such as an incore guide tube and a neutron monitor dry tube, are attributed to IASCC. Based on field experience and laboratory tests, IASCC does not occur below a certain threshold level of fast fluence; the threshold depends on material, stress levels, geometries, and environment. The threshold for Type 304 stainless steel components with high stresses is 5×10^{20} n/cm^2; with low stresses the threshold is 2×10^{21} n/cm^2. A relatively high level of dissolved oxygen and a crevice geometry can lower the threshold and accelerate any existing IASCC damage. Development of specialized equipment and techniques for inspection, use of hydrogen water chemistry, and the underwater weld repair discussed above for the BWR vessel welds are also applicable here to manage IASCC damage. Modified heat treatment and IASCC-resistant materials are being developed for replacement of damaged reactor internal components. For example, preirradiation solution annealing treatments in the 1200 to 1300°C temperature range can make Type 304 stainless steel more resistant to IASCC. High-purity Type 348 stainless steel is also found to be IASCC-resistant.

LESSONS LEARNED FROM FATIGUE FAILURES IN MAJOR LWR COMPONENTS[1]

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Fatigue is one of the leading degradation mechanisms affecting major light water reactor (LWR) components. Fatigue has caused surface cracks and, in some cases, through-wall cracks and coolant leakage. This paper evaluates the fatigue failure experience in the field and discusses the lessons learned that can be employed in managing fatigue damage. The lessons relate to stressors causing the fatigue damage; sites susceptible to the damage; and inspection, monitoring, and mitigation of the damage.

Investigation of field fatigue experience has identified stressors and degradation mechanisms that contributed to failures but were not accounted for in the designs. Examples of such stressors are thermal stratification and striping acting on PWR surge lines, high-pressure safety injection lines, and feedwater nozzles. Design calculations for the fatigue usage factors for several affected components are being revised to account for these stressors. Examples of the mechanisms, which were not accounted for in the design, are environmentally assisted fatigue and high-cycle thermal fatigue. Environmentally assisted fatigue has caused cracking in PWR steam generator tubes, girth welds, and feedwater nozzles. The need for revising the ASME fatigue design curves to include environmental effects should be evaluated. High-cycle thermal fatigue has caused crack initiation at the inside radius of the BWR feedwater nozzles and in the PWR high-pressure injection lines.

The field fatigue failures have revealed several sites susceptible to fatigue damage that were not originally considered vulnerable to fatigue. Examples of such sites include welds and elbow base metal in PWR surge lines, safety injection lines, and residual heat removal piping, and include PWR steam generator girth welds. Base metal sites in elbows are susceptible to fatigue damage because of the ovalization of the elbow cross section caused by inplane bending; the most susceptible site is the inside surface of the flank of an elbow. Fatigue test results for pipe bends and finite element analyses for surge lines subject to thermal stratification have identified elbow base metal as a susceptible site for fatigue damage.

Current inservice inspection requirements concentrate on the inspection of welds; however, as discussed above, several field failures have occurred in the base metal (away from welds) not included in the inservice inspection program. ASME Code Section XI requirements might be expanded to include sites where high

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[1] Work sponsored by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under DOE Contract No. DE-AC07-76ID01570; Dr. G. H. Weidenhamer, Technical Monitor.

fatigue usage is predicted (including base metal sites) or where there has been a history of failures. Such sites would include surge line elbows and feedwater nozzles. Inservice inspection requirements in some other countries include examination of susceptible base metal sites. For example, the inservice inspection requirements for the newest generation of nuclear power plants in Germany include inspection of the elbow base metal.

Inservice inspection experience indicates that the detection of thermal fatigue cracks is difficult because these cracks generally are tightly closed during the inspection period when pressure and thermal stresses are reduced. The conventional inservice inspection techniques and procedures satisfying the ASME Code minimum requirements are inadequate to detect the presence of such tight cracks. Inspections with conventional ultrasonic methods were unable to detect a through-wall crack on the Farley safety injection line. Reliable inservice inspection for the fatigue cracks may be performed during hydrotests when the cracks are opened sufficiently to be detected. Advanced methods, such as the time-of-flight diffraction technique, need to be developed and field tested to inspect sites susceptible to fatigue cracking. Acoustic emission monitoring for fatigue crack growth needs to be tested in the field.

Uncertainty in the amplitude of the stressors and limited access for inspection at certain susceptible sites have resulted in a need for the development of on-line fatigue monitoring techniques. Utilities are applying these techniques to more accurately determine the severity and numbers of cycles for the transients that contribute to fatigue damage at susceptible locations. Further development of these techniques will be of potential benefit in better estimating the fatigue usage at critical sites, in supplying useful information in making inspection and repair/replacement decisions, and in modifying operating procedures to mitigate fatigue damage. The present generation of nuclear plants was not instrumented to acquire all the information needed for fatigue monitoring; thus, indirect measurements from the existing plant instrumentation must be used. Some newly constructed nuclear power plants in Germany, France, and Japan are being instrumented with additional local sensors for the express purpose of fatigue monitoring. This should be considered in the design and construction of the advanced LWRs.

Understanding the stressors that cause fatigue damage can lead to the development of mitigation techniques, which include modified designs and changes in operating procedures. For example, the cladding has been removed from the inside radius of BWR feedwater nozzles to eliminate stresses caused by differential thermal expansion between the cladding and the base metal. Some plants now have preheating tanks to raise the temperature of auxiliary feedwater closer to that of steam generator coolant, thus mitigating the thermal shock loads. Procedural requirements also can be implemented, such as limiting the differences in pressurizer and reactor coolant system temperatures to no more than 200°F during heatups and cooldowns to reduce the effects of thermal stratification in surge lines.

A COMPREHENSIVE APPROACH TO MANAGE AGING IN NUCLEAR SERVICE WATER SYSTEMS

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Pacific Northwest Laboratory,[a] in support of the Nuclear Plant Aging Research (NPAR) Program, has conducted aging assessments of service water systems (SWSs) in nuclear plants. The objectives of the assessments are to provide guidance to manage and mitigate aging of SWSs and to address system-specific options for water chemistry control, cleaning, and surveillance. The scope of the study encompasses the following elements:

- review of databases for SWS failure histories
- interfaces with experts in SWS chemistries and corrosion
- visits by PNL teams to three nuclear power plants for comprehensive assessments of SWS age-related degradation
- participation in NRC case studies of SWSs at two nuclear power stations
- reviews of relevant literature
- publications that summarize results of the NPAR SWS aging assessment.

SWS designs at nuclear plants operating in the United States include three general types:

- open type
- recirculating type
- closed type.

These designs differ in functions and capabilities for water chemistry control, cleaning methods, and, in some details, surveillance. The following characteristics regarding SWSs should be considered when developing a comprehensive approach to manage the effects of age-related degradation:

- Major portions of the systems are critical to safety, including service to key safety-related equipment.
- The systems are relatively large and have numerous components (e.g., ~40 or more heat exchangers, four to ten major pumps, and over 200 valves).

(a) Pacific Northwest Laboratory is operated for the U.S. Department of Energy under Contract DE-AC06 76RLO 1830. Work is conducted under NRC FIN B2911.

- The systems are subject to a variety of aging mechanisms that impose challenges to determine causes of degradation and to select appropriate counter measures.

- A given system is subject to a variety of service conditions; for example, some surfaces are wetted, some are exposed only to air, and some are exposed to soils; a given component may operate stagnant, under flow, or intermittently; wetted surfaces may be subject to abrasive particulates, biota, and detritus.

- Many of the corrosion-prone surface areas are not readily accessible for visual inspection.

- The vulnerabilities to degradation are system-specific and component-specific, depending on the materials, designs, water compositions, etc., of each system.

- The degradation rates and mechanisms have seasonal variations.

- Unexpected phenomena, such as encroachment of previously undetected biological species, can occur.

Effective SWS management should begin with the premise that active strategies involving timely monitoring and control are preferable to reactive measures associated with recovery from highly degraded states. The key elements of effective SWS management are as follows:

A. sound design

B. appropriate materials selection

C. system-specific water chemistry control

D. systematic monitoring, inspection, and surveillance

E. system-specific cleaning on a planned schedule

F. systematic and timely maintenance (Note: D and E could be regarded as elements of maintenance.)

G. Attention to operating history, including trending.

Decisions concerning corrective actions should be based on analysis of the root cause(s) of observed degradation, proposed resolutions, and the effects of the proposed resolutions on the total system. The need for corrective actions may arise from errors in design or materials selection, from changes in operating conditions after the design was completed, or from degradation anticipated during operation.

ASSESSMENT OF DIAGNOSTIC METHODS FOR DETERMINING DEGRADATION OF MOTOR-OPERATED VALVES*

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SUMMARY

Motor-operated valves (MOVs) are located throughout nuclear power plant fluid systems. Their failures have resulted in significant maintenance efforts and, on occasion, have led to the loss of operational readiness of safety-related systems. The Oak Ridge National Laboratory (ORNL) has carried out a comprehensive aging assessment of MOVs in support of the Nuclear Plant Aging Research (NPAR) program. This paper provides a summary of the ORNL MOV aging assessment with emphasis on the identification, evaluation, and application of MOV monitoring methods and techniques.

Beginning in 1985, the diagnostic information available from many MOV measurable parameters was evaluated by ORNL using MOVs that were mounted on test stands. Those parameters included valve stem position, valve stem velocity, valve stem strain, torque and limit switch actuation times, internal and external motor temperatures, vibration, torque switch angular position, and motor current. Those tests led to the conclusion that the single most informative MOV measurable parameter was also the one which was most easily acquired, namely the motor current. Motor current signature analysis (MCSA) was found to provide detailed information related to the condition of the motor, motor operator, and valve across a wide range of levels from mean values and gross variations during a valve operation to information which characterizes transients and periodic occurrences.

A detailed discussion of the application of MCSA to MOVs is provided, including examples of time waveform analysis, frequency spectral analysis, and additional techniques found useful in determining MOV performance and condition.

As part of the MOV aging assessment, several tests were carried out by ORNL on MOVs having implanted defects and degradations. Tests were also carried out on many MOVs located within a nuclear power plant. In addition, ORNL participated in the Gate Valve Flow Interruption Blowdown Test program carried out at Wyle Laboratories in Huntsville, Alabama. Results from all of these tests are summarized in this paper and several selected examples are given.

Other areas covered in the paper include descriptions of relevant regulatory issues and activities, other related diagnostics research at ORNL (including the application of MCSA to other equipment), and interactions ORNL has had with outside organizations for the purpose of disseminating research results.

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LIFE TESTING OF A LOW VOLTAGE AIR CIRCUIT BREAKER*

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SUMMARY

A DS-416 low voltage air circuit breaker manufactured by Westinghouse was mechanically cycled to identify age-related degradation in the various breaker subcomponents, specifically the power-operated mechanism. This accelerated aging test was performed on one breaker unit for over 36,000 cycles. Three separate pole shafts, one with a 60-degree weld, one with a 120-degree weld, and one with a 180-degree weld in the third pole lever were used to characterize cracking in the welds. In addition, during the testing three different operating mechanisms and several other parts were replaced as they degraded to an inoperable condition.

Pole shafts used in this test program were found to have substandard welds. Examination of a fractured trip shaft lever suggested inadequacies in electroplating techniques. Newly purchased reset springs had sharper bends at the neck of the hooks than an older design, which led to early spring failures. Testing of the hardness of the oscillator surface showed a 30% reduction for the newly procured units.

Wear, fracture, distortion, and normal fatigue dominated the aging process with wear being the largest contributor. Excessive wear was evident in the ratchet wheel, holding pawls, oscillator, drive plate, motor crank and handle, cam segments, main roller, and the stop roller. Structural components and contact assembly parts indicated that there was little aging due to mechanical cycling. A pole shaft with a reduced size weld could fail at a cycle as low as 3000.

The ultimate life of various breaker parts was found to be 10,000 cycles, except for the newly procured reset spring, which was 2000 cycles. One commercial grade lubricant was found to perform better than those recommended. The current plant maintenance practices need to incorporate the experience of aging problems associated with the power-operated mechanisms. The scheduling of the breaker maintenance should be dependent primarily on the number of cycles experienced by the breaker, with some consideration given to time in service.

The maintenance and manufacturing recommendations, obtained from the test results, should help mitigate aging problems. When procuring a new breaker or spare parts, careful attention should be given to their design adequacy. For the indicator and reset springs, it is recommended that a smooth transition bend should be made instead of a sharp bend to minimize the damage to the surface, and

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

to reduce the tensile stresses on the inside surface of the bend area of the end hooks. For the pole pin connecting Phase A of the breaker contacts, smooth corners should be machined, free from surface defects. Improper welding practices at the pole shaft levers should be avoided. Assuming a factor of safety of 2, the life of a DS-416 (or DS-206) breaker is estimated to be 5000 cycles. Based on an assumption that a breaker, such as reactor trip, is typically subjected to 250 cycles annually; this translates to a breaker life of 20 years.

Aging and Low-Flow Degradation of Auxiliary Feedwater Pumps*

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SUMMARY

Auxiliary feedwater (AFW) pumps are used in safety-related AFW Systems at pressurized water reactor plants. The function of AFW pumps is to deliver water from either a condensate storage tank or, as a backup, from the emergency service water system, to the steam generators. The water that is pumped to the steam generators is evaporated, thereby removing decay heat from the reactor coolant system.

The pumps are automatically started in response to several emergency conditions, such as low steam generator level, a safety injection signal, and emergency bus undervoltage. However, many plants also use the pumps in support of normal shutdown and startup sequences, since the main feedwater system pump capacity greatly exceeds demand during these conditions. The other principal service seen by the pumps is during testing.

The AFW pumps are multistage (normally 5 to 9 stages) high-head centrifugal pumps, normally driven by motors or turbines. Rotating speeds for the motor-driven pumps are nominally 3550 rpm, while those for turbine-driven pumps are routinely closer to 4000 rpm. Rated pump deliveries range from roughly 200 to 1200 gpm.

Much of the operation of AFW pumps is at low-flow conditions. Once the reactor has been shut down, the reactor decay heat generation rate and the attendant AFW pump flow requirement drop exponentially. For example, if full pump capacity is required to remove all decay heat at ten minutes after a reactor shutdown, just over a third of the pump capacity would be required for heat removal after five hours. Of course the heat removal requirements during reactor startup following an outage are insignificant relative to typical pump capacity. As a result, much of the pump operation in support of these startup and shutdown evolutions is essentially at minimum flow.

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Many of the AFW pumps are provided with relatively low minimum flow line capacities, typically in the range of 5 to 10% of best efficiency point flow. The pumps are normally tested under minimum flow conditions. Testing is conducted, per plant Technical Specifications, on a monthly or quarterly basis. In addition, automatic start testing is normally required to be conducted every eighteen months.

The minimum flow provisions for many of these pumps are not adequate to support continuous operation. The installed miniflow lines were, in most cases, intended by the pump manufacturers to be used just during pump starting and stopping. However, the required testing as well as the necessity of using these pumps for plant startup and shutdown support has resulted in a considerable amount of operation at low-flow conditions. Operation at low flow results in accelerated wear of the pumps due to vibration associated with the hydraulically unstable conditions. The wear can be manifested in a number of ways, such as impeller or diffuser breakage, thrust bearing and/or balance device failure due to excessive loading, cavitation damage on suction stage impellers, increased seal leakage or failure, seal injection piping failure, shaft or coupling breakage, and rotating element seizure.

This paper discusses pump design, historical operating experience, and testing and inspection methods. Examples of operating problems that have been experienced and potential design modifications are provided.

Aging Evaluation of Nuclear Plant RTDs and Pressure Transmitters

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ABSTRACT

Resistance Temperature detectors (RTDs) and pressure, level, and flow transmitters provide almost all the vital signals that are used for the control and safety of nuclear power plants. Therefore, it is crucial to ensure that the performance of these sensor remain adequate as they age in the process under normal operating conditions.

Two comprehensive research projects were conducted for the NRC to evaluate the effects of normal aging on calibration stability and response time of RTDs and pressure transmitters of the types used for safety-related measurements in nuclear power plants. Each project was conducted over a three year period. The projects involved laboratory testing of representative RTDs and pressure transmitters aged in simulated reactor conditions. The main purpose of these projects was to establish the degradation rate of the sensors and use the information to determine if the current testing intervals practiced by the nuclear power industry is adequate for management of aging of the sensors. The results have indicated that the current nuclear industry practice of testing the response time and calibration of the sensors once every fuel cycle is adequate. This is provided that all the safety-related sensors are tested for both calibration and response time as opposed to testing one out of every two or four of the redundant sensors.

In addition to working to identify degradation rates and testing intervals of the sensors, a few outstanding issues regarding the performance of nuclear plant RTDs and pressure transmitters were addressed. These included a comprehensive assessment of the acceptability of the cross calibration method for on-line testing of calibration of installed RTDs, the oil loss phenomenon in Rosemount pressure transmitters, and the effects of sensing line blockages on the overall response time of nuclear plant pressure sensing systems.

For both RTDs and pressure transmitters, the LER and NPRDS databases were searched and analyzed to augment the experimental data we generated by laboratory tests.

POWER REACTOR EMBRITTLEMENT DATA BASE (PR-EDB):
USES IN EVALUATING RADIATION EMBRITTLEMENT OF REACTOR VESSELS*

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Investigations of regulatory issues such as vessel integrity over plant life, vessel failure, and sufficiency of current Codes, Standard Review Plans (SRPs), and Guides for license renewal can be greatly expedited by the use of a well-designed, computerized data base. Also, such a data base is essential for the validation of embrittlement prediction models by researchers. The PR-EDB is such a comprehensive collection of data for U.S. power reactors. The compilation of the Test Reactor Embrittlement Data Base (TR-EDB) is in progress and will supplement the data in the PR-EDB. More analyses of test and power reactor results are needed to establish the applicability of test reactor data to power reactors. "User-friendly" software programs to access, process, and manipulate the data have been written and further developments are expected.

A list of studies using the PR-EDB follows:

1. Analysis of the correlation monitor materials (A302B and A533B) in surveillance capsules of commercial power reactors (NUREG/CR-4947);

2. Differences between the measured transition temperature shifts (ΔRT_{NDT}) and the values predicted by Regulatory Guide 1.99 (Rev. 2) for base and weld materials;

3. Comparison of ΔRT_{NDT} and the reduction in the Charpy-V upper-shelf energy (ΔUSE) for specimens fabricated in the longitudinal orientation (LT) with those in the transverse orientation (TL);

4. Comparison of USE from LT specimens reduced to 65% of their value with the USE values from TL specimens;

5. Comparison to see if Regulatory Guide 1.99 (Rev. 2) underpredicts the Charpy shift for low-copper material when the phosphorus content is high;

6. Comparison of the two standard deviation value (2σ) for the residual [observed value of ΔRT_{NDT} minus value calculated by Regulatory Guide 1.99 (Rev. 2)] from Guthrie's data base and the PR-EDB;

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7. Comparison of the Regulatory Guide 1.99 (Rev. 2) model with the French FIM model in predicting ΔRT_{NDT} for base materials and welds;

8. Comparison of the TANH fitting program in PR-EDB with one from the Electric Power Research Institute (EPRI);

9. Analysis of instrumented Charpy data based on model by D. Pachur;

10. Study of plate materials similar to the Yankee Rowe Reactor vessel and to materials in their test reactor experimental program; and

11. Generation of a data file that has similar radiation and annealing environments as the Yankee Rowe Reactor to test model prediction.

Aging Impact on the Safety and Operability of Nuclear Reactor Pressure Vessels*

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Structural integrity of the reactor pressure vessel must be assured throughout the operating life of a nuclear power plant in order to maintain the capability to cool the nuclear core. This assurance is achieved by requiring that the reactor vessel maintain specified fracture-prevention margins throughout its operating life. Fracture-prevention margins are calculated using fr. re mechanics technology in conjunction with a material property known as fracture toughness.

Fracture toughness is a measure of the ability of a material containing sharp edged cracks to sustain stress. Fracture-mechanics based structural integrity assessments differ therefore from those obtained from the more familiar stress-strength analysis in that they account in a quantitative manner for the effect of cracks, which are unavoidably present in all engineering structures.

Irradiation damage is the aging mechanism of dominant concern in pressurized-water-reactor pressure vessels. Irradiation exposure causes atom displacements in the vessel material lattice structure which have the effect of increasing its strength but decreasing its ductility and fracture toughness. It follows therefore that fracture-prevention margins will progressively decrease as the vessel material absorbs increasing amounts of irradiation damage throughout its operating life.

Regulatory requirements limit the permissible accumulation of irradiation damage in the material of a given reactor vessel. Irradiation damage limits are set such that required fracture-prevention margins are maintained throughout the nuclear plant licensed operating period. The regulatory requirements are based on fracture mechanics technology and utilize materials aging data drawn from mandatory reactor vessel irradiation damage surveillance programs. They address both normal operation of the reactor system and potential accident loading. For normal operation, the regulatory requirements dictate that plant technical specifications be periodically adjusted to preclude operating conditions which could reduce the fracture prevention margins. For accident loading, they set regulatory limits on the acceptable level of irradiation damage in the vessel material. They also define the scope and acceptance criteria for fracture-margin assessments which must be performed to support any proposal for continued operation of the plant once the regulatory irradiation damage limits have been exceeded.

In recent years it has become evident that a number of nuclear plants will exceed the regulatory limits on irradiation damage to the reactor vessel material before the end of their current licensing period. One result of this development is that a number of nuclear industry organizations have gained experience in the application of fracture-margin-assessment technology. This experience has resulted in the identification of a number of issues with the technology in its present form. Data from irradiation testing programs, operating plant surveillance programs and large scale fracture technology validation tests have identified additional issues. The NRC-funded Heavy-Section Steel Technology program at Oak Ridge National Laboratory is performing the researc' required to resolve these issues and further develop and refine the fracture-margin assessment technology.

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This paper will present a brief overview of the current status of fracture prevention regulatory requirements and the associated fracture-margin assessment technology. Issues identified with the technology will be reviewed. Research programs implemented to resolve the issues will be described. Potential impacts of this ongoing research on the fracture-margin assessment process will be discussed.

The Application of Probabilistic Fracture Analysis to Residual Life Evaluation of Embrittled Reactor Vessels*

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Irradiation damage is the aging mechanism of dominant concern for pressurized-water reactor (PWR) pressure vessels. Cumulative irradiation exposure has the effect of making the vessel material more brittle (decreased fracture toughness) and therefore more susceptible to cleavage fracture. For such embrittled vessels, pressurized-thermal shock (PTS) is the major challenge confronting vessel integrity. Concern with PTS results from the combined effects of (1) pressure and thermal-shock loadings, (2) embrittlement of the vessel material due to irradiation, and (3) the possible existence of sharp, crack-like defects at or near the inner surface of the vessel.

The PTS issue has been under investigation for many years. Most of the early PTS analyses were of a conservative deterministic nature. In an effort to establish more realistic limiting values of vessel embrittlement, the NRC funded the Integrated Pressurized Thermal Shock (IPTS) Program, which developed a comprehensive probabilistic approach to the PTS vessel integrity issue. A major element of the probabilistic approach is performing probabilistic fracture mechanics (PFM) analyses to predict the probability of vessel failure assuming a specified PTS event occurs at a specified time in the operating life of the plant. This paper describes the application of PFM to predict the residual life of embrittled vessels in accordance with regulatory requirements.

Current regulatory requirements for vessel integrity are based on the probabilistic methodology and Regulatory Guide 1.154 which provides guidance to utilities on how to perform PTS probabilistic vessel integrity evaluations. Regulatory Guide 1.154 references OCA-P and VISA-II as acceptable PFM computer codes for performing the probabilistic fracture mechanics portion of the evaluations. These codes predict the increase in vessel failure probability that occur as the vessel material accumulates irradiation damage as a function of time. Such results, when compared with limits of acceptable failure probabilities, provide an estimation of the residual life of a specific vessel. These codes can be used to evaluate the benefits of plant-specific mitigating actions which have the potential to extend the residual life of a vessel.

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Managing Irradiation Embrittlement in Aging Reactor Pressure Vessels*

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Maintaining integrity of the reactor pressure vessel (RPV) in light-water-cooled nuclear power plants is crucial in preventing severe accidents and their potential for major contamination releases. This requires a quantitative understanding of irradiation-induced degradation of the RPV's fracture resistance, a methodology for assessing the impact of reduced toughness on integrity, and a means of deciding under what conditions the vessel can continue to operate safely. For reactor vessels recently fabricated from radiation-damage-resistant steels or with very limited service, there is little concern over challenges to their integrity since without significant radiation damage, it is virtually impossible to postulate a realistic scenario resulting in RPV fracture. However, for aging reactors containing pressure vessels fabricated before the mid-1970s, when metallurgical controls were instituted to control embrittlement in vessels, the irradiation-induced loss-of-fracture resistance could unacceptably compromise vessel integrity under severe loading conditions. Unfortunately, with the United States having pioneered civilian nuclear power, many of our older reactors fall into this later category. It is therefore imperative to understand and predict the effects of irradiation on RPV steels to ensure the continued safe operation of these aging reactors. For this reason, the Heavy-Section Steel Irradiation (HSSI) Program has been established by the U.S. Nuclear Regulatory Commission (USNRC) to provide a thorough, quantitative assessment of the effects of neutron irradiation on the material behavior and, in particular, the fracture toughness properties of typical pressure vessel steels as they relate to light-water reactor pressure-vessel integrity. Results from the HSSI studies provide information needed to aid in resolving major regulatory issues facing the USNRC, which involve RPV irradiation embrittlement, such as pressurized-thermal shock, operating pressure-temperature limits, low-temperature overpressurization, and the specialized problems associated with low upper-shelf (LUS) welds. The results of these studies, when used in conjunction with results from other USNRC-sponsored programs that provide comparable information on the detection and evaluation of flaws and overall

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fracture assessment methodologies, provide guidance and bases for both managing irradiation-induced embrittlement in aging vessels and evaluating the potential for their extended-life operation.

Major program elements within the HSSI Program include: experimental investigation and verification of the irradiation-induced loss of fracture resistance in critical reactor vessel materials, with an emphasis on applicability to the thick sections used in RPV construction; experimental verification and expansion of existing data on annealing recovery and subsequent reembrittlement in critical pressure vessel materials to provide a validated basis for remedial actions for severely embrittled vessels; coordinated, advanced microstructural examinations and physically-based theoretical model development of controlling microstructural mechanisms to provide improved predictions of macroscopic embrittlement; verification of predictions of radiation-induced damage by examination of materials exposed during actual service; and maintaining the supply of correlation monitor material used for validating results of world-wide irradiation surveillance programs. The principal materials examined within the HSSI Program are high-copper welds since their postirradiation properties are most frequently limiting in the continued safe operation of commercial RPVs. The irradiation-induced shifts and changes in shape in fracture toughness were examined for two high-copper welds. The irradiation-shifted fracture toughness fell slightly below the ASME K_{Ic} curve even when it was shifted according to Revision 2 of Regulatory Guide 1.99, including its margins, indicating that the current method for assessing fracture toughness reductions may be nonconservative. The high-copper beltline weld removed from the Midland reactor is being examined to establish the effects of irradiation on a commercial LUS weld. A wide variation in the unirradiated fracture properties of the Midland weld, with values of RT_{NDT} ranging from -23 to 20°C and copper contents from 0.16 to 0.46 wt %, were determined. Experiments have been initiated to examine the irradiation embrittlement of the LUS weld as well as provide initial data in the annealed and reirradiated conditions. Annealing experiments on critical high-copper materials have been initiated to provide the first quantitative comparison of Charpy-V notch versus fracture toughness recovery and reembrittlement. Embrittlement modelling studies have shown that the dose required for point-defect concentrations, which contribute to irradiation embrittlement, to reach steady-state values can be comparable to component or irradiation-experiment lifetime. Thus, embrittlement models that rely on the assumption of steady-state at these conditions are not valid. Consequently, simple direct comparisons of embrittlement generated under different rates of exposure (e.g., test reactors verses power reactors) may be misleading because the relative state within the initial defect production transient will likely be different. Since the effect of displacement rate is different in the transient and steady-state regimes, unqualified data extrapolation from test-reactor irradiations may cross mechanism boundaries and lead to poor estimates of material response at low displacement rates.

DETECTING AND MITIGATING AGING IN COMPONENT COOLING WATER SYSTEMS*

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SUMMARY

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

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

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

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

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

To determine the most effective methods of managing aging, a second phase of the CCW system aging study was performed[2]. In this part of the study inspection, surveillance, monitoring, and maintenance (ISM&M) practices were investigated. Information on ISM&M practices currently used at plants was obtained from a survey, along with actual plant visits and personnel interviews. In addition, various advanced practices were identified through literature searches and discussions with component manufacturers. The findings provided an excellent overview of what methods are available to properly control aging degradation.

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

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SNUBBER AGING ASSESSMENT

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The Pacific Northwest Laboratory (PNL),[a] in support of the Nuclear Plant Aging Research (NPAR) Program, has conducted aging assessments of snubbers used in commercial nuclear power plants. The objectives of the assessments were to characterize snubber aging and to provide a basis for managing and mitigating its effects. Phase I of the study was completed and the results are published in NUREG/CR-4279. Phase II was also completed by PNL and its subcontractors, Lake Engineering Company (LEC) and Wyle Laboratories. During Phase II, we conducted a feasibility study (NUREG/CR-5386) and identified specific aging research needs. The remainder of Phase II involved in-plant research with several nuclear plant utilities, where we interviewed plant maintenance and engineering staff, and analyzed plant operating data.

Thirteen plants at eight different sites were visited during a three-month period. Snubbers at five of the sites were primarily mechanical; snubbers at the remaining three sites were primarily hydraulic. Laboratory research was also conducted at the LEC facility, and general information pertaining to aging was obtained from LEC's files. For this study, we defined snubber aging as "showing the effects of time and use on the physical characteristics of a snubber." By distinguishing between snubber failures related to aging and failures related to nonaging causes, we concluded that approximately half of all snubber failures were attributable to aging influences that involve the operating environment (temperature, humidity, etc.), dynamic transients, and vibration. Nonaging-related failures are associated with snubber design or manufacturing inadequacies, improper assembly, or damage incurred during plant construction or during snubber installation.

Heat, vibration, and moisture can degrade the performance of mechanical snubbers by increasing drag and breakaway forces, and by changing activation acceleration thresholds. Data from mechanical snubbers in one plant indicated a slightly increasing trend in drag force with service time. For hydraulic snubbers, high temperatures in isolated operating areas can rapidly degrade seals. At one boiling-water plant, the incidence of seal leakage in hydraulic snubbers was higher in the drywell (higher temperature) than in other areas of the plant. Radiation probably contributes less significantly to seal degra-

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(a) Pacific Northwest Laboratory is operated for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830. Work was conducted under NRC FIN B2911.

dation than was originally hypothesized. Aging degradation mechanisms for elastomeric seals include extrusion, embrittlement, permanent set, wear, and adhesion to mating surfaces. However, most seal leaks are not directly attributable to long-term service degradation. Fluid leakage in hydraulic snubbers is commonly attributable to leaking fittings, improper assembly, or design inadequacies. Several plants have implemented seal life evaluation studies.

Many plants routinely evaluate the causes of snubber failures, often identifying specific designs or applications that are prone to degradation; augmented surveillance for such applications is common. However, not all plants have implemented formal service-life monitoring programs. For those that have, service-life monitoring generally involves only ensuring that seals in hydraulic snubbers are replaced at prescribed intervals.

Based on results of the Phase II investigation, we developed several service-life monitoring recommendations for mechanical and hydraulic snubbers:

- Information about the snubber operating environment should be identified and can be obtained from the snubbers themselves by visual examination (in-situ and during disassembly), by evaluation of functional test data, by fluid sampling, and by conducting "hands-on" checks.
- In-situ monitoring is useful for verifying specific operating conditions; various equipment is available for this purpose.
- Root causes of failures should be determined. Determining the root cause of degradation in snubbers removed from service is also advisable. Data on degradation caused by non-service-related influences should be interpreted separately from data used to verify service life.
- Snubbers subjected to severe environmental influences should be identified and managed on a case-by-case basis.
- Service life for the general snubber population (snubbers that do not require augmented maintenance) should be established by trending relevant degradation parameters such as seal compression set for hydraulic snubbers, or drag force for mechanical snubbers.
- Augmented evaluation methods, such as hand-stroking, are useful to identify some forms of snubber degradation, such as degradation caused by dynamic load transients.
- Service-life projections based on data from snubbers exposed to actual plant operating environments are preferable to analytical service-life projections.
- Scheduled maintenance should be based on realistic expectations pertaining to service-related degradation. Unnecessary maintenance can increase the potential for snubber failure and result in unnecessary personnel radiation exposure.

Managing the Aging of BWR Control Rod Drive Systems*

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SUMMARY

This Phase I Nuclear Plant Aging Research (NPAR) study examines the aging phenomena associated with BWR control rod drive mechanisms (CRDMs) and assesses the merits of various methods of "managing" this aging. Information for this study was acquired from (1) the results of a special CRDM aging questionnaire distributed to each U.S. BWR utility, (2) a first-of-its-kind workshop held to discuss CRDM aging and maintenance concerns, (3) an analysis of the Nuclear Plant Reliability Data System (NPRDS) failure cases attributed to the control rod drive (CRD) system, and (4) personal information exchange with nuclear industry CRDM maintenance experts. The report documenting the findings of this research, NUREG-5699, will be published this year.

Utilities evaluate the operability of their CRD systems by performing individual CRDM scram time testing (as required per plant technical specifications) and weekly to monthly step insertion and withdrawal tests. When a CRDM fails to meet test timing specifications or begins to show symptoms such as double-notching (erroneously moves two steps instead of one), frequently becomes uncoupled from the control rod blade, exhibits high operational temperatures, or requires excess drive pressure to move, it is usually selected for changeout during a plant refueling or maintenance outage. During an outage, utilities typically replace nearly 16% (on the average) of a unit's CRDMs with new or rebuilt units.

Nearly 23% of the NPRDS CRD system component failure reports were attributed to the CRDM. The CRDM components most often requiring replacement due to aging are the Graphitar seals. The predominant causes of aging for these seals are mechanical wear and thermal embrittlement. Premature aging of these seals is also caused by excessive amounts of crud (dirt particles, debris, and foreign materials found in the reactor coolant). This material becomes entrapped between the seals in the CRDM and creates uneven force distributions at the seal's contact surfaces during scrams, causing them to break. Some utilities are vacuuming their reactor vessels in and around the guide tubes during refueling outages to remove and reduce the amounts of crud.

More than 59% of the NPRDS CRD system failure reports were attributed to components that comprise the hydraulic control unit (HCU). Each CRDM has a companion HCU that contains numerous valves which regulate the flow of coolant that controls the movement of the respective CRDM. The predominant HCU valve components experiencing the effects of service wear and aging are the packing, seals, discs, seats, stems, and diaphragms. The HCU valves reporting the most maintenance activity (due to aging) in the NPRDS are the accumulator nitrogen charging cartridge valve, the scram discharge riser isolation valve, the inlet and outlet scram valves, and the scram pilot valve assemblies and their solenoids.

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The scram water accumulator on pre-BWR-6 HCUs has a carbon steel tank liner which has experienced a large amount of corrosion caused by low pH water conditions at various plants. Several utility specialists believe that corrosion flakes from these affected accumulators have also caused the Teflon seats in the scram valves to erode and plug the companion CRDM's cooling water orifice. To avoid this degradation, many utilities are replacing these carbon steel accumulators with a improved component design that features a stainless steel liner.

Throughout the course of this study, it also became evident that as-low-as reasonably achievable (ALARA) dose reduction techniques used during CRD system maintenance have become an issue of concern and interest to many utilities. CRDM changeout and rebuilding is one of the highest dose, most physically demanding, and complicated maintenance activities routinely accomplished by BWR utilities. Recent innovations in CRDM handling equipment and rebuilding tools have allowed some utilities to make significant reductions in exposures obtained by personnel during the performance of CRDM maintenance activities.

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THE ROLE OF MONITORING AND TRENDING APPLIED TO DIESEL GENERATOR AGING

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Results of aging studies of emergency diesel generators (EDG) have shown that techniques based on practical system condition monitoring results could improve average system conditions. Condition monitoring of about 25 operating parameters of the engine and generator effectively indicates the functional status of important engine components and the location of and when to apply maintenance efforts. In contrast, intrusive engine inspections and overhauls, based on time periods alone, tend to reduce engine and system reliability. Pacific Northwest Laboratory[a] under the Nuclear Plant Aging Research (NPAR) program, has completed these aging studies of nuclear service diesel generators.

Based on the results of the aging assessments, it is recommended that a reliability centered program for managing emergency diesel generators integrating testing, inspection, monitoring, trending and maintenance activities be considered. A reliability centered program will 1) reduce the aging stressors associated with present EDG test requirements, while providing improved confidence in the diesel generator's capability to respond to accident situations, and 2) identify degraded and failing systems and components needing replacement or repair before a failure actually occurs.

Based on the aging studies, it is recommended that monthly engine testing be changed to a new approach that involves monitoring and trending. Monthly surveillance testing should involve data acquisition on performance parameters that indicate trends in component and subsystem condition. Such data would provide short-term and long-term information on degraded performance and can be used for practical reliability improvements. Systematic analysis of the data offers a basis to improve basic engine reliability and mitigate aging effects.

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Aging Studies of Batteries and Transformers in Class 1E Power Systems[a]

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The Idaho National Engineering Laboratory (INEL) performed aging studies of two components in the Class 1E Power System in support of the United States Nuclear Regulatory Commission's (USNRC) Nuclear Plant Aging Research (NPAR) Program. Both Phase I and Phase II aging research was performed for batteries while transformer research was limited to a Phase I study. The Phase I research consisted of : 1) an identification of the materials of construction, 2) an identification of the aging stressors and the significant aging mechanisms, 3) an examination of operational data to identify the dominant failure mechanisms that are being observed in nuclear power plants, 4) a review of current maintenance practices to determine if all significant aging degradation is being detected and managed, and 5) an identification of the needs (if any) for a Phase II study. The Phase II study involved performing testing of naturally aged batteries to determine if adequate seismic ruggedness of well maintained, naturally aged, batteries was retained to withstand the most severe earthquakes expected in the U.S.

The Phase I battery study noted that cracking of the containers and oxidation of the lead are the two most significant aging related mechanisms leading to battery failures. Oxidation causes the lead to swell, leading to either cracking of containers or degraded positive plates and decreased electrical capacity. Oxidation also leads to embrittlement of the lead which, if allowed to continue, will ultimately result in decreased seismic ruggedness. Results of previous battery tests, results of NRC investigations of batteries at nuclear plant with brittle lead components, and the fact that there are no in-plant tests that identify decreased seismic ruggedness pointed to a concern that seismic ruggedness could decrease to less than required to withstand a safe shutdown earthquake (SSE) and yet have acceptable electrical capacity as measured by periodic discharge tests.

Phase II seismic tests were conducted on twelve naturally aged batteries obtained from a nuclear power plant. The batteries were about 13 year old and had been maintained in accordance with the practices specified in IEEE Std 450 and NRC Regulatory Guide 1.1.129, Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants. The batteries were mounted in a seismically approved battery rack supplied by the battery vendor and subjected to four seismic levels, with the final level approximating the most severe that

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is required of batteries in the U.S. None of the batteries experienced any loss of electrical capacity as a result of the seismic testing and only minor damage was caused of battery containers as a result of the batteries rubbing on the battery rack. The study concluded that when batteries are maintained and operated in accordance with IEEE Std 450 and Regulatory Guide 1.129, little if any electrical capacity will be lost as a result of seismic shaking at levels that are typical of the most severe levels required for SSE in the U.S.

The Phase I transformer study found that degradation or loss of electrical insulation is the most significant failure mode for transformers. Failures of the windings (turn to turn, winding to winding or winding to ground faults) and failures in the bushings that provide the interface between the transformer and transmission lines are primarily caused by a failure of the insulation system. The study concluded that the failure rate of transformers does not show a trend that indicates an increased failure rate with age of transformers. However over 95% of the transformers are less than 20-years old and 75% are under 15-years old. Because transformers are normally considered to be a long life item (40 years or greater), a significant trend with age would not be expected at this time. Because transformers in nuclear facilities are relatively young, it is recommended that a periodic review of operating experience of transformers be performed to determine whether significant trends, not previously identified, are developing. Transformer reliability can be improved and maintained by the use of a thorough and continual program of inspection, surveillance, and maintenance. Such a program will detect and reduce stressors that shorten transformer life, prevent stressors before they cause degradation, and detect degradation in the early stages so that preventive and corrective action can be taken prior to transformer failure.

Operating Experiences and Degradation Detection*
for Auxiliary Feedwater Systems

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SUMMARY

The Auxiliary Feedwater (AFW) System has historically been recognized as critical to successful mitigation of pressurized water reactor (PWR) plant transients and accidents. A report prepared for the Nuclear Regulatory Commission (NRC) by Oak Ridge National Laboratory (ORNL) in 1990, NUREG/CR-5404, ORNL-6566/V1, "Auxiliary Feedwater System Aging Study", reviewed the AFW system design, testing requirements and practices, and historical operating experience. This study was carried out under the auspices of the NRC's Nuclear Plant Aging Research (NPAR) program. The results of the research are presented in this paper.

<u>Failure Data Review</u>
Failure data from the Institute of Nuclear Power Operation's Nuclear Plant Reliability Data System (NPRDS) and Licensee Event Reports were reviewed in detail. Various parameters were evaluated during the review, including the component and subsystem affected, method of detection, and the effect of the failure on the system. The numbers of failures of specific components found in the databases were not deemed to be reliable indicators of actual failure experience; however, the *distribution* of failures was considered to be meaningful.

Five major categories of components for the AFW system were designated for the review:

- Pump drives
- Pumps
- Valve operators
- Valves
- Other

It was found that 37% of the system degradation was attributable to failures of pump drives, including turbines, motors, and diesels. The bulk of the drive degradation was found to occur in turbine drives, which accounted for 27% of overall system degradation. Valve power operators, including motor, air, and electrohydraulic operators, were the second leading major category of system degradation, accounting for 28% of overall degradation.

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Three categories of methods of detection were designated:

- Programmatic monitoring
- Routine observation
- Demand

Failures detected during planned, periodic inspections or tests, such as surveillance tests, were deemed to have been detected by programmatic monitoring. Failures detected by operators or others during the course of normal plant operation, such as the observation of valve stem leakage or a control board annunciator, were deemed to have been detected by routine observation. Failures detected when the system was called upon to function in support of normal or emergency operations, such as failure of a pump to start on demand following a reactor trip, were deemed to have been detected on demand.

It was found that the distribution of system degradation associated with programmatic monitoring, routine observation, and demand was 42%, 39%, and 18%, respectively.

Individual Plant Design and Monitoring Practices Review

The design of a reference AFW system for a cooperating utility's plant, and the associated surveillance, operating, and maintenance procedures were reviewed. A principal purpose of the review was to determine the extent to which potential sources of failure would be detectable by the exisiting monitoring programs.

It was found that there are two general categories of failure sources which would not have been detected by the programmatic monitoring practices of the reference system:

- Failures of various instrumentation and control components that, for whatever reason, are not tested periodically, and
- Failures of components to perform under design-basis type conditions (although performance under less stringent conditions may be demonstrated periodically).

It was also found during the review of testing practices that some components appear to be tested excessively, possibly leading to accelerated aging. This was attributed to be largely due to the difficulty of coordinating a variety of technical specification test requirements to minimize testing of individual components (the focus of the testing program was naturally more oriented toward ensuring that all technical specification surveillances were met, not on minimizing the numbers of tests).

Conclusions

It was concluded that additional focus on turbine drives, specifically turbine governors and control systems, was merited. A phase I study is currently underway. It was also concluded that the potential for optimizing test requirements related to AFW systems should be explored. This study is scheduled to begin in FY 1992.

AGING ASSESSMENT OF RESIDUAL HEAT REMOVAL SYSTEMS IN BOILING WATER REACTORS*

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SUMMARY

The Nuclear Plant Aging Research (NPAR) program was established by the U.S. Nuclear Regulatory Commission to address concerns related to aging effects on the safety and reliability of nuclear power plants. As part of the NPAR program, various studies have been performed, the goals of which are to characterize aging and service wear effects and identify methods of detecting and mitigating them.

Work under the NPAR program is structured into two phases. In phase I, aging effects are characterized by identifying predominant failure causes, modes and mechanisms, along with the components most susceptible. The second phase then uses phase I results to assess current surveillance and monitoring practices, and develop functional indicators to mitigate aging effects. This paper presents preliminary phase I results for the Residual Heat Removal (RHR) system study.

In this study, the RHR system for Boiling Water Reactors was analyzed. Various designs are used, however, the most typical includes two loops. Each loop includes two pumps and one heat exchanger, along with numerous valves and instrumentation. The RHR system is capable of operating in several different modes. The two most common are Low Pressure Coolant Injection (LPCI) and Shutdown Cooling (SDC), which are the subject of this study.

The aging analysis included a review and evaluation of numerous RHR failure records from various national data bases, as well as actual plant records. Results showed that approximately 70% of the failures reported were related to aging degradation. The dominant failure cause was normal service, while the dominant mechanism was wear.

An evaluation of the component failures showed that valves were the component most frequently failing followed by instrumentation and controls. The valve failures involved mostly motor-operated valves and typically were characterized by leakage from the valve or failure of the valve to transfer. The instrumentation/control failures were predominantly switch malfunctions where the device was out of calibration or failed to operate. All components were found to have a large fraction of failures related to aging.

The failure records were also reviewed to determine the effect of the failures. At the system level it was found that 53% of the reported failures

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

resulted in degraded system performance, while 21% resulted in a loss of redundancy. At the plant level it was found that a significant number of RHR failures were the direct cause of engineered safety feature actuations, plant shutdowns, automatic plant scrams, power reductions , and extension of outages. In addition, several aging related RHR failures were found which resulted in radiological releases.

This work has shown that aging degradation is a concern for RHR systems and that it can adversely effect system performance, as well as plant safety. To mitigate the effects of aging, good functional indicators are required. They should be capable of detecting aging degradation in the incipient stage to ensure that system reliability is not compromised. Potential functional indicators for the RHR system have been identified and are presented.

These results have also provided a technical basis for evaluating monitoring, inspection and maintenance practices. A preliminary review of current practices has been made, including a survey of several utilities, and results are presented.

Corrosion and Erosion Effects on Valve Friction and Operability.[a]

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The Idaho National Engineering Laboratory (INEL) has studied the aging mechanisms that could affect motor-operated valves (MOVs) operability. These studies support the U.S. Nuclear Regulatory Commission's (NRC) Nuclear Plant Aging Research (NPAR) programs. These studies focused on evaluating aging's effect on friction in MOVs and on the effects of valve wall thinning on MOV operability. Our first friction study reviewed corrosion, deposition, and erosion-corrosion mechanisms and their potential effect on friction, analyzed the failure data for MOVs, and inspected MOV conditions at operating facilities. This study showed that further study of aging and friction was necessary. As a result, a second study is progressing which will corrode samples of valve materials and measure the changes in friction factors for these materials. The structural analysis of wall thinning on MOV operability is currently issued as an informal report to the NRC. This study used a finite-element model of a 16-inch carbon steel globe valve to evaluate the potential for erosion-related wall thinning to compromise the operability of these MOVs.

A draft report, NUREG/CR-5735, "An Evaluation of the Effect of Aging on Friction Factors of Motor-Operated Valves" has been completed. The report is the product of a study that investigated the effects of three aging mechanisms, corrosion, erosion-corrosion, and deposition, on the friction coefficients of the valve disk along the valve guides and seats of MOVs. The study was a Phase I NPAR study to determine the extent of frictional related failures in MOVs and to assess the need for further studies in this area. Three areas were evaluated in the study: 1) a review of existing research of the three mechanisms to determine if their effects on friction had previously been evaluated, and if not, their likely effect on friction, 2) a review of MOV operating history to evaluate the extent of the failures due to these mechanisms, and 3) observation of the as-found condition of MOVs during maintenance activities and interviews with key MOV industry personnel to corroborate the findings of the operating history review.

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[a]Work sponsored by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under DOE contract number DE-AC07-76ID01570; Dr. G. H. Weidenhamer, Technical Monitor.

The aging-friction factor study reached the following conclusions:

- The rates of corrosion of and deposition on valve sliding surfaces is not likely to obstruct the mechanical tolerances of the valves.

- It is likely that corrosion and deposition products on valve sliding surfaces will cause increases in the friction factors of these components, but the magnitude of the problem is indeterminate requiring further study in this area.

- Insufficient theoretical information is available regarding deposition on valve surfaces. The operating history review indicated a few problems related to deposition have been reported, therefore, further study is needed.

The INEL is presently conducting initial friction testing of material samples to provide the answers to the questions raised by the previous study. The initial friction tests will focus on carbon steel samples and carbon steel clad with Stellite. These materials are representative of the materials used in the valves of concern to the NRC's Generic Issue 87 (Failure of HPCI Steam Line Without Isolation). Separate friction tests are planned on both uncorroded and corroded samples to assess the impact of corrosion products on sliding friction. The initial results of these tests are expected later this year.

The report EGG-SSRE-10039, "An Evaluation of the Effects of Valve Body Erosion on MOV Operability" is the product of a study on erosion damage to MOVs that could compromise the valve's operability. The purpose of the study was to determine if wall thinning could reduce structural strength so design basis stresses could deform the body preventing valve operation due to disk binding. A finite-element model was used to simulate the effect of thinning on the structural integrity of the valve body. The valve chosen for modeling was a 16-inch carbon steel globe valve typical of ones used in the RHR systems at BWRs. A review of existing NRC operating history reports and sponsored research showed this system to be subject to the erosion damage of concern and it was a first-line safety system in the event of an accident. The specific valve selection was based on the worst-case observed erosion damage that has been reported by the industry.

The evaluation using the finite-element model used two sets of damage conditions: 1) a duplication of the actual damage that occurred in the worst-case reported event for a first-line safety system valve, and 2) the maximum damage that could occur when discovered by normal surveillance methods. The second conditions reduced the wall thickness of the first set of conditions to simulate through-wall erosion of the bridge and valve body. For both of the condition sets, the analysis showed that peak stresses remained well below the yield stresses for the body material. For all areas of the valve body, stresses peaked at approximately 50% of yield stress. Given these results, we concluded it is unlikely that thinning of the valve walls will result in deformation of valve materials under design basis conditions.

EFFECT OF COMPONENT AGING ON PWR CONTROL ROD DRIVE SYSTEMS*

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SUMMARY

An aging assessment of PWR control rod drive (CRD) systems has been completed as part of the USNRC Nuclear Plant Aging Research (NPAR) Program. The CRDs are flange mounted on top of the reactor pressure vessel head, and serve to position the control rod assemblies in the core in response to automatic or manual reactivity control signals. Additionally, the systems are designed to provide a rapid insertion of the control rods upon loss of AC power. This study examined the design, construction, maintenance, and operation of the Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse (W) systems to determine the potential for degradation as the system ages. The CRD system boundaries consisted of the drive mechanisms, power and control, rod position indication, and cooling system components. The individual absorber rods, fuel assembly and upper internal guide tubes were also examined since failure of these components may prevent control rod insertion.

Operating experience data were evaluated to identify the predominant failure modes, causes, and effects. This evaluation, coupled with an assessment of the materials of construction and operating environment, demonstrate that each design is subject to degradation, which if left unchecked, could affect its safety function as the plant ages.

Failures of the power and control, and rod position indication systems; and degradation of the control rod drive mechanisms and seals accounted for the majority of system failures. These failures have resulted in significant operational effects including primary coolant leakage, dropped or slipped rods, power reductions, plant scrams, and in some instances, emergency safety system actuation. Since 1980, the NRC has issued six Information Notices alerting plant operators to various CRD system failures.

An industry survey, conducted with the assistance of EPRI and NUMARC, identified current CRD system maintenance and inspection practices. The results of this survey indicate that some plants have performed system modifications, replaced components, or augmented existing preventive maintenance practices in response to system aging. The survey results also supported the operating experience data, which concluded that the timely replacement of degraded components prior to failure was not always possible using existing condition monitoring techniques. Therefore, the recommendations presented in this study also include a discussion of more advanced monitoring techniques, which provide trendable results capable of detecting aging.

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

AGING ASSESSMENT OF REACTOR INSTRUMENTATION AND PROTECTION SYSTEM COMPONENTS, PHASE I

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Summary

A study of the aging-related operating experiences throughout a five year period (1984-1988) of six generic instrumentation modules (indicators, sensors, controllers, transmitters, annunciators, and recorders) was performed as a part of the USNRC Nuclear Plant Aging Research (NPAR) Program. The effects of aging from operational and environmental stressors on these instrumentation modules were characterized using data reported in nationwide industry databases such as Licensee Event Reports (LERs), Nuclear Plant Reliability Data Systems (NPRDS), and Nuclear Plant Experience (NPE). Other material used in the research included NRC Daily Headquarters Reports, NRC Daily Operating Events Reports, NRC Regional Inspection Reports, and published literature for related investigations of instrumentation aging. By examination of operational history, those safety-related instrumentation modules most subject to aging were identified. The data are graphically displayed as frequency of events per plant year for operating plant ages from 1 to 28 years to determine aging-related failure trend patterns.

Three main conclusions were drawn from this study:

1. Instrumentation and control (I&C) modules make a modest contribution to safety-significant events.

   - 17% of all LERs issued during 1984-1988 dealt with malfunctions of the six I&C modules studied.

   - 28% of the LERs dealing with these I&C module malfunctions were aging related (other studies show a range of 25-50%).

2. Of the six modules studied, indicators, sensors, and controllers account for the bulk (83%) of aging-related failures.

3. Infant mortality appears to be the dominant aging-related failure mode for most I&C module categories (with the exception of annunciators and recorders, which appear to fail randomly).

Additional observations are made from examination of the module failure data, and three

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*Managed by Martin Marietta Energy Systems, Inc., for the U.S. Department of Energy under Contract DE-AC05-84OR21400.

recommendations were derived from the overall investigation:

1. Consideration should be given to methods that would be helpful in reducing the incidence of infant mortality, particularly for the indicators and sensors categories, which dominate aging-related I&C module malfunctions and failures. For example, modules could be subjected to extremes of operational environment (e.g., temperature, humidity, operating voltages) and perhaps cycled over an extended time period to identify marginal components prior to installation. Military standards provide good indication of those practices likely to be beneficial.

2. Consideration should be given to testing selected I&C modules for synergistic effects of aging stressors. The purpose would not be qualification of specific equipment but rather identification and quantification of generic stress and failure relationships. Tests would not be intended to demonstrate operating envelopes of any specific brand of equipment.

3. Consideration should be given to creating an industry-wide database dedicated to aging-related information. Earlier studies pointed out that existing databases, while reporting stressors, do not adequately indicate the root cause failure mechanisms; this current study also encountered difficulty in drawing conclusions as a result of this deficiency. A good source of information for aging studies would be the maintenance records of the individual plants. An industry-wide, readily accessible database devoted specifically to aging-related events and information would provide a most helpful and efficient service for those interested in plant and equipment aging.

OPERATING EXPERIENCE REVIEW OF FAILURES OF POWER OPERATED RELIEF VALVES AND BLOCK VALVES IN NUCLEAR POWER PLANTS

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SUMMARY

This paper contains a review of report NUREG/CR-4692 "Operating Experience Review of Failures of Power-operated Relief Valves and Block Valves in Nuclear Power Plants" which compiled nuclear power plant operating events involving failures of power-operated relief valves (PORVs) and associated block valves (BVs). Of the 230 events identified, 101 involved PORV mechanical failure, 91 were attributable to PORV control failure, 6 events involved design or fabrication of the PORVs, and 32 events involved BV failures. The report contains a compilation of the PORV and BV failure events, including failure cause and severity. The events are identified as to plant and valve manufacturer. An assessment of the need to upgrade PORVs and BVs to safety-grade status concludes that such action would improve PORV and BV reliability. The greatest improvement in reliability would result from using newer, more reliable PORV designs and improving testing, diagnostics, and maintenance applied to PORVs and BVs, particularly the BV motor operator. A summary of interviews conducted with four PORV manufacturers is also included in the report.

NUREG/CR-4692 was prepared by the Nuclear Operations Analysis Center (NOAC) in response to a request from the Nuclear Regulatory Commission (NRC) Division of Engineering Technology (DET) for a survey of power-operated relief valve (PORV) and block valve (BV) operating experience. The information was provided under the Nuclear Plant Aging Research (NPAR) Program to support the resolution of Generic Issue 70 (GI-70) "PORV and Block Valve Reliability."

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ASSESSMENT OF DIAGNOSTIC METHODS FOR SOLENOID-OPERATED VALVES

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Summary

Solenoid-operated valves (SOVs) were studied at Oak Ridge National Laboratory as part of the USNRC Nuclear Plant Aging Research (NPAR) Program. The primary objective of the study was to identify, evaluate, and recommend methods for inspection, surveillance, monitoring, and maintenance of SOVs that can help ensure their operational readiness–that is, their ability to perform required safety functions under all anticipated operating conditions.

Solenoid-operated valves are available, both with and without nuclear qualification, from a number of different manufacturers and are found throughout nuclear power plants in relatively large numbers, oftentim [illegible] omponent of larger, more complex, and clearly safety-related systems such as contai [illegible] ve actuators, BWR control rod scram systems, and PWR safety injection syste [illegible] They [illegible] ly simple devices, with a long history of satisfactory operation in a variety of [illegible] n-nuclear industrial applications. However, their presence in systems imp [illegible] res an especially high degree of assurance that they are ready to perform their [illegible] nder all anticipated operating conditions, since failure of one of these small and re [illegible] ve devices could have serious consequences under certain circumstances.

An earlier (Phase I) NPAR program study described SOV failure modes and causes and identified measurable parameters thought to be linked to the progression of ever-present degradation mechanisms that may ultimately result in functional failure of the valve. Using this earlier work as a guide, the present (Phase II) study focused on devising and then demonstrating the effectiveness of techniques and equipment with which to measure performance parameters that show promise for detecting the presence and trending the progress of such degradations before they reach a critical stage.

Intrusive techniques requiring the addition of magnetic or acoustic sensors or the application of special test signals were investigated briefly, but major emphasis was placed on the examination of condition-indicating techniques that can be applied with minimal cost and impact on plant operation (see accompanying Table 1). These include remotely monitoring coil mean temperature, determining valve plunger position and verifying unrestricted SOV plunger movement, and detecting the presence of shorted turns or insulation breakdown within the solenoid coil. The first of these techniques, though perhaps the simplest conceptually, will likely benefit the nuclear industry most because SOVs have a history of failure in service as a result of unwitting operation at excessive temperatures.

Experimental results are presented that demonstrate the technical feasibility and practicality of the monitoring techniques assessed in the study, and recommendations for further work are provided.

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Table 1. SOV monitoring methods evaluated in this study

Method	Degradation(s) or malfunction(s) addressed	Attributes	Promise for in-plant use
Measurement of SOV temperature, via coil resistance or impedance	Electrical failure of coil and degradation of elastomers resulting from prolonged operation at excessively high temperatures	• Nonperturbative to plant operations • No new sensors or signal cables are required • No permanent instrumentation required; can be applied as needed from a remote location • Applicable to ac- and dc-powered SOVs	High; ready for immediate use
Indication of valve position and change of state upon application of power, via change in coil impedance	Mechanical binding, sluggishness, or failure to shift as a result of worn or improper parts or the presence of foreign materials inside the valve	• No need for add-on sensors or signal cables • Valve position readout from a remote location • Static method does not disturb SOV	High; some additional development work required
Indication of mechanical binding, by tracking changes in current and voltage at SOV pull-in and drop-out points	Mechanical binding and sluggish shifting caused by worn, swollen, or improper parts or the presence of foreign materials inside the valve	• Detects simultaneously degradation of magnetic or spring forces, and increase in frictional forces • No need for add-on sensors or cables or access to SOV • Applicable to ac- and dc-powered SOVs	Medium; further testing needed to ascertain cause of poor repeatability of test results
Indication of shorted coil turns or insulation breakdown, based on characteristics of electrical transient generated upon deenergizing a dc SOV	Electrical failure of solenoid coil, caused by high-voltage turn-off transients in combination with insulation weakened by prolonged operation at high temperatures	• Detects presence of defects within coil that cannot be revealed by other means	Low; useful for laboratory post-mortem tests
Indication of mechanical binding, by analyzing the time-varying characteristics of the inrush current accompanying application of electrical power to the SOV	Mechanical binding and sluggish shifting caused by worn, swollen, or improper parts or the presence of foreign materials inside the valve	• No need for add-on sensors, signal cables, or access to SOV • Information could be obtained as a result of everyday valve operation	Minimal; investigation of method abandoned early in the study
Indication of mechanical looseness within ac-powered valves, via electrical detection of humming or chattering of the plunger assembly (frequency decomposition of steady-state coil current)	Wear of internal valve parts, improper assembly, or replacement with incorrect parts	• No need for add-on sensors, signal cables, or access to SOV • Nonperturbative to plant operations	Minimal; investigation of method abandoned early in the study. Addition of miniature acoustic sensor to SOV might prove worthwhile

SHIPPINGPORT STATION AGING MANAGEMENT LESSONS

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The decommissioned Shippingport Atomic Power Station has been a major source of naturally aged equipment for the U.S. Nuclear Regulatory Commission's Nuclear Plant Aging Research Program and other aging research studies. The decommissioning of the Shippingport Station provided a unique opportunity to conduct in situ assessments at an aged reactor and to obtain a variety of naturally aged components and samples for detailed laboratory evaluation.

As the first U.S. large-scale, central-station nuclear plant, the Shippingport Station paralleled commercial pressurized-water reactors in reactor, steam, auxiliary, support, and safety systems. Its 25-year service life (1957 to 1982) overlapped the construction and initial operating period of most reactors currently operating. Also, because of su tantial modifications during the mid-1960s and 1970s, the Shippingport Station offered unique examples of identical or similar equipment used side-by-side, but representing different vintages and degrees of aging.

As part of the Pacific Northwest Laboratory[a] Shippingport Station aging evaluation work, more than 200 items, ranging in size from small instruments and materials samples to one of the main coolant pumps, were removed and shipped to various laboratories for evaluation. These items included battery chargers, inverters, relays, breakers, switches, power and control cables, electrical penetrations, check valves, solenoid valves, and motor-operated valves. Samples of piping from various plant systems also were acquired for radiological characterization studies, and samples from the primary system check valves, main stop valves, and main coolant pumps were removed for materials degradation studies. In situ assessments of Shippingport Station components also were conducted, including the pre-removal visual and physical examination of components, the testing of electrical circuits, and special measurements to assist in selecting specific components for further evaluation.

Naturally aged components and materials are subjected to the actual in-plant environments, operating conditions, testing procedures, and maintenance practices. Thus, their evaluation is an important way to verify expected degradation mechanisms and failure modes, to ensure that other components will be minimally affected by aging, to validate aging projections based on the extrapolation of accelerated test data, and to detect unexpected

(a) Pacific Northwest Laboratory is operated for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830. Work is conducted under NRC FIN B2911.

aging mechanisms (surprises) that could significantly impact component or system performance. The following are examples of the types of aging management information that have been derived from the studies of the naturally aged Shippingport Station components and materials.

Identification of Equipment Demonstrating Satisfactory Long-term Performance with Minimal Aging Effects

Naturally aged inverters and battery chargers from the Shippingport Station were tested by Brookhaven National Laboratory. Although some aging-induced changes were noted, it was concluded that aging had not substantially affected equipment operation. Similarly, Wyle Laboratories conducted operability and performance tests of Shippingport Station circuit breakers and relays, demonstrating that these components still met the original specifications.

Comparison Data for Validating Aging Projections Based on Accelerated Aging Studies

An Argonne National Laboratory investigation of the microstructural characteristics of cast stainless steel from Shippingport Station primary system valves and pump volutes verified that the low-temperature thermal embrittlement mechanism for this naturally aged material is the same as that of laboratory-aged material. This provided a unique opportunity to validate and benchmark the laboratory studies.

Basic Insights on Degradation Mechanisms

An Argonne National Laboratory evaluation of samples from the inner wall of the Shippingport Station neutron shield tank, which represent base metal and weld material exposed to different neutron flux levels, has provided valuable information on possible low-temperature low-flux embrittlement processes in reactor pressure vessel support structures.

Detection of Unexpected Aging Mechanisms (Surprises)

An Oak Ridge National Laboratory evaluation of a piston lift check valve from the Shippingport Station found significantly more wear than would be expected based on the valve's normal service environment. Similarly, an evaluation of an 8-in. diameter gate valve and operator from Shippingport Station revealed a previously unrecognized cable sizing problem.

Identification of Condition and Performance Parameters to Detect and Monitor Aging

An Idaho National Engineering Laboratory in-situ evaluation of Shippingport Station electrical circuits confirmed the effectiveness of the measurement system for detecting degradation of circuit connections and splices.

Effectiveness of Surveillance Methods for the Class 1E Power and Reactor Protection Systems[a]

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

The operational data review showed that approximately 50% of all Class 1E Power System failures were not detected by routine IS&MM. Batteries, circuit breakers, inverters, and relays accounted for over 80% of all 1E System failures. Furthermore, these components dominate the failures in each of the four subsystems in the Class 1E Power system; the Plant AC (PAC), the Instrument AC (IAC), the DC Power (DCP), and the Emergency Power (EMP) subsystems. This indicates that the performance of a subsystem can be significantly improved by concentrating on the detection methods of just one or two types of components that are significant for that respective subsystem. The data showed that, for the DCP subsystem, batteries had the largest number of failures. Routine IS&MM for these batteries were not completely effective in detecting failures caused by short/ground, open circuit, burned circuit, wear, and aging/cyclic fatigue. In the PAC subsystem circuit breakers had the highest numbers of failures; its routine methods did not effectively detect failures caused by open circuits, weld faults, out-of-adjustment, and mechanical damage. Inverters had the largest number of failures in the IAC subsystem; short/ground circuit and open circuit failures were not effectively detected by the routine methods. For EMP subsystem relays and circuit breakers had the highest number of failures. Routine IS&MM was not effective in detecting open circuit, burned circuit, out-of-adjustment, and wear relay

a. Work sponsored by the U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, under DOE Contract No. DE-AC07-76ID01570; Dr. G. H. Weidenhamer, Technical Monitor.

failures. Open circuit, wear, burned circuit, and out-of-adjustment failures were not effectively detected by routine IS&MM for the circuit breakers.

The operational data review also revealed that approximately 40% of all Reactor Protection System failures were not detected by routine IS&MM. Over 70% of all RPS events were accountable to transmitters, integrators, and bistables. For transmitters routine methods were not effectively able to detect failures caused by wear and aging/cyclic fatigue. Open and dirty circuit failures were not detected effectively by routine IS&MM for bistables. Aging/cyclic fatigue and defective connections caused the failures not effectively detected by these routine methods for integrators.

A standard Probabalistic Risk Assessment (PRA) was used to determine the risk significant Class 1E components. Truncation of the fault tree cut sets left the circuit breakers, transformers and buses as the risk significant components. The effects of aging were determined by simply applying aging rates, as determined by expert opinion, to the initial failure rates used in the PRA to determine if significant changes occurred as a function of time. The failure rate for the circuit breaker increased the most at 20%, followed by the transformer (6%) and the bus work (1%). Requantification of the PRA, using the new failure rates, revealed a less than 1% increase in the core damage frequency. This was verified by using operational data and statistical methods to formulate an actual aging failure rate for the circuit breakers, since they had the highest theoretical failure rate increase.

An in-plant evaluation was performed to study ways to improve current IS&MM practices for both the Class 1E and Reactor Protection Systems. The cooperating plant currently uses a variety of advanced IS&MM. One method uses electromagnetic signals to measures various parameters of de-energized electrical circuits. This system determines insulation integrity of the system through measurements of capacitance, dissipation factor, impedance, and insulation resistance. Time Domain Reflectometry is used to measure circuit integrity. This method collects baseline data and uses it in a comparison on later tests of the same piece of equipment; this has proven especially effective for circuits. Other improved methods include the use of state-of-the-art infrared technology. This method can be used on a "one time only" basis to reveal hot spots and to collect data for long term trending. The basic principle of this device is to measure the heat generated by electrical equipment; the test data is compared to baseline criteria to determine whether that component is within a certain operating range. Data obtained from this system is stored on computers for analysis. A third system used at this plant is Redundant Instrumentation Monitoring. This system is used to verify calibration of on-line instrumentation. A computer reads instrument output data from the plant computers. The system then allows comparison of redundant channel outputs over time or trending of one channel. The ability of all these systems to store data on computers (in an easily retrievable form) allows plant personnel the ability to trend the 1E Power and Reactor Protection Systems for aging related degradation.

EFFECTIVE AGING MANAGEMENT OF CIRCUIT BREAKERS AND RELAYS

by

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It is the challenge of a good preventive maintenance program to be sensitive to the effects of aging. Early identification of age related degradation increases the probability that the safety significance of this aging is minimized. An effective inspection, surveillance and monitoring program enhances mitigation of the impact of age related degradation on the safety of nuclear plant operations.

This paper describes the results of a comprehensive aging assessment of relays and circuit breakers that was completed as part of the NRC Nuclear Plant Aging Research (NPAR) Program. Relays and circuit breakers were analyzed because they are important safety-related equipment which perform critical functions in the operation and control of nuclear power plants and have experienced age related degradation.

The research investigated current and improved inspection, surveillance and monitoring methods (ISM) for circuit breakers and relays. The effort was accomplished in four major elements. These were an investigation into current and advanced ISM methods, tests of aged relays and circuit breakers, tests of degraded relays and circuit breakers and in-situ tests.

The results are significant in that :

- o Improved inspection, surveillance and monitoring (ISM) methods for relays and circuit breakers have been identified which are more effective at detecting aging degradation than current nuclear plant practice,

- o Less intrusive ISM methods, which have the potential for providing predictive maintenance and condition based maintenance have been determined, and

- o Important modifications of specific maintenance practices for some common relays and circuit breakers are presented where the research showed superior and more cost effective ISM methods for these devices.

Current and advanced ISM methods were ascertained by soliciting information from nuclear and non-nuclear utilities, relay and circuit breaker manufacturers and maintenance facilities.

Testing of aged devices was performed. Test specimens for each of the five relay (auxiliary, control, electronic, protective and timing) and two circuit breaker (molded case and metal clad) types were solicited from nuclear and non-nuclear utilities and manufacturers. A total of 39 specimens were tested to the current and improved ISM methods.

Eleven specimens of relays and circuit breakers were purposely degraded and the ISM methods performed after each degraded condition. The purpose of these degradation tests was to evaluate the effectiveness of the method to detect and/ or predict the level of degradation. This also provided some quantifiable parameters of the extent of degradation. The degradations chosen for each relay and circuit breaker type were purposely severe, but for the most part, did not cause total loss of operability of the device. Thus, it was an attempt to simulate the worst state of deterioration or degradation prior to failure to operate. The degradations were chosen based on a review and evaluation of the failure mode and mechanisms reported in NPAR reports, specified by the nuclear and non-nuclear utilities, manufacturers, and experiences of the research team.

These degradation condition evaluations showed that generally accepted current nuclear maintenance practices do not always detect significant aging mechanisms. This result provides insight into the reason why failures of safety-related relays have occurred in service in spite of a comprehensive maintenance program.

The practicability of the effective methods was also evaluated at Duke Power Company's Catawba Nuclear Station and Niagara Mohawk Power Corporation's Nine Mile Point Unit 1 Nuclear Plant.

While at the plants, the research team witnessed plant maintenance personnel performing routine plant maintenance on relays and circuit breakers. Copies of procedures were obtained, results of plant maintenance tests were reviewed, and engineering and maintenance personnel were interviewed. Additionally, non-intrusive ISM methods of infrared pyrometry, infrared scanning and vibration testing were demonstrated to the plant personnel. The plant maintenance personnel and engineering staff at both plants were found to be cooperative, professional, experienced, knowledgeable and eager to discuss potential improved techniques.

UNDERSTANDING AND MANAGING EFFECTS OF BATTERY CHARGER AND INVERTER AGING*

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SUMMARY

An aging assessment of battery chargers and inverters was conducted under the auspices of the NRC, Nuclear Plant Aging Research (NPAR) Program. The intentions of this program are to resolve issues related to the aging and service wear of equipment and systems at operating reactor facilities and to assess their impact on safety.

Inverters and battery chargers are used in nuclear power plants to perform significant functions related to plant safety and availability. The specific impact of a battery charger or inverter failure varies with plant configuration. Operating experience data have demonstrated that reactor trips, safety injection system actuations, and inoperable emergency core cooling systems have resulted from inverter failures; and dc bus degradation leading to diesel generator inoperability or loss of control room annunciation and indication have resulted from battery and battery charger failures. For the battery charger and inverter, the aging and service wear of subcomponents have contributed significantly to equipment failures.

To identify aging and service wear effects and appropriate inspection/surveillance/monitoring techniques, it was necessary to examine potential failure modes, mechanisms, and causes. This was achieved by reviewing battery charger and inverter design and materials of construction, by establishing the stressors that are both operational and accident related, and by reviewing existing failure related data.

The primary contributors to inverter and battery charger failures are overheating, electrical transients, and personnel errors. In many cases, the stresses induced from these occurrences results in an accelerated aging of critical components. Electrolytic capacitors, fuses, magnetics (inductors and transformers) and semiconductors such as silicon controlled rectifiers, are susceptible to aging degradation resulting from these stresses.

Based on the testing of a naturally aged inverter and battery charger, surveys of the current maintenance practices at nuclear power plants, and an assessment of the present equipment technology, it was found that methods exist for detecting aging degradation. The detection techniques found to be most valuable in determining inverter and battery charger aging are component and

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

equipment temperature monitoring, periodic observation of circuit waveforms, and component parameter measurements.

Mitigating the effects of aging degradation can be accomplished through a combination of maintenance, design changes, and personnel training. It is recommended that a comprehensive maintenance program be established for inverters and battery chargers that addresses four areas: inspection, testing, predictive maintenance, and corrective maintenance. Guidelines for each of these areas have been provided in research reports NUREG/CR-4564, 5051, and 5192.

Current maintenance practices at nuclear power plants were evaluated to assess utility programs for addressing inverter and charger aging. Two independent surveys, one by BNL and one by the Electric Power Research Institute (EPRI) found that all of the surveyed plants specified some maintenance activities for chargers and inverters, and most utilities are cognizant of this equipment's importance to safety and availability. However, the wide range in the type of maintenance performed, and the varying levels of success achieved could reflect inadequacies at some plants. While emphasis has been placed on maintenance practices for mitigating equipment aging, the potential improvement in vital bus reliability through design improvements should not be overlooked. As plants age, there should be an awareness of improvements in equipment to take advantage of advances which could minimize the plant's aging effects and maintain or enhance its established performance and safety goals. One recommended design improvement is the automatic transfer switch. The device reduces the impact of inverter degradation by sensing the failure and switching the vital bus to an alternate electrical source without interrupting power to safety related instrumentation, controls, and equipment.

Other recommended improvements resulting from this research include the application of equipment for detecting and suppressing electrical bus transients experienced regularly in power plants, the use of higher voltage and temperature rated components in the inverter circuitry, and the addition of forced air cooling to reduce the overheating problems experienced.

With personnel induced stresses accounting for approximately 15% of the age related inverter and charger failures, it is recommended that training be provided and procedures be established for the operation and maintenance of this fairly complex equipment.

Maintenance must be performed periodically to refurbish and/or replace components which exhibit aging. In addition to discrete components such as capacitors, transformers, and semiconductors, the integrity of other entities such as cable connectors, wiring, and structural fasteners must also be maintained to assure proper equipment operation under normal operating and postulated accident conditions. While it is not possible to prevent all component failures, preventive maintenance activities and condition monitoring techniques can be effective in reducing the number of failures.

SAND92-0206A

Aging Assessment of Cables*

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Abstract

This paper summarizes the results and conclusions of aging, condition monitoring, and accident testing of Class 1E cables used in nuclear power generating stations. The primary objectives of this experimental program were to determine the life extension potential of popular cable products used in nuclear power plants and to determine the potential of condition monitoring (CM) for residual life assessment. Cable insulation materials that were tested include cross-linked polyethylene (XLPE), ethylene propylene rubber (EPR), polyimide, and silicone rubber (SR). Cable jacket materials that were tested include neoprene, chlorosulfonated polyethylene, and fiberglass braid.

Three sets of cables were aged using simultaneous thermal (≈100° C) and radiation (≈0.10 kGy/hr) conditions. One set of cables was aged for 3 months, one set was aged for 6 months, and one set was aged for 9 months. After aging, each set of cables was exposed to a simulated loss-of-coolant accident (LOCA) consisting of high dose rate irradiation (≈6 kGy/hr) followed by a high temperature steam exposure. A fourth set of cables, which were unaged, was also exposed to the LOCA conditions. The cables that were aged for 3 months and then LOCA tested were subsequently exposed to a high temperature steam fragility test (up to 400° C), while the cables that were aged for 6 months and then LOCA tested were subsequently exposed to a 1000-hour submergence test in a chemical solution.

The experimental program provided several improvements upon most previous efforts by employing considerably less accelerated, simultaneous thermal and radiation aging conditions; by employing many more condition monitoring measurements during aging; by performing similar accident tests on cables aged to four different nominal lifetimes (including unaged cables); by obtaining data on the submergence behavior of cables that had been exposed to aging and LOCA testing; and by providing information on the ultimate thermal fragility of many different cable products after exposure to aging and LOCA testing. The test program generally followed the guidance of IEEE 323-1974 and IEEE 383-1974.

The accelerated aging temperature was determined by equating the 6-month exposure to a 40-year life and assuming an activation energy of 1.15 eV and a plant ambient temperature of 55° C. The accelerated radiation aging dose rate was determined by assuming a 40-year radiation dose of 400 kGy. The 3-month chamber was therefore nominally equivalent to 20 years of aging and the 9-month chamber was nominally equivalent to 60 years of aging.

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

During the aging exposure, various electrical and mechanical condition monitoring measurements were performed on the cables. Some of the measurements were performed on completed cable samples and others were performed on smaller samples that were aged together with the complete cable specimens and were removed from the test chambers during aging. The parameters measured included insulation resistance and polarization index at three different voltages, capacitance and dissipation factor over a wide range of frequencies, elongation and tensile strength at failure, modulus profiles, cable indenter modulus tests (using a cable indenter developed at Franklin Research Center under Electric Power Research Institute funding), hardness, and bulk density.

The results of the tests indicate that the feasibility of life extension of many popular nuclear power plant cable products is promising and that mechanical measurements (primarily elongation, modulus, and density) are more effective than electrical measurements for monitoring age-related degradation. In the high temperature steam test, ethylene propylene rubber (EPR) cable materials generally survived to higher temperatures than crosslinked polyolefin (XLPO) cable materials. In dielectric testing after the submergence testing, the XLPO materials performed better than the EPR materials.

The paper will summarize the preliminary conclusions from the overall test program, divided into six categories: general conclusions regarding aging and condition monitoring, general conclusions regarding the accident performance of aged cables, specific conclusions from accident testing of each of the three specific cable types (XLPO, EPR, and miscellaneous), and general conclusions from the submergence and high temperature steam tests.

ASSESSMENT OF DIAGNOSTIC METHODS FOR DETERMINING DEGRADATION OF CHECK VALVES*

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SUMMARY

Check valves are used extensively in nuclear plant safety systems and balance-of-plant (BOP) systems. Their failures have resulted in significant maintenance efforts and, on occasion, damage to other flow system components. Check valve failures have largely been attributed to severe degradation of internal parts resulting from instability (flutter) of check valves under normal plant operating conditions.

The Oak Ridge National Laboratory (ORNL) has carried out a comprehensive aging assessment of check valves in support of the NRC Nuclear Plant Aging Research (NPAR) program. This paper provides a summary of that assessment with emphasis on the identification, evaluation, and application of check valve monitoring methods and techniques.

As part of the ORNL Advanced Diagnostic Engineering Research and Development Center, ORNL has developed two novel nonintrusive methods that are useful for monitoring the position and motion of check valve internal parts. These methods are based on the use of externally-applied magnetic fields from permanent magnets and from electromagnet coils driven by either alternating or direct current.

Descriptions and evaluations of several check valve monitoring methods are provided in this paper including:

- Acoustic emission monitoring
- Ultrasonic inspection
- Magnetic flux signature analysis (MFSA)
- External ac- and dc-magnetic techniques

A major conclusion reached was that none of these methods examined could, by themselves, monitor the position and motion of valve internals and valve leakage; however, the combination of acoustic emission with either of the other methods yields a monitoring system that succeeds in providing the means to determine vital check valve operational information.

Other areas covered in the paper include descriptions of relevant regulatory issues and activities, other related diagnostics research at ORNL, and interactions ORNL has had with outside organizations for the purpose of disseminating research results.

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AN OVERVIEW OF THE STRUCTURAL AGING PROGRAM*

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ABSTRACT

The Structural Aging (SAG) Program was initiated at ORNL in mid-1988 and has the overall objective of preparing a report which will provide NRC license reviewers and licensees with the following: (1) identification and evaluation of the structural degradation processes; (2) issues to be addressed under nuclear power plant continued service reviews, as well as criteria, and their bases, for resolution of these issues; (3) identification and evaluation of relevant inservice inspection or structural assessment programs in use, or needed; and (4) quantitative methodologies for assessing current, or predicting future, structural safety margins. The SAG Program consists of three technical tasks: Materials Property Data Base, Structural Component Assessment/Repair Technology, and Quantitative Methodology for Continued Service Determinations.

The objective of the materials property data base task is to develop a computer-based structural materials property data base which will contain information on the time variation of material properties under the influence of pertinent environmental stressors and aging factors. Two complementary data base formats, hardcopy and electronic, have been developed and contain information on the performance of concrete, conventional steel reinforcement, prestressing steel, and structural steel materials. Also under this task a state-of-the-art report has been prepared which identifies and evaluates models and accelerated aging techniques and methodologies which can be used in making predictions of the remaining service life of concrete in nuclear power plants.

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The overall objectives of the structural component assessment/repair technology task are to (1) develop a systematic methodology which can be used to make a quantitative assessment of the presence, magnitude, and significance of any environmental stressors or aging factors, (2) provide recommended inservice inspection or sampling procedures, and (3) identify and evaluate techniques for mitigation of any environmental stressors or aging factors which may act on critical concrete components. Under this task an aging assessment methodology has been developed which can be used to identify and rank concrete structures as well as the degradation factors which can impact the performance of these structures. Also, a state-of-the-art report has been prepared which reviewed and assessed inservice inspection techniques and methodologies for application to concrete structures in nuclear power plants.

The overall objective of the quantitative methodology for continued service determinations task is to develop procedures that can be used to perform condition assessments and make reliability-based life predictions of critical concrete components in nuclear power plants. A draft report has been completed which describes development of a probabilistic method for condition assessment and life prediction of concrete structures.

The results of the SAG Program will provide an improved basis for the NRC staff to evaluate requests for continued operation beyond the nominal 40-year design life of a nuclear power plant. Potential regulatory applications of this research include: (1) improved predictions of long-term material and structural performance and available safety margins at future times, (2) establishment of limits on exposure to environmental stressors, (3) reduction in total reliance by licensing on inspection and surveillance through development of a methodology which will enable the integrity of structures to be assessed (either pre- or post-accident), and (4) improvements in damage inspection methodology through potential incorporation of results into national standards which could be referenced by standard review plans.

DATA BASE ON STRUCTURAL MATERIALS AGING PROPERTIES*

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ABSTRACT

The U.S. Nuclear Regulatory Commission (USNRC) has initiated a Structural Aging Program at the Oak Ridge National Laboratory to identify potential structural safety issues related to continued service of nuclear power plants and to establish criteria for evaluating and resolving these issues. One of the tasks in this program focuses on the establishment of a Structural Materials Information Center (SMIC) where data and information on concretes and other related materials used in nuclear power plant construction are being collected and assembled into a structural materials property data base. This data base will be used to establish current properties for materials in existing concrete structures and to predict future performance of these materials.

The Structural Materials Information Center is being presented in two complementary formats. The *Structural Materials Handbook* is published in four volumes as an expandable, hard-copy, reference document. The *Structural Materials Electronic Data Base* is formatted for use on an IBM-compatible personal computer. The SMIC contains baseline data, reference properties, and environmental information that are presented as tables, notes and graphs. The handbook contains a complete set of data and information for each material included in the data base and serves as the reference information source for the electronic data base. The electronic data base enhances the use of the handbook by providing an efficient means for searching the various data base files to locate materials with similar characteristics. The handbook and the electronic data base will be used by USNRC reviewers who perform structural assessments for continued service.

Material properties, data, and information are being collected at the Structural Materials Information Center from open literature and published references, and from laboratory tests conducted on material samples removed from existing concrete structures. Currently, the data base includes properties for portland cement concrete, metallic reinforcement, prestressing tendon and structural steel materials. As data and information for other materials such as rubbers, plastics, and nonferrous metals are obtained, the data base will be expanded and updated.

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Reliability-based Condition Assessment of Concrete Structures in Nuclear Power Plants

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Introduction

During the next 15 years, the operating licenses for a number of nuclear power plants in the United States will expire. Faced with the prospect of having to replace the lost generating capacity from other sources and substantial shutdown and decommissioning costs, many utilities are expected to seek renewals of their plant operating licenses. Reinforced or prestressed concrete structures provide essential safety-related functions in a nuclear plant. Studies have shown that replacement of or major repairs to certain concrete structures in the plant as a condition of continued service would be economically unfeasible. Concrete structures may be affected by aging, or changes in strength and stiffness beyond the baseline conditions assumed in initial structural design. Some of these aging effects are benign; others may cause strength to degrade over extended periods of time, particularly when the concrete is exposed to an aggressive environment. Thus, the evaluation of safety-related concrete structures for continued service should provide quantitative evidence that their strength will be sufficient to withstand future extreme events within the proposed service period with a level of reliability sufficient for public safety.

Technical Approach

A methodology is being developed as part of the Structural Aging Program to evaluate the time-dependent reliability of a reinforced or prestressed concrete structure. This methodology takes into account the stochastic nature of past and future loads due to plant operating conditions and the environment, and randomness in those physical processes and environmental stressors that may lead to degradation in strength. The role of periodic inspection and maintenance in meeting a target reliability level over a period of continued service is also included.

Events giving rise to significant structural loads occur randomly in time. When viewed on a timescale of 40 to 100 years, however, such events occur infrequently, have a relatively short duration, and occupy only a small fraction of the total life of a structure. Based on these observations, many of the operating, environmental and accidental loads that act on nuclear plant structures can be modeled as Poisson renewal or pulse processes. Statistical data to describe the rates of occurrence, duration and intensity of these load events are being synthesized from other research programs.

Changes in engineering properties of steel and concrete over an extended service life are modeled as time-dependent random functions. Strength degradation mechanisms related to corrosion of reinforcement, detensioning of prestressing tendons, and aggressive chemical attack from sulfates and acids appear potentially important for concrete structures in nuclear power plants. Statistical data to describe initial strength are available from previous probability-based code studies; strength changes in time are based on data provided from another task in the Structural Aging Program.

Structural reliability analysis integrates the probabilistic descriptions of strength and loads to describe time-dependent reliability and deterioration of concrete structural components and systems subjected to stochastic loads. The reliability functions can be used as a basis for selecting appropriate periods for continued service or for intervals of inspection and maintenance necessary to maintain reliability at an acceptable level.

Summary of Results

The methodology is illustrated using simple parametric representations of strength degradation and load process models. Comprehensive data currently are being developed as part of the Structural Aging Program; here, we attempt to identify those parameters that have a particularly significant impact on time-dependent reliability so as to guide subsequent data acquisition. Results to date indicate that the function describing the mean strength degradation and the mean occurrence rate of significant load events appear to be most important. Factors that apparently are less important include the correlation between component strengths in a system and variability in the initial strength. The hazard function, or conditional failure rate, is clearly nonlinear for the degradation mechanisms studied, and increases sharply after a prolonged service life for the degradation functions assumed in the analysis. Thus the assumption of a linear failure rate would lead to an overly optimistic appraisal of reliability.

Prediction of Aging Degradation of Cast Stainless Steel Components in LWR Systems*

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Cast stainless steels used in light water reactor (LWR) systems for primary pressure-boundary components such as valve bodies, pump casings, and primary coolant piping are susceptible to thermal embrittlement at reactor operating temperatures, i.e., 280–350°C. Thermal aging of cast stainless steels at these temperatures increases hardness and tensile strength and decreases ductility, impact strength, and fracture toughness of the material. Investigations at Argonne National Laboratory and elsewhere have shown that thermal embrittlement of cast stainless steels can occur during the reactor design lifetime of 40 y. Current assessment of thermal embrittlement of cast stainless steels involve simulation of end-of-design-life reactor conditions by accelerated aging at higher temperatures, viz., 400°C. Estimates of mechanical-property degradation of cast stainless steel components are based on an Arrhenius extrapolation of high-temperature data to reactor operating conditions.

A procedure and correlations have been developed for predicting the mechanical properties of cast stainless steel components during thermal aging in LWRs. The analysis focused on developing correlations for fracture properties in terms of material information in certified material test records and on ensuring that the estimated mechanical properties are adequately conservative for cast stainless steels defined by ASTM Specification A 351. Fracture toughness of a specific cast stainless steel is estimated from the extent and kinetics of thermal embrittlement. The extent of embrittlement is characterized by the room-temperature Charpy-impact energy. A correlation for the extent of embrittlement at "saturation," i.e., the minimum impact energy that can be achieved for the material after long-term aging, is given in terms of chemical composition. Extent of thermal embrittlement as a function of time and temperature of reactor service is then estimated from the saturation impact energy and correlations describing the kinetics of embrittlement, which are given in terms of chemical composition and the aging behavior at 400°C. The fracture toughness J-R curve is obtained from the correlation between fracture toughness parameters and room-temperature impact energy. A common lower-bound J-R curve for cast stainless steels of unknown chemical composition is also defined for a given material grade and temperature. Increase in tensile flow stress is determined from the kinetics of thermal embrittlement; the initial tensile properties must be known in order to determine the flow stress of the aged material. Fracture toughness J_{IC} and tearing modulus are obtained from the estimated J-R curve and tensile flow stress. The correlations do not consider the effect of metallurgical differences that may arise from differences in production heat treatment or casting process and, therefore, may be conservative for some steels.

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Effect of Aging on the Predicted Maximum Load-Carrying Capacity of Circumferentially Cracked Cast Stainless Steel Pipe

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ABSTRACT

Cast stainless steel used in LWR primary system components such as valve bodies, pump castings, and piping is susceptible to thermal embrittlement at reactor operating temperatures, 280-320 C (536-608 F). This process of thermal aging causes an increase in the hardness and ultimate tensile strength of the steel and at the same time a decrease in toughness. Work at Argonne National Laboratories (ANL) has shown that such thermal embrittlement due to changes in the microstructure can occur even during the reactor lifetime of 40 years. An Arrhenius type kinetic model has been used by Argonne to develop a correlation for predicting the mechanical properties (J-R curve and strength properties) of aged cast stainless steel. The effect of this thermal degradation on the load-carrying capacity of circumferentially cracked piping is the subject of this work.

In this study, both lower bound and average values of the J-R curve and the tensile properties for CF8M and CF8A cast stainless steel aged at 300 C for 20, 40, and 60 years were used to predict the maximum load-carrying capacity of cracked pipe. Both through-wall-cracked (TWC) and surface-cracked (SC) pipe have been considered. The effect of aging, that is, reduced toughness and increased strength, for different pipe diameters and crack lengths has been investigated. Three analyses methods have been used to estimate the maximum load-carrying capacity of pipes: (1) a J-estimation scheme for TWC pipes developed by Paris that uses the J-R curve as well as the yield strength and flow stress of the material, (2) a Plastic-Zone-Screening Criteria (DPZP) developed at Battelle which is applicable to both TWC and SC pipe which uses J_i and the flow stress, and (3) the R-6 Option 1 method developed by CEGB which uses the J-R curve, yield strength, and flow stress and is applicable for both TWC and SC pipe.

Evaluation of Aging Degradation of Structural Components*

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Structural materials exposed to reactor environments undergo microstructural changes that can influence mechanical and corrosion properties and thus compromise the integrity of structure components. Current assessments of end-of-design-life mechanical properties of reactor components are mostly based on simulations by accelerated test conditions. Aged materials from reactor components offer an excellent opportunity to validate the laboratory studies. Mechanical properties of neutron shield tank (NST) material (A212 Grade B steel) from the Shippingport reactor, as well as of cast stainless steel (CF-8 steel) components from Shippingport and KRB reactors, have been characterized. The results are compared with estimates based on laboratory studies.

Increases in Charpy transition temperature (CTT), yield stress, and hardness of the NST material in the low-temperature low-flux environment are consistent with the test and Army reactor data for irradiations at <232°C (<450°F). The shift in CTT is not as severe as that observed in High Flux Isotope Reactor (HFIR) surveillance samples; however, it shows very good agreement with the results for HFIR A212-B steel irradiated in the Oak Ridge Research Reactor (ORR). The results suggest that radiation damage in Shippingport NST and HFIR surveillance samples may be different because of the neutron spectra and/or Cu and Ni content of the two materials. High-flux low-temperature irradiation experiments are in progress on materials from the Shippingport NST and the HFIR vessel to evaluate the possible effects of compositional and metallurgical differences between the two materials.

Mechanical-property data from cast stainless steel components indicate relatively modest decreases in fracture toughness and Charpy-impact properties and an increase in tensile strength. The procedure and correlations based on laboratory-aging studies for estimating mechanical properties of cast stainless steel components in LWR systems, predict accurate or slightly conservative values for Charpy-impact energy, tensile flow stress, fracture toughness J-R curve, and J_{IC} of the Shippingport and KRB materials. The kinetics of thermal embrittlement and degree of embrittlement at saturation, i.e., the minimum impact energy that would be achieved after long-term aging, were established from materials that were aged further in the laboratory at temperatures between 320 and 400°C. The results were consistent with the estimates. The correlations successfully predict the mechanical properties of the Ringhals 2 reactor hot- and crossover-leg elbows (CF-8M steel) after service of ≈15 y.

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