Assessment of the Quality of Selected NRC Research Projects by the Advisory Committee on Reactor Safeguards - FY 2007

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ABOUT THE ACRS

The Advisory Committee on Reactor Safeguards (ACRS) was established as a statutory Committee of the Atomic Energy Commission (AEC) by a 1957 amendment to the *Atomic Energy Act* of 1954. The functions of the Committee are described in Sections 29 and 182b of the Act. The *Energy Reorganization Act* of 1974 transferred the AEC's licensing functions to the U.S. Nuclear Regulatory Commission (NRC), and the Committee has continued serving the same advisory role to the NRC.

The ACRS provides independent reviews of, and advice on, the safety of proposed or existing NRC-licensed reactor facilities and the adequacy of proposed safety standards. The ACRS reviews power reactor and fuel cycle facility license applications for which the NRC is responsible, as well as the safety-significant NRC regulations and guidance related to these facilities. On its own initiative, the ACRS may review certain generic matters or safety-significant nuclear facility items. The Committee also advises the Commission on safety-significant policy issues, and performs other duties as the Commission may request. Upon request from the U.S. Department of Energy (DOE), the ACRS provides advice on U.S. Naval reactor designs and hazards associated with the DOE's nuclear activities and facilities. In addition, upon request, the ACRS provides technical advice to the Defense Nuclear Facilities Safety Board.

ACRS operations are governed by the *Federal Advisory Committee Act* (FACA), which is implemented through NRC regulations at Title 10, Part 7, of the *Code of Federal Regulations* (10 CFR Part 7). ACRS operational practices encourage the public, industry, State and local governments, and other stakeholders to express their views on regulatory matters.

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ABSTRACT

In this report, the Advisory Committee on Reactor Safeguards (ACRS) presents the results of its assessment of the quality of selected research projects sponsored by the Office of Nuclear Regulatory Research (RES) of the NRC. An analytic/deliberative methodology was adopted by the Committee to guide its review of research projects. The methods of multi-attribute utility theory were utilized to structure the objectives of the review and develop numerical scales for rating the project with respect to each objective. The results of the evaluations of the quality of the three research projects are summarized as follows:

- Cable Response to Live Fire (CAROLFIRE) Testing Program
  - This project was found to be satisfactory. The results meet the research objectives.

- Fatigue Crack Flaw Tolerance in Nuclear Plant Piping
  - This project was found to be more than satisfactory. The results meet the research objectives.

- Technical Review of Online Monitoring Techniques for Performance Assessment
  - This project was found to be satisfactory. The results meet the research objectives.

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ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ASME	American Society of Mechanical Engineers
AEC	Atomic Energy Commission
BWR	Boiling Water Reactor
CAROLFIRE	Cable Response to Live Fire
CFR	Code of Federal Regulations
DOE	Department of Energy
EPRI	Electric Power Research Institute
ESFAS	Engineered Safety Features Actuation System
FACA	Federal Advisory Committee Act
FY	Fiscal Year
GPRA	Government Performance and Results Act
MAUT	Multi-Attribute Utility Theory
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OLM	On-Line Monitoring
PDI	Performance Demonstration Initiative
PNNL	Pacific Northwest National Laboratory
POD	Probability of Detection
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RES	Office of Nuclear Regulatory Research
RIS	Regulatory Issue summary
SNL	Sandia National laboratories
UMD	University of Maryland

1 INTRODUCTION

The Nuclear Regulatory Commission (NRC) maintains a safety research program to ensure that the agency's regulations have sound technical bases. The research effort is needed to support regulatory activities and agency initiatives while maintaining an infrastructure of expertise, facilities, analytical tools, and data to support regulatory decisions.

The Office of Nuclear Regulatory Research (RES) is required to have an independent evaluation of the effectiveness (quality) and utility of its research programs. This evaluation is required by the NRC Strategic Plan that was developed as mandated by the Government Performance and Results Act (GPRA). Since fiscal year 2004, the Advisory Committee on Reactor Safeguards (ACRS) has been assisting RES by performing independent assessments of the quality of selected research projects [1-3]. The Committee has established the following process for conducting the review of the quality of research projects:

- RES submits to the ACRS a list of candidate research projects for review because they have reached sufficient maturity that meaningful technical review can be conducted

- The ACRS selects no more than four projects for detailed review during the fiscal year.

- A panel of three to four ACRS members is established to assess the quality of each research project.

- The panel follows the guidance developed by the ACRS full Committee in conducting the technical review. This guidance is discussed further below.

- Each panel assesses the quality of the assigned research project and presents an oral and a written report to the ACRS full Committee for review. This review is to ensure uniformity in the evaluations by the various panels.

- The Committee submits an annual summary report to the RES Director.

An analytic/deliberative decisionmaking framework was adopted for evaluating the quality of NRC research projects. The definition of quality research adopted by the Committee includes two major characteristics:

- Results meet the objectives
- The results and methods are adequately documented

Within the first characteristic, ACRS considered the following general attributes in evaluating the NRC research projects:

- Soundness of technical approach and results
  - Has execution of the work used available expertise in appropriate disciplines?

- Justification of major assumptions
  - Have assumptions key to the technical approach and the results been tested or otherwise justified?

- Treatment of uncertainties/sensitivities
  - Have significant uncertainties been characterized?
  - Have important sensitivities been identified?

Within the general category of documentation, the projects were evaluated in terms of following measures:

- Clarity of presentation
- Identification of major assumptions

In this report, the ACRS presents the results of its assessment of the quality of the research projects associated with:

- Cable Response to Live Fire (CAROLFIRE) Testing Program
- Fatigue Crack Flaw Tolerance in Nuclear Plant Piping
- Technical Review of Online Monitoring Techniques for Performance Assessment

These three projects were selected from a list of candidate projects suggested by RES.

The methodology for developing the quantitative metrics (numerical grades) for evaluating the quality of NRC research projects is presented in Section 2 of this report. The results of assessment and ratings for the selected projects are discussed in Section 3.

2 METHODOLOGY FOR EVALUATING THE QUALITY OF RESEARCH PROJECTS

To guide its review of research projects, the ACRS has adopted an analytic/deliberative methodology [4-5]. The analytical part utilizes methods of multi-attribute utility theory (MAUT) [6-7] to structure the objectives of the review and develop numerical scales for rating the project with respect to each objective. The objectives were developed in a hierarchical manner (in the form of a "value tree"), and weights reflecting their relative importance were developed. The value tree and the relative weights developed by the full Committee are shown in Figure 1.

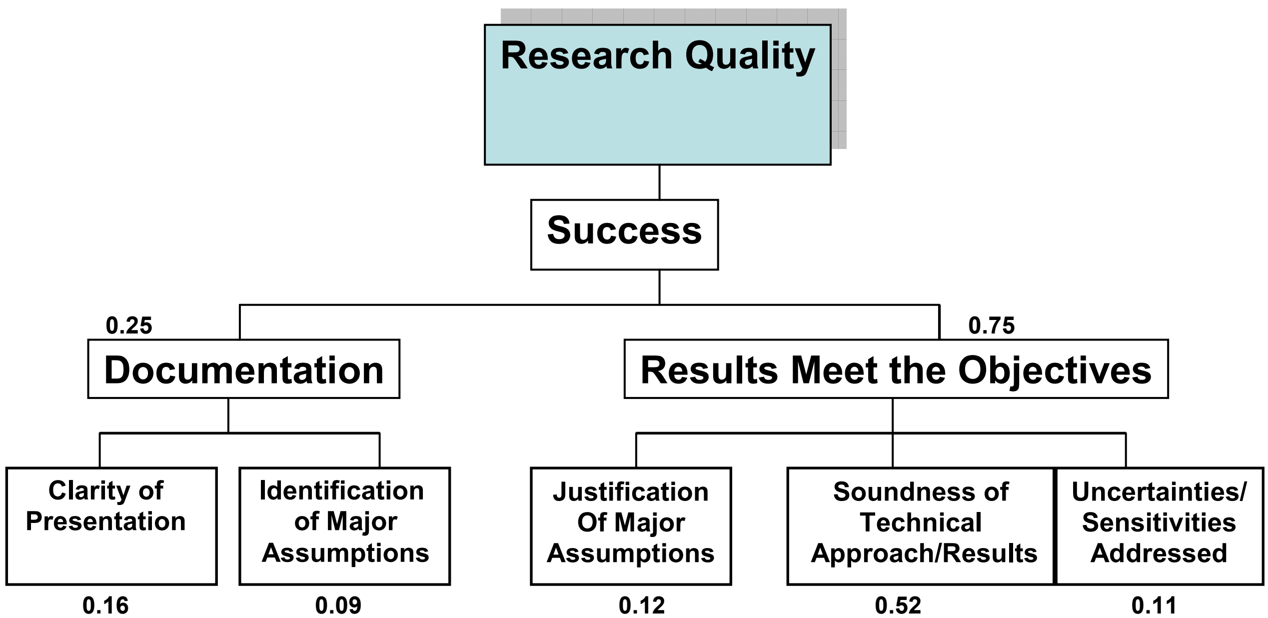

Figure 1 The value tree used for evaluating the quality of research projects

The quality of projects is evaluated in terms of the degree to which the results meet the objectives of the research and of the adequacy of the documentation of the research. It is the consensus of the ACRS that meeting the objectives of the research should have a weight of 0.75 in the overall evaluation of the research project. Adequacy of the documentation was assigned a weight of 0.25. Within these two broad categories, research projects were evaluated in terms of subsidiary "performance measures":

- justification of major assumptions (weight: 0.12)
- soundness of the technical approach and reliability of results (weight: 0.52)
- treatment of uncertainties and characterization of sensitivities (weight: 0.11)

Documentation of the research was evaluated in terms of the following performance measures:

- clarity of presentation (weight: 0.16)
- identification of major assumptions (weight: 0.09)

To evaluate how well the research project performed with respect to each performance measure, constructed scales were developed as shown in Table 1. The starting point is a rating of 5, Satisfactory (professional work that satisfies the research objectives). Often in evaluations of this nature, a grade that is less than excellent is interpreted as pejorative. In this ACRS evaluation, a grade of 5 should be interpreted literally as satisfactory. Although innovation and excellent work are to be encouraged, the ACRS realizes that time and cost place constraints on innovation. Furthermore, research projects are constrained by the work scope that has been agreed upon. The score was, then, increased or decreased according to the attributes shown in the table. The overall score of the project was produced by multiplying each score by the corresponding weight of the performance measure and adding all the weighted scores.

The value tree, weights, and constructed scales were the result of extensive deliberations of the whole ACRS. As discussed in Section 1, a panel of three ACRS members was formed to review each selected research project. Each member of the review panel independently evaluated the project in terms of the performance measures shown in the value tree. The panel deliberated the assigned scores and developed a consensus score, which was not necessarily the arithmetic average of individual scores. The panel's consensus score was discussed by the full Committee and adjusted in response to ACRS members' comments. The final consensus scores were multiplied by the appropriate weights, the weighted scores of all the categories were summed, and an overall score for the project was produced. A set of comments justifying the ratings was also produced.

Table 1. Constructed Scales for the Performance Measures

SCORE	RANKING	INTERPRETATION
10	Outstanding	Creative and uniformly excellent
8	Excellent	Important elements of innovation or insight
5	Satisfactory	Professional work that satisfies research objectives
3	Marginal	Some deficiencies identified; marginally satisfies research objectives
0	Unacceptable	Results do not satisfy the objectives or are not reliable

3. RESULTS OF QUALITY ASSESSMENT

3.1 CABLE RESPONSE TO LIVE FIRE (CAROLFIRE) TESTING PROGRAM

The ability to determine risk due to fire damage to electrical power, control and instrumentation cables in nuclear power plants has been a concern for many years. It has been assumed that any system that was dependent on electrical cables that passed through a compartment exposed to fire would be unusable for its intended safety function. Also, the effects of hot-shorts (including spurious actuation) should be taken in account (10CFR50 – Appendix R, III.G.2). This second issue has been a source of uncertainty for the NRC and licensees. To reduce the uncertainties, a series of cable fire damage tests were jointly conducted by NRC, NEI and EPRI. Data from these tests, as well as previous test data, resulted in the publication of an NRC Regulatory Issue Summary (RIS) 2004-003 Rev.1 entitled, "Risk-Informed Approach for Post-Fire Safe Shutdown Circuit Inspections" [8]. This RIS presented guidance for NRC inspectors decide which causes of hot-shorts to consider, which hot-short/circuit failures are credible, and related issues. The RIS specifically described four categories of concerns that should be considered during inspections (called 'Bin-1 items'). Other issues whose importance still needed to be determined were also described in the RIS and were referred to as 'Bin-2 items'. These included inter-cable shorting for thermo-set cables, inter-cable shorting between thermo-plastic and thermo-set cables, configurations involving three or more cables, multiple spurious actuations in control circuits, and prolonged fire-induced hot-shorts that could impair the ability of a plant to achieve hot shutdown. NRC identified the need for empirical testing to provide additional data in certain cable configurations to support further development of guidance and modeling capabilities.

The CAROLFIRE Testing Program was undertaken in response to the needs identified. CAROLFIRE consisted of 78 small-scale cable fire tests and 18 intermediate-scale open-burn fire tests conducted in 2006 and early 2007. These empirical tests were designed to complete and complement previous testing with the following objectives:

1. Provide empirical data to elucidate the regulatory issues involved in the Bin-2 items noted in the RIS.
2. Provide test data for improvement of models for responses of cables exposed to fire.

The test results are documented in two volumes of a draft report (NUREG/CR-6931) prepared by Sandia National Laboratories (SNL). The first volume, "CAROLFIRE Test Report Volume 1: General Test Descriptions and the Analysis of Circuit Response Data" [9], contains the small-scale and intermediate-scale test results for electric circuit failures (i.e., "hot shorts" data). The second volume, "CAROLFIRE Test Report Volume 2: Cable Fire Response Data for Fire Model Improvement" [10], contains thermal test data from these same experiments aimed at improvements in cable fire response models

The proposed research project was peer reviewed before initiation and monitored during its evolution by the participants (SNL, NRC-RES, NRC-NRR, NIST, and the University of Maryland). Revisions to a preliminary draft of the experimental project plan were reported in summer 2006 and documented in Reference 11. The investigators at SNL then modified the test plan to respond to the peer review in fall 2006. Testing and documentation of the results were completed in March 2007. Use of the results for regulatory guidance, model development and validation is still under way.

General Observations

This research project, which was ambitious and encompassed both elucidation of ‘Bin-2’ regulatory issues and provision of data to improve cable fire response models, met much of its original test objectives. It has provided an expanded empirical database, including simultaneous measurement of hot-short failure phenomena, as well as thermal response data for electrical cable fires (thermo-set and thermo-plastic) under a range of conditions. It has explored some phenomena of interest under a reasonably wide range of conditions and provided significant data for both ‘Bin 2’ items and model development. The project was empirical in its test design and, as mentioned earlier, had an ambitious set of objectives that were difficult to meet within the allotted time and budget. Nonetheless, satisfactory progress was made. Furthermore, based on an initial review of the project test plan by a peer review committee, the principal investigators modified the test plan appropriately and completed the experiments over the modified range of conditions within the allotted time period. The qualitative understanding derived from the empirical test data provided improved understanding and characterization of ‘Bin-2 Items’. In addition, the tests provided a wider database for cable fire response modeling by other researchers in the field.

The consensus scores for this project are shown in Table 2. The score for the overall assessment of the work was found to be 5.23, which should be interpreted as “a professional job that satisfies the research objectives.”

Table 2. Summary Results of ACRS Assessment of the Quality of Cable Response to Live Fire (CAROLFIRE) Project

Performance Measures	**Consensus Scores**	**Weights**	**Weighted Scores**
Clarity of presentation	6.0	0.16	0.96
Identification of major assumptions	5.2	0.09	0.47
Justification of major assumptions	5.0	0.12	0.60
Soundness of technical approach/results	5.2	0.52	2.70
Treatment of uncertainties/sensitivities	4.5	0.11	0.50
		Overall Score	**5.23**

Comments and conclusions within the evaluation categories are:

Documentation

- Clarity of Presentation **(Consensus Score – 6.0)**

  The score in this category was raised somewhat above satisfactory because of the above-average clarity of presentation. The experimental test setup, procedures and result descriptions are clear and sufficiently detailed. The text generally presents a coherent narrative of an ambitious undertaking over a short span of time (less than one year). Some specific examples are:

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  - Good discussion of past work and background information, though somewhat lacking in discussion of the physicochemical phenomena impacting cable response to fires
  - Discussion of role of other collaborators (e.g. NIST and UMD modelers)
  - Good qualitative description of the experiments
  - Reasonable description of the empiricisms inherent in the tests (both their positive and negative aspects), though somewhat lacking in the discussion of sources of uncertainty

  A minor point is that the table of cable physical properties (Table 2.2 of draft Volume 1 of NUREG/CR-6931) is really a table of geometric data and should be augmented to include thermo-physical properties (e.g., thermal conductivity, density, specific heat, and emissivity) of jacket and insulation materials needed to interpret results and for PRA use. Adding the physical property data and the derived parameter values will help to support proper use of the data during fire PRA applications. Additionally, the report should be edited to correct typographical errors and eliminate inconsistencies between the two volumes.

- Identification of Major Assumptions **(Consensus Score – 5.2)**

  This was an empirical test program and, as such, the major assumptions were appropriately discussed in draft NUREG/CR-6931, Volume 1, in parallel with the description of the small-scale and the intermediate-scale experiments. Some significant assumptions related to the heat flux and its spatial profile for the cable-fire simulations (radiant heating for Penlight and discrete fires for the intermediate tests) were made. Other key assumptions were the simplifications (e.g., with regard to cabling materials, cable bundle geometry) to the test matrix that the researchers had to make, given the practicalities related to design, fabrication, and performance of the experiments.

Results Meet the Objectives

- Justification of Major Assumptions **(Consensus Score – 5.0)**

  The major assumptions that investigators discussed and justified were the experimental simplifications that were required and the scores in this category were considered to be average. This discussion was satisfactory, but no detailed justification of assumptions for the test matrix or testing, indicating how they were reasonable for the cable fire scenarios of interest, was provided.

- Soundness of Technical Approach and Results **(Consensus Score – 5. 2)**

  The design and conduct of the small-scale and intermediate-scale tests were somewhat better than satisfactory given the application of an innovative technique for simultaneous resistance and cable temperature measurement during the testing. This required that the two separate measurements be made in two separate cables placed in symmetric locations where they would be expected to be exposed to similar external heat flux conditions.

  Additional strengths in the approach were that:

  - The test matrix initially planned was allowed to evolve in light of the results obtained.
  - The test matrix was complementary to the NIST/EPRI (prior) testing, and the complementary nature of these tests was well described.

  On the other hand, no attempts were made to directly measure or calculate (by detailed simulations) the heat flux to which the cables were exposed in both the small and intermediate scale experiments.

- Treatment of Uncertainties/Sensitivities **(Consensus Score – 4.5)**

  The test results had uncertainties, that could have been addressed better, which resulted in this “lower than average” rating. For example, an important finding of the Penlight series was that the copper-to-plastic cable content has a first-order influence on thermal response and short-circuiting failure. The written discussion uses vague terminology such as “copper is more massive” or “larger thermal mass”, that adds uncertainty to the interpretation of the results but does not quantify it. This could be improved by listing the conductor mass per unit cable length (in addition to total mass per cable length) and the actual heat capacities of conductor and jacket/insulation in terms of energy per unit length of cable per degree (calculated from the product of density, specific heat, and area), and at least use a lumped parameter thermal model to interpret the results somewhat more quantitatively.

  Conclusions regarding the Penlight single cable data are also sparse. Modelers will attempt to use such data to quantify the short-circuit failure threshold with respect to various criteria such as a failure temperature. A test matrix focused on determining the appropriate criteria, containing significantly larger database to reduce uncertainties would have been desirable. Data described in Volume 2 should also be summarized to indicate the observed temperature range of failure for the cable insulation/jacket combinations used. In some cases the temperature at which failure was observed is listed in the text, in other cases it is not. The physicochemical aspects of the degradation of insulating materials suggest that kinetics of the process could be important and a simple temperature criterion may not be sufficient to cover failure in all scenarios, though the rapid increase in degradation rate with increasing temperature implied by Arrhenius kinetics could lead to an apparent temperature criterion in many scenarios with high heating rates. In any case, if the authors do not want to draw any conclusions regarding failure criteria, that opinion should be clearly stated.

  The discussion of combustion efficiency and the heat release rate for the intermediate scale experiments also leaves the uncertainties in these experiments vaguely defined.

Furthermore, there were some results which suggested an oscillatory combustion rate so the combustion may not have been diffusion controlled. No heat and species balances were attempted, which could have been helpful in quantifying the uncertainties in the conditions to which the cables were exposed. The effects of the pulsating nature of the flame were not explained or quantified.

3.2 FATIGUE CRACK FLAW TOLERANCE IN NUCLEAR PLANT PIPING

American Society of Mechanical Engineers (ASME) first incorporated Appendix L, "Operating Plant Fatigue Assessment," in the 1996 Addenda of Section XI Boiler and Pressure Vessel Code. ASME developed Appendix L to provide guidance for evaluating plant operating events and determining the acceptability of components for which fatigue usage limits may have been exceeded or for which fatigue stresses may be a concern. A key part of this Appendix was a damage tolerance analysis, which postulated a flaw at the fatigue location of interest and then performed a fatigue crack growth analysis to determine inspection intervals that can detect fatigue cracks before they exceed Code-allowable sizes. Experience in applying the initial version of Appendix L raised the following concerns:

- The crack growth model needed to account for the initiation and coalescence of multiple fatigue cracks observed in the field.
- Flaw detection and sizing performance needed to reflect newly available performance demonstration data.
- Application of Appendix L was found to be impractical for plant license renewal.

The Office of Nuclear Regulatory Research sponsored a research project to address the above concerns and provide an independent technical assessment. The research was conducted at the Pacific Northwest National Laboratory (PNNL). The results of this research project that the ACRS reviewed were documented in a NUREG/CR report entitled, "Fatigue Crack Flaw Tolerance in Nuclear Plant Piping — A Basis for Improvements to ASME Code Section XI Appendix L."

General Observations

The reported research provided a technical basis for improvements in the flaw tolerance approach of Section XI Appendix L of the ASME Boiler and Pressure Vessel Code and the assessment of the effectiveness of nondestructive examination (NDE) strategies in preventing leakage due to the growth of fatigue cracks. ASME has revised Appendix L, based substantially on the technical work described in the NUREG/CR report, and ASME has approved the revised Appendix for use.

The report covers a very broad scope of work including:

- The background and need for improvement of the ASME Section XI Appendix L code governing the treatment of fatigue cracks
- A review of nuclear industry thermal fatigue cracking experience and its implications for the definition of cracks for flaw tolerance analyses
- Quantitative evaluation of in fatigue crack detection capability
- Use of probabilistic fracture mechanics codes to evaluate the effectiveness of NDE capability and strategies on leakage probability
- Effects of aspect ratio on growth of cracks
- Influence of initiation and linking of multiple fatigue cracks on flaw tolerance

The consensus scores for this project are shown in Table 3. The score for the overall assessment of the work was found to be 5.7, which is more than satisfactory. A satisfactory score should be interpreted as “a professional job that satisfies the research objectives.”

Table 3. Summary Results of ACRS Assessment of the Quality of the Project on Fatigue Crack Flaw Tolerance in Nuclear Plant Piping

Performance Measures	**Consensus Scores**	**Weights**	**Weighted Scores**
Clarity of presentation	6.0	0.16	1.0
Identification of major assumptions	5.5	0.09	0.5
Justification of major assumptions	5.5	0.12	0.7
Soundness of technical approach/results	5.8	0.52	3.0
Treatment of uncertainties/sensitivities	4.7	0.11	0.5
		Overall Score	**5.7**

Comments and conclusions within the evaluation categories are:

Documentation

- Clarity of Presentation **(Consensus Score – 6.0)**

  The authors have done an excellent job in presenting the results of several years of research in a clear, understandable, and readable document.

- Identification of Major Assumptions **(Consensus Score – 5.5)**

  The authors clearly identified and discussed the major assumptions involved in the analyses presented in the report.

Results Meet the Objectives

- Justification of Major Assumptions **(Consensus Score – 5.5)**

  The authors provided sufficient data and analyses to justify the major assumptions in this work with the exception of the development of the probability of detection (POD) curves that are used to express the effectiveness of NDE inspections. The POD curves are developed from data from the nuclear industry Performance Demonstration Initiative (PDI). The report provides a clear statement of the limitations of the data from the PDI and the processes used to develop POD curves from the PDI and the scope of the PDI database (numbers of cracks, diameters of pipes, and types of cracks). However, in order to protect the integrity of the data, the only information given on the distribution of crack sizes within the database is that the vast majority of cracks are greater than 10% in depth. Thus, it is impossible to make an independent assessment of the adequacy of the database to support the development of the POD curves. Ultimately one has to rely on the expert judgment of the authors to accept these POD curves as reasonably reflective of current NDE capabilities.

  The POD curves are logistic regressions on the detection results as a function of crack depth. No measures of goodness of fit are given. Confidence limits (95th percentile) are given without a clear statement that if the POD is actually given by a logistic fit, then this represents the uncertainty in the fit, but the limits do not reflect model uncertainty, i.e., whether the POD is actually logistic. The text notes "In general, the PDI data were not very good at determining the "breakdown" flaw sizes for the inspection procedures; that is, the flaw sizes for which the POD begins to rapidly decrease," and "The initial NDE performance demonstration evaluations and POD curves developed in this report represent the first step in trying to answer a very basic question: "How reliable are nuclear system piping inspections in the field?", but Table 4.9 of the report, which gives POD results from 0 to 100% through-wall to 5 decimal places, and the figures which extrapolate the curves to zero depth give a misleading picture of the POD curve. However, for use in the analysis the authors do truncate the statistical POD curves at a depth more consistent with the stated limitations of the PDI database.

- Soundness of Technical Approach and Results **(Consensus Score – 5. 8)**

  There are two major sets of results in the report: an assessment of the impact of NDE effectiveness and NDE strategy (frequency of inspection) on the likelihood of cracking, and a technical basis for the selection of cracks for a flaw tolerance analysis to assess the potential impacts of cracks that are likely to be missed by in-service NDE.

  The technical approach in the report is sound, making good use of fracture mechanics analysis, empirical data on probability of detection of cracks, and research into the nature of fatigue initiation and crack growth. The authors have developed an insightful way to characterize the nature of cracks in terms of a single equivalent crack with an aspect ratio that is a function of pipe size and the ratio of the membrane stress to the gradient stress.

- Treatment of Uncertainties/Sensitivities **(Consensus Score – 4.7)**

The report provides extensive sensitivity studies that provide an adequate justification for the use of the results to define a more realistic approach to flaw tolerance analyses for Appendix L of Section XI. However, although the report recognizes the weaknesses of the PDI database in defining the flaw sizes for which the POD begins to rapidly decrease, the choice made in the report for this “breakdown size” seems somewhat arbitrary. Lacking data, it would have been useful to use a more formal expert judgment process with a somewhat larger group of experts to ensure that the choice does indeed reflect the current consensus.

3.3 TECHNICAL REVIEW OF ONLINE MONITORING TECHNIQUES FOR PERFORMANCE ASSESSMENT

Nuclear power plant operators are required to periodically perform surveillances and channel calibrations on safety related instrumentation that feeds into the reactor protection system and other safety related systems such as the engineered safety features actuation system (ESFAS). These surveillances and calibrations are time consuming and may involve some degree of radiation exposure to the workers. For redundant parameters, the on-line monitoring (OLM) systems being proposed will periodically compare the output of the redundant instruments (typically 3 or 4) and then compare their differences using a statistical analysis to determine if one or more of the instruments deviate significantly from the others. For non-redundant instrumentation the OLM systems utilize other statistical methods discussed later. The results would then be used to determine when a more detailed surveillance or calibration is needed.

In 2004, the NRC entered into a cooperative agreement with Ohio State University to investigate current research and development efforts for OLM of important nuclear power plant instrumentations. Principal investigators were Ohio State University, University of Tennessee, and the University of Virginia. The results of the cooperative agreement effort are being documented in NUREG/CR-6895.

The proposed research will investigate current OLM research and development efforts. It will investigate what information is needed to determine the adequacy of these systems in predicting system calibration and system failures. The research will also investigate the possibility of developing tools for the analysis and review of OLM methods. It will include a review of current and proposed methods, both currently available commercial and proposed in the literature, and provide requirements necessary for the justification of the methods used. A theoretical basis for the analysis of uncertainty including assumptions and their significance will be developed.

Volume 1 of NUREG/CR-6895, “State of the Art” [13] offers a general overview of current sensor calibration monitoring technologies and their uncertainty analysis, a review of the supporting information necessary for assessing these techniques, and a cross reference between the literature and the requirements previously outlined by the NRC. This is the only part of the research project completed when the ACRS started its review and is the only part of the research project assessed in this report. Volume 2 “Theoretical Issues” will present an in-depth theoretical study and independent review of the most widely used OLM techniques. It will include a presentation of the theory and further explanation of the assumptions inherent in empirical models and uncertainty quantification techniques. Volume 3 “Limiting Case Studies” will present the results of applying OLM models to a wide variety of plant data. Specifically, it will summarize six case studies investigating the effects of model development and assumptions on model performance and will offer recommendations for identifying and handling these limiting cases.

The consensus scores for this project are shown in Table 4. The score for the overall assessment of the work was found to be 5.42, which should be interpreted as “a professional job that satisfies the research objectives.” Note that the framework adopted by the ACRS for evaluating the quality of NRC research projects requires consensus scores for five performance measures. As discussed in Section 2, the five performance measures represent two categories, “Documentation” and “Results meet the objectives”. Volume 1 of this research project focused on review and documentation of the current state-of-the-art for OLM techniques. This type of research project is difficult to evaluate using the five performance measures under the two primary categories. Therefore, consensus scores were given for the two primary categories and the overall score was the weighted average of these two scores.

Table 4 Summary Results of the ACRS Assessment of the Quality of the Project on Technical Review of On-Line Monitoring Techniques for Performance Assessment

Performance Measures	Consensus Scores	Weights	Weighted Scores
Documentation	4.67	0.25	1.2
Results meet the objectives	5.67	0.75	4.2
		Overall Score	**5.4**

Comments and conclusions within the evaluation categories are:

- **Documentation (Consensus score = 4.67)**

  The report identifies and discusses methodologies that have been proposed for use in domestic plants as well as those currently used in foreign plants. It provides references to additional documentation and evaluations associated with OLM. Detailed information is also included on additional reviews of uncertainty analyses and techniques used by developers of the various OLM methodologies. It provides detailed discussions of the various algorithms and methodologies used in the available techniques.

  The report provides a good discussion and evaluation of uncertainties associated with the proposed methodologies for OLM techniques. Although this project focused primarily on the methods used to evaluate the plant data once it has been entered into the program, it would have been beneficial to have included a better discussion on the methods to collect the data from the plant and transfer it to the program. This collection and transfer process can add uncertainty to the overall process and the discussion could have provided the reader with a more complete overview of the entire process.

  Although the report provides the appropriate references and technical information, it would have been beneficial to have included some of the referenced information in the text of the report. For example, the safety evaluation discussed in the appendix contained information that would have been beneficial to the reader if included in the main text of the report.

- **Results meet the objectives (Consensus score = 5.67)**

  The objectives for this part of the project were to review the current state-of-the-art in OLM techniques and to provide additional reviews of the uncertainty analyses provided by developers of the current OLM methodologies. These objectives were accomplished with the collection and review of the techniques currently proposed or used. The independent uncertainty analyses were thorough and provide additional technical bases for future reviews and evaluations. This part of the overall research project provides the baseline information needed to complete the remaining phases which will be documented in Volumes 2 and 3 of NUREG/CR-6895.

  This phase of the project reviewed two of the redundant modeling techniques and three of the non-redundant techniques commonly used in the Nuclear Power Industry for OLM. The redundant techniques reviewed were a simple averaging algorithm (Instrumentation and Calibration Monitoring Program) and an advanced factor analysis method (Independent Component Analysis). The non-redundant methods were a kernel-based method (Multivariate State Estimation Technique), neural network-based methods (Process Evaluation and Analysis by Neural Operators and the University of Tennessee Auto Associative Neural Network), and a transformation method (Non-Linear Partial Least Squares).

  The detailed review and evaluation of these algorithms and methodologies as they are applied to OLM techniques should provide a good foundation for future regulatory review and evaluation of applications for license amendments.

4. REFERENCES

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