UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

ACRSR-2160

November 4, 2005

Dr. Carl J. Paperiello
Director
Office of Nuclear Regulatory Research
Washington, D.C. 20555-0001

SUBJECT: ACRS ASSESSMENT OF THE QUALITY OF SELECTED NRC RESEARCH PROJECTS - FY 2005

Dear Dr. Paperiello:

Enclosed is our report on the quality assessment of the following research projects:

- Station Blackout Risk Evaluation for Nuclear Power Plants performed as a part of Standardized Plant Analysis Risk (SPAR) Models Development Program
  - This project was found to be more than satisfactory. The results meet the research objectives.
- Steam Generator Tube Integrity Program at the Argonne National Laboratory
  - This project was found to be satisfactory. The results meet the research objectives.
- Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag Experiments at the Penn State University
  - This project marginally satisfied the research objectives. The Committee identified significant deficiencies.

These projects were selected from a list of candidate projects suggested by the Office of Nuclear Regulatory Research.

A fourth research project on reactor containment performance being conducted at Sandia National Laboratories will be evaluated later, once a particularly pivotal report on the research becomes available.

We anticipate receiving your list of candidate projects for review during the next 12 months.

Sincerely,

/RA/

William J. Shack
Acting Chairman

Enclosure: As stated

ACRS Assessment of the Quality of Selected NRC Research Projects

October 2005

U.S. Nuclear Regulatory Commission
Advisory Committee on Reactor Safeguards
Washington, DC 20555-0001

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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ACKNOWLEDGMENT

The Committee would like to acknowledge the contributions of Dr. Hossein Nourbakhsh and Mr. Sam Duraiswamy of the ACRS Staff to the development of this assessment.

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:

- Station Blackout Risk Evaluation for Nuclear Power Plants performed as a part of Standardized Plant Analysis Risk (SPAR) Models Development Program

  - This project was found to be more than satisfactory . The results meet the research objectives.

- Steam Generator Tube Integrity Program at the Argonne National Laboratory

  - This project was found to be satisfactory. The results meet the research objectives.

- Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag Experiments at the Penn State University

  - This project marginally satisfied the research objectives. The Committee identified significant deficiencies.

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ACRONYMS

Acronym	Definition
ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
ANL	Argonne National Laboratory
B&W	Babcock & Wilcox
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
DOE	Department of Energy
EDG	Emergency Diesel Generator
FACA	Federal Advisory Committee Act
FY	Fiscal Year
GPRA	Government Performance and Results Act
LOCA	Loss-of-Coolant Accident
MAUT	Multi-Attribute Utility Theory
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk assessment
PWR	Pressurized Water Reactor
RES	Office of Nuclear Regulatory Research
SBO	Station Blackout
SG	Steam Generator
SPAR	Standardized Plant Analysis Risk

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). The Advisory Committee on Reactor Safeguards (ACRS) has agreed to assist RES by performing independent assessments of the quality of selected research projects. Quality assessment of individual research projects constitutes a new undertaking for the Committee; one that is quite different in scope and depth in comparison to the ACRS biennial review of the overall NRC research activities. During fiscal year (FY) 2004, the ACRS conducted a trial review of the quality of selected research projects [Ref. 1]. Based on the outcome of this trial review , the Committee has established the following review process:

- 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 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 revises these reports, as needed, and provides them to the cognizant research manager, as appropriate.
- 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:

- Station Blackout Risk Evaluation for Nuclear Power Plants performed as a part of Standardized Plant Analysis Risk (SPAR) Models Development Program

- Steam Generator Tube Integrity Program at the Argonne National Laboratory

- Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag experiments at the Penn State University

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

A fourth research project on reactor containment performance being conducted at Sandia National Laboratories will be evaluated during FY-2006, once a particularly pivotal report on this research becomes available.

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 [Ref. 2 and 3]. The analytical part utilizes methods of multi-attribute utility theory (MAUT) [Ref. 4 and 5] 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.

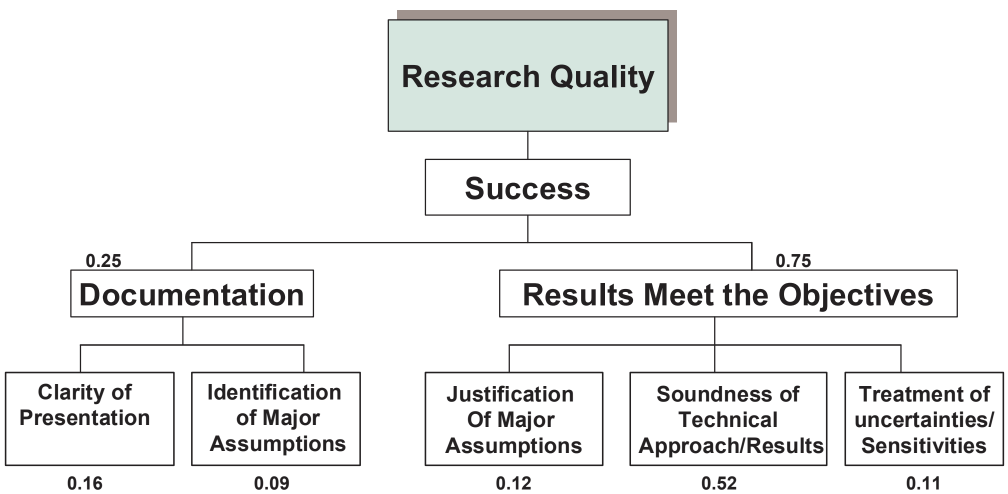

Figure 1 The value tree used for evaluating the quality of research projects

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	LABEL	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 STATION BLACKOUT RISK EVALUATION FOR NUCLEAR POWER PLANTS PERFORMED AS A PART OF SPAR MODELS DEVELOPMENT PROGRAM

In 1988, the NRC issued the Station Blackout Rule, 10 CFR 50.63, and the associated Regulatory Guide 1.155 establishing requirements and guidance to ensure decay heat removal for the period following loss-of-offsite power. Subsequent Probabilistic risk assessments (PRAs) indicated that compliance with these regulatory documents resulted in appropriately small core damage frequencies for station blackout (SBO) scenarios. On August 14, 2003, a widespread grid-related loss-of-offsite power event resulted in the controlled shut down of nine nuclear power plants. The NRC initiated a program to reevaluate the frequencies and durations of loss-of-offsite power, as well as the SBO risk contribution. The results of this study are documented in Reference 6. This report that the Committee reviewed is an update of previous reports analyzing the risk from loss-of-offsite power and subsequent SBO events in all operating U.S. power plants.

The SPAR models were used to evaluate the core damage frequency from internal events only for each plant during power operation. A number of enhancements to the SPAR models had to be made for this evaluation. The reliability estimates for diesel generators were also updated using recent data. Updated data were also collected for turbine-driven pumps, high-pressure core spray motor-driven pumps, and diesel-driven pumps. For the pressurized water reactors (PWRs), pump-seal failure models were selected based on the most recent developments.

The scope of this quality review is limited to the above report rather than a broader assessment of the quality of the updated SPAR models requested by RES. The Committee judged that it would have been overly ambitious to undertake such an evaluation in a single step and within the time constraints of the present review. The ACRS decided to have its Reliability and Probabilistic Risk Assessment Subcommittee perform a much broader review of the SPAR models during the upcoming year. Thus, in evaluating this report, the Committee has not considered the validity of the SPAR models that form the basis for the study.

GENERAL OBSERVATIONS

This report is an excellent example of the value of the SPAR Models Development Program and of the contribution that RES can make to the understanding of the safety of operating plants. The independent capability to evaluate risk issues across the population of operating plants has great value. By utilizing the same model and assumptions for all types of reactors in the fleet, the staff has been able to reach several conclusions regarding the effects of plant-specific design features on the risk from SBO. The availability of these models allows for periodic reevaluation of issues and trends associated with, for example, the effect of deregulation on grid reliability, and the effect of online maintenance on SBO.

The consensus scores for this project are shown in Table 2. This project was found to be more than satisfactory with a number of elements of excellence present. Comments and conclusions within the evaluation categories are:

Table 2. Summary Results of ACRS Assessment of the Quality of the Project on Station Blackout Risk Evaluation for Nuclear Power Plants

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	7.0	0.16	1.12
Identification of major assumptions	7.0	0.09	0.63
Justification of major assumptions	6.33	0.12	0.76
Soundness of technical approach/results	6.66	0.52	3.46
Treatment of uncertainties/sensitivities	6.0	0.11	0.66
		Overall Score:	**6.63**

Documentation

- Clarity of presentation **(Consensus score = 7.0)**

  The report is clearly written and well organized. It provides a good description of prior work and describes in detail the logic utilized in the selection of databases and assumptions. It presents the results in the context of previous evaluations, provides good explanation of changes, and discusses important trends and insights.

- Identification of major assumptions **(Consensus score = 7.0)**

  Assumptions are clearly stated, and the report does a good job of explaining the logic behind these assumptions.

Results Meet Objectives

- Justification of major assumptions **(Consensus score = 6.33)**

  Major assumptions are generally well justified, for example the use of industry-average data rather than plant-specific data for component unreliability, train test and maintenance outage probabilities, and initiating event frequencies.

In some instances, a full explanation is not provided. For example, no argument is provided for not modifying the Babcock & Wilcox (B&W) seal leakage model, except that there is no pending submittal to the NRC. From that statement, the reader is left with no insights regarding the quality of the B&W seal leakage model. Another example is the choice of a factor of two in the emergency diesel generator (EDG) performance sensitivity study. It is not clear why a factor of two was chosen.

- Soundness of technical approach and results **(Consensus score = 6.66)**

  There is nothing novel about the approach (this is not a criticism). The event trees are borrowed from those that had been developed previously.

  The use of industry-wide data to place all nuclear power plants on a common basis helped in determining the relative effectiveness of general features of electric power systems and backup safe shutdown modes in reducing the risk from SBO.

- Treatment of uncertainties and characterization of sensitivities **(Consensus score = 6.0)**

  The report includes the results of an uncertainty analysis and of a sensitivity study.

  The sensitivity results are point estimates, i.e., no uncertainty analyses were performed for the sensitivity cases. It is this last point that generated discussion among the panel members. What does a "sensitivity analysis" mean in the probabilistic world? In traditional engineering analysis where all the calculations were done on a "point estimate" basis, a sensitivity study usually means to vary, more or less arbitrarily, various parameters and evaluate their impact on the final answer. In probabilistic analyses, this approach must be reconsidered. Possible variability in parameter values should be included in the uncertainty distributions of these parameters. The focus should be on the assumptions and parameters that drive the results. An example is the use of the risk achievement worth to identify events that may have a significant impact on the core damage frequency calculated in a PRA. The ACRS acknowledges that this issue should be discussed in a broader context with the staff and that, perhaps, it would be unfair to judge the authors of this report harshly on an issue that has not been widely debated.

3.2 STEAM GENERATOR TUBE INTEGRITY PROGRAM AT THE ARGONNE NATIONAL LABORATORY[1]

The overall objective of the steam generator tube integrity research program is to provide experimental data and predictive correlations and models needed to permit the NRC staff to independently evaluate the integrity of steam generator tubes as plants age and degradation proceeds, new forms of degradation appear, and as new defect-specific management schemes are implemented. This program builds upon the results of NRC steam generator tube integrity and inspection research conducted since 1977.

The objectives of the specific project (task 3, Research on Tube Integrity and Integrity Predictions) selected for quality assessment were to:

- Determine if the flow stress of MA Nickel Alloy 600 tube material exhibits dependence on the stress rate or the strain rate (i.e.: the rate of internal pressurization).

- Determine the relationship between crack or ligament size (width, depth, and length), orientation, geometry, morphology, and number of ligaments and the tube leak rate and burst pressure.

- Confirm the validation of the tube leak rate correlation model and its relevance to choked two-phase flow expected at operating temperatures and pressures, including the relative uncertainties involved under various conditions.

- Compare laboratory leak rate and burst pressure models with the results of tests of samples of defective steam generator tubes removed from a decommissioned steam generator from McGuire Nuclear Plant.

These studies were conducted at the Argonne National Laboratory. The results of studies that the ACRS reviewed were documented in References 7 and 8.

The consensus scores for this project are shown in Table 3. This project was found to be satisfactory. The results meet the research objectives. Comments and conclusions within the evaluation categories are:

Documentation

- Clarity of presentation **(Consensus score = 4.7)**.

  The manuscripts documenting the results of this project [Ref. 7 and 8] are exceptionally informal. These documents read like laboratory reports prepared by technicians and sent to professional staff to be used in the preparation of a more formal report. Both manuscripts are rather more summary in nature. This terse informality of documentation makes the reports more readable though incomplete.

Table 3 Summary Results of the ACRS Assessment of the Quality of the Project on Steam Generator Tube Integrity

[1] Dr. William J. Shack, ACRS member, did not participate in the Committee's deliberations regarding this matter.

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	4.7	0.16	0.75
Identification of major assumptions	4.7	0.09	0.42
Justification of major assumptions	4.7	0.12	0.56
Soundness of technical approach/results	5.0	0.52	2.6
Treatment of uncertainties/sensitivities	4.3	0.11	0.47
		Overall Score:	**4.8**

The reports are inadequate for the archival documentation of expensive tests. Experimental methods are mentioned in casual ways with no effort, even by reference, to show that these methods are adequate or produce reliable, reproducible results.

Calibration and qualification of instruments are not discussed at all.

Theoretical models and even data analysis methods are mentioned without reference.

Figures showing data and correlations are exceptionally difficult to interpret since minimal legends and labeling are employed despite the figures being quite "busy." For the leak rate studies (page 34 of Ref. 7),except for specimen SLG900, no results are provided. The discussion on page 44 is not clear when correlating L/D ratios and choked flow.

A reader who does not routinely examine reports from this laboratory and is not intimately familiar with the equipment and methods of the laboratory will have difficulty in understanding the documentation. (Only after reading Ref 8 did one come to understand that the unlabeled scale in some photos in Ref. 7 was an inch scale and not a centimeter scale despite all the text on lengths referring to millimeters!) In the end, one can understand the points the authors are trying to make in Ref. 7, but with difficulty. Clarity of presentation is not of high quality, but adequate to understood the work.

It is dubious that the experimental results could ever be used directly in a regulatory process involving licensees. The qualification of methods and calibration of instruments simply will not be acceptable for such direct use.

- Identification of major assumptions **(Consensus score = 4.7)**

  The major assumptions employed are not separately and explicitly stated but some of these assumptions are embedded in the text. In a complex report such as this, it is an acceptable and appropriate practice to state assumptions in the context of the issues where they are used or evaluated and rejected.

  As noted above, identification and justification of assumptions are difficult to evaluate. There is not a coherent effort to do this in the document largely because it is not evident that results have any applicability. It is not evident that the results for the notched specimens discussed in the document will be used to infer the behavior of real cracks in tubes under accident conditions.

  The investigators have done a better job in identifying factors that will affect the experimental results and including their sensitivities in test programs.

  The documentation does not provide adequate justification for sensitivities that are included nor does it include discussions concerning the sensitivities of other factors that has not been considered.

  The document fails completely to address uncertainties in measurements or to provide adequate descriptions of parametric uncertainties in reporting results of fitting the data to correlations. Presumably, if needed, these uncertainties as well as uncertainties in measurements could be extracted. Therefore, only a modest reduction in the score has been imposed.

Results Meet Objectives

- Justification of major assumptions **(Consensus score = 4.7)**

  Certain assumptions are implicit in the statement of scope. However, the work plan and scope were designed so that the major assumptions would be tested experimentally to verify the validity of these assumptions. An example was the assumption that flow stress is virtually independent of the rate at which stress and strain are applied to the specimen. This assumption had its origins in earlier test work performed by others prior to the in-depth study undertaken in this project. ANL could not confirm the validity of this assumption and undertook an effort to determine why a rate effect was observed in their tests and not in the earlier tests. Other examples of implicit assumptions involved issues such as ligament linkage and its relationship to both leakage and burst pressure, the quantification of choke flow leakage through cracks with two-phase flow, and the existence of a correlation between leakage and crack growth. The investigators did not make an explicit effort to identify and justify these assumptions.

In connection with the development of failure ‘maps’, it is asserted that the complex ligament geometries of real cracks can be idealized as either solely axial, solely circumferential, or radial. The report does not include any discussion on how close those assumptions are to reality. As noted above, an assumption about application of correlation developed for two cracks being applicable to configurations with four and six cracks is neither articulated nor justified.

In some cases, the assumed level of familiarity with previous work limits the discussion to the extent that the bases for assumptions are not clear. For example, in the predictions of ligament rupture against the McGuire tests, the ligament rupture pressure of each test was predicted by the equivalent rectangular crack methods. There is no explanation of why this is the appropriate model. An explanation would be worthwhile given that the benchmark is only partially successful. The abstract states that this is the “latest correlation.” But some additional explanation would have contributed to a better understanding.

Much of the work on main steamline break effects on damaged tubes (Ref.8) relies on analytical simulation with TRAC-M and RELAP-5 codes. The ability of these codes to model appropriately pressure drops in complex geometries such as those of steam generator tube bundles and tube support plates has been questioned. The report does not discuss this issue. There are good comparisons of results from the two codes and finite-element analysis results, but applicability of these models is an important issue that deserves some discussion.

- Soundness of technical approach and results **(Consensus score = 5.0)**

The scope of work was thorough in identifying the major steps and the technical approach to be used by the investigators. The investigators used sound scientific and engineering methods to conduct these investigations. In addition, it is clear that the investigators followed up on anomalies and results that differed from prior assumptions to gain insights into the phenomenon that they were investigating. These new insights were factored into the analytical models under development to the extent that they could be, and uncertainties were estimated for data that had a range of numerical results. The investigators stated that the models provided conservative predictions.

Though quibbles abound in the review of the technical approach, no flaws were identified that would detract from the value of the results in any major way. On the other hand, the technical approaches adopted in the following four efforts were not inspired, so no bases for higher scores were identified either.

Pressurization rate effects

The first reported task was the confirmation of claims that rupture of flawed tubes is dependent on the rate of pressurization. The approach undertaken was to test a variety of flawed tubes similar to those used by investigators making the claim of a pressurization rate effect. The testing was, however, done in a consistent fashion unlike the testing done by those making the claims.

Testing was done at pressurization rates that varied from quasi-static to greater than 69 MPa/s. This range included, apparently, the pressurization rate used by those making the claims of a pressurization rate effect. Whether it includes prototypic pressurization rates is not stated, but it appears likely that it did. Tests were done at enough pressurization rates that it should be possible to infer by interpolation results for any pressurization rate likely to be of practical interest. This appears to be a technically sound and defensible approach.

In addition, tests are planned on cracks that were formed by a stress corrosion cracking process. The results of these tests will be presumably used to relate the results of tests with machined flaws to more realistic cracks. Again, this seems a prudent and reasonable approach.

o Development of failure maps

To prepare failure maps, the authors have correlated data on the ligament ruptures of two types of flaws in tubes. A simple polynomial model has been used for correlation and it does not seem to have been selected based on some theoretical considerations. Details of the procedure for fitting the data to correlations are not spelled out to any extent. It is apparent that the polynomial is a very approximate description of the data and the parametric values must be changed for different crack lengths. Fitting apparently neglected the uncertainties in the data. Had these uncertainties been recognized, it might have been possible to use simpler correlation expressions. A similar polynomial correlation was developed for rupture pressure for the case of two cracks separated by a circumferential ligament. It appears that the data used for correlation may have come from room temperature tests, but documentation is not definitive on this point, and salient references have not yet been retrieved.

The correlations were then used to develop maps of crack length versus ligament width showing behavior for various pressure differences and crack geometries assuming 80 and 90% through-wall cracks. This approach is common and technically sound for maps involving two cracks separated by an axial or a radial ligament, provided that the correlations developed from test data are applicable at the assumed 300°C.

Maps were also prepared for cases with four and six cracks. There seems to be no demonstration that the correlations of ligament rupture and tube rupture obtained for two cracks are applicable to cases with four or more cracks. To be sure, there is an extrapolation taking place here that is not especially well highlighted in the documentation. Nevertheless, one must concede that if this extrapolation is palatable, the approach adopted in preparing the maps is a widely accepted one. Use of the maps, on the other hand, would demand a great deal more than is attempted in this limited effort. A reader would benefit from some comparison of the map predictions to data for the multiple crack cases.

Leak Rate Studies

The leak rate studies were undertaken to determine the limits of applicability with respect to the through-wall crack length and crack tightness of the simple orifice model for predicting leak rates of cracked tubes. The effort undertaken focused on conditions that will lead to "flashing" of the coolant within the crack. Crack length divided by the hydraulic diameter of the crack was used as the metric for cracks in tubes used in the tests. This is acceptable because realistic cracks are used in the test program. Analysis of the results was supplemented by data from the literature concerning flow through better instrumented slits in plates. The technical approach appears to be adequate to the task.

Results obtained in the effort only address conditions for subcooling in the range of 50-60^{o}C . Such a subcooling range corresponds to cold leg conditions. A plausibility argument is advanced that "conservative" results will be predicted for hot leg conditions that are more appropriate for issues associated with steam generator tube leakage. Thus, results only marginally meet the objective if the objective is to find limits of applicability of the orifice model for conditions where it is likely to be of interest to apply.

Rupture and Leak Rate Predictions for McGuire Steam Generator Tubes

The technical approach for this effort involved acquisition of flawed tubes form the McGuire plant and characterization of the flaws first by nondestructive examianation methods and later by fractography. The tubes were then tested for leakage in a facility that is presumably well established and well described in some other publications. Unfortunately, no references were provided to validate this presumption. No description of the method for measuring leak rates was provided. Presumably, a well established method exists and the authors could have informed the reader about this method by means of a reference. Though poorly documented, the technical approach appears sound.

- Treatment of uncertainties and characterization of sensitivities **(Consensus score = 4.3)**

The comparison of predictive models of leak rate and rupture as applied to actual tubes removed from a retired McGuire steam generator with leakage and burst test data of these tubes showed reasonable agreement. In the discussion, explanations were provided as to why the predictive models differed from the actual test results. A range of uncertainty and the degree of conservatism between the models and observed results were estimated, in order to establish the degree of usefulness of the correlations developed. Because of the complex nature of stress corrosion cracks, predictive uncertainty exists and has been estimated and factored into the resulting conclusions.

The investigators do a rather good job in developing their experimental projects in considering sensitivities such as sensitivity to the number of cracks, ligament sizes, crack orientation and the like. The investigators have not estimated uncertainties associated with any measured value that they report. Where they have fit data to a parametric correlation, they have failed to cite any uncertainties in the parametric values and certainly have not reported covariance matrices for models involving more than two parameters. They do not report on the uncertainties of predictions derived from correlations. Episodically, the authors report linear correlation coefficients that are essentially useless in the interpretation of the quality of a fit of a parameterized equation to data without a great deal more information about the fitting results.

The adequacy of the investigators' treatments of sensitivities in the development of their research efforts is acknowledged. Neglect of uncertainties in reports of measurements is the basis for reduction of the score in this category.

3.3 ANALYSIS OF ROD BUNDLE HEAT TRANSFER FACILITY TWO-PHASE INTERFACE DRAG EXPERIMENTS AT THE PENN STATE UNIVERSITY

The objective of a task at the Penn State University was to analyze data that had been collected in the Rod Bundle Heat Transfer Facility in order to gain insights to be used in the development and validation of the TRACE computer code. The specific set of data was collected to examine level swell under reflood conditions. The rod bundle in the experimental setup simulates a PWR fuel assembly with spacer grids, as in the standard 17x17 Westinghouse array. The experimental bundle involves a 7x7 array of full length, electrically heated fuel pins. The principal data collected in the experiments were the pressure drop along the length of the pins with varied reflood flow rate, power level, and inlet subcooling. Other properties of the flow, such as void fraction, interfacial drag force, and the product of interfacial area and friction factor, were determined by inference from a simplified model of energy conservation.

The data are said to be “more detailed” than previous data, but no comparisons are made to illustrate why, or to show consistency (or otherwise) with previous work.

The review is based on the only report [Ref. 9] that was provided to the Committee of results from the test program. It is entitled “Analysis of Rod Bundle Heat Transfer Test Facility Two-Phase Interfacial Drag Experiments.” It has no number and is believed to be a draft. The title of the report is somewhat misleading, since there were no measurements of interfacial drag. The only parameter measured, apart from those defining the boundary conditions of the experiment, such as flow rate, power supplied etc., was the pressure drop over several lengths of a rod bundle.

The broader experimental program, which represents a substantial undertaking, with extensive measurement of parameters such as temperature, droplet size, and velocity, was not part of this review.

The Committee also had the benefit of an earlier report describing the test facility and of the RES Thermal-hydraulics Research Plan, dated March 1, 2005. RES provided a memo dated June 6, 2005 entitled, “Usage of Data from the Rod Bundle Heat Transfer Test”.

The consensus scores for the project are shown in Table 4. This project marginally satisfied the research objectives. The Committee identified important deficiencies. Comments and conclusions within the evaluation categories are:

Documentation

- Clarity of presentation **(Consensus score = 4.33)**

  The report is readable and it is reasonably clear on what was done. However, the objectives of the work are not clearly stated.

Table 4 Summary Results of the ACRS Assessment of the Quality of the Project on Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag Experiments at the Penn State University

Performance Measures	**Consensus Scores**	**Weights**	**Weighted Scores**
Clarity of presentation	4.33	0.16	0.69
Identification of major assumptions	4.0	0.09	0.36
Justification of major assumptions	3.33	0.12	0.40
Soundness of technical approach/results	3.33	0.52	1.73
Treatment of uncertainties/sensitivities	0.66	0.11	0.07
		Overall Score:	**3.25**

Figures are mostly clear but some lack essential details. Descriptions of the location of pressure taps are inconsistent.

The report requires substantial manipulation of pressure drop data to infer void fraction, interfacial drag force, and the product of interfacial area and friction factor but the main report does not explain how these properties are obtained. The reader has to study the appendices to determine the assumptions and theory applied.

- Identification of major assumptions **(Consensus score = 4.0)**

“Correction” of data is described but insufficiently to provide understanding of how spacers were treated, or why certain flow regimes were used to predict terms needed to convert from pressure drop to void fraction. These assumptions prejudice the eventual use for TRACE development, since they are in parallel to the comparisons with TRACE. It would be better to have TRACE predict the raw data.

The assumption that the pressure drop does not influence fluid properties appears to be used but is not identified.

The assumption that the only source of vapor generation is the addition of heat ignores the significant effect of flashing that is not identified.

The assumption that “the total pressure drop is small” is incorrect. Since the pressure drop along the bundle can be substantial (almost 6psi), specification of a single “pressure” (e.g. 20psia) for each experiment is inadequate without identifying clearly where it is measured.

Results Meet Objectives

- Justification of major assumptions **(Consensus score = 3.33)**

  Several inappropriate flow regimes are used.

  The energy balance is erroneous, omitting an important “flashing” term, leading to inaccurate prediction of quality.

  Property changes along the bundle due to pressure drop are ignored, though they are influenced by pressure and temperature changes.

  The effect of spacers on the flow pattern, pressure drop, and void fraction is not explained. In “correcting” the pressure drop measurements to compute a void fraction, some justification is provided for the friction pressure drop correction, but none for the acceleration pressure drop correction.

- Soundness of technical approach and results **(Consensus score = 3.33)**

  It is doubtful if the results are useful for TRACE development. There is no discussion of models currently in TRACE or direct comparison with these models.

  The presentation and reduction of data contain errors and there is no investigation of the effects of assumptions.

  Several of the comparisons with theory are inappropriate. There is no critical examination of features of the data, such as large fluctuations in the pressure drop data and the apparent lack of steady state in some tests.

  Since the intent of the report is to derive interfacial drag, there should be more information on how this was done, the sources of error, the effect of parameters, the effect of spacers, etc. Only one example is given, and it appears to have a basic flaw, since the large spikes of extreme values that are predicted indicate the flow to be close to homogeneous, which is inconsistent with evidence provided by the void fraction results.

- Treatment of uncertainties and characterization of sensitivities **(Consensus score = 0.66)**

  There is no treatment or discussion of uncertainties.

4. REFERENCES

1. Letter Dated November 18, 2004, from Mario V. Bonaca, Chairman, ACRS, to Carl J. Paperiello, Director, Office of Nuclear Regulatory Research, NRC, Subject: ACRS Assessment of the Quality of Selected NRC Research Projects.

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5. Keeney, R.L., and H. Raiffa, *Decisions with Multiple Objectives: Preferences and Value Tradeoffs*, Wiley, New York, 1976.

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7. Majumdar, S., K. Kasza, S. Bakhtiari, J. Oras, J. Franklin, and C. Vulyak, Jr., "Pressurization Rate Effect on Flawed Tube Rupture, Failure Maps for Complex Multiple Cracks, Validation of Leak Rate Correlation Model and Rupture and Leak Rate Predictions for McGuire Steam Generator Tubes," Draft, NUREG/CR-xxxx, Argonne National Laboratory, April 2004.

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9. Hochreiter, L. E., F. B. Cheung, T. F. Lin, and D.J. Miller, "Analysis of Rod Bundle Heat Transfer Test Facility Two-Phase Interfacial Drag Experiments," The Pennsylvania State University, June 2005.

November 4, 2005

Dr. Carl J. Paperiello
Director
Office of Nuclear Regulatory Research
Washington, D.C. 20555-0001

SUBJECT: ACRS ASSESSMENT OF THE QUALITY OF SELECTED NRC RESEARCH PROJECTS - FY 2005

Dear Dr. Paperiello:

Enclosed is our report on the quality assessment of the following research projects:

- Station Blackout Risk Evaluation for Nuclear Power Plants performed as a part of Standardized Plant Analysis Risk (SPAR) Models Development Program
  - This project was found to be more than satisfactory. The results meet the research objectives.
- Steam Generator Tube Integrity Program at the Argonne National Laboratory
  - This project was found to be satisfactory. The results meet the research objectives.
- Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag Experiments at the Penn State University
  - This project marginally satisfied the research objectives. The Committee identified significant deficiencies.

These projects were selected from a list of candidate projects suggested by the Office of Nuclear Regulatory Research.

A fourth research project on reactor containment performance being conducted at Sandia National Laboratories will be evaluated later, once a particularly pivotal report on the research becomes available.

We anticipate receiving your list of candidate projects for review during the next 12 months.

Sincerely,

William J. Shack
Acting Chairman

Enclosure: As stated

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