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

ACRSR-2217

October 17, 2006

Dr. Brian Sheron
Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington DC 20555-0001

SUBJECT: ACRS ASSESSMENT OF THE QUALITY OF SELECTED NRC RESEARCH PROJECTS- FY 2006

Dear Dr. Sheron:

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

- Melt Coolability and Concrete Interaction (MCCI) Program at the Argonne National Laboratory
  - This project was found to be satisfactory. The results meet the research objectives.
- Containment Integrity Research at Sandia National Laboratories
  - This project was found to be more than satisfactory. The results meet the research objectives.

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

We anticipate receiving your list of candidate projects for ACRS prior to our December 2006 Full Committee meeting.

Sincerely,

/RA/

Graham B. Wallis
Chairman

Enclosure: As stated

Assessment of the Quality of Selected NRC Research Projects by the Advisory Committee on Reactor Safeguards for FY 2006

October 2006

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.

MEMBERS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Dr. Said Abdel-Khalik, Southern Nuclear Distinguished Professor, George W. Woodruff School of Mechanical Engineering, Georgia Institute of Technology, Atlanta, Georgia

Dr. Joseph S. Armijo, Adjunct Professor of Materials Science and Engineering at the University of Nevada, Reno

Dr. George E. Apostolakis, Professor of Nuclear Science and Engineering, Professor of Engineering Systems, Massachusetts Institute of Technology, Cambridge, Massachusetts

Dr. Sanjoy Banerjee, Professor of Chemical Engineering, University of California at Santa Barbara, Santa Barbara, California

Dr. Mario V. Bonaca, Retired Director, Nuclear Engineering Department, Northeast Utilities, Connecticut

Dr. Michael Corradini, Professor and Chairman of Department of Engineering Physics, University of Wisconsin, Madison, Wisconsin

Dr. Thomas S. Kress, Retired Head of Applied Systems Technology Section, Oak Ridge National Laboratory, Oak Ridge, Tennessee

Mr. Otto L. Maynard, Retired Chief Executive Officer, Wolf Creek Generating Station, Kansas

Dr. Dana A. Powers, Senior Scientist, Sandia National Laboratories, Albuquerque, New Mexico

Dr. William J. Shack, **(Vice-Chairman)**, Associate Director, Energy Technology Division, Argonne National Laboratory, Argonne, Illinois

Mr. John D. Sieber, **(Member-at-Large)**, Retired Senior Vice-President, Nuclear Power Division, Duquesne Light Company, Pittsburgh, Pennsylvania

Dr. Graham B. Wallis, (Chairman), Sherman Fairchild Professor Emeritus, Thayer School of Engineering, Dartmouth College, Hanover, New Hampshire

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 two research projects are summarized as follows:

- Melt Coolability and Concrete Interaction (MCCI) Program at the Argonne National Laboratory

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

- Containment Integrity Research at Sandia National Laboratories

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

CONTENTS

Page

ABSTRACT iv
FIGURES vi
TABLES vi
ABBREVIATIONS vii

1. INTRODUCTION 1

2. METHODOLOGY FOR EVALUATING THE QUALITY OF RESEARCH PROJECTS 3

3. RESULTS OF QUALITY ASSESSMENT 5

3.1 Melt Coolability and Concrete Interaction (MCCI) Program at the Argonne National Laboratory 5

3.2 Containment Integrity Research at Sandia National Laboratories 15

4. REFERENCES 20

FIGURES

Page

1. The value tree used for evaluating the quality of research projects 3

TABLES

1. Constructed Scales for the Performance Measures 4

2. Summary Results of ACRS Assessment of the Quality of the Project on Melt Coolability and Concrete Interaction (MCCI) 6

3. Summary Results of the ACRS Assessment of the Quality of the Project on Containment Integrity Research at Sandia National Laboratories 9

ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
ANL	Argonne National Laboratory
BWR	Boiling Water Reactor
CCFL	Counter Current Flooding Limit
CCI	Core Concrete Interactions
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
DOE	Department of Energy
FACA	Federal Advisory Committee Act
FY	Fiscal Year
GPRA	Government Performance and Results Act
LOCA	Loss-of-Coolant Accident
MAUT	Multi-Attribute Utility Theory
MACE	Melt Attack and Coolability Experiments
MCCI	Melt Coolability and Concrete Interaction
MET	Melt Eruption Test
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Cooperation and Developmen
PRA	Probabilistic Risk assessment
PWR	Pressurized Water Reactor
RES	Office of Nuclear Regulatory Research
SSWICS	SMALL Scale Water Ingression and Crust Strength

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 [1,2]. 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 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:

- Melt Coolability and Concrete Interaction (MCCI) Program at the Argonne National Laboratory
- Containment Integrity Research at Sandia National Laboratories

These two 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 [Ref. 3 and 4]. The analytical part utilizes methods of multi-attribute utility theory (MAUT) [Ref. 5 and 6] 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.

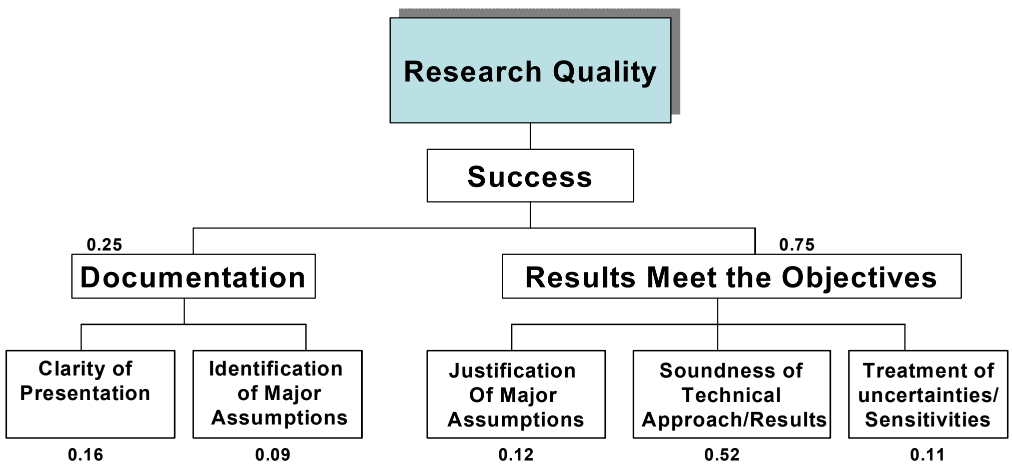

Figure 1 The value tree used for evaluating the quality of research projects

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 MELT COOLABILITY AND CONCRETE INTERACTION PROGRAM AT THE ARGONNE NATIONAL LABORATORY

The Melt Coolability and Concrete Interaction (MCCI) research was conducted at the Argonne National Laboratory (ANL). The research was part of an international collaborative effort under the auspices of the Organization for Economic Cooperation and Development (OECD). Thirteen OECD countries including the United States participated in the research program. As the host organization, NRC coordinated the program with the following objectives:

1. Provide both confirmatory evidence and test data on coolability mechanisms identified in previous integral effect tests and resolve the ex-vessel coolability issue through an understanding of the synergistic effects of these coolability mechanisms and through development of analytical models.

2. Address remaining uncertainties related to long-term two-dimensional melt-concrete interaction under dry as well as flooded cavity conditions.

The MCCI experimental efforts built upon previous OECD sponsored (MACE) integral effect tests program that attempted to define conditions under which water was able to quench core debris interacting with concrete. This previous effort identified some mechanisms of debris cooling that had not previously been recognized.

The MCCI project consisted of three experimental efforts:

- Small-scale Water Ingression and Crust Strength Tests
- Melt Eruption Tests.
- Core Concrete Interaction Tests

The Small Scale Water Ingression and Crust Strength (SSWICS) tests were intended to measure the ability of water to cool the molten core material by mechanisms other than conduction limited heat transfer, and to measure the strength of the crust formed during flooding of the melt.

The Melt Eruption Test (MET) was intended to measure the influence of gas sparging on melt entrainment and cooling and to determine the effect of melt ejection on the core-concrete interaction.

The Core Concrete Interaction (CCI) tests were intended to resolve uncertainties in axial versus lateral power splits and respective concrete ablation rates. The tests were intended to replicate as closely as possible conditions at plant scale and contribute data to verify and validate predictive codes. These tests were augmented by flooding with water after partial ablation to obtain debris coolability at later stages in the accident process.

The MCCI project was completed in December 2005 and an OECD final report [7] was issued in February 2006.

GENERAL OBSERVATIONS

The project has met several of its goals. It has successfully demonstrated several valuable test techniques to simulate the complex phenomena that occur during molten core concrete interactions. It has explored phenomena of interest and provided data consistent with scoping level tests. The project however has been too ambitious in its scope (or claims), and has failed to work with fully prototypic materials. The study was more exploratory than confirmatory. Analysis and deductions are somewhat weak, tenuous, and may be wrong in some aspects. Though some qualitative understanding has been achieved and possible theoretical approaches have been developed, the work is far from having established reliable predictive tools for resolving any of the issues that led to the proposed research.

The consensus scores for this project are shown in Table 2. The score for the overall assessment of the work was found to be 5.22 which should be interpreted as "a professional job that satisfies the research objectives." The Committee identified areas for improvement in all of the evaluation categories. Comments and conclusions within the evaluation categories are:

Documentation

- Clarity of presentation **(Consensus score = 6.5)**

    The Committee is generally pleased with the documentation which is challenging for a long-term, multifaceted effort such as the MCCI project.

    The writing and descriptions are generally clear. Many observations were made. Some seem inconsistent (e.g. pictures and data plots do not confirm the text). Some necessary details and dimensions are missing. The text generally presents a reasonable story of an ambitious undertaking that was partially successful.

    The report is generally well written and understandable. However it was difficult to find the actual composition of the particular "thermite" mixture used to produce the molten core material in the tests. The information was found in supporting documents. The compositions of the various melts used in the tests were provided, although some (Table 3-4) were mislabeled as ‘thermite compositions”.

Table 2. Summary Results of ACRS Assessment of the Quality of the Project on Melt Coolability and Concrete Interaction (MCCI)

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	6.5	0.16	1.04
Identification of major assumptions	4.75	0.09	0.43
Justification of major assumptions	4.0	0.12	0.48
Soundness of technical approach/results	5.5	0.52	2.86
Treatment of uncertainties/sensitivities	3.75	0.11	0.41
Overall Score:			**5.22**

The authors are overly familiar with the results derived from the SSWICS tests and do not present these results as clearly as possible. However, they do a poor job explaining legends of figures such as the legends of Figure 2-2 and 2-4. It is especially difficult to ascertain the meaning of “F-integrated” in the legend of Figure 2-4. The pair-wise comparisons of test results discussed on page 11 and following would have been far more effective if figures of corresponding pair-wise results had been provided rather than the jumbles of multiple results found in Figures 2-9 and 2-10. Scatter and variations in the plotted results make it quite challenging to understand what the authors mean by cooling plateaus they discuss at length in the text. Indeed, on page 15 the authors acknowledge that the identification of a plateau is subjective. Why, then, don’t they share explicitly with the reader their identifications of the plateaus so the reader can judge for himself what has drawn the authors’ interests and attentions?

The authors do not provide the readers with any parametric quantities used for the evaluation of models they compare to data or parametric values used for the plant scale comparisons. This places an enormous burden on the reader to independently assess the asserted models and the experimental data. In some cases, the reader is quite challenged to do this. For instance, in the discussion of radiation heat transfer (p. 52) the authors invoke the core debris melting temperature as a point value even though they previously went to pains to note that core debris freezes over a very large temperature range (Figure 2-13 and associated text). What value did the authors in fact use for the radiation heat transfer calculations? Similar conceptual questions in addition to simple material properties questions arise in connection with many other correlations.

- Identification of major assumptions **(Consensus score = 4.75)**

Assumptions appear in the way in which physical models were hypothesized, developed, and accepted. For example, CCFL was claimed to limit heat transfer in a porous crust, gas was assumed to create eruptions and entrain particles, gas bubbles were assumed to prevent crust formation, the crust was modeled as being broken by its own weight and the weight of water above it without being supported from below. Though these assumptions were identified and described, they were not critically examined.

No explicit statement of assumptions for the tests was provided. The assumptions could only be inferred from the tests themselves, and the subsequent analyses of results. If major assumptions had been stated and reviewed by the parties planning and authorizing the tests, perhaps they would have been modified to provide prototypic test melts and base mats.

Results Meet Objectives

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

No explicit justification of assumptions for any of the tests was provided. It is not clear how so many test parameters could have been varied in so few tests without justification. This suggests that the choice of variables was rather informal and consistent with exploratory testing and equipment checkout.

Assumptions are justified as being deduced from empirical evidence, mechanistic models and previous work. Critical examination of them by clear comparisons with data and evidence is weak. Equations are written down based on word descriptions which are sometimes vague and would benefit from a clearer, more rigorous approach, using sketches and control volumes. Radiation is invoked as a mechanism of heat transfer but is not evaluated numerically, so it is unclear what role it plays. CHF and CCFL are invoked rather loosely without being fully described or evaluated analytically, so it is not clear exactly how they were used and how well their appropriateness was validated.

- Soundness of technical approach and results **(Consensus score = 5.5)**

Design and conduct of the tests were quite good and represent an engineering achievement. The system "worked" (except for the MET test), allowing observations to be made and data obtained. Theoretical and "engineering modeling" conclusions are not so strong. On balance this is a reasonable piece of exploratory investigation, but it does not really meet the work-scope objectives, which seem overly ambitious. No mechanism nor theory is developed for the actual process of erosion of the concrete (it is hypothesized that the rate is proportional to heat flux, as if this were a phase change reaction, but this mechanism is not confirmed and the heat flux to the concrete is not predicted).

The approach to the simulation of real molten core concrete interaction phenomena was sound, but attempted to answer too many questions with too few experiments. Further, neither the most likely composition of the real molten cores, nor the structure and composition of real concrete base mats were simulated. Consequently the applicability of the test results to real events is limited. There is no reason why the approach taken cannot be improved by better simulation of core melts and base mats and by a more disciplined approach in the definition of each future test, the limitation of variables in each test, the performance of duplicate tests, and improvements in redundancy and reliability of instrumentation.

The absence of pre-test predictions for the various tests is troubling. Certainly the models exist, as well as some data. It would have been expected some pre-test analyses to help pinpoint the parameters with greatest uncertainty and to focus on the primary objectives of each experiment.

The absence of a thorough ceramographic examination of the solidified crust to understand the structure and composition (on a microscopic scale) of this highly heterogeneous material was a significant shortcoming in the approach. Gross chemical composition measurements were made indicating some variability across the solidified melt, but this level of analysis provides little if any knowledge of its physical properties. In the absence of a detailed understanding of the microstructure of the solidified corium, analytical models will have to rely on crude approximations of the property data required for predictive models.

The SSWICS tests were undertaken to evaluate the rate of core debris cooling by an overlying water pool and to obtain samples of solidified core debris for strength measurements. The "core debris" simulant used in the tests was laced with various fractions of concrete to simulate the effects of some period of concrete ablation prior to exposing the molten core debris to water. The tests were, however, conducted in an apparatus composed of cast zirconia and magnesia. The tests did not involve active attack on the concrete and the vigorous melt stirring and sparging that accompanies such attack. The assumption that the concrete attack would not affect cooling or crust formation cries out for justification but none is provided. The further, implicit, assumption that attention can focus on the oxide phase of core debris[1] and not consider the voluminous metallic phase that would be present in core debris that had penetrated a reactor vessel also calls for some justification and none is provided

One of the objectives of the SSWICS tests was to obtain solidified core debris samples for strength measurements. There is, of course, an implicit assumption that the mechanical properties of the solidified materials produced in so unprototypical a way are somehow similar to what would be expected of core debris. Ceramic materials are notorious for having mechanical properties that are quite sensitive to the details of microstructure. Indeed, the authors find that crack structure has more a bearing on strength than composition, but they do not explain why they think the crack structures of their samples are indicative of the crack structures of solidified core debris. Certainly, the challenges faced by those removing solidified core debris (which, of course, had zero concrete content but was quenched by water) from the Three Mile Island vessel suggest that real core debris may be much stronger than suggested by the test results for the samples from the SSWICS tests.

More troubling about SSWICS is the implicit assumption that a room temperature strength measurement is somehow useful in the prediction of the strength of a solid with a thermal gradient that goes from the saturation point of water where the solid should be brittle to the melting point and a zone where the crust will be quite plastic. No explanation is provided on why the authors think that a crack will propagate from the cool regions through this plastic zone which might be quite thick. Indeed, the authors in section 6.0 seem to feel a brittle failure model is appropriate even though they acknowledge the underside of the melt will be very close to the melting point of the core debris.

In the authors' defense, some of the material that is the basis of their technical approach for the SSWICS tests is to be found in ancillary documentation so well referenced in the report. Examination of this material does not resolve the issues raised here. The question is so central to the thesis of the document that it deserves exposition.

Results obtained by the authors show that the concrete content of the core debris affects the crust strength and fracture behavior. The exposition would have been greatly enhanced if the authors had demonstrated based on accident analyses that they were working in a relevant and meaningful range. Discussion below concerning the initial transient when core debris first contacts concrete suggests that they are not.

---

[1] The tests involved some small fraction of chromium metal in the core debris ~6 to 8 weight percent which is much less than the fraction of metal usually expected in ex-vessel core debris.

Overall, the soundness of the technical approach for the SSWICS tests is arguable, but not demonstratively flawed.

The MET failed because of inadequate tests apparatus design. Indeed, experience has shown it unlikely that the test would have met its objectives had it been possible to retain the melt within the experimental cavity. The tests were to examine melt entrainment by gas sparging. Gas was supplied not by the attack of the melt on concrete but through a porous plate at the base of the apparatus. Such porous plate designs seldom yield a uniform gas flux. The variable flow resistances near the wall cause preferential flow through regions of the melt. The investigators did no simulation tests to see if they could get uniform flow. Without a reasonably uniform flow across the diameter of the melt, entrainment results are difficult to interpret and nearly impossible to scale up to reactor dimensions. To get uniform flow through a porous plate, rather difficult variations in plate porosity must be engineered across the diameter.

The authors conclude, however, that the objectives of the MET were met by examining data from other experiments. They do this with no attention to uncertainty. They compare results to a model that has an uncertainty of (-25% to +50%) which is large enough. As noted below, it is remarkable that the authors were able to avoid comparing results to the widely used Kataoka-Ishii correlation.

The CCI tests were undertaken to ascertain the split between the horizontal heat flux to concrete from molten core debris and the downward heat flux to concrete. A rich literature on this issue developed very shortly after publication of WASH-1400 and it is unfortunate that the authors do not provide a precise of this literature involving both experiments and analyses. The technical approach adopted in the experimental effort minimizes the important effect of the metallic phase of core debris to the downward (and sideward) heat flux. The authors acknowledge this and even note their results "... may not be directly applicable to reactor accident sequences ..." They hope instead that the results may be useful for code validation but do not provide any evidence that this is the case such as might be derived by discussing the extent models of heat flux from core debris to concrete. It is consequently not evident at all why these tests were undertaken.

The CCI experiments were done in the direct electrical heating apparatus used for the MACE program. Core debris was used that contains some fraction of concrete as might be expected following the vigorous initial interaction with concrete by core debris containing some amount of metallic zirconium. The initial configuration of the core debris has not similarly been modified to reflect such an initial interaction and it is not at all evident why. Modern reactor analysis codes predict for many of the risk dominant accident sequences that core debris penetrating a reactor vessel will contain an important fraction of unoxidized zirconium metal. This is especially so for accidents at boiling water reactors that have a much higher initial core inventory of zirconium metal than do pressurized water reactors. Hot, metallic zirconium even when alloyed with very substantial quantities of steel structures from the reactor vessel is quite reactive. Heat liberated by the chemical reactions of metallic zirconium with the gaseous and condensed products of concrete decomposition is often predicted to raise the core debris temperature to quite high levels leading to more gas generation and more reaction with zirconium in the core debris[2]. There is the further radiation heat transfer to concrete not contacted by core debris. This concrete spalls and melts into the core debris. The geometry of the region occupied by core debris in an accident will be quite different than the regular geometry of the test and this will affect the heat flux partitioning between the horizontal and axial directions. The investigators could not simulate the initial vigorous interaction of core debris with concrete because of limitations of experimental methods. It is unclear why they did not address the geometric issues.

The CCI tests yielded results that indicate that the ratio of axial to radial ablation of concrete depend on concrete type. This, of course, has been known since the first tests of melt interactions with concrete were done in 1977. Most models attribute the differences to the higher gas production per unit of calcareous concrete ablated than gas production associated with ablation of siliceous concrete. Liquid concrete films at the interface also affect the heat transfer split. The authors draw attention to the core debris interface with concrete and note the differences in the interfaces for siliceous and calcareous concretes - differences that have also been known for 30 years. Siliceous concretes typically melt at lower temperatures to yield a more viscous product than do limestone concrete. Furthermore, the decomposition of calcium carbonate yield a decrepitated product. The heat transfer models, especially that developed by Bradley for the CORCON code, take these well known observations concerning the interface into account.

The report concludes with sections dealing with correlations of results and applications to reactor accidents. The titles of these sections are somewhat misleading. The authors do not really correlate their results. They assert models and compare model predictions to the results with scant attention to the uncertainty of the model predictions as a result of parameter uncertainty nor uncertainties in their experimental results. This technical approach has its merits, but it does not recognize that there are several models of core debris interactions with concrete being used today for accident analyses. It is remarkable that the authors elected not to compare their results to predictions of these models (CORCON, WECHSL, MAAP, etc.) Similarly, for plant applications, the authors did not analyze accidents. They set up stylized situations to examine how their correlations would relate to a larger scale. Again, more interesting would be to compare predictions of extant models modified to account for the new data to actual plant accidents. Do the new results change any of our current perceptions concerning accidents? Were this the first investigation of core debris interactions with concrete, the technical approach adopted for this work would be satisfactory. In fact, core debris

[2] The authors of the report seem not to recognize this prediction of modern accident analysis models as evidenced by their discussion of the initial transient interactions presented in section 5.1 concerning bulk cooling where they emphasize the temperature fall during the initial interaction.

interactions with concrete is a well-trodden field and there is a lot of both experimental and theoretical work that has been done that is not recognized in this documentation.

The technical approach does not defend the model selection. This is left to ancillary documentation. In many cases, this is quite acceptable. But, in other cases models of core debris interactions with concrete used in accident analyses are using different models. A comparison of the data to these models now in use would be most illuminating. For instance, it is not clear why the Kataoka-Ishii liquid entrainment correlation used in so many places is neglected in favor of the Rico-Spalding correlation.

- Treatment of uncertainties and characterization of sensitivities **(Consensus score = 3.75)**

There is no formal treatment of uncertainties. The conclusions are all qualitatively uncertain, since they are mostly descriptive. More efforts could have been usefully placed on giving the reader a crisper evaluation of uncertainty, based on the very small number of tests and many speculative effects. In particular, anyone wanting to use the theory and coefficients “C” and “E” would benefit from more direct warning about how uncertain they are, as well as the preliminary status of the equations in which they appear.

Obvious uncertainties were simply ignored in the analyses and conclusion. For example on pages 16 through 19 of the final report, data from the SSWICS tests were used to verify the Lister/Epstein dryout heat flux model. This required various thermal and mechanical property data for the solidified crust. There was no explicit discussion of the methods used to create the necessary data for such a heterogeneous material and consequently no treatment of uncertainties.

The report states that the crust mechanical property data were approximated using a volume-weighted method based on the properties of the individual constituents (uranium and zirconium oxides, chromium and concrete). However, there were no ceramographic examinations of the solidified melts identifying the microstructures and quantifying the compositions and amounts of the various phases present. Since the mechanical properties of a heterogeneous material are generally not controlled by a simple volumetric weighted average of constituents, this was an overly simplistic assumption.

As in all experiments there are uncertainties in the reliability of individual pieces of equipment and instruments used in the test system. These uncertainties are generally estimated by careful pre-test analyses, and ultimately confirmed by repeated testing of the entire test system with minimum variation of test parameters. The report does not indicate that any duplicate tests were performed in this project. Consequently it is not clear how much confidence one can have in the reproducibility of the results. An example is the behavior of the CCI-2 test in which the average melt temperature rose by approximately 100 C° after water addition (Figure 0-5). This was either an experimental error or a real phenomenon. A rationalization was provided in the report that suggested that the bulk melt temperature could have increased due to the quenching of the surface, formation of an insulating crust and loss of conductive heat transfer, but there were no redundant thermocouples available to support this supposition or resolve the question of experimental error. The fact that the phenomenon was not observed in the CCI-1 and CCI-3 tests leaves the issue open. This uncertainty undermines confidence in the temperature measurement throughout the CCI-2 experiment, and propagates through to any analysis that uses the data.

3.2 CONTAINMENT INTEGRITY RESEARCH AT SANDIA NATIONAL LABORATORIES

For nearly 30 years, significant research has been performed at Sandia National Laboratories (SNL), primarily under the sponsorship of the NRC, to improve the understanding of performance of nuclear power plant steel and concrete containment structures under severe accident pressure and temperature loads that exceed the design bases of containments. This work has consisted of experimental programs and analytical studies to investigate the response and capacity of containment structures for a wide variety of loading conditions with a primary emphasis on internal overpressurization. The report [8] selected for the present review and quality evaluation does not document a specific research effort, but summarizes the results obtained from all the earlier research activities and identifies common themes that have emerged. As stated in its foreword, the primary focus of the report is to comment on and tie the results of earlier experiments and analyses into current research. The scope also includes documenting the lesson learned during the containment model testing and analytical simulations, which are directly applicable to regulating and licensing the operation of the current fleet of nuclear power plants, as well as the design of new plants.

GENERAL OBSERVATIONS

This report summarizes the results of increasingly large and complex tests of scale models of containment structures and sub-components conducted at SNL between 1983 and 2001.These tests were intended to model existing containments in operation in the US. For most of these tests, various international nuclear research agencies and designers, operators, universities and consultants were invited to participate in test planning, pre-test predictions and post-test analysis. These efforts were meant to improve the ability to predict containment performance up to and including failure. The report effectively describes the results of these evaluations. The report describes a number of lessons learned and provides detailed recommendations and cautions regarding analytical modeling techniques of containments and their subcomponents.

The tests described in the report took place over a period of 20 years, with containment models increasing over time in scale and level of detailed simulation of critical components (penetrations, hatches, stiffeners, etc.) Although one can infer from the text the main reasons for the increasing complexity of the tests, it would have been valuable to have in the report a more explicit discussion of how lessons learned from tests were utilized to design future tests, to reduce uncertainties and to improve simulation and applicability of results to full scale containments. But it is recognized that such level of detail may have been beyond the intent of this report.

This report does not explicitly address uncertainties and their treatment. There are occasional discussions about uncertainties inherent in containment testing and analysis and sensitivity to certain effects and parameters, but an explicit discussion of uncertainties is lacking. Twenty-five years of research on containment deserve some discussion of uncertainties, or why such discussion is not provided or cannot be provided. The absence of such a section is a major detractor to the value of this report.

Table 3 Summary Results of the ACRS Assessment of the Quality of the Project on Containment Integrity Research at Sandia National Laboratories

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	6.5	0.16	1.04
Identification of major assumptions	5.5	0.09	0.495
Justification of major assumptions	5.0	0.12	0.60
Soundness of technical approach/results	6.0	0.52	3.12
Treatment of uncertainties/sensitivities	4.5	0.11	0.495
		Overall Score:	**5.75**

The consensus scores for this project are shown in Table 3. The score for the overall assessment of the work was found to be 5.75 which is more than satisfactory. Comments and conclusions within the evaluation categories are:

Documentation

- Clarity of presentation **(Consensus score = 6.5)**

  The report is well written and understandable, results are well communicated and explained. The sheer volume of information provided is a challenge to clarity that was well met by the authors.

  The scope of the work that is being summarized is enormous. The accompanying detailed report on the reinforced concrete containment test gives an example of the extraordinary amount of detailed information available on these tests. The presentation overall is very clear and readable. However, at least in the draft that the Committee was reviewing there were some annoying editorial problems. Through most of section 3 (text page 24) the references to figures in the text differed from the actual figure number by 6. In section 4 the references were again consistent.

Sometimes the summary can be somewhat misleading. On page 137 and again on page A–11, the authors state "The element choices must be assessed by verifying that the solutions do not violate fundamental mechanics (for example, force equilibrium), …." In a finite element solution, force equilibrium is never exactly satisfied (because finite element solutions are only approximate), and as the authors correctly point out on page A–20 "a lack of internal element force convergence is not necessarily a good measure of the quality of solution, and in fact, the philosophy of ignoring internal element force convergence (but still enforcing global external force convergence and displacement convergence) has led to many good pretest predictions of containment large scale tests for many years."

Sometimes the presentation is too complete. On page 115 there is a fairly extensive reviews of work done on models for leakage through cracked concrete. It is not until page A–9 that we find out that even for intact concrete there is an enough shrinkage cracks and other defects that for containments with liners (all U.S. containments), the leakage is controlled for all practical purposes by the leakage through the liner.

In the course of the testing, fundamental issues (e.g.: What constitutes "failure?") needed to be defined. Other compromises (e. g. the choice of testing medium, construction details of the test models, lack of temperature and dynamic impulse effects (detonation and deflagration) and construction details (penetrations, hatches, and joints) were discussed but not explicitly treated. Further, the effect of large displacements of the containment on other structures and systems was not treated.

- Identification of major assumptions **(Consensus score = 5.5)**

Major assumptions utilized in the reported tests and analytical simulations are generally identified, discussed and documented.

A number of explicit and implied assumptions arise in the research and these were identified satisfactorily. In some cases, the effect of these simplifications and assumptions were estimated. On other cases, these simplifications were identified but not further evaluated. Considering the scope and limitations of these combined projects, the authors of the report properly identified these issues.

The report is very good at pointing out the limitations and assumptions of the testing program and the capability to analyze containment behavior.

Results Meet Objectives

- Justification of major assumptions **(Consensus score = 5.0)**

  Assumptions are well discussed and justified throughout the report

  Although the major assumptions were discussed in some detail, often the effect of these assumptions upon the analytical and/or test results were not numerically estimated. Many of these assumptions were driven by the practicalities inherent in scale model testing of complex phenomena. Other assumptions were driven by the practicalities and limitations of the analytical modeling tools used. Overall, insights into the effects of these assumptions led to a testing and analytical program that is least likely to be distorted by the effect of these assumptions.

  The report provides good justifications for major assumptions and limitations (e.g., the decision to not include temperature effects in the model test program). There is also a good discussion of how the intended use of an analytical model affects the modeling assumptions that must be made.

- Soundness of technical approach and results **(Consensus score = 6.0)**

  In general, the technical approach was appropriate for this type of study. The program gave reasonable results consistent with intuitive and experiential expectations and produced results that confirm the regulatory expectations of the program.

  While our capability to analyze containment behavior is not yet complete, the work done in the programs at Sandia and in complementary industry programs have provided a wealth of data with which to benchmark analyses of containment behavior. In particular the development and benchmarking of materials models for the behavior of reinforced concrete have been important contributions to the capability to analyze containment behavior.

  The work on steel containments where the failure is controlled by tensile instabilities probably would have benefited from input from processing engineers. Structural engineers tend to focus on load capability. Simple elastic–plastic models work quite well to predict loads. However, in cases where one is interested in the detailed understanding of the ductility of the material, the work done by processing engineers, such as for example, those doing sheet forming would be relevant. It would probably still be impossible to know the geometries and the material properties well enough to compute local failure but such interactions would help better understand the role of triaxiality and strain hardening on the failures.

  The approach taken by the authors is generally appropriate to meet the stated objectives. However, a more explicit discussion of how lessons learned from tests were used to design future tests to reduce uncertainties and to improve simulation and applicability of results to full scale containments would have further enhanced the value of this report.

- Treatment of uncertainties and characterization of sensitivities **(Consensus score = 4.5)**

  This report does not explicitly address uncertainty in the results.

  Although there is no quantitative discussion of uncertainties, there is extensive discussion of the potential limitations of our capability to do containment analyses and what we can compute with relative small uncertainty (gross structural failure) and what we can only compute with large uncertainties (onset and amount of leakage).

  The uncertainties in the results were estimated. Due to the cost of this research and the substantial margin to failure that the results appear to demonstrate, there is probably not a need to define these uncertainties in a more rigorous manner. However, uncertainties that arise from the lack of treatment of temperature effects and dynamic (impulse loading) may be important.

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.

2. Letter Dated November 5, 2005, from William J. Shack, Acting Chairman, ACRS, to Carl J. Paperiello, Director, Office of Nuclear Regulatory research, NRC, Subject: ACRS Assessment of the Quality of Selected NRC Research Projects - FY 2005.

3. National Research Council, *Understanding Risk: Informing Decisions in a Democratic Society.* National Academy Press, Washington, DC, 1996.

4. Apostolakis, G.E., and Pickett, S.E., "Deliberation: Integrating Analytical Results into Environmental Decisions Involving Multiple Stakeholders," *Risk Analysis*, 18:621-634, 1998.

5. Clemen, R., *Making Hard Decisions*, 2nd Edition, Duxbury Press, Belmont, CA, 1995.

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

7. Farmer, M.T., S. Lomperski, D. J. Kilsdonk, and R.W. Aeschlimann, "MCCI Project," OECD/MCCI-2005-TR06, February 28, 2006.

8. Hessheimer, M. F. , and F. A. Dameron, " Containment Integrity Research at sandia National Laboratories," NUREG/CR-xxxx, Draft March 2006.