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

November 25, 2014

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 2014

Dear Dr. Sheron:

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

- Characterization of Thermal-Hydraulic and Ignition Phenomena in Prototypic, Full-Length Boiling Water Reactor Spent Fuel Pool Assemblies After a Postulated Complete Loss-of-Coolant Accident, NUREG/CR-7143
    - This project was found to be satisfactory, a professional work that satisfies research objectives.

- Confirmatory Battery Testing: The Use of Float Current Monitoring to Determine Battery State-of-Charge, NUREG/CR-7148
    - This project was found to be satisfactory, a professional work that satisfies research objectives.

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

We anticipate receiving a list of candidate projects for quality assessment in FY-2015 prior to our February 5-7, 2015 meeting.

Sincerely,

/RA/

John W. Stetkar
Chairman

Enclosure: As stated

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OFFICE	ACRS	SUNSI Review	ACRS	ACRS	ACRS
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DATE	11/19/14	11/19/14	11/19/14	11/25/14	11/25/14

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Assessment of the Quality of Selected NRC Research Projects by the Advisory Committee on Reactor Safeguards - FY 2014

November 2014

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. The ACRS also provides advice on radiation protection, radioactive waste management and earth sciences in the agency's licensing reviews for fuel fabrication and enrichment facilities and waste disposal facilities. On its own initiative, the ACRS may review certain generic matters or safety-significant nuclear facility items. The Committee also advises the Commission on safety-significant policy issues, and performs other duties as the Commission may request. Upon request from the U.S. Department of Energy (DOE), the ACRS provides advice on U.S. Naval reactor designs and hazards associated with the DOE's nuclear activities and facilities. In addition, upon request, the ACRS provides technical advice to the Defense Nuclear Facilities Safety Board.

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

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ABSTRACT

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

- Characterization of Thermal-Hydraulic and Ignition Phenomena in Prototypic, Full-Length Boiling Water Reactor Spent Fuel Pool Assemblies after a Postulated Complete Loss-of-Coolant Accident, NUREG/CR-7143:
    - This project was found to be satisfactory, a professional work that satisfies research objectives.

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CONTENTS

Page

FIGURES

Page

TABLES

ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
ANS	American Nuclear Society
BWR	Boiling Water Reactor
CFR	Code of Federal Regulation
FACA	Federal Advisory Committee Act
FY	Fiscal Year
GPRA	Government Performance and Results Act
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MAUT	Multi-Attribute Utility Theory
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
PWR	Pressurized Water Reactor
RES	Office of Nuclear Regulatory Research
SFP	Spent Fuel Pool
SNL	Sandia National Laboratories
SOW	Statement of Work
U.S.	United States

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 (FY) 2004, the Advisory Committee on Reactor Safeguards (ACRS) has been assisting RES by performing independent assessments of the quality of selected research projects [1-10]. The Committee 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 a maximum of four projects for detailed review during the fiscal year.

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

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

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

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

Based on our later discussions with the RES, the ACRS made the following enhancements to its quality assessment process:

- After familiarizing itself with the research projects selected for quality assessment, each panel holds an informal meeting with the RES project manager and representatives of the User Office to obtain an overview of the project and the User Office's insights on the expectations for the project with regard to their needs.

- In addition, if needed, an additional informal meeting would be held with the project manager to obtain further clarification of information prior to completing the quality assessment.

The purposes of these enhancements were to ensure greater involvement of the RES project managers and their program office counterparts during the review process and to identify objectives, user office needs, and perspectives on the research projects.

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, the 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 the 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:

- Characterization of Thermal-Hydraulic and Ignition Phenomena in Prototypic, Full-Length Boiling Water Reactor Spent Fuel Pool Assemblies after a Postulated Complete Loss-of-Coolant Accident

- Confirmatory Battery Testing: The Use of Float Current Monitoring to Determine Battery State-of-Charge

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 the 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 [11-12]. The analytical part utilizes methods of multi-attribute utility theory (MAUT) [13-14] 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.

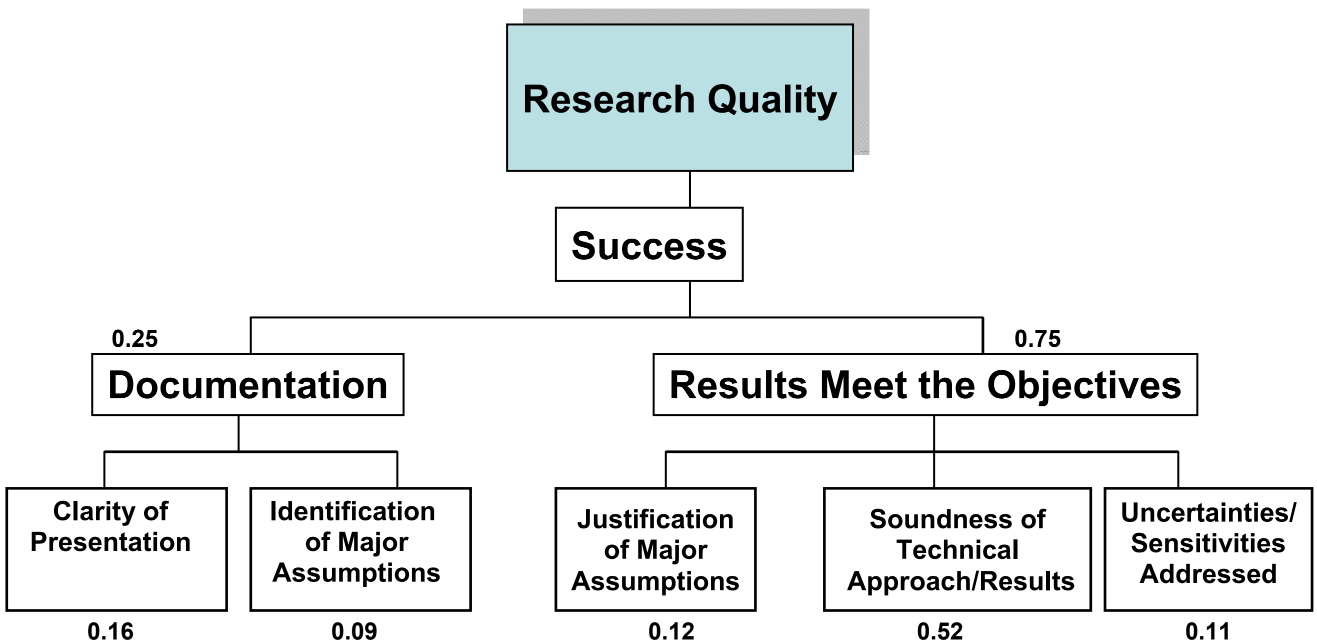

Figure 1. The value tree used for evaluating the quality of research projects

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.

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 Characterization of Thermal-Hydraulic and Ignition Phenomena in Prototypic, Full-Length Boiling Water Reactor Spent Fuel Pool Assemblies after a Postulated Complete Loss-of-Coolant Accident

All nuclear power plants have a spent fuel pool (SFP) in which used reactor fuel assemblies are allowed to cool before being transferred to dry storage. These pools are robust constructions made of reinforced concrete several feet thick, with steel liners. The water is typically about 40 feet deep, and serves both to shield the radiation and cool the fuel rods. The SFP structures have been assessed to have a low likelihood of a complete loss of coolant under traditional accident scenarios. However, in the wake of the terrorist attacks of September 11, 2001, SFP accident progression was reevaluated using system level computer codes.

In 2001, the NRC staff performed an evaluation of potential SFP accident risk at decommissioned plants in the United States. This evaluation is documented in NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants." Some assumptions in the accident progression were known to be conservative, especially fuel damage estimates. The NRC expanded SFP accident research by applying system level computer codes to predict severe accident progression following various postulated accident initiators in SFPs of operating plants. These code calculations identified various modeling and phenomenological uncertainties. The NRC initiated an experimental and analysis program at Sandia National Laboratories (SNL) to address thermal-hydraulic issues associated with complete loss-of-coolant accidents in SFPs of boiling water reactors (BWRs). One objective of this program was to simulate accident conditions of interest for the SFP in a full-scale prototypic fashion (electrically-heated, prototypic assemblies in a prototypic SFP rack). A major impetus for this work was to facilitate code validation (primarily the MELCOR computer code) and reduce modeling uncertainties. The results of this program are documented in NUREG/CR-7143, "Characterization of Thermal-Hydraulic and Ignition Phenomena in Prototypic, Full-Length Boiling Water Reactor Spent Fuel Pool Assemblies after a Postulated Complete Loss-of-Coolant Accident" [15].

As shown in Table 2, SNL used a phased approach with three basic types of experiments to complete this program. As a proof of concept, two heater design tests were first performed to determine the suitability of the electrically heated, Zircaloy-clad spent fuel rod simulators. Next, three separate effects tests were conducted to study and understand specific phenomena independently. In the separate effects tests, the experiments were designed to investigate a specific heat transfer or flow phenomenon such as thermal radiative coupling or induced natural convective flow. These tests were non-destructive and involved some non-prototypic materials (e.g., stainless steel and Incoloy). Finally, two prototypic assembly configurations were heated to ignition in the integral effects test series. The phased approach for this experimental program involved testing, measurement, and analytical evaluation of results to be applied to improve subsequent work in the overall program. This approach resulted in several required improvements to the experimental design and operation of the integral effects tests. It also improved the interpretation of the testing results and of the code analysis predictions.

Three configurations of fuel assemblies were developed for these different testing phases. The first two heater design tests, which were conducted using a 12 Zircaloy-clad rod bundle configuration, demonstrated that Zircaloy ignition could be achieved when an appropriate test design was used that minimized heat loss and maximized gas pre-heating and bundle power. The second test configuration was a single, full-length 'highly prototypic' BWR 9x9 fuel rod assembly used to measure thermal-hydraulic response and determine appropriate loss coefficients as a function of bundle mass flow under adiabatic conditions. The third configuration used for integral effects testing was (1) a single full-length assembly and (2) five Zircaloy partial (1/3) length assemblies in a 3x3 pool rack. This short array of assemblies was designed to simulate the power profile and performance in a slice from the middle to upper portion of an array of full-length assemblies.

Table 2. Test Elements Evaluated in the NUREG/CR-7143 Experimental Program

Test Description	Purpose	Assembly	Rod Cladding Material
Heater Design	Electrical heater performance - Obtain preliminary Zircaloy fire data, conducted at normal and reduced oxygen concentrations	12 rod bundle	Zircaloy
Separate Effects	Hydraulics – Determine viscous and form loss coefficients for laminar volumetric flowrates	Prototypic	Stainless Steel
	Thermal hydraulics – Determine input conditions for partial length experiments	Prototypic	Incoloy
	Thermal radiation – Determine radiation coupling in a 1x 4 arrangement	Prototypic – Partial length	Incoloy
Integral Effects	Axial Ignition – Determine temperature profiles, induced flow, axial oxygen profile, nature of fire	Prototypic - Single full length assembly	Zircaloy
	Radial Propagation – Determine nature of radial fire propagation in a 1 x 4 arrangement	Prototypic – Partial length	Zircaloy

General Observations

A unique aspect of this project was the deliberate close coupling of the experiments with computer code analysis. This project demonstrates the benefits of carefully planned and staged experiments that are coupled with detailed pre- and post-analytical evaluations for each test. The primary system computer code used was the severe accident code MELCOR. At each step in the experimental program, MELCOR was used (1) as a tool for the experimental design, (2) for the pre-test results prediction, and (3) for post-test analysis of the calculated and measured responses. The post-test analysis helped identify and assess important response parameters, which often improved the conduct of subsequent testing and enhanced the modeling approach.

The primary objective, as stated in NUREG/CR-7143, is to document results from an experimental test series examining the heat up and oxidation of a BWR spent fuel assembly as well as the associated pre- and post-test modeling. To achieve this objective, researchers identified and incorporated measures into initial tests to ensure that Zircaloy cladding ignition temperatures were attained in each test protocol. Additional objectives were accomplished and documented. The authors resolved several unexpected technical challenges related to thermocouple attachment, the choice of appropriate input power in the heated design test that would cause ignition, and the addition of a heater on the bottom plate to reduce unwanted cooling of gas entering the assembly. This is the first project to provide thermal-hydraulic data to support detailed MELCOR analysis of BWR spent fuel pool assemblies during a complete loss-of-coolant accident. The experiments and the computer modeling were integrated to assist and to improve the related MELCOR analyses for the broader spent fuel pool research program. In the final integral test, an 'untuned' MELCOR model predicted ignition in the center and peripheral assemblies to within 30 and 15 minutes, respectively. The error in ignition timing between the simulations and experiment is approximately 10%. The investigators attributed the difference in timing to the inability of the lumped parameter approach used in MELCOR to account for steep radial temperature gradients.

The consensus scores for this project are shown in Table 3. The score for the overall assessment of this work was evaluated to be 5.4 (satisfactory, a professional work that satisfies research objectives).

Performance Measures	**Consensus Scores**	**Weights**	**Weighted Scores**
Clarity of presentation	5.3	0.16	0.85
Identification of major assumptions	5.0	0.09	0.45
Justification of major assumptions	4.5	0.12	0.54
Soundness of technical approach/results	5.8	0.52	3.03
Treatment of uncertainties/sensitivities	4.3	0.11	0.48
		Overall Score	**5.4**

Table 3. Summary Results of ACRS Assessment of the Quality of the Project, "Characterization of Thermal-Hydraulic and Ignition Phenomena in Prototypic, Full-Length Boiling Water Reactor Spent Fuel Pool Assemblies after a Postulated Complete Loss-of-Coolant Accident"

Comments and conclusions within the evaluation categories are provided below.

Clarity of Presentation *(Consensus Score: 5.3)*

This report is well organized and well written. We commend the authors for their clear description of the experimental setup with photographs, diagrams, and tables that illustrate the experimental setup and placement of instrumentation.

The document could have been improved in certain areas. For example, although the program was a well designed set of experiments, the results presentation appears as a collection of 'reports' (as evidenced from the first sentence in Sections 3 through 8). Editing the document to emphasize the connections between experimental designs would have presented a better integrated report without unnecessary repetition of common experimental or analysis features. Some items that would have improved the clarity of this document include:

- Selection of a consistent set of units (e.g., English, metric, or dual units).

- A final edit to prevent grammatical errors, incorrect figure citations, inappropriate significant figures, and undefined acronyms and symbols.

  References should be provided for items, such as "Hottel's crossed-string method," the "Nertz Equation," RADGEN, and COBRA-SFS.

- Since "Inconel" and "Incoloy" refer to patented alloys, identification of the alloy of Inconel or Incoloy (e.g., Inconel 600, Inconel 625, Incoloy Alloy 800, Incoloy 825, etc.) used and why the type was selected.

Identification of Major Assumptions *(Consensus Score: 5.0)*

The authors identified major assumptions for each of the separate-effects and integral test series and discussed the associated limitations in the experimental setup and associated modeling choices. In particular, the authors noted their primary assumptions were that integral and separate-effects tests adequately represented prototypic scale effects (e.g., number of rods, number of assemblies, and length of fuel assemblies in truncated experiments).

The report could have been improved if the authors had identified other important assumptions, such as the following:

- The selected nodalization for MELCOR and COBRA was adequate to represent the phenomena in these experiments.

- The effects of selected phenomena, such as a center–peaked axial power profile in the fuel assemblies and power in peripheral assemblies, can be analytically addressed and not affect modeling validation efforts. For example, if peripheral assemblies had been heated, would self-sustaining ignition occur?

Justification of Major Assumptions *(Consensus Score: 4.5)*

As noted by the authors, a major limitation of this test series is the inability to repeat more complex and expensive tests, especially those in which the experimental apparatus was destroyed. The authors justify the assumption that their approach is adequate by stressing the importance of using a phased testing approach that is closely coupled with test analysis. Many assumptions in the document are provided with no references to support them (the authors only included four references in this report). Our review identified the following assumptions associated with the analysis that could be better justified:

- State the author assumptions about the suitability of the MELCOR computer code for modeling the various phenomena. As noted by the authors, the steep radial gradients in the peripheral assemblies and observed differences between measured temperatures and MELCOR predicted values call into question the validity of the MELCOR lumped thermal analysis approach for this particular application.

- The assumption that it was appropriate to use different Zircaloy and stainless steel emissivity values in the MELCOR and COBRA-SFS analyses. In Figure 8.28, the authors present the assumed Zircaloy emissivity with a high temperature correction and indicate that this differs from the default emissivity in the MELCOR code. It would have been appropriate for the authors to provide some insights about the experimental bases for the assumed MELCOR correlation (especially for the high temperature correction which apparently differs from the 'default' MELCOR correlation). The justification for emissivity seems especially important when one considers that the authors identified radiation heat transfer as the most important heat transfer mechanism prior to ignition.

Soundness of Technical Approach/Results *(Consensus Score: 5.8)*

We commend the authors for applying a systematic, phased experimental program that allowed them to detect and address experimental issues in less complicated and less expensive tests before conducting larger, integral tests. As indicated in Table 2, Sections 2 through 8 of the document present the necessary information on the various types of tests, including the test assembly design, the test procedure history, test results, and additional assumptions that must be made in order to compare data with predictions from MELCOR (as well as COBRA-SFS, in certain cases).

While researchers were able to successfully complete this complex experimental program and satisfy the project objectives, we identified additional items that could enhance this report:

- The authors present comparisons between results from the experiments and results obtained from COBRA-SFS and MELCOR, but do not discuss why COBRA-SFS predictions more closely match data than MELCOR predictions for some parameters (e.g., Figures 5.17 and 5.18) and MELCOR predictions are closer to the data than COBRA-SFS in other cases (e.g., see Figure 5.16). The general conclusions of the separate effects thermal-hydraulic testing are reasonable. However, except for the hydraulic input information derived for MELCOR modeling, the authors do not provide additional insights for these model comparisons or recommendations for improvement.

- Differences in ignition time predictions versus data were attributed to the inadequacies in a new Zircaloy oxidation kinetics model. Results from a sensitivity calculation led the authors to postulate that these differences were due to the reaction rate and the inability of the MELCOR model to capture the radial temperature gradient in peripheral assemblies. However, the authors did not perform any additional tests (with additional heating of the peripheral assemblies to offset the radial temperature gradient) or provide analyses to address discrepancies between the predicted and observed peripheral assembly ignition times.

Treatment of Uncertainties/Sensitivities *(Consensus Score: 4.3)*

In this experimental program, there are uncertainties associated with repeatability, the adequacy of the setup to represent phenomena of interest at prototypic conditions, and instrumentation measurement uncertainties. Although some tests were repeated to demonstrate consistency in test results, this was not possible for destructive testing or thermal testing where instrumentation failed. Therefore, evaluated measurement uncertainty relies primarily on the instrument capabilities reported by manufacturers or suppliers. No integrated assessments of the accuracy of the derived experimental uncertainties were performed. Instrumentation attributes incorporated in the derivation of this uncertainty are described in detail and quantified using accepted practices for instrument uncertainty computation in Appendices B and C. In addition, the authors discuss differences in computed values and measured results.

Our review suggests that this aspect of the research would be improved if the following items were included:

- Discussion of the uncertainties introduced by assuming the experimental setup is representative of BWR fuel assemblies in a spent fuel pool.

- Discussion of the uncertainty introduced when sensors were subjected to conditions outside their operating envelope.

- Explanation of why differences between code predictions and data are larger than values estimated in the Appendix B error analysis.

- An uncertainty analysis, using the model to quantify what fraction of the differences could be attributed to uncertainties in the input parameters and what fraction was due to modeling error. This was likely outside the scope of this work but would have been insightful for parameters such as the breakaway oxidation and the associated ignition threshold in the Zircaloy clad fuel assembly integral effects tests.

- 3.2 Confirmatory Battery Testing: The Use of Float Current Monitoring to Determine Battery State-of-Charge

Nuclear power plant electrical batteries play an essential role in maintaining the ability of plants to control and monitor operations. Typical nuclear power plant Technical Specifications require the measurement of specific gravity to determine the state-of-charge of the batteries, based on Regulatory Guide 1.129 Rev.1 and IEEE Std. 450-1975. A more recent version of this standard, IEEE Std. 450-2002, suggests that either float charging current or specific gravity could be used for determining a vented lead-calcium battery's state-of-charge.[1] Thus, the primary objective of this research project was to determine whether float current monitoring is a useful indicator for determining a vented lead-calcium battery's state-of-charge. A secondary objective was to evaluate the criteria for selecting the point when a battery can be returned to service and meet its design requirements.

The NRC sponsored a testing program at Brookhaven National Laboratory. Three sets of nuclear qualified batteries were procured from three battery vendors. Each battery set consisted of 12 battery cells. These cells are the same models that are typically used in a Class IE dc system application. Two suitably sized battery chargers and a load bank were used in the tests. The test setup was similar to a typical nuclear power station's Class 1E battery design. The testing program used a series of 4-hour performance tests to validate this approach. Comparisons were made of the recharge/float current and the specific gravity responses as the cells were charged following the four-hour performance test. These test results are only applicable to vented lead-calcium batteries. The primary finding of the study is that both float current and specific gravity provide adequate means to determine battery state-of-charge.

The results of this effort are documented in NUREG/CR-7148, "Confirmatory Battery Testing: The Use of Float Current Monitoring to Determine Battery State-of-Charge" [16].

General Observations

The report is very well written. The objectives were clearly stated; the testing process was carefully described; and the results were concisely presented. The results and the analyses presented are adequate with regard to meeting the stated objectives, but only limited insights have been derived from the data. For example, the development of an alternative return to service protocol (based on three time constants) was tested, but was only presented as an approach that "may be a more practical method" than the IEEE standard criterion of three hours of stable float current. In fact, should time be essential, this approach could allow return to service in as little as half the time required by the method of the standard.

The consensus scores for this project are shown in Table 4. In summary, this was good professional work that satisfies the research objectives, but did not demonstrate the kinds of innovation and insight that deserve scores of excellent or outstanding as describe in Section 2.

Comments and conclusions within the evaluation categories are provided below.

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[1] Although not explained in the contractor's report, NRC staff advised us that they found the basis cited by the standard inadequate for nuclear power plant regulation, which indicated a need for additional research, before adopting the float charging current approach for operating nuclear power plants.

Table 4. Summary Results of ACRS Assessment of the Quality of the Project, "Confirmatory Battery Testing: The Use of Float Current Monitoring to Determine Battery State-of-Charge"

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	7.0	0.16	1.12
Identification of major assumptions	3.7	0.09	0.33
Justification of major assumptions	3.0	0.12	0.36
Soundness of technical approach/results	6.7	0.52	3.48
Treatment of uncertainties/sensitivities	3.0	0.11	0.33
		Overall Score	**5.6**

Clarity of Presentation (*Consensus Score*: 7)

The report was a pleasure to read. The text was clearly written and easy to follow. The discussion and figures confirmed the objectives that were specified. The presentation could have been improved, if graphs or tables had highlighted the time difference between the stable float current and the three time constant protocol for determining return to service of the battery. Also, as part of the work on the secondary objective, an alternative return to service (three time constant) protocol was explored, but there was insufficient quantitative discussion of its advantages and disadvantages. The test equipment and testing protocol were well described, but the batteries themselves were only identified by the brand name and model numbers for the cells. While one could research the technical specifications for the cells, if the report were to stand alone, it would have been useful to have more information regarding what was being tested. With regard to the presentation of the results, it would have been useful to have had more discussion, e.g., information regarding the energy extracted in each discharge test in addition to the voltage-time plot. As another example, the tabulated results (e.g., in Table 3-1) are presented without much comment, though there are wide variations in the recharge time.

Identification of Major Assumptions (*Consensus Score*: 3.7)

No major assumptions were stated. Some of the implicit assumptions should have been made explicit. For example, an implicit assumption was that the testing could usefully emulate what would happen in power-plant periodic testing over years, i.e., could the tests produce useful results in spite of aging effects like corrosion. Similarly, there was an implicit assumption about the applicability of repeated testing before the electrolyte density stratification equilibrated, as it would in actual plant tests.

Justification of Major Assumptions (*Consensus Score*: 3.0)

There was no explicit justification of major assumptions. However, there was some discussion of the implicit assumptions.

Soundness of Technical Approach/Results (*Consensus Score*: 6.7)

The technical approach was sound with regard to addressing the stated objectives. More could have been done with the results as discussed earlier; and the technical approach could have encompassed analysis of several interesting observations, such as the large variations in recharge time and the return-to-service time advantage of the three time constant approach.

Treatment of Uncertainties/Sensitivities (*Consensus Score*: 3.0)

There was no explicit treatment of uncertainties, either the statistical uncertainty in the experimental results or the uncertainty in extrapolating these results to batteries in service in all nuclear plants and regulatory decisions that might be based on these results.

The replicated tests developed the necessary database that would enable an analysis of the uncertainty in the test results. However, the report only showed data and figures from representative test cycles that we are told are consistent with the other tests. Neither figures displaying the range of results in the separate tests nor statistical analyses of the test results are presented. In particular, it would have been of interest to analyze the data with regard to the recharge times.

The authors should have identified areas where batteries operating in real power plants might behave differently than those in the tests. They did state that aging effects would not be studied, but did not discuss the ways, if any, in which age could affect the results.

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9. Letter Dated October 22, 2012, from J. Sam Armijo, Chairman, ACRS, to Brian Sheron, Director, Office of Nuclear Regulatory Research, NRC, Subject: ACRS Assessment of the Quality of Selected NRC Research Projects - FY 2012.

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