Assessment of the Quality of Selected NRC Research Projects by the Advisory Committee on Reactor Safeguards - FY 2019

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ABOUT THE ACRS

The Advisory Committee on Reactor Safeguards (ACRS) was established as a statutory Committee of the Atomic Energy Commission (AEC) by a 1957 amendment to the *Atomic Energy Act* of 1954. The functions of the Committee are described in Sections 29 and 182b of the Act. The *Energy Reorganization Act* of 1974 transferred the AEC's licensing functions to the U.S. Nuclear Regulatory Commission (NRC), and the Committee has continued serving in 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. Navy reactor designs and hazards associated with 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*, which is implemented through NRC regulations at Title 10, Part 7, of the *Code of Federal Regulations*. 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:

- Overview of Nuclear Data Uncertainty in SCALE and Application to Light Water Reactor Uncertainty Analysis, NUREG/CR-7249, ORNL/TM-2017/706

    - This project was found to be satisfactory. With minor limitations, the results meet the research objectives.

- Response of Nuclear Power Plant Instrumentation Cables Exposed to Fire Conditions, NUREG/CR-7244, SAND2017-10346R

    - This project was found to be more than satisfactory, a professional work that satisfies research objectives

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ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
CE	Combustion Engineering
DOE	Department of Energy
ENDF	evaluated nuclear data file
FACA	Federal Advisory Committee Act
FY	fiscal year
HRA	Human Reliability Analysis
LWR	light water reactor
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PRA	Probabilistic Risk Analysis
RES	Office of Nuclear Regulatory Research
TP	thermoplastic
TS	thermoset
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. 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 [1-15]. 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 ACRS submits an annual summary report to the RES Director.

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

- After familiarizing itself with the research project 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 is 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 decision-making framework was adopted for evaluating the quality of NRC research projects. The definition of quality research adopted by the ACRS 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:

- NUREG/CR-7249: Overview of Nuclear Data Uncertainty in SCALE and Application to Light Water Reactor Uncertainty Analysis

- NUREG/CR-7244: Response of Nuclear Power Plant Instrumentation Cables Exposed to Fire Conditions

These 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 QUALITY OF RESEARCH PROJECTS

To guide its review of research projects, the ACRS has adopted an analytic/deliberative methodology [16-17]. The analytical part utilizes methods of multi-attribute utility theory [18-19] 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"), with weights reflecting their relative importance. The value tree and the relative weights developed by the Full Committee are shown in **Figure *1***.

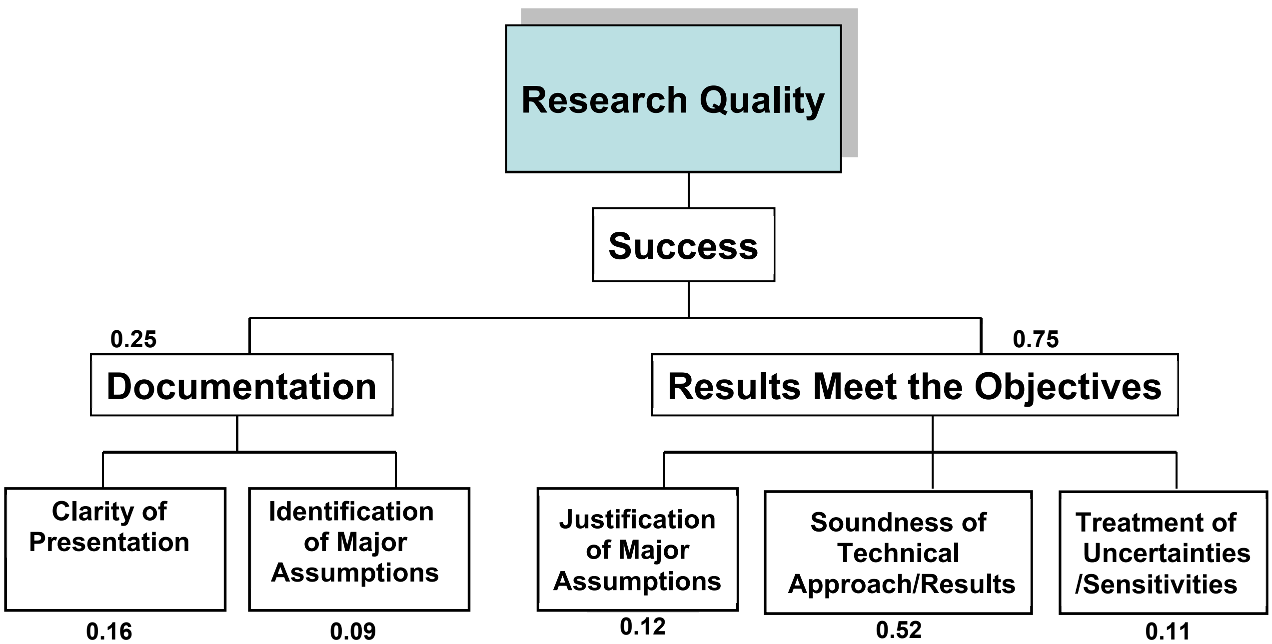

Figure 1. The value tree used for evaluating the quality of research projects

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 was 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 to four 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 Overview of Nuclear Data Uncertainty in SCALE and Application to Light Water Reactor Uncertainty Analysis (NUREG/CR-7249, ORNL/TM-2017/706)

The objective of this project was to evaluate the current state of the art in the SCALE code suite for nuclear data uncertainty analysis capability. It also provides an overview of the uncertainty in multigroup cross sections, fission product yields, and decay data. The results of this effort are documented in NUREG/CR-7249 report [20].

The nuclear data used in a typical SCALE calculation are processed from evaluated nuclear data files (ENDF/B) produced by the National Nuclear Data Center using the SCALE AMPX code system. The ENDF/B data are based on a large set of nuclear data measurements that have been evaluated by nuclear data experts and compiled into a cohesive, consistent database. Ideally, ENDF/B data should include an associated uncertainty indicating the accuracy of the measurement, as well as any uncertainty introduced in the evaluation due to modeling fits, consistency adjustments, or other causes. This uncertainty in the nuclear data can be propagated to uncertainty in calculated quantities of interest for the nuclear analyst; for example, k_{eff} for criticality safety applications or a void coefficient of reactivity for reactor physics applications. However, ENDF/B uncertainty information is not available for many nuclides or for some types of nuclear data relevant to nuclear engineering applications. To address this need, supplemental data have been developed within SCALE to provide complete uncertainty data sets for fission yield data, multigroup cross section data, and decay data.

The authors evaluate the effect of nuclear data uncertainty using a typical light water reactor (LWR) depletion analysis problem involving a Combustion Engineering (CE) 14 × 14 assembly irradiated in Calvert Cliffs Unit 1. A single fuel rod from assembly D047, designated as MKP109, has been subjected to destructive radiochemical assay to measure the isotopic contents and was simulated with SCALE to evaluate its accuracy.

<u>General Observations</u>

The authors introduce the relevant model equations used to explicitly define the data parameters, discuss the background of uncertainty quantification, and describe the sampling-based propagation technique as implemented in SCALE. The report describes the uncertainty data in SCALE, presents an application uncertainty problem in LWR analysis, provides general recommendations for interpreting results with uncertainty, and provides recommendations for future work.

SCALE includes two uncertainty qualification methods: a conventional sampling method, 'Sampler'; and a perturbation theory–based method, which requires an adjoint calculation and is limited in practice to estimate only k_{eff} uncertainty. The report analysis used the Sampler methodology.

The main application of the analysis is related to a calculation of nuclear parameter uncertainty for multiple axial locations of rod MKP109 in Calvert Cliffs Unit 1 reactor assembly D047, a CE 14 ×14 design with known operating history and high-precision radiochemical assay data available.

The authors conclude that uncertainties in the macroscopic cross sections, reactivity, and power distributions are generally low, in the few percent range. The SCALE results show a high uncertainty for the effective delayed neutron fraction, β_{eff}, which is not necessarily an unexpected result but is not consistent with integral benchmarks of power-excursion experimental data.

The consensus scores for this project are shown in Table 2. The score for the overall assessment of this work was found to be satisfactory (a 4.9 score), a professional work that satisfies its research objectives with some limitations. However, the report does not clarify how their research results will be applied. More detailed comments and conclusions within the evaluation categories are provided below.

Table 2. Summary Results of ACRS Assessment of the Quality of the Project, "Overview of Nuclear Data Uncertainty in SCALE and Application to Light Water Reactor Uncertainty Analysis"

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	5.7	0.16	0.9
Identification of major assumptions	5.3	0.09	0.5
Justification of major assumptions	4.3	0.12	0.5
Soundness of technical approach/results	4.8	0.52	2.5
Treatment of uncertainties/sensitivities	4.5	0.11	0.5
		Overall Score	4.9

<u>Clarity of Presentation (Consensus Score: 5.7)</u>

The report is well-organized and well-written. The objective of the report was to document results from SCALE analyses regarding the uncertainty on calculated nuclear data. That objective was achieved.

The authors' description of the SCALE models is thorough and adequate. The description of the input data uncertainty is categorized by cross section, fission yield, and decay uncertainties. The methodology used for each category is clearly described in the report. Their presentation and explanation of their data are thorough. However, the Introduction appears to be a repeat of the SCALE manual on transport theory and methods, which may have been addressed with a reference. Chapters 2 and 3 have some interesting data on isotopic uncertainties; however, the large calculation uncertainty (or unexpected bias) includes some elements that are used only in calculations for burnable poisons (Eu and Gd) and fission product poisons (Sm).

In general, the report is adequately illustrated. The tables and figures are easily readable, properly labeled, and easily provide the intended information. The report is complete and informative; it avoids being unduly long by avoiding excessive detail. This contributes to the overall readability, and a non-expert reader can understand the information and major conclusions.

Identification of Major Assumptions (Consensus Score: 5.3)

The authors identify adequately all major assumptions in their analysis. All sources of input data uncertainty are identified and described. The methodology and its limitations are described in the report.

Justification of Major Assumptions (Consensus Score: 4.3)

The authors describe and document their assumptions, but do not attempt to justify them. The main assumptions deal with the value of the uncertainty of the basic input parameters. The authors simply state the source of the value used (either ENDF/B or previous SCALE studies) but did not attempt to review or identify deficiencies in this input data.

In particular, one of the main conclusions of the report is the anomalous uncertainty on the delayed neutron fraction, β_{eff}, which can be as large as 20 to 100% for some cases. The authors call this large error unexpected, but it was reported earlier [21]. The results, however, are counterintuitive because benchmarks against multiple integral experiments where β_{eff} is a key parameter reproduce the power excursions. A possible explanation is that the uncertainty for decay input data is lacking covariance terms that correlate the error in different decay groups. It is possible that the reported data errors on each decay group are conservatively bounded in the database without accounting that when group "i" yield is high, the group "j" yield may be low (also known as covariance terms). The authors did not attempt to justify the use of these data. During the informal meeting with the ACRS evaluation panel, the authors agreed that the results may be contrary to experimental integral benchmark data. However, the authors did not investigate their finding and simply reported their calculated values.

Soundness of Technical Approach/Results (Consensus Score: 4.8)

The technical approach and the design of this study are sound and adequate. SCALE has built-in features to generate uncertainty qualification – either by using adjoint-based perturbation theory or brute-force sampling. The authors used these features to perform their analysis and thoroughly documented their results.

The authors completed a comprehensive effort in which they evaluated the propagation of uncertainties in SCALE input. The effort considered uncertainties in basic input data (from either ENDF/B or previous SCALE data) to nuclear parameters of interest (including cross sections and isotopic composition for prompt and delayed neutrons).

The authors properly identified the unusually high calculated β_{eff} uncertainties. However, the multitude of benchmarks on integral tests that suggest that the β_{eff} uncertainty is not as high as suggested, requires further investigation by the authors. At a minimum, the authors should have suggested areas for future work to identify deficiencies in this area. Especially considering that the magnitude of errors in β_{eff}, should they be accurate, could have serious safety consequences for reactivity insertion accidents.

The analysis in Chapter 3 of the NUREG/CR-7249 report is somewhat perfunctory. The major result (the high uncertainties discovered in β_{eff}) is only briefly discussed in Chapter 5 of the NUREG/CR-7249 report, but implications and importance are not expanded upon. This is a concern because, especially for advanced reactor concepts wherein β_{eff} and kinetic feedback is critical, a 20 to 100% error in β_{eff} could be a major issue. The authors recognize the error, but do not expand on the implications of the issue.

The authors considered areas for future work. However, they approached only future code modifications to SCALE and not the impact of their results on safety-related confirmatory calculations. The staff should have considered this impact when recommending future activities. In particular, the impact of the calculated β_{eff} uncertainty and its possible sources of error should be considered.

Treatment of Uncertainties/Sensitivities (Consensus Score: 4.5)

The objective of the report was to identify and document the uncertainty of SCALE calculations of nuclear data used for LWR safety calculations. The objective was accomplished, and these uncertainties are the main output of this effort. However, the authors failed to address the uncertainty in their calculations. At a minimum, they should have performed sensitivity studies to identify the source of the possibly-anomalous β_{eff} uncertainty to shed light on future activities.

3.2 Response of Nuclear Power Plant Instrumentation Cables Exposed to Fire Conditions (NUREG/CR-7244, SAND2017-10346R)

This report presents the results of a series of small-scale instrumentation cable tests sponsored by NRC and performed at Sandia National Laboratories (SNL). These tests were performed as a follow up to instrumentation cable fire tests done in 2001, by the Nuclear Energy Institute (NEI) and the Electric Power Research Institute (EPRI). Both test series were designed to address specific issues of signal degradation prior to complete loss of signal. The 2001 tests observed significant differences between the failure of the thermoplastic (TP) and thermoset (TS) cables: TP cables generally displayed no characteristics of signal degradation prior to complete loss of signal, while TS cables displayed a substantial amount of signal degradation. In the worst case, the signal degradation lasted close to ten minutes prior to the total signal loss. Based on these observations, it could be postulated that a fire affecting a TS cable could lead to some misleading instrument indications potentially causing operators to take an action based on faulty information. On the contrary, a fire affecting a TP cable would be far less likely to mislead operators who are likely to diagnose the instrumentation failure.

The conclusions made in the 2001 study were based on a limited series of testing. To obtain better insights into this type of failure behavior, a wider variety of cables and circuit configurations were tested, and the results of these tests are presented in the NUREG/CR -7244 report [22].

The goal of the NUREG/CR-7244 test series was to assess how instrumentation cable fire exposure, simulated by severe radiant heating, affects current or voltage signals in an instrumentation circuit. A total of thirty-nine tests were conducted, on ten different instrumentation cables, ranging from one conductor to eight-twisted pairs. Based on the 2001 study, the NUREG/CR-7244 series focused on TS cables: eight of ten cables had TS insulation and jacket material; only two cables had TP insulation and jacket material. Two instrumentation cables from previous cable fire testing were included, one TS and one TP. Three test circuits were used to simulate instrumentation circuits present in nuclear power plants: a 4–20 mA current loop, a 10–50 mA current loop and a 1–5 VDC voltage loop.

The main objectives of the work presented in this report could be summarized as follows:

1) To follow up on 2001 tests by performing tests on a wider variety of cables and circuit configurations, and reexamining conclusions from the previous tests
2) To comprehensively document the bench-scale testing efforts, and to identify focus areas for further testing necessary to address the research goals
3) To provide better understanding of the fire-induced failure modes of instrumentation cables and to use a regression analysis to determine key variables affecting signal leakage time.

<u>General Observations</u>

The results from the tests presented in this report did not support the conclusions from the 2001 tests:

- *2001 conclusion that TP cables had no signal leakage characteristics prior to signal loss*: During this series of testing, TP cables were found to have a smaller leakage time when compared with TS cables and most TP cables failed instantly; however, one TP test had a leakage time of 2.6 minutes. Therefore, TP cables may have some signal degradation prior to failure.
- *2001 conclusion that TS cables displayed some amount of signal leakage before the signal failed*: During this series of testing, twelve out of the thirty-two TS tests had less than one minute of signal leakage before failure and four of these tests experienced no signal leakage. Four other tests had signal leakage for longer than ten minutes. Therefore, TS cables may not always experience signal leakage before failure.

Testing efforts and results are comprehensively documented; potential focus areas for further testing are identified, including extending tests to full scale, to different configurations, positions, testing heating impact on digital transmitters and receivers, testing a larger variety of TS and especially, TP cables.

A regression analysis was performed on the test data to determine key variables that contributed to longer leakage times. Nine variables were considered, in addition to TS or TP insulation/jacket material, they include manufacturer, number of conductors, circuit type, presence or absence of shielding, and circuit fuse or grounding. The fitted model coefficient methodology concluded that there is a significant relationship between the number of conductors and the signal leakage time.

A final goal of such testing would be to better understand fire-induced failure modes of instrumentation cables and to try to determine an impact of these failures on Human Reliability Analysis (HRA) and Fire Probabilistic Risk Assessment (PRA). If a signal does not fail immediately, it could cause indicators to display an intermediate, but not obviously erroneous value due to signal degradation. This misleading information could cause operators to take a wrong action based on a faulty display. The report recommends performing Fire PRA and HRA evaluation in order to analyze potential impacts of a long signal delay to operators.

The consensus scores for this project are shown in Table 3. The score for the overall assessment of this work was found to be 6.0 (above satisfactory; slightly higher than the grade of a professional work that satisfies its research objectives). Our main conclusion is that the report very clearly presents this testing project, results and underlying insights. More detailed comments and conclusions within the evaluation categories are provided below.

Table 3 Summary Results of ACRS Assessment of the Quality of the Project, "Response of Nuclear Power Plant Instrumentation Cables Exposed to Fire Conditions”

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	7.0	0.16	1.12
Identification of major assumptions	5.0	0.09	0.45
Justification of major assumptions	5.0	0.12	0.60
Soundness of technical approach/results	6.3	0.52	3.28
Treatment of uncertainties/sensitivities	5.0	0.11	0.55
		Overall Score	6.0

Clarity of Presentation (Consensus Score: 7.0)

When considering the main goal of this report, to provide a follow up on 2001 tests by performing tests on a wider variety of cables and circuit configurations, it could be concluded that this goal is fully achieved by comprehensively documenting the new testing efforts and results. The report is well written, organized and easy to follow.

The document is composed in a logical manner with the scope of each section clearly defined. In Section 2 the report provides background of instrumentation circuits, configurations, and types of failure, which enhanced the clarity of the report. The detailed test process is clearly described in Section 3 and Appendix C, including test facilities, protocols, the actual instrument loop circuits, and the cables to be tested, with description of their characteristics. Figures and photographs of test setups were incorporated for clarity. A complete test matrix is provided in Section 4 with tables showing the specific cable types, circuit configuration used, thermal exposure to be applied and dates completed. The test results are presented in Section 5 and Appendix A. Additional clarity is provided by including test result graphics in Appendix C.

It is very clearly illustrated in this report that the new test results do not support main conclusions from the 2001 tests; namely, that TP cables may have some signal degradation prior to failure, and that TS cables may not always experience signal leakage before failure.

The report could have been improved if it had discussed results from the regression analysis in more detail. The regression analysis was performed to determine key variables that contributed to longer leakage times. Results from the more recent series of tests show a more significant relationship between the number of conductors and the signal leakage time, rather than between TP/TS cable isolation and the signal leakage time, as concluded in 2001 tests. This result seems to be used more as an illustration than as a new conclusion, and this could explain why it did not get more attention.

We observe that it would be interesting if the report went further in some areas, for example, considering other conclusions from test results, such as timing to the first signal degradation, or, in the recommendations, discussing more specifics regarding follow-on activities, such as future tests or how results should be applied in a Fire PRA and the related HRA.

Nevertheless, as discussed above, the main objectives of the report are fully achieved, with a clear presentation of new insights. This report is a well-written, high quality professional work. Therefore, “Clarity of Presentation” is evaluated as above “satisfactory”.

Identification of Major Assumptions (Consensus Score: 5.0)

No assumptions were explicitly identified in the report.

The report “inherited” some assumptions from the previous report on the 2001 tests. One of these is an important underlying assumption related to the signal leakage times: it is assumed that the indicators are likely to start displaying erroneous values after the signal has dropped 0.25 percent from its starting value. Origins of this assumption are identified, but not discussed in detail.

Similarly, data regression analysis usually relies on a number of standard assumptions; for example, that the analyzed sample is representative of the population at large, or that the independent variables are measured with no error, etc. These assumptions are not identified in the report.

In this study, it is not clear that there are any major assumptions that needed to be identified to complete the testing. For that reason, a score of 5 was assigned.

Justification of Major Assumptions (Consensus Score: 5.0)

No assumptions were explicitly identified in the report; and thus, no justifications were provided. Similarly, as in the previous category, a score of 5 was used.

Soundness of Technical Approach/Results (Consensus Score: 6.3)

The report provides a strong basis for presented tests and results.

Test cases were run before the actual experiments to validate their proposed technical approach. Some adjustments were made, such as adding the stainless steel rods to validate the use of radiant heating. A significant effort was put into conducting a review of the proposed test plan to assure that the appropriate testing requirements were considered. Major goals were identified and incorporated into the test plan. For example, the need to fully understand TS cables and the differences between them and TP cables were highlighted in the Phenomena Identification and Ranking Table (PIRT) exercises conducted in 2012.

Treatment of Uncertainties/Sensitivities (Consensus Score: 5.0)

Uncertainties in this report are not explicitly analyzed, and no sensitivity studies were performed. Uncertainties in the test measurements or in a definition of the signal leakage times are not considered.

However, the project employed standard quality assurance procedures to reduce uncertainties associated with data obtained from these tests. Uncertainties related to data regression analysis are identified; for example, the conclusion about a significant relationship between the number of conductors and the signal leakage time was made with a 95 percent level of statistical confidence in that specific regression method.

In this study, it is not clear that any additional uncertainties or sensitivities needed to be included. For that reason, a score of 5 was assigned.

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