Assessment of the Quality of Selected NRC Research Projects by the Advisory Committee on Reactor Safeguards - FY 2010

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ABOUT THE ACRS

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

The ACRS provides independent reviews of, and advice on, the safety of proposed or existing NRC-licensed reactor facilities and the adequacy of proposed safety standards. The ACRS reviews power reactor and fuel cycle facility license applications for which the NRC is responsible, as well as the safety-significant NRC regulations and guidance related to these facilities. 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.

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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:

- Modeling a Digital Feedwater Control System Using Traditional Probabilistic Risk Assessment Methods (NUREG/CR-6997)
  - This project was found to be of superior professional quality with some elements of innovation
- Human Factors Considerations with Respect to Emerging Technology in Nuclear Power Plants (NUREG/CR-6947)
  - ACRS concluded that NUREG/CR-6947 by itself is not a technical research report suitable for quality review by ACRS. Even as a summary/management report, it could be substantially improved by clarifying priorities, interdependencies, and intended continued activities

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ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
BNL	Brookhaven National Laboratory
BTP	Branch Technical Position
CFR	Code of Federal Regulations
DFWCS	Digital Feedwater Control System
DOE	Department of Energy
FACA	Federal Advisory Committee Act
FMEA	Failure Modes and Effects Analysis
FWP	Feedwater Pump
FY	Fiscal Year
GPRA	Government Performance and Results Act
HBM	Hierarchical Bayesian Method
HFE	Human Factors Engineering
I&C	Instrumentation and Control
MAUT	Multi-Attribute Utility Theory
MFRV	Main Feedwater Regulating Valve
MFV	Main Feedwater Valve
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PIRT	Phenomena Identification and Ranking Table
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RES	Office of Nuclear Regulatory Research
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 2004, the Advisory Committee on Reactor Safeguards (ACRS) has been assisting RES by performing independent assessments of the quality of selected research projects [1-6]. 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 more recent discussions with the RES, the ACRS has 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 would be 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:

- Modeling a Digital Feedwater Control System Using Traditional Probabilistic Risk Assessment Methods
- Human Factors Considerations with Respect to Emerging Technology in Nuclear Power Plants

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. It should be noted that for the second project, our evaluation panel encountered serious problems in attempting to evaluate the research described in NUREG/CR-6947 against the methodology adopted for evaluating the quality of NRC research projects. Given the significance of the research summarized in the report, the panel agreed that evaluating the report against the adopted criteria described in Section 2 would serve no useful purpose.

2 METHODOLOGY FOR EVALUATING THE QUALITY OF RESEARCH PROJECTS

To guide its review of research projects, the ACRS has adopted an analytic/deliberative methodology [7-8]. The analytical part utilizes methods of multi-attribute utility theory (MAUT) [9-10] 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.

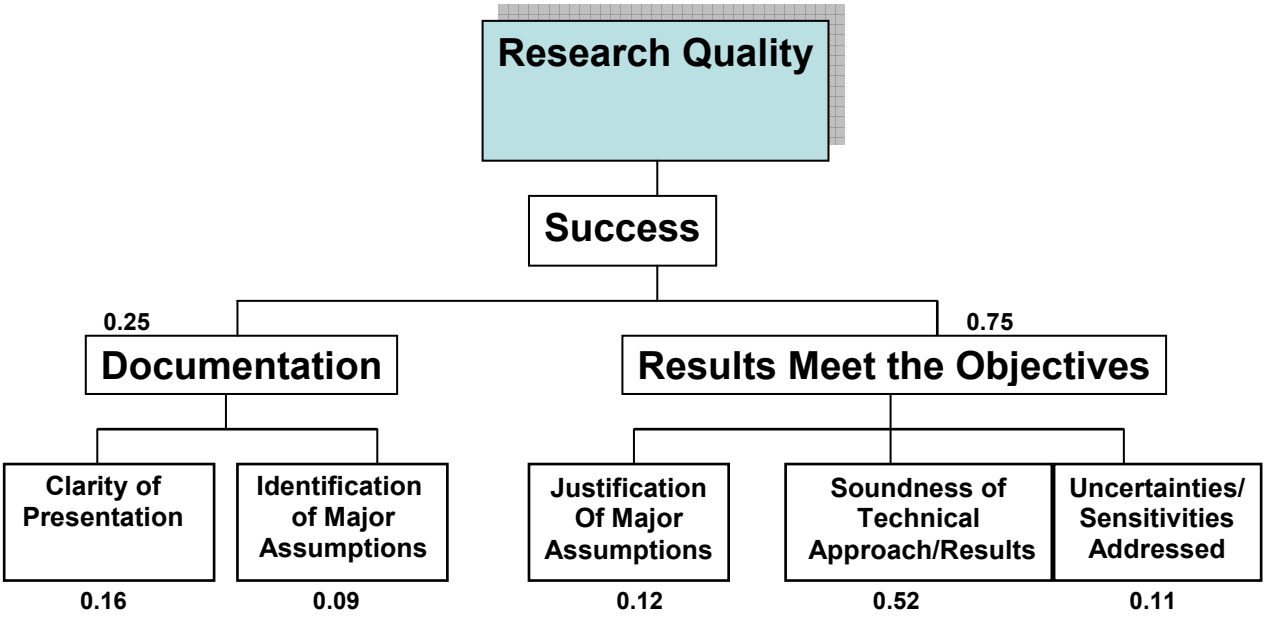

Figure 1 The value tree used for evaluating the quality of research projects

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 Modeling a Digital Feedwater Control System Using Traditional Probabilistic Risk Assessment Methods

Nuclear power plants have traditionally relied on analog systems for their protection and control functions. As a result of analog system obsolescence and digital functional advantages, existing plants have begun to replace some analog equipment with digital instrumentation and control (I&C) systems, while new plant designs fully incorporate digital systems.

Though many activities are carried out in the life cycle of digital systems to ensure a high-quality product, there are no consensus methods at present for quantifying the reliability of digital systems. This has been an impediment to developing a risk-informed analysis process for digital systems. To address this limitation, the NRC is currently performing research on the development of models and data for digital I&C systems to facilitate their integration into nuclear plant probabilistic risk assessments (PRAs).

The NRC has sponsored a research project at Brookhaven National Laboratory (BNL) to determine the capabilities and limitations of using traditional (i.e., static) reliability methods to develop and quantify digital system reliability models. In parallel, the NRC is also sponsoring other research projects for evaluating non-traditional (i.e., dynamic) methods. For the purposes of this research, traditional methods are defined to include modeling techniques such as event trees, fault trees, and Markov models. Dynamic methods are defined as those which explicitly attempt to model the interactions between a digital I&C system and physical processes (i.e., the changing values of plant process parameters) and the timing of those interactions.

A previous report (NUREG/CR-6962 [11]) documents the initial BNL work in the area of traditional PRA methods for digital systems, including the identification of desirable characteristics for evaluating digital system reliability models and the development of a general process for performing a reliability study of a digital feedwater control system (DFWCS) using two traditional modeling methods (i.e., the event tree / fault tree method and the Markov modeling method). The scope of the present quality review is limited to NUREG/CR-6997, "Modeling a Digital Feedwater Control System Using Traditional Probabilistic Risk Assessment Methods" [12], which documents the application of these methods to the DFWCS. Since this project was not intended to substantially advance the state of the art in modeling digital I&C systems, it does not include methods or guidance for detailed analysis and quantification of software reliability.

General Observations

The original objectives of this project were to perform a benchmark analysis of a DFWCS using two traditional methods (i.e., event tree / fault tree models and Markov models) and to compare each method with the desirable attributes that are delineated in NUREG/CR-6962. During the course of its development, the project ultimately evolved into a product that does not directly satisfy these objectives. In particular, the project team concluded that an advanced simulation tool was needed to appropriately identify subtle sequential order and timing considerations in their failure modes and effects analyses (FMEAs) that were not evident in other FMEA techniques. Development and application of this simulation tool represents an advancement in the state of the art of digital I&C analysis that extends beyond the original objectives of the

research program. Because the project team concluded that these order and timing considerations could affect the completeness of the FMEA and the resulting reliability models, their analyses use only a simplified Markov modeling technique. The study did not examine the use of event tree and fault tree methods to analyze the DFWCS, and it did not compare those methods with the selected Markov modeling techniques. Although Markov techniques are characterized as a "traditional" modeling methodology, they are not used in any PRA software suites that are currently in widespread use by the nuclear industry or the NRC. Therefore, integration of the FMEA simulation tool and the Markov modeling methods that are described in this report into current PRAs requires substantial additional effort and demonstration of practical feasibility.

The project team has acknowledged this evolution of the original research objectives and has documented the limitations of the described methods. The review of this project accounts for these redefined objectives, and it evaluates the project solely on its inherent qualities, without regard to practical considerations of near-term integration of these methods with current PRA modeling techniques.

The consensus scores for this project are shown in Table 2. The score for the overall assessment of the work was found to be 7, i.e., it is of superior professional quality with some elements of innovation.

Table 2. Summary Results of ACRS Assessment of the Quality of the Project "Modeling a Digital Feedwater Control System Using Traditional Probabilistic Risk Assessment Methods" (NUREG/CR-6997, BNL-NUREG-90315-2009)

Performance Measures	**Consensus Scores**	**Weights**	**Weighted Scores**
Clarity of presentation	7.3	0.16	1.17
Identification of major assumptions	6.7	0.09	0.60
Justification of major assumptions	5.5	0.12	0.66
Soundness of technical approach/results	7.3	0.52	3.81
Treatment of uncertainties/sensitivities	6.2	0.11	0.68
		Overall Score	**7.0**

Comments and conclusions within the evaluation categories are provided below.

Clarity of Presentation (Score = 7.3)

The document is very well written. It provides adequate background information and is scrupulous in explaining what was done and what the motivations for the activities were. Results are presented in a very clear concise manner. Text and tables align rather well. The authors are careful to define the limits of their work. They describe not only their results but also the insights that they have gained in the conduct of the research. They suggest and justify future work.

Although the report ultimately describes what was and was not accomplished during this project, it is somewhat difficult to follow the technical rationale for each of the departures from its original objectives. The introductory sections of the report would benefit from more explicit discussions of these departures so the reader is not forced to readjust expectations as new material is presented.

Identification of Major Assumptions (Score = 6.7)

The authors have done a good job in identifying the major assumptions that affect their work. Section 3 clearly describes the detailed assumptions that were used to simplify and bound the FMEA and the automated simulation tool. Section 8 identifies key assumptions that affect uncertainty in the applied models and completeness of the evaluations.

Justification of Major Assumptions (Score = 5.5)

Justification is generally provided for all major assumptions. However, some assumptions are simply stated, without additional explanatory context to understand their potential importance.

A very important implicit assumption in this project is that it is essential for any digital I&C reliability model to explicitly account for the identified effects from the order and sequential timing of specific failures. This assumption dictates the adoption of the automated FMEA simulation tool and the associated Markov modeling techniques. It is simply noted that event trees and fault tress do not typically account for these variations in failure sequencing. However, practical examples or comparative analyses are not provided to explain why alternate or simplified treatment in a different modeling context is not justified.

Soundness of Technical Approach / Results (Score = 7.3)

The development of new automated simulation methods to inform the FMEA for a digital I&C system is a significant advancement in the state of the art for understanding subtle interactions among specific hardware failures. Further development of this tool may also facilitate an integrated assessment of hardware and software interactions, which are beyond the scope of this project. The development of these new methods and the attempt to extend them for more generic applicability are the primary bases for the exceptional rating in this review category.

The study presents an extremely detailed analysis of a relatively simple feedwater control function, which is simplified further by the applied assumptions and boundary conditions (e.g., only loss of automatic control, one-year steady-state power operation, no examination of "too much" or "too little" feedwater, no repairs). It is not clear how these constraints and simplifications affect the practicality of these methods and models for actual PRA applications, which generally require analyses of short-term transient response under a large number of input

conditions. This is especially true for evaluations of modern, fully integrated, digital reactor protection and safeguards actuation platforms, including their attendant software, which must respond to a variety of plant transients and accidents under multiple, possible, input signal states.

From the block diagram in Figure 2-1, it is apparent that failures of the MFV (Main Feedwater Valve) controller, MFRV (Main Feedwater Regulating Valve) positioner, FWP (Feedwater Pump) controller, and FWP turbine controller may be the determining items for overall DFWCS reliability, because they do not have installed design redundancy. The comparison with operating experience also indicates that the most important causes for actual system problems may be outside the scope of this model (i.e., valve controller, maintenance-caused loss of power, solenoid valve). This is important information from an integrated PRA perspective. Thus, unless the detailed CPU models that are developed in this study identify failures that are both very subtle and potentially important from an integrated plant risk perspective, it is not clear what practical benefit is obtained from these advanced analysis techniques.

The results from the applied Markov models do not readily identify the specific contributors to system failure and their relative importance. The authors acknowledge that this is a major weakness of the tools in their current form. Examination and discussion of the analyzed failure contributors would support objective conclusions about the potential benefits of the automated FMEA tool and the Markov model, compared with other FMEA techniques and modeling methods. The authors fully acknowledge that “an assessment of the challenges and potential solutions for integrating the DFWCS model with a PRA is beyond the scope of this study, but will need to be addressed in the future.”

Treatment of Uncertainties/Sensitivities (Score = 6.2)

The uncertainty analysis provided in the report is essentially at the state of the art. The researchers directly addressed parameter uncertainty. They also attempted to address the issues of model uncertainty and completeness uncertainty, and they clearly identified important assumptions that contribute to those uncertainties. The treatment of parameter uncertainties is consistent with recommended methods, including explicit consideration of state-of-knowledge correlations in the parameter values.

It is not evident that the parametric data uncertainties are evaluated consistently. The applied Hierarchical Bayesian Method (HBM) derives much larger uncertainties (e.g., error factors of 14 to 88) when several data sources are available, compared with the assigned lognormal uncertainties (e.g., error factors of 3 to 10) when only one data source is used. This seems contrary to normal experience, unless the single data sources are supported by extensive documented operating experience. Since the data for most parameters in the study are derived from single sources with these assigned uncertainties, it is likely that the overall uncertainty in the results is underestimated.

There is no discussion of the reason why the "mean" frequency from the uncertainty analyses (i.e., 0.069) deviates substantially from the "point estimate" frequency (i.e., 0.079). The direction of this deviation is also unexpected, considering the lognormal nature of most parameter data and the explicit treatment of the data correlations. It is noted that only 1000 samples were used for the uncertainty calculations, but there is no other explanation for this very unusual situation.

3.2 Human Factors Considerations with Respect to Emerging Technology in Nuclear Power Plants

The increased use of automation and other technologies in existing, new and advanced nuclear power plant designs has the potential to introduce new Human Factors Engineering (HFE) challenges. The NRC sponsored a research project at BNL to identify human performance research that may be needed to support the review of a licensee's implementation of new technology in nuclear power plants (NPPs). The specific objective of this study was to identify potential human performance issues related to the role of personnel in new NPPs and the technological advances that will support that role. In this study, the term "issue" refers to:

- An aspect of new NPP development or evaluation for which available information suggests that human performance may be negatively impacted

- An aspect of new reactor development or design for which it is suspected that human performance may be impacted, but additional research and/or analysis is needed to better understand and quantify that impact

- A technology or technique that will be used for new plant design or implementation for which there is little or no review guidance

To identify the research issues, current industry developments and trends in the areas of reactor technology, instrumentation and control technology, human-system integration technology, and human factors engineering methods and tools were evaluated. Sixty-four potential human performance research issues associated with the introduction of emerging technologies in nuclear power plants were identified. These potential research issues were organized into seven high-level HFE topic areas:

- Roles of personnel and automation
- Staffing and training
- Normal operations management
- Disturbance and emergency management
- Maintenance and change management
- Plant design and construction
- Human factors engineering methods and tools

The research issues were then prioritized into four categories using a Phenomena Identification and Ranking Table (PIRT) methodology based on evaluations provided by 14 independent subject matter experts. The experts, representing vendors, utilities, research organizations, and regulators, were knowledgeable in a variety of disciplines. Twenty issues were categorized into the top priority category. A summary of the high-level topic areas and the issues in each is documented in NUREG/CR-6947, "Human Factors Considerations with Respect to Emerging Technology in Nuclear Power Plants." The scope of the present quality review was limited to this report.

General Observations

Our evaluation panel encountered serious problems in attempting to evaluate the research described in NUREG/CR-6947 against the methodology adopted for evaluating the quality of NRC research projects. Given the significance of the research summarized in the report, the panel agreed that evaluating the report against the adopted criteria described in Section 2 would serve no useful purpose. It would score slightly below the standard for professional work that satisfies the research objectives for documentation and well below that standard for presenting results that meet the research objectives.

NUREG/CR-6947 is a nice primer of the importance of human performance to reactor safety, and it provides a simple model of the impacts of operators on plant safety based on a general information processing model. It summarizes what has been learned in more than twenty years of research about the causes and impacts of human errors and sets this information within the milieu of plant operations. The objective of the study was to identify potential human performance issues related to the role of personnel in nuclear reactor plants. The concept of human performance issues is defined, and a catalog of identified issues is presented. Almost no detailed explanation of the research process is provided. Little or no documentation of the bases of the work is provided. The researchers used an expert elicitation process, but no discussion of controls on bias is provided and no convincing explanation of the uncertainties is provided. During the review process, it was learned that a companion BNL technical letter report [14] provides details on the study methodology, issue analysis, and results. NUREG/CR-6947 actually provides a table in Appendix A that carefully ties the potential issues to detailed sections of the BNL report. However, the scope of the present quality review was limited to NUREG/CR-6947 as originally proposed by RES.

Unfortunately, even as a summary/management report, NUREG/CR-6947 falls short. Identifying sixty-four potential human performance issues in new plants is little more than a catalog. Priorities 1-4 are assigned to each issue, but there is no overall explanation of which of these issues in which topical areas should be studied more intensively. There is no plan for organizing and attacking these new issues. There is no clear identification of the interdependence of the issues (many are not separable from others) and how that should affect further research and application of the results.

In summary, NUREG/CR-6947 by itself is not a technical research report suitable for quality review by the ACRS. Even as a summary/management report, it could be substantially improved by clarifying priorities, interdependencies, and intended continued activities.

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