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

October 22, 2008

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 2008

Dear Dr. Sheron:

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

- Assessment of Predictive Bias and the Influence of Manufacturing, Model, and Power Uncertainties in NRC Fuel Performance Code Predictions

    - This project marginally satisfied the research objectives. The Committee identified methods for assessment of biases and uncertainties that could yield superior insights into the FRAPCON and FRAPTRAN computer codes.

- Study of Remote Visual Methods to Detect Cracking in Reactor Components

    - This project was found to be satisfactory. With some limitations, the results meet the research objectives. The Committee notes the scope of research could be expanded to fully meet the research needs.

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

We anticipate receiving your list of candidate projects for quality assessment in FY-2009 prior to our December 4-6, 2008, meeting.

Sincerely,

/RA/

William J. Shack
Chairman

Enclosure: As stated

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Letter to Dr. Brian Sheron, NRC/RES, from William J. Shack, Chairman, NRS/ACRS, dated October 22, 2008

SUBJECT: ACRS ASSESSMENT OF THE QUALITY OF SELECTED NRC RESEARCH PROJECTS - FY 2008

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Assessment of the Quality of Selected NRC Research Projects by the Advisory Committee on Reactor Safeguards - FY 2008

October 2008

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

ABOUT THE ACRS

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

The ACRS provides independent reviews of, and advice on, the safety of proposed or existing NRC-licensed reactor facilities and the adequacy of proposed safety standards. The ACRS reviews power reactor and fuel cycle facility license applications for which the NRC is responsible, as well as the safety-significant NRC regulations and guidance related to these facilities. On its own initiative, the ACRS may review certain generic matters or safety-significant nuclear facility items. The Committee also advises the Commission on safety-significant policy issues, and performs other duties as the Commission may request. Upon request from the U.S. Department of Energy (DOE), the ACRS provides advice on U.S. Naval reactor designs and hazards associated with the DOE's nuclear activities and facilities. In addition, upon request, the ACRS provides technical advice to the Defense Nuclear Facilities Safety Board.

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

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ABSTRACT

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

- Assessment of Predictive Bias and the Influence of Manufacturing, Model, and Power Uncertainties in NRC Fuel Performance Code Predictions
  - This project marginally satisfied the research objectives. The Committee identified methods for assessment of biases and uncertainties that could yield superior insights into the FRAPCON and FRAPTRAN computer codes.

- Study of Remote Visual Methods to Detect Cracking in Reactor Components
  - This project was found to be satisfactory. With some limitations, the results meet the research objectives. The Committee notes the scope of research could be expanded to fully meet research needs.

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ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ASME	American Society of Mechanical Engineers
AEC	Atomic Energy Commission
AOOs	Anticipated Operational Occurrences
BWR	Boiling Water Reactor
CFR	Code of Federal Regulation
CSAU	Code Scaling, Applicability, and Uncertainty
COD	Crack Opening Displacement
DOE	Department of Energy
FACA	Federal Advisory Committee Act
FY	Fiscal Year
GPRA	Government Performance and Results Act
MAUT	Multi-Attribute Utility Theory
NDE	Nondestructive Examination
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PNNL	Pacific Northwest National Laboratory
PWR	Pressurized Water Reactor
RES	Office of Nuclear Regulatory Research
U.S.	United States
VT	Visual Testing

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 [1-4]. The Committee has established the following process for conducting the review of the quality of research projects:

- RES submits to the ACRS a list of candidate research projects for review because they have reached sufficient maturity that meaningful technical review can be conducted.

- The ACRS selects 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.

An analytic/deliberative decisionmaking framework was adopted for evaluating the quality of NRC research projects. The definition of quality research adopted by the Committee includes two major characteristics:

- Results meet the objectives
- The results and methods are adequately documented

Within the first characteristic, ACRS considered the following general attributes in evaluating the NRC research projects:

- Soundness of technical approach and results
    - Has execution of the work used available expertise in appropriate disciplines?

- Justification of major assumptions
    - Have assumptions key to the technical approach and the results been tested or otherwise justified?

- Treatment of uncertainties/sensitivities
    - Have significant uncertainties been characterized?
    - Have important sensitivities been identified?

Within the general category of documentation, the projects were evaluated in terms of 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:

- Assessment of Predictive Bias and the Influence of Manufacturing, Model, and Power Uncertainties in NRC Fuel Performance Code Predictions
- Study of Remote Visual Methods to Detect Cracking in Reactor Components

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

The methodology for developing the quantitative metrics (numerical grades) for evaluating the quality of NRC research projects is presented in Section 2 of this report. The results of assessment and ratings for the selected projects are discussed in Section 3.

2. METHODOLOGY FOR EVALUATING THE QUALITY OF RESEARCH PROJECTS

To guide its review of research projects, the ACRS has adopted an analytic/deliberative methodology [5-6]. The analytical part utilizes methods of multi-attribute utility theory (MAUT) [7-8] 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.

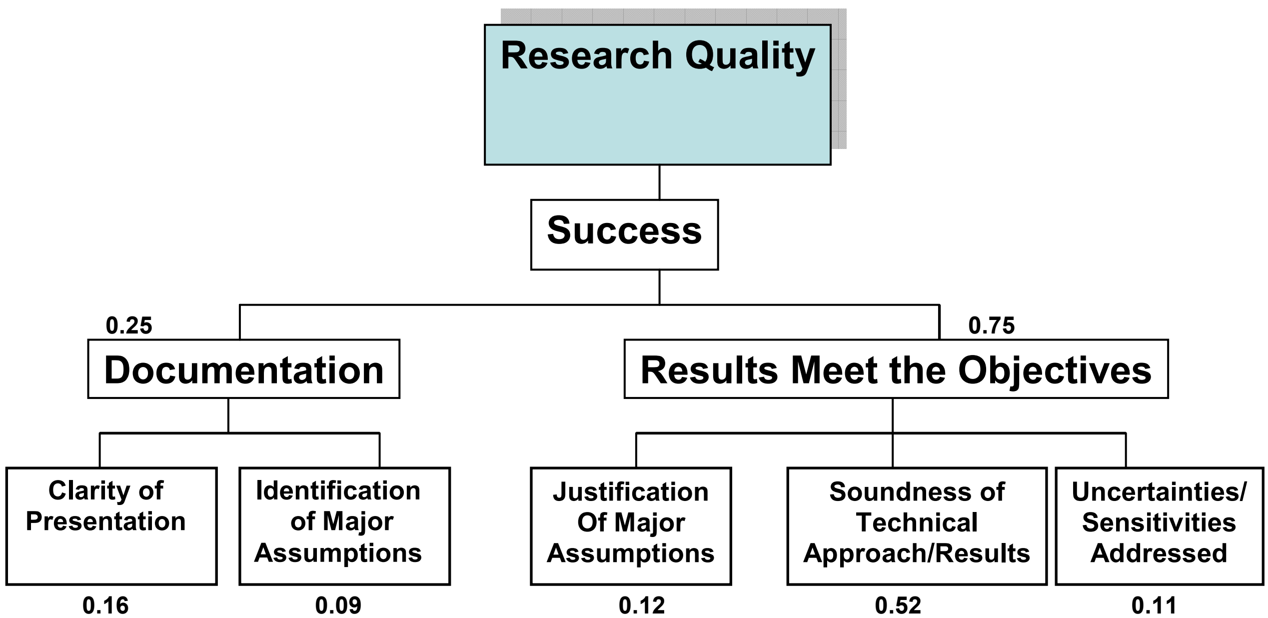

Figure 1 The value tree used for evaluating the quality of research projects

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 ASSESSMENT OF PREDICTIVE BIAS AND THE INFLUENCE OF MANUFACTURING, MODEL, AND POWER UNCERTAINTIES IN NRC FUEL PERFORMANCE CODE PREDICTIONS

The fuel rod thermal-mechanical performance computer codes FRAPCON and FRAPTRAN have been maintained by the NRC Office of Nuclear Regulatory Research for many years. FRAPCON-3 predicts fuel rod performance in pressurized water reactors (PWRs) and boiling water reactors (BWRs) by modeling the material responses of both the fuel and the cladding under normal operating conditions and anticipated operational occurrences (AOOs) with a duration of several minutes or greater. The transient fuel performance code FRAPTRAN predicts fuel rod performance in PWRs and BWRs by modeling the material responses of both the fuel and the cladding under fast transient and accident conditions. These computer models are used to verify that the fuel rod design, in combination with the Technical Specification power operating limits, is capable of satisfying fuel rod design criteria. Design calculations include fission gas release, rod internal pressure, fuel temperature, fuel thermal expansion, and cladding strain. The NRC sponsored a research project at Pacific Northwest National Laboratory (PNNL) to re-examine the fuel performance codes FRAPCON-3 and FRAPTRAN to determine if these codes are intrinsically conservative. The results of this study are documented in a Draft NUREG/CR report entitled “Assessment of Predictive Bias and Influence of Manufacturing, Model, and Power Uncertainties in NRC Fuel Performance Code Predictions” [9]. The scope of the present quality review is limited to this report.

<u>**General Observations**</u>

The draft NUREG/CR report is an ambitious effort to assess the FRAPCON-3 and FRAPTRAN computer codes for predicting the behaviors of power reactor fuels. The document describes assessments of the accuracy of predictions of primarily material property models in the codes, suggests some code improvements, and sensitivity assessments of the FRAPCON computer code to tolerances in the manufacturing of fuel and cladding. The intent appears to be to ensure the conservatism of predicted fuel behavior.

The consensus scores for this project are shown in Table 2. The score for the overall assessment of the work was found to be 3.5, which should be interpreted as “a project that marginally satisfied the research objectives.” The Committee identified deficiencies in methodology for assessing bias and uncertainty. The summary evaluation of the document is provided below:

Clarity of Presentation (Score = 4.0)

The document is readable. In the presentation of comparisons of data and code predictions, the authors have not facilitated the readers’ understanding of the comparisons of code predictions and data through the choice of scales. In most cases, the authors do not provide elementary, quantitative metrics of the comparison such as bias, offset, and standard deviation. Often, the authors summarily assert the acceptable quality of the agreement between code predictions and data without providing the bases for their conclusions. Not all acronyms and abbreviations are defined. Some terms of art such as "thermal racheting", "liftoff", and "hydride reorientation" are not adequately defined. In many instances, assertions by the authors are not substantiated by text or reference.

Table 2. Summary Results of ACRS Assessment of the Quality of the Project on Assessment of Predictive Bias and the Influence of Manufacturing, Model, and Power Uncertainties in NRC Fuel Performance Code Predictions (NUREG/CR-XXXX, PNNL-17644)

Performance Measures	**Consensus Scores**	**Weights**	**Weighted Scores**
Clarity of presentation	4	0.16	0.64
Identification of major assumptions	3.5	0.09	0.315
Justification of major assumptions	3.5	0.12	0.42
Soundness of technical approach/results	3.5	0.52	1.82
Treatment of uncertainties/sensitivities	3.0	0.11	0.33
		Overall Score	**3.5**

Identification of Major Assumptions (Score = 3.5)
Justification of Major Assumptions (Score = 3.5)

The authors do not do an adequate job identifying the assumptions that they have made in undertaking the work. As will be discussed further later, there are underlying assumptions throughout much of the work. One such unidentified assumption is that variability of code predictions is adequately revealed by varying only one factor at a time. That is, there are no physical correlations and therefore there are no synergisms among the factors being investigated. Another underlying assumption is that uncertainties are uncorrelated. It is by no means obvious that these unstated assumptions are correct and that the work represents an adequate exploration of bias and uncertainty for quantities of importance, which, according to the Forward to the report, include temperature distribution and stored energy in the fuel. There is, of course, no effort made to justify the unstated underlying assumptions of the work.

Soundness of Technical Approach/Results (Score = 3.5)

The NRC has pioneered methods for assessing both bias and uncertainty in computer code predictions. The code scaling, applicability, and uncertainty (CSAU) evaluation methodology, that is rigorously followed for thermal hydraulic code assessment, is noteworthy with respect to the task under review here. The authors have not adopted any of these well-established methods for assessing bias, sensitivity, or uncertainty. The authors do not define a figure of merit pertinent to the uses made of FRAPCON-3 and FRAPTRAN in the regulatory context. The Forward to the report suggests that either temperature distribution or stored energy might be an appropriate figure of merit for the assessment of code bias. Without such a figure of merit, there is not a consistent basis for assessing the adequacy of the models investigated in the work. Instead, the authors rely on comparisons of model predictions and data for individual processes and phenomena. The assessment of adequacy is then based on the authors' engineering judgments. Readers are assured at numerous points in the document of the adequacy of model predictions with no articulated basis for the assurance. In some cases, this assurance is made even though it is apparent that the model and the data have different functional forms (See for example Figure 2.29 of the NUREG/CR Report).

As noted above, the authors do not attempt in most cases to quantify the comparisons they make between model predictions and data. In some cases, the authors do note a standard deviation between model predictions and data but it is evident that the deviations are not stochastic but are instead systematic. The distinctions between stochastic and systematic deviations are not well drawn as would be expected given the state of the art for assessment of NRC computer codes. The means by which standard deviations are estimated is not always apparent.

In their examinations of the bias of models and manufacturing tolerances, the authors have considered variations in code inputs one at a time. They have not recognized any mechanistic correlation among variations of fuel and cladding properties though it is by no means obvious that there should not be important correlations. They do not recognize that variations of inputs may have synergistic effects on the code predictions. They avoid exploration for such synergistic effects even though they note qualitatively that some inputs have similar effects on code predictions. (See for example, "A tendency for rod internal pressure, peak fuel centerline temperature and cladding permanent hoop strain to increase and decrease with pellet roughness was observed." and "...a tendency for rod internal pressure, peak fuel centerline temperature, and cladding permanent hoop strain to vary inversely with cladding oxide conductivity was observed.") The code may not be configured to facilitate propagation of uncertainties. It should be possible, however, to make first order assessments of the combined effects of input variations:

$$\sigma_f^2 = \left(\frac{\partial f}{\partial x_1}\right)^2 \sigma_1^2 + \left(\frac{\partial f}{\partial x_2}\right)^2 \sigma_2^2 + \ldots\ldots$$

Even such simplistic evaluations of the combined effects of individual variations could facilitate comparisons to overall effects observed for integral evaluations.

The authors have avoided much explanation of individual models in the exposition of their work. They have relied instead on references to larger works devoted to model development. This has made some explanations they do provide difficult to understand and even confusing.

Treatment of Uncertainties /Sensitivities (Score = 3)

> The treatment of uncertainties and sensitivities is not consistent with the treatment that is being demanded of NRC computer codes in other contexts. Indeed, as noted above, a much more appropriate analysis is possible even with the constraints of current code capabilities.

3.2 A STUDY OF REMOTE VISUAL METHODS TO DETECT CRACKING IN REACTOR COMPONENTS

The U.S. nuclear industry has proposed replacing volumetric and/or surface examination methods currently required for some components by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI, “In-service Inspection of Nuclear Power Plant Components,” with remote visual testing (VT). Expanded use of visual methods is being proposed to reduce occupational exposures resulting from volumetric inspections in high radiation environments, and to overcome space and geometry constraints preventing volumetric inspections.

Piping and pressure vessel components in nuclear power stations are already being examined using remote VT. Remote VT with radiation-hardened video components is currently being used in the inspection of vessel cladding in pressurized water reactors and core internals in boiling water reactors. VT inspections employ a wide variety of procedures, radiation hardened cameras, lighting systems, and standards.

The Office of Nuclear Regulatory Research sponsored a research project at Pacific Northwest National Laboratory (PNNL) to evaluate the reliability and effectiveness of VT. PNNL conducted a parametric study that examined the important variables influencing the effectiveness of remote VT. The variables addressed in the research included lighting techniques, camera resolution, scanning speed and magnification, and material surface conditions. PNNL also conducted a limited laboratory test using a commercial VT camera system to quantify the ability to detect cracks of various widths under ideal conditions. The results of this research project that the ACRS reviewed are documented in NUREG/CR-6943, PNNL-16472 “A Study of Remote Visual Methods to Detect Cracking in Reactor Components” [10].

General Observations

As stated in its introduction to NUREG/CR-6943, this research was intended to provide a basis for describing the technical issues that must be addressed when applying remote VT to detect cracking phenomena by highlighting the inherent capabilities and limitations associated with current system deployment. With some limitations, the research and the report have met that goal.

The consensus scores for this project are shown in Table 3. The score for the overall assessment of the work was found to be 4.1, which is midway between a 5.0 “a professional job that satisfies the research objectives” and 3.0 “some deficiencies identified; marginally satisfies research objectives.”

Table 3. Summary Results of ACRS Assessment of the Quality of the Project on A Study of Remote Visual Methods to Detect Cracking in Reactor Components
(NUREG/CR-6943, PNNL-16472)

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	5.0	0.16	0.8
Identification of major assumptions	4.0	0.09	0.36
Justification of major assumptions	3.5	0.12	0.42
Soundness of technical approach/results	4.0	0.52	2.08
Treatment of uncertainties/sensitivities	4.0	0.11	0.44
		Overall Score	**4.1**

Comments and conclusions within the evaluation categories are:

Clarity of Presentation (Score = 5.0)

The report is well written and contains chapters on relevant background information, parametric studies, laboratory tests, conditions in nuclear plant components, discussions, and conclusions. Tables in the report provide concise summaries of the results and the figures provide visual examples of the different effects observed during the various VT tests. Technical details are discussed in a manner that can be understood by those who may not be expert in crack detection and visual examination methods. Examples include discussions of crack tortuosity, brightfield and darkfield imaging as well as effects of camera and lighting variables. High quality photographs are provided in the text and appendices. In addition, findings from comparable research in Sweden and Finland are presented and compared with those from the PNNL study.

The major conclusions of this research were that crack opening displacement (COD) is the most important variable governing crack detectability by remote VT methods, that cracks with COD’s significantly smaller than 100 microns will be difficult to detect, and cracks with CODs greater than 100 microns should be readily detectable.

Identification of Major Assumptions (Score = 4.0)

No explicit assumptions were provided in the PNNL report. The major (implicit) assumptions were that the experimental setup and test environments used in this research adequately represented conditions in nuclear plants, and those findings and conclusions would apply to power plant inspections.

This research was performed under ideal laboratory conditions far different from those in a nuclear power plant. During laboratory testing at PNNL, lighting conditions were ideal, and camera vendor representatives were present to ensure that the cameras were operated at optimum levels of performance. Inspectors did not work in hot/humid environments typically encountered during nuclear plant outages and cameras and lights were not deeply submerged or exposed to high radiation fields. All test samples were examined either in air or under a few inches of water. Consequently, the following major conclusion of the report is not adequately supported.

"*The current radiation-hardened video cameras being used in the field can be expected to reliably find cracks with CODs greater than 100 μm (0.004 in.), provided surface conditions are not overly unfavorable, adequate lighting is achieved, and sufficiently slow scan rates are applied.*"

This conclusion may be correct but was not demonstrated by the research. As shown in Table 4 (reproduced from Table 2.1 of the PNNL report), the report did show that many cracks found in nuclear plants exhibited maximum CODs greater than 100 microns. However the report did not state whether these cracks were found by VT or by volumetric inspection methods or by coolant leakage. The research should have provided direct evidence that remote VT methods would have detected all or most or these cracks.

Table 4 Crack Widths in Stainless Steel Components
(Reproduced from NUREG/CR-6943, Table 2.1, pp. 2.2)

	IGSCC SS	**IGSCC Ni**	**IDSCC Ni**	**TGSCC SS**	**Thermal Fatigue**	**Mechanical Fatigue**	**Hot Cracks**
Total Cracks	65	14	14	25	29	15	17
Minimum (μm)	3	4	0	3	5	3	2
Maximum (μm)	160	260	120	500	380	450	250
Mean (μm)	37.7	42.4	33.4	49.9	51.4	79.4	38.6
Median (μm)	30	17.5	21	20	30	16	25
RMS (μm)	47.2	77.8	48.5	110	85.4	144	67.3
Standard Deviation (μm)	28.7	67.7	36.4	99.6	69.3	125	56.8

Justification of Major Assumptions (Score = 3.5)

The assumption that crack detection probabilities determined in this research directly apply to nuclear plant inspections was not justified. This research demonstrated the best crack detection probabilities currently achievable under ideal conditions. A statement that these probabilities have not been demonstrated in actual plants would have been appropriate. Alternately, one or more inspections could have been performed at a nuclear inspection training facility using the best practices described in the report.

Soundness of Technical Approach / Results (Score = 4.0)

This research program included extensive efforts to identify and evaluate the effects of lighting, camera scanning speed, and other variables that could affect the effectiveness of visual examinations. The researchers used various methods to quantify the influence of subjective variables. One example was the use of the 1951 U. S. Air Force Resolution Target to quantify camera performance. Another was the use of test specimens with well defined crack dimensions.

Although qualitatively consistent, there were quantitative discrepancies between the findings from the Parametric Study and those from the Laboratory Tests. As shown in Table 5 and 6 (reproduced from Tables 4.3 and 5.2 of the PNNL report) crack detection probabilities were significantly different in the two studies even for the widest cracks examined (cracks with greater than100 micron CODs). These discrepancies should have been discussed in the report.

Table 5 Effects of Crack COD on Crack Detectability
(Parametric Test Matrix)
(Reproduced from NUREG/CR-6943, Table 4.3, pp4.11) [10]

COD (μm)	Excellent	Good	Fair	Poor
<20	2.1%	14.6%	10.4%	72.9%
20–40	31.3%	22.9%	16.7%	29.2%
40–100	22.9%	27.1%	16.7%	33.3%
100+	47.9%	16.7%	12.5%	22.9%

Table 6 Probability of Detection Versus Crack COD Results Using Fixed-Focal Length Camera (Laboratory Tests)
(Reproduced from NUREG/CR-6943, Table 5.2, pp. 5.3) [10]

	Probability of Detection	
Crack Size	**Lenient**	**Strict**
<20 μm	6 ± 6%	0 ± 6%
20–40 μm	37 ± 11%	11 ± 7%
40–100 μm	42 ± 11%	32 ± 11%
100–150 μm	92 ± 8%	92 ± 8%

Treatment of Uncertainties/Sensitivities (Score = 4.0)

Sensitivity of crack detection probability to several inspection variables was evaluated. These variables included camera and lighting variables, scanning speeds, material surface conditions, crack lengths, crack opening displacement, surface deposits, and human factors.

The use of qualified industry technicians as well as PNNL technicians provided some interesting comparisons. Although the results obtained were consistent with other research programs, using only four inspectors limited the assessment of human performance uncertainties. For example, as shown in Table 7 (reproduced from Table 5.1 of the PNNL report), there appeared to be significant differences in crack detection capabilities among inspectors. Extra inspection time did not help the PNNL staff to find more cracks than industry inspectors, but did increase the number of false calls. This variability (which the authors called good agreement) suggests the need for more carefully controlled and broadly based experiments to understand the influence of the inspector on VT inspection uncertainty.

Moreover, the effects of the human element as a key part of the visual testing process are minimized in Chapter 8 conclusions, which dwell most heavily on the characteristics of cameras and lighting systems.

Table 7 Probability of Detection by Inspector Using Fixed-Focal Length Camera
(Reproduced from NUREG/CR-6943, Table 5.1, pp. 5.1) [10]

Fixed Focus Camera	**Strict**	**Lenient**	**False Calls**	**Time taken**
PNNL Inspector 1	29%	53%	9	4 hr
PNNL Inspector 2	29%	29%	11	4.5 hr
Contractor 1	18%	35%	1	2.5 hr
Contractor 2	29%	53%	1	2.5 hr

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