UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Title: AFFIRMATION/DISCUSSION AND VOTE

Location: ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND

Date: THURSDAY, AUGUST 4, 1988

Pages: 1-5

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The transcript is intended solely for general informational purposes. As provided by 10 CFR 9.103, it is not part of the formal or informal record of decision of the matters discussed. Expressions of opinion in this transcript do not necessarily reflect final determination or beliefs. No pleading or other paper may be filed with the Commission in any proceeding as the result of, or addressed to, any statement or argument contained herein, except as the Commission may authorize.

1 UNITED STATES OF AMERICA

2 NUCLEAR REGULATORY COMMISSION

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4 AFFIRMATION/DISCUSSION AND VOTE

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6 PUBLIC MEETING

7 ***

8 Nuclear Regulatory Commission

9 One White Flint North

10 Rockville, Maryland

11

12 Thursday, August 4, 1988

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14 The Commission met in open session, pursuant to

15 notice, at 3:30 p.m.., the Honorable LANDO W. ZECH, Chairman of

16 the Commission, presiding.

17 COMMISSIONERS PRESENT:

18 LANDO W. ZECH, Chairman of the Commission

19 THOMAS M. ROBERTS, Member of the Commission

20 KENNETH CARR, Member of the Commission

21 KENNETH ROGERS, Member of the Commission

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1 STAFF AND PRESENTERS SEATED AT THE COMMISSION TABLE:

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3 S. CHILK

4 W. PARLER

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1 P R O C E E D I N G S

2 [3:30 p.m.]

3 CHAIRMAN ZECH: Good afternoon, ladies and gentlemen.
4 This is an affirmation session. We have three items to come
5 before us this afternoon.

6 Before I ask the Secretary to go through the items,
7 do any of my fellow Commissioners have any comments to make?

8 [No response.]

9 CHAIRMAN ZECH: If not, Mr. Secretary, please
10 proceed.

11 MR. CHILK: The first paper, Mr. Chairman, is SECY
12 88-162 entitled "Revision of the ECCS Rule" contained in
13 Appendix K and Section 50.46 of 10 CFR Part 50. In this paper,
14 the Commission is being asked to act on a revision to the ECCS
15 rule contained in Appendix K and Section 50.46 of 10 CFR 50.

16 All Commissioners have approved the revision of the
17 rule with modifications which were attached to our memorandum
18 of August the 4th from Commissioner Roberts.

19 Would you please affirm your votes?

20 CHAIRMAN ZECH: Aye.

21 COMMISSIONER ROBERTS: Aye.

22 COMMISSIONER ROGERS: Aye.

23 COMMISSIONER CARR: Aye.

24 MR. CHILK: The second paper is 88-164. It deals
25 with the allocation between the Commission and Illinois of

1 regulatory authority over West Chicago waste materials. In
2 this paper, the Commission is being asked to act on an order
3 which resolves the uncertainty as to whether Illinois has
4 jurisdiction over certain materials in Kress Creek and at other
5 locations at or near the West Chicago facilities.
6 All Commissioners have approved first an order as
7 modified by Commissioners Roberts and Carr, which is attached
8 to our memorandum of August the 3rd, which holds that NRC
9 retains jurisdiction over Kress Creek materials and that the
10 Commission has relinquished jurisdiction over other materials
11 in dispute.
12 In addition, the Commission approved an order denying
13 the NRC Staff's July 13th petition for review of ALAB-867.
14 Would you please affirm your votes?
15 CHAIRMAN ZECH: Aye.
16 COMMISSIONER ROBERTS: Aye.
17 COMMISSIONER ROGERS: Aye.
18 COMMISSIONER CARR: Aye.
19 MR. CHILK: The last item, Mr. Chairman, is SECY 88-
20 184. It's a Licensing Board decision on the Senior Reactor
21 Operator License for David W. Held.
22 The Commission here is being asked to act on an order
23 which completes its consideration of the January 11th and the
24 February 2, 1988 decisions of the Administrative Judge who
25 resided over the request of Mr. Held for a hearing on the

1 denial of the Senior Reactor Operator's License.

2 All Commissioners have approved the order, which was

3 attached to our memorandum of August the 4th, which remands the

4 case to the Administrative Judge for a proceeding on the

5 specific issue of whether Mr. Held should have been found to

6 have passed or failed the simulator examination.

7 Chairman Zech has additional views which are attached

8 to the order.

9 Would you please affirm your votes?

10 CHAIRMAN ZECH: Aye.

11 COMMISSIONER ROBERTS: Aye.

12 COMMISSIONER ROGERS: Aye.

13 COMMISSIONER CARR: Aye.

14 MR. CHILK: I have nothing further, sir.

15 CHAIRMAN ZECH: Thank you very much. We are

16 adjourned.

17 [Whereupon, at 3:34 o'clock, p.m., the Commission

18 meeting was adjourned.]

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CERTIFICATE OF TRANSCRIBER

This is to certify that the attached events of a meeting of the U.S. Nuclear Regulatory Commission entitled:

TITLE OF MEETING: AFFIRMATION/DISCUSSION AND VOTE

PLACE OF MEETING: Washington, D.C.

DATE OF MEETING: THURSDAY, AUGUST 4, 1988

were transcribed by me. I further certify that said transcription is accurate and complete, to the best of my ability, and that the transcript is a true and accurate record of the foregoing events.

Suzanne Young

Ann Riley & Associates, Ltd.

June 9, 1988

RULEMAKING ISSUE
(Affirmation)

SECY-88-162

For: The Commissioners

From: Victor Stello, Jr.
Executive Director for Operations

Subject: REVISION OF THE ECCS RULE CONTAINED IN APPENDIX K AND SECTION 50.46 OF 10 CFR PART 50

Purpose: To obtain Commission approval for publication in the Federal Register of final amendments revising the ECCS rule contained in Appendix K and Section 50.46 of 10 CFR Part 50.

Category: This paper covers a major policy question.

Issue: Should the final text of the proposed ECCS rule be approved by the Commission. The proposed amendments would:

a. allow the use of best-estimate evaluation methods through the utilization of more recent information gained on the performance of ECC systems,

b. relax certain reanalysis requirements that do not contribute to safety,

c. permit the continued use of current methods for those licensees and applicants wishing to do so, and

d. delete from Appendix K the reference to the Dougall-Rohsenow heat transfer correlation as an acceptable model.

Summary: Section 50.46 of 10 CFR Part 50 requires that calculations be performed to show that the emergency core cooling systems (ECCS) will adequately cool the reactor in the event of a loss-of-coolant accident (LOCA). Appendix K sets forth certain required and acceptable features that the evaluation models, used to perform these calculations, must contain.

Contact:
Louis M. Shotkin, RES
49-23530

The results of these calculations are used to determine the acceptability of the ECCS performance. In many instances, these calculations result in technical specification limits on reactor operation (e.g., peak local power) in order to comply with the 2200°F cladding temperature limit and other limits of §50.46. These limits may restrict the total power output and optimal operation of many reactors (e.g., most Westinghouse plants) in terms of efficient fuel utilization, maneuvering capability and surveillance requirements. Removing unnecessary restrictions on operation will allow increased U. S. electricity production, worth several hundred million dollars a year, without loss of benefit to the public health and safety.

NRC, DOE (including AEC and ERDA), U. S. nuclear industry and foreign research on ECCS performance since the present ECCS rule was issued provides a technical understanding which shows that the existing ECCS rule restrictions are more stringent than necessary for safety. Thus, the staff recommends that the ECCS rule be amended to reflect this more realistic safety assessment and to remove unnecessary operating restrictions. This is consistent with the 1973 Commission opinion published with the existing rule. As a result of the large body of research into the behavior of emergency core cooling systems during a loss-of-coolant accident, the Nuclear Regulatory Commission has proposed to amend its requirements (52 FR 6334) to allow licensees and applicants to use best-estimate calculations accompanied by an uncertainty quantification to demonstrate compliance with the acceptance criteria specified in 10 CFR 50.46(b). The rule changes were published as a proposed rule on March 3, 1987. Based on the generally favorable public response to the proposed rule, the staff recommends that the Commission approve the final rule for publication with no changes.

Discussion: On March 3, 1987, the Nuclear Regulatory Commission published in the Federal Register proposed amendments (52 FR 6334) to 10 CFR Part 50 and Appendix K (Enclosure A). These proposed amendments were motivated by the fact that since the promulgation of Section 50.46 of 10 CFR Part 50, "Acceptance Criteria for Emergency Core Cooling Systems (ECCS) in Light Water Power Reactors", and the acceptable and required features and models specified in Appendix K to 10 CFR Part 50 for performing ECCS performance analyses, considerable research has been performed that has greatly increased the understanding of ECCS performance during a LOCA. We have now confirmed that the methods specified in Appendix K, combined with other analysis methods currently in use, are highly conservative and that the actual cladding temperatures which

would occur during a postulated LOCA would be much lower than those calculated using Appendix K methods. In addition, the large body of research available has provided a method to both estimate the degree of conservatism in Appendix K calculations and, to determine to a reasonable extent, the uncertainty associated with that estimate. In light of these factors, the Commission approved the publication of the proposed rule which would permit licensees and applicants to make realistic calculations of ECCS performance during a LOCA in the regulatory process, as well as to require an estimate of the uncertainty of the calculation to assure that there is a high probability that the acceptance criteria in 10 CFR 50.46 (b) (e.g., calculated peak cladding temperature shall not exceed 2200°F) would not be exceeded. The Commission paper which transmitted the proposed rule (SECY 86-318) contained a discussion of the Staff's perception of the inherent conservatism in the acceptance criteria in §50.46. Enclosure J provides further development of the Staff's views concerning the amount of margin in these criteria.

In considering the staff's recommendation to approve publication of the proposed rule, the Commission directed the staff to subject the methodology for evaluating the uncertainty in NRC codes that was developed by NRC to both peer and ACRS review and to solicit public comment on several specific questions (Enclosure B). The staff was directed to seek public comment on an ACRS question concerning the indefinite grandfathering of plants with acceptable Appendix K models. The staff was also instructed to submit three questions posed by Commissioner Asselstine for public comment. Namely:

1. Should this rule change include an explicit degree of conservatism that must be applied to the evaluation models?

2. Should this rule change explicitly prohibit any increase in approved power levels until all severe accident issues and unresolved safety issues are resolved?

3. Should the technical basis for this proposed rule change be reviewed by an independent group; such as the American Physical Society?

In accordance with the Commission's request, the ACRS reviewed the NRC methodology for ECCS code uncertainty evaluation on September 10, 1987 and again on May 5-7, 1988. (Enclosure C). In addition, review of the NRC methodology by an independent group of experts was also conducted in January 1988. The summary report of this independent panel,

which was chaired by Professor Todreas from MIT, has been received and is also provided in Enclosure C. Both the ACRS and the expert's panel made generally favorable comments concerning the methodology; however, both groups recognized that a complete demonstration (i.e., application to small break LOCA and the reflood portion of large break LOCA) has not yet been accomplished and certain reviewers questioned whether such a demonstration could be performed successfully. The only objectives of the NRC methodology demonstration are to demonstrate feasibility, to develop an audit tool, and to provide the necessary experience to audit licensee submittals. The staff does not believe that NRC demonstration of the methodology is a prerequisite to this rulemaking. Licensees wishing to adopt the best estimate approach permitted as a result of this rule are neither required to use this methodology nor to model their own methodologies on it. The NRC has determined through twenty years of experience that independent analysis with independent methodologies is the most effective way to intelligently review new vendor or licensee methodologies. It is therefore appropriate that this new methodology be subjected to stringent technical scrutiny, as directed by the Commission. The NRC Staff is committed to completing this demonstration by the time that it will be needed to review licensee submittals and is confident that such a demonstration will be successful.

The comment period for the proposed rule revision expired on July 1, 1987. Twenty-seven letters addressing the proposed rule change were received. In addition, six letters commenting on the proposed rule were received after the expiration of the comment period. These were also considered by the staff to the extent that new and substantial comments were provided. A list of the rule commenters, and a summary of the comment letters are provided in Enclosure D.

The bulk of the comments received on the rule came from the nuclear industry and were largely supportive of the action proposed. These comments suggested a number of minor revisions to the rule, which are not discussed here, but can be found in Enclosure E. One negative comment was received from an anonymous commenter but specific objections were not provided. Two commenters recommended that the rule not be implemented until other safety issues are resolved. A detailed analysis of the public comments may be found in the Federal Register Notice (Enclosure E) including proposed responses to the public comments on questions posed by the

ACRS and Commissioner Asselstine, as well as several other comments that recommended major changes.

In conjunction with the publication of the proposed amendments to 10 CFR 50.46 and Appendix K, the NRC staff prepared a regulatory guide to set forth the staff's views concerning acceptable procedures for compliance. This regulatory guide, entitled "Best Estimate Calculations of Emergency Core Cooling System Performance," was also released for public comment to assist the public in understanding the proposed revisions and to allow public participation in their development. This guide describes features that a realistic ECCS evaluation model should contain and guidance on performing the uncertainty evaluation. The regulatory guide lists a number of models and corresponding experimental data that are considered suitable for use. The commenters on the regulatory guide largely supported the inclusion of experimental data that was acceptable and models and correlations that fit the data acceptably well. Therefore, these references have been included in the final guide with the statement that other models and correlations will be considered but must be justified with appropriate experimental data. This regulatory guide is provided as Enclosure F and a list of commenters on the guide and a paraphrased summary of guide comments is provided in Enclosure G.

In order to apprise licensees and applicants of the large body of research which supports realistic calculations of ECCS performance, the staff has prepared NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis." This report identifies the relevant ECCS research performed and describes the NRC developed methodology for the estimation of the uncertainty of thermal-hydraulic safety analysis codes. This report also contains documentation of NRC studies of effects of reactor power increases on risk, background information on the ECCS rule, and a description of the NRC developed methodology for estimating thermal-hydraulic code uncertainty. The staff feels that this document will provide valuable guidance to licensees and applicants wishing to avail themselves of the benefits of performing realistic LOCA analyses to evaluate ECCS performance.

Based on the comments received and considered and the large experimental data base available at this time, the staff recommends that the Commission approve for publication the final rule as contained in the Federal Register Notice (Enclosure E). This final rule does not alter the

requirements found in the notice of proposed rulemaking (Enclosure A). A regulatory analysis of the final rule is provided in Enclosure I.

Resource Estimates: The NRC staff resources to implement this rule are thought to be negligible under the assumption that no unusual or special rulemaking procedures (e.g., adjudicatory hearings) will be established by the Commission. If the Commission chooses to hold hearings, resources would have to be diverted from other high priority activities.

The major staff resources required under the rule change will be to review the realistic models and uncertainty analysis that may be submitted by licensees or applicants wishing to utilize the revised ECCS Rule. Based on previous experience with the General Electric Co. SAFER model and the learning that has resulted from this effort, it is estimated that approximately one staff year will be required to review each generic model submitted. There are four major reactor vendors (GE already has a revised evaluation model approved under the existing Appendix K for both jet-pump and non-jet pump plants that is consistent with SECY 83-472 and meets the requirements of the revised ECCS rule proposed herein) and several fuel suppliers and utilities which perform their own analyses and potentially might submit generic models for review. However, it is expected that only 3 or 4 generic models would be submitted since not all plants would benefit from the rule. Thus, approximately three to four staff years will be required to review the expected generic models. Once a generic model is approved, the plant-specific review is expected to be very straightforward.

Recommendations: That the Commission:

1. Approve the publication of final amendments, as set forth in Enclosure E, which would permit the use of realistic evaluation models and the accompanying estimate of the uncertainty of the calculation to demonstrate that the acceptance criteria contained in Section (b) of 10 CFR 50.46 are not exceeded.

2. Note:

   a. That the notice of final rulemaking in Enclosure E will be published in the Federal Register to be effective 30 days after publication.

   b. That the regulatory guide in Enclosure F will be published concurrent with this rule.

c. That pursuant to §§ 51.21 and 51.31 of 10 CFR Part 51 of the Commission's regulations, an environmental assessment and finding of no significant impact is attached as Enclosure H.

d. That the reporting requirements in connection with analyses required by the rule (Enclosure E) impose information collection requirements that are subject to the Paperwork Reduction Act. The requirements were approved by OMB.

e. That pursuant to the Regulatory Flexibility Act of 1980 the rule contains a statement that the Commission certifies that the rule will not, if promulgated, have a significant economic impact upon a substantial number of small entities and a copy of this certification will be forwarded to the Chief Counsel for Advocacy, SBA by the Division of Rules and Records, ADM.

f. That the Subcommittee on Nuclear Regulation of the Senate Committee on Environment and Public Works, the Subcommittee on Energy and the Environment of the House Committee on Interior and Insular Affairs, the Subcommittee on Energy Conservation and Commerce, and the Subcommittee on Environment, Energy and Natural Resources of the House Committee on Government Operations will be informed.

g. That a Regulatory Analysis is attached as Enclosure I.

h. That a public announcement will be issued.

i. That copies of the Notice of Final Rulemaking will be distributed by TIDC, ADM to each affected licensee and other interested parties.

j. That the staff recommends the paper be placed in the PDR.

k. That this paper has been reviewed with the ACRS Subcommittee on Thermal-Hydraulic Phenomena on April 21, 1988, and they have indicated that they have no objection to this rule. The ACRS has indicated that it wishes to be apprised of the progress made on the demonstration of the uncertainty methodology. The staff will continue

to brief the ACRS until the demonstration is complete.

l. That this proposed rule change has been concurred in by the Offices of NRR and ARM.

m. OGC has reviewed the rule and regulatory guide and has no legal objections.

Scheduling: The staff recommends that this action be affirmed through a notation vote. No specific circumstance is known to the staff which would require Commission action by any particular date in the near future.

Victor Stello, Jr.
Executive Director
for Operations

Enclosures:

A. Notice of Proposed Rulemaking
B. Memorandum Chilk to Stello, dtd 1/9/87
C. ACRS and Peer Review Summaries
D. Summary of Public Comment
E. Notice of Final Rulemaking
F. Draft Regulatory Guide
G. Summary of Public Comment on Guide
H. Environmental Assessment
I. Regulatory Analysis
J. Margin Inherent in 2200°F Limit

Commissioners' comments should be provided directly to the Office of the Secretary by c.o.b. Tuesday, June 28, 1988.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Monday, June 20, 1988, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

This paper is tentatively scheduled for affirmation at an Open Meeting during the Week of June 27, 1988. Please refer to the appropriate Weekly Commission Schedule, when published, for a specific date and time.

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ENCLOSURE A

Proposed Rules

Federal Register
Vol. 52, No. 41
Tuesday, March 3, 1987

This section of the FEDERAL REGISTER contains notices to the public of the proposed issuance of rules and regulations. The purpose of these notices is to give interested persons an opportunity to participate in the rule making prior to the adoption of the final rules.

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Emergency Core Cooling Systems; Revisions to Acceptance Criteria

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is proposing an amendment that would allow the use of alternative methods to demonstrate that the emergency core cooling system (ECCS) would protect the nuclear reactor core during a postulated design basis loss-of-coolant accident (LOCA). The amendment is proposed because research, performed since the current rule was written, has shown that calculations performed using current methods and in accordance with the current requirements result in estimates of cooling system performance that are significantly more conservative than estimates based on the improved knowledge gained from this research. In addition, the operation of some nuclear reactors is being unnecessarily restricted by the rule, resulting in increased costs of electricity generation. The proposed rule, while continuing to allow the use of current methods and requirements, would also allow the use of more recent information and knowledge to demonstrate that the ECCS would protect the reactor during a LOCA. The proposed amendment, which would apply to all applicants for and holders of construction permits or operating licenses for light water reactors, would also relax requirements for certain reanalyses which do not contribute to safety.

DATES: Comment period expires July 1, 1987. Comments received after that date will be considered if it is practical to do so, but assurance of consideration cannot be given except as to comments received on or before that date.

ADDRESSES: Submit written comments to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention Docketing and Service Branch. Hand deliver comments to Room 1121, 1717 H Street NW., Washington, DC between 8:15 a.m. and 5:00 p.m. Examine comments received, the environmental assessment and finding of no significant impact, and the regulatory analysis at the Commission's Public Document Room at 1717 H Street, NW., Washington, DC. Obtain single copies of the environmental assessment and finding of no significant impact and the regulatory analysis from L.M. Shotkin, Office of Nuclear Regulatory Research, Washington, DC 20555, telephone (301) 443-7825.

FOR FURTHER INFORMATION CONTACT: L.M. Shotkin, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 443-7825.

SUPPLEMENTARY INFORMATION:

Background

Section 50.46 of 10 CFR Part 50 provides "Acceptance Criteria for Emergency Core Cooling Systems (ECCS) in Light Water Nuclear Power Reactors." This section requires that calculations of loss-of-coolant accidents (LOCA) be performed to show that the ECCS will maintain cladding temperatures, cladding oxidation and hydrogen generation to within certain specified limits. It also requires that a coolable core geometry be maintained and that long term decay heat removal be provided. Appendix K to 10 CFR Part 50 sets forth certain required and acceptable features of the models used to perform these calcuations. The criteria of 10 CFR 50.46 and the calculational methods specified in Appendix K were formally issued in January 1974 after extensive rulemaking hearings and are based on the understanding of ECCS performance available at that time.

In the thirteen years following the rulemaking, the NRC, the Department of Energy (including the Atomic Energy Commission and the Energy Research and Development Agency), U.S. nuclear industry and foreign researchers have obtained considerable information on ECCS performance. The majority of this LOCA research is complete and has greatly improved of understanding of ECCS performance during a LOCA. The methods specified in Appendix K, combined with other analysis methods currently in use, are now known to be highly conservative; that is, the actual temperatures during a LOCA would be much less than the temperatures calculated using Appendix K methods. The ECCS research has gone beyond confirming that Appendix K is conservative; it has allowed quantification of that conservatism. The results of experiments, computer code development, and code assessment now allow more realistic calculations of ECCS performance during a LOCA, along with reasonable estimates of uncertainty, than is possible using current evaluation models.

It is also known that some plants are now restricted in operating flexibility by limits resulting from conservative calculations using current models and Appendix K requirements. These restrictions may be preventing optimal operation of these plants. Based on research performed, it is now known that these restrictions can be relaxed because of improved knowledge of safety margins.

On December 6, 1978, the NRC published an advance notice of proposed rulemaking (43 FR 57157) calling for a two-phase approach to the revision of 10 CFR Part 50 and Appendix K. The first step would have been to make procedural changes and to permit minor technical changes which would not have reduced the conservatism contained in Appendix K. The second phase would have made further technical changes based on research results and operating experience.

NRC activity on the ECCS rulemaking was severely curtailed as a result of the high priority efforts required by the TMI-2 accident. This rulemaking was dormant until July 1981 when it was revived in the context of simplifying and streamlining the regulatory process.

The NRC has reviewed the comments made by outside organizations on the advance notice of proposed rulemaking as well as a number of other comments received since that time. In general, the commenters support a rule change that would permit greater flexibility in meeting the regulations and would incorporate the use of presently available research information. Many felt that the Phase 1 scope should be expanded to allow the use of additional

information available from completed ECCS research.

Because of the delay in changing the ECCS rule, the NRC has used an interim approach, described in SECY-83-472,[1] to accommodate requests for improved evaluation models, submitted by reactor vendors, for the purpose of reducing reactor operating restrictions. This interim approach requires a realistic calculation, with an evaluation of the uncertainty in the calculation, to demonstrate that the improved evaluation model maintains an adequate conservatism of safety factor.

The NRC has decided to proceed with the rulemaking, but in the form of a more comprehensive amendment based on (1) the comments received since the publication of the 1978 notice of proposed rulemaking, (2) the additional research conducted and experience gained since the 1978 notice, and (3) recent experience applying the methods of SECY-83-472.

A report, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG 1230 is being prepared. It summarizes the extensive ECCS research that has been conducted. A draft version of this report will be available at the NRC Public Document Room, 1717 H Street NW., Washington, DC 20555, 30 days following publication of this proposed rule in the Federal Register.

Summary of Proposed Rule Changes

Section 50.46 Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors

Section 50.46(a)(1) would be amended and redesignated § 50.46(a)(1)(i) to delete the requirement that the features of Section I of Appendix K to 10 CFR Part 50 be used to develop the evaluation model. This section would require that an acceptable evaluation model have sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a LOCA. The staff expects that the analytical technique will, to the extent practicable, utilize realistic methods and be based upon applicable experimental data. The amended rule would also require that the uncertainty of the calculation be estimated and accounted for when camparing the results of the calculation to the temperature limits and other criteria of § 50.46(b) so that there is a high probability that the criteria would not be exceeded. The staff expects the realistic evaluation model to retain a degree of conservatism consistent with the uncertainty of the calculation. The proposed rule would not specifically prescribe the analytical methods or uncertainty evaluation techniques to be used. However, guidance would be provided in the form of a Regulatory Guide.[2] It should be noted, as discussed in SECY-83-472, that the NRC has, in the past, found acceptable a method for estimating the uncertainty that was judged to be at least at the 95% probability level. This probability level of 95% is considered by the staff to meet the high level of probability required by the rule. It is also recognized that the probability cannot be determined using totally rigorous mathematical methods due to the complexity of the calculations. However, the staff expects that any simplifying assumptions will be stated so that the staff may evaluate them to ensure that they are reasonable. Appendix K, Section II, "Required Documentation," would remain generally applicable, with only minor revisions made to be consistent with the amended rule.

A new § 50.46(a)(1)(ii) would be added to allow the features of Section I of Appendix K to be used in evaluation models as an alternative to performing the uncertainty evaluation specified in the amended § 50.46(a)(1)i). This method would remain acceptable because Appendix K is conservative with respect to the realistic method proposed in the amended § 50.46(a)(1)(i). This would allow both current and future applicants and licensees to use existing evaluation models if they did not need or desire relief from current operating restrictions.

In § 50.46 paragraphs (a)(2) and (3) would be totally revised to eliminate portions of those paragraphs concerned with historical implementation of the current rule. These provisions would be replaced as described in the following paragraphs.

Section 50.46(a)(2) would be revised to indicate that restrictions on reactor operation may be imposed by the Director of Nuclear Reactor Regulation if the ECCS cooling performance evaluations are not consistent with the requirements of §§ 50.46(a)(1) (i) and (ii). Because of this revision, the last sentence of the existing § 50.46(a)(1) has been deleted in the redesignated § 50.46(a)(1)(i).

The current rule contains no explicit requirements concerning reporting and reanalysis when errors in evaluation models are discovered or changes are made to evaluation models. However, current practice has required reporting of errors and changes. The proposed rule would explicity set forth requirements to be followed in the event of errors or changes. The definition of a significant change is currently taken from Appendix K, Section II.1.b which defines a significant change as one which changes calculated cladding temperature by more than 20° F. The revised § 50.46(a)(3) would state specific requirements for reporting and reanalyses when errors in evaluation models are discovered or changes are made to evaluation models. It would require that all changes or errors in approved evaluation models be reported at least annually and would not require any further action by the licensee until the error is reported. Thereafter, although reanalysis is not required solely because of such minor error, any subsequent calculated evaluation of ECCS performance requires use of a model with such error, and any prior errors, corrected. The staff needs to be apprised of even minor errors or changes in order to ensure that they agree with the applicant's or licensee's assessment of the significance of the error or change and to maintain a general knowledge of modifications made since staff review of the evaluation model. However, past experience has shown that many errors or changes to evaluation models are very minor and the burden of immediate reporting cannot be justified for such minor errors because they do not affect the immediate safety or operation of the plant. The staff has therefore proposed periodic reporting to satisfy the need to be apprised of changes or errors without providing undue burden on the applicant or licensee. Such report is to be filed within one year of discovery of the error and shall be reported each year thereafter until a revised evaluation model or a revised evaluation correcting such errors is approved by the NRC staff.

Significant errors may require more timely attention since they may be important to the safe operation of the plant. The proposed rule revision would define a significant error or change as one which results in a calculated peak fuel cladding temperature different by

[1] SECY-83-472, "Emergency Core Cooling System Analysis Methods," November 13, 1983, is available for inspection and copying for a fee at the Commission's Public Document Room at 1717 H Street NW., Washington, DC.

[2] Draft Regulatory Guide, "Best Estimate Calculations of Emergency Core Cooling Systems Performance," will be issued to all licensees and will be available for inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington DC, 30 days following publication of this proposed rule in the Federal Register. Requests for single copies of the draft guide, which many be reproduced, should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Technical Information and Document Control.

more than 50° F, or an accumulation of errors and changes such that the sum of the absolute magnitude of the temperature changes is greater than 50° F. More timely reporting (30 days) would be required for such errors or changes. This definition of a significant change is based on staff judgement concerning the importance of errors and changes typically reported to the staff in the past. The proposed rule revision would also allow the staff to determine the schedule for reanalysis based on the importance to safety relative to other applicant or licensee requirements. Errors or changes that would result in the calculated plant performance exceeding any of the criteria of § 50.46(b) would mean that the plant is not operating within the requirements of the regulations and would require immediate reporting as required by § 50.55(e), § 50.72 and § 50.73 and immediate steps to bring the plant into compliance with § 50.46.

Appendix K ECCS Evaluation Models. Amendments would be made to Appendix K, Section I.C.5.b, to modify post-CHF heat transfer correlations listed as acceptable. The "McDonough" reference would be replaced with a later paper which is more generally available and which includes additional data.

The heat transfer correlation of Dougall and Rohsenow, listed as an acceptable heat transfer correlation in Appendix K, paragraph I.C.5.b, would be removed under the proposed rule revision. Research performed since Appendix K was written has shown that this correlation overpredicts heat transfer coefficients under certain conditions and therefore can produce nonconservative results. Since the Dougall-Rohsenow correlation is now known to be nonconservative under certain conditions, it is appropriate to no longer reference it as a generally acceptable correlation. A number of applicants and licensees currently use the Dougall-Rohsenow correlation in approved evaluation models. Because of this, the staff has considered how this change should be implemented. There is no justification on grounds of safety for requiring that applicants and licensees making use of Dougall-Rohsenow revise their evaluation models at this time. This is appropiate (even though part of the approved evaluation model, Dougall-Rohsenow, is now known to be nonconservative) because the existing evaluation models are known to contain a large degree of overall conservatism even while using the Dougall-Rohsenow correlation. This large overall conservatism has been demonstrated through comparisons between evaluation model calculations and calculations using NRC's best estimate computer codes. The cost of revising the evaluation models would be high with no real benefit to safety. Thus requiring that the applicants and licensees remove the Dougall-Rohsenow correlation from their evaluation models could not be justified on a cost-benefit basis.

A new Section I.C.5.c would be added to Appendix K to state the Commission's requirements regarding continued use of the Dougall-Rohsenow correlation in existing evaluation models. Evaluation models which make use of the Dougall-Rohsenow correlation and have been approved prior to the effective date of this proposed rule revision may continue to use this correlation as long as no changes are made to the evaluation model which significantly reduce the current overall conservatism of the evaluation model. If the applicant or licensee submits proposed changes to an approved evaluation model, or submits corrections to errors in the evaluation model which significantly reduce the existing overall conservatism of the model, continued use of the Dougall-Rohsenow correlation under conditions where nonconservative heat transfer coefficients result would no longer be acceptable. For this purpose, a significant reduction in overall conservatism has been defined as a "net" reduction in calculated peak clad temperature of at least 50° F from that which would have been calculated using existing evaluation models. A reduction in calculated peak clad temperature could potentially result in an increase in the actual allowed peak power in the plant. An increase in allowed plant peak power with a known nonconservatism in the analysis would be unacceptable. This definition of a significant reduction in overall conservatism is based on staff judgement regarding the size of the existing overall conservatism in evaluation model calculations relative to the conservatism required to account for overall uncertainties in the calculations.

Appendix K, Section II.1.b, would be removed since this requirement would be clarified under the amended § 50.46(a)(3). Likewise, Appendix K, Section II.5, would be amended to account for the fact that not all evaluation models will be required to use the features of Appendix K, Section I. These minor changes to Appendix K will not affect any existing approved evaluation models since the changes are either "housekeeping" in nature or are changes to "acceptable features," not "required features."

With respect to the proposed rule changes identified in this summary, the Adivsory Committee on Reactor Safeguards requests the public's comments on whether the existing rule should be "grandfathered" indefintely. That is:

1. Should the conservative ECCS evaluation method of Appendix K be permitted indefinitely or should this aspect of the ECCS rule be phased out after some period of time?

Further, Commissioner Asselstine requests the public's comments on the following:

2. Should this rule change include an explicit degree of conservatism that must be applied to the evaluation models?

3. This rule change would allow a 5 to 10 percent increase in the fission product inventory that could be released from any core meltdown scenario. Should this rule change explicity prohibit any increase in approved power levels until all severe accident issues and unresolved safety issues are resolved?

4. Should the technical basis for this proposed rule change be reviewed by an independent group such as the American Physical Society?

Finding Of No Significant Environmental Impact: Availability

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR Part 51, that this rule, if adopted, would not be a major Federal actions significantly affecting the quality of the human environment and therefore an environmental impact statement is not required. The primary effect of the rule would be to allow an increase in the peak local power in the reactor. This could be used to either tailor the power shape within the reactor or increase the total power. Changing the power shape without changing the total power would have a negligible effect on the environmental impact. The total power could also be increased, but would be expected to be increased by no more than about 5% to 10% due to hardware limitations in existing plants. This 5% to 10% power increase is not expected to cause difficulty in meeting the existing environmental limits. The only change in non-radiological waste would be an increase in waste heat rejection commensurate with any increase in power. For stations operating with an open (once through) cooling system, this additional heat would be directed to a surface water body. Discharge of this heat is regulated under the Clean Water Act administered by the U.S. Environmental Protection Agency (EPA) or designated state agencies. It is not

intended that NRC approval of increased power level affect in any way the responsibility of the licensee to comply with the requirements of the Clean Water Act. The environmental assessment and finding of no significant impact on which this determination is based are available for inspection at the NRC Public Document Room, 1717 H Street, NW., Washington, DC. Single copies of the environmental assessment and the finding of no significant impact are available from L.M. Shotkin, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington DC. 20555, telephone (301) 443-7825.

Paperwork Reduction Act Statement

This proposed rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). This rule has been submitted to the Office of Management and Budget for review and approval of the paperwork requirements.

Regulatory Analysis

The Commission has prepared a draft regulatory analysis for this proposed regulation. The analysis examines the cost and benefits of the alternatives considered by the Commission. The draft regulatory analysis is available for inspection and copying for a fee at the NRC Public Document Room, 1717 H Street NW., Washington, DC. Single copies of the analysis may be obtained from L.M. Shotkin, Office of Nuclear Regulatory Research, Washington, DC 20555, telephone (301) 443-7825.

The Commission requests public comment on the draft analysis. Comments on the draft analysis may be submitted to the NRC as indicated under the ADDRESSES heading.

Regulatory Flexibility Certification

As required by the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission certifies that this rule, if adopted, will not have a significant economic impact upon a substantial number of small entities. This proposed rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration in 13 CFR Part 121. Since these companies are dominant in their service areas, this proposed rule does not fall within the purview of the Act.

Backfit Analysis

Although a backfit analysis is not required by 10 CFR 50.109 because the proposed rule does not require applicants or licensees to make a change but only offers additional options, the factors in 10 CFR 50.109(c) have been analyzed as indicated below. More detailed information relevant to this backfit analysis may be found in the regulatory analysis referenced above.

1. Statement of the specific objectives that the proposed backfit is designed to achieve

The objective of the proposed rule is to modify 10 CFR 50.46 and Appendix K to permit the use of realistic ECCS evaluation models. More realistic estimates of ECCS performance, based on the improved knowledge gained from recent research on ECCS performance, would remove unnecessary operating restrictions.

2. General description of the activity that would be required by the licensee or applicant in order to complete the backfit

The proposed amendment would allow alternative methods to be used to demonstrate that the ECCS would protect the nuclear reactor core during a postulated design basis loss-of-coolant accident (LOCA). While continuing to allow the use of current Appendix K methods and requirements, the proposed rule would also allow the use of more recent information and knowledge currently available to demonstrate that the ECCS would perform its safety function during a LOCA. If an applicant or licensee elected to use a new realistic model they would have to provide sufficient supporting justification to validate the model and include comparisons to experimental data and estimates of uncertainty. In accounting for the uncertainty, the analysis would have to show, with a high level of probability, that the ECCS performance criteria are not exceeded.

3. Potential change in risk to the public from the accidental offsite release of radioactive materials

The proposed rule could result in increased local power within the reactor core and possibly increases in total power. Power increases on the order of 5-10% will have an insignificant effect on risk. One effect of increased power would be to increase the fission product inventory. A five percent power increase would result in a five percent increase in fission products. Thus, five percent more fission products could be released during core melt scenarios and potentially released to the environment during severe accidents.

The proposed rule would still require that fuel rod peak cladding temperature (PCT) remain below 2200°F. Because research indicates that significant fuel damage will not occur until 2600°F, a 400°F safety margin will remain. However, reactors choosing to increase power by five to ten percent would be operating with less margin between the PCT and the 2200°F limit than previously. The increased risk represented by this decrease in margin and increase in fission product inventory is negligible and falls within the uncertainties of PRA risk estimates. In addition, other safety limits, such as departure from nucleate boiling (DNB), and operational limits, such as turbine design, would limit the amount of margin reduction permitted under the revised rule. The proposed rule could also potentially reduce the risk from pressurized thermal shock by allowing the reactor to be operated in a manner which reduces the neutron fluence to the the vessel.

4. Potential impact on radiological exposure to facility employees

Since the primary effect of the proposed rule involves the calculational methods to be used in determining the ECC cooling performance, it is expected that there will be an insignificant impact on the radiological exposure to facility employees. Because of the reduced LOCA restrictions resulting from the new calculations it is possible for the plant to achieve more efficient operation and improved fuel utilization with improved maneuvering capabilities. As a result, it is conceivable that there could be a reduction in radiological exposure if the fuel reloads can be reduced. This effect is not expected to be very significant.

5. Installation and continuing costs associated with the backfit, including the cost of facility down times or the cost of construction delay

LOCA considerations resulting from the present rule are restricting the optimum production of nuclear electric power in some plants. These restrictions can be placed into the following three categories:

(1) Maximum plant operating power,
(2) Operational flexibility and operational efficiency of the plant, and
(3) Availability of manpower to work on other activities.

The effect of the proposed rule will vary from plant to plant. Some plants may realize savings of several million dollars per year in fuel and operating costs. Significantly greater economic benefit would be realized by plants able to increase total power as a result of the proposed rule. The regulatory analysis cited above indicates that the total present value of the energy replacement cost savings for a five percent power upgrade would vary between 10 and 150 million dollars depending on the plant. Additional information concerning these potential cost savings are included in the regulatory analysis.

6. The potential safety impact of changes in plant or operational complexity including the effect on other proposed and existing regulatory requirements

There are safety benefits derivable from alternative fuel management schemes that could be utilized if the proposed changes were implemented. The higher power peaking factors that would be allowed with the revised rule could provide greater flexibility for fuel designers when attempting to reduce neutron flux at the vessel. This can result in a corresponding reduction in risk from pressurized thermal shock.

The reduced cladding temperatures that would be calculated under the proposed rule offers the possibility of other design and operational changes that could result from the lower calculated temperatures. ECCS equipment numbers, sizes or surveillance requirements might be reduced and still meet the ECCS design criteria (if not required to meet other licensing requirements). Another option may be to increase the diesel/generator start time duration.

In summary, the effect of the proposed rule on safety would have both potential positive and negative aspects. The potential for reduction of ECC systems in existing or new plants is present. However, several positive aspects may also be realized under the proposed rule. While the net effect on safety would be plant specific, the effect is believed to be small.

7. The estimated resource burden on the NRC associated with the proposed backfit: and the availability of such resources

The major staff resources required under the proposed rule change would be to review the realistic models and uncertainty analysis required by the revised ECCS Rule. Based on previous experience with the General Electric Co. SAFER model and the learning that has resulted from these efforts, it is estimated that approximately one staff year would be required to review each generic model submitted. There are four major reactor vendors (GE already has a revised evaluation model approved under the existing Appendix K for jet pump plants but is currently working on a new evaluation model for non-jet pump plants and may update their methodology under a new rule) and several fuel suppliers and utilities which perform their own analyses and potentially might submit generic models for review. However, it is expected that only 3 or 4 generic models would be submitted since not all plants would benefit from the rule change. Thus, about 3–4 staff years would be required to review the expected generic models. Once a generic model is approved, the plant specific review is very short. In addition, several vendors are currently planning to submit realistic models in conjunction with the use of SECY–83–472. Therefore, staff resources would be expended to review these models in any event. Since these models would not change as a result of the revised ECCS rule, there should be no net increase in resources required over that already planned to be expended. In summary, while it is difficult to accurately estimate, it is expected that the proposed rule change will have a small overall impact on NRC resources.

8. The potential impact of differences in facility type, design or age on the relevancy and practicality of the proposed backfit

The degree to which the proposed rule would affect a particular plant depends on how limited the plant is by the LOCA restrictions. The Babcock and Wilcox (B&W) and Combustion Engineering (CE) companies have informally indicated that they do not feel that the plants which they design are limited by LOCA and, therefore, B&W and CE plants would not be affected. General Electric Co. (GE) plants do tend to be limited in operation by LOCA restrictions and would benefit from relief from LOCA restrictions. However, this relief is already available for most GE plants through the recently approved SAFER evaluation model. Any additional relief due to a rule change would be of little further benefit. Westinghouse (*W*) plants would appear to directly benefit from relaxation of LOCA limits. *W* plants represent the largest number of plants being constructed. *W* indicates that most of these plants are limited by LOCA considerations.

9. Whether the proposed backfit is interim or final and if interim, the justification for imposing the proposed backfit on an interim basis

The proposed rule, when made effective, would be done so in final form and not on an interim basis. It would continue to permit the performance of ECCS cooling calculations using either realistic models or models in accord with Appendix K.

List of Subjects in 10 CFR Part 50

Antitrust, Classified information, Fire prevention, Incorporation by reference, Intergovernmental relations, Nuclear power plants and reactors, Penalty, Radiation protection, Reactor siting criteria, Reporting and Recordkeeping requirements.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 553, the NRC is proposing to adopt the following amendments to 10 CFR Part 50.

PART 50—DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for Part 50 is revised to read as follows:

Authority: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1248, (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95–601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185, 68 Stat. 936, 955, as amended (42 U.S.C. 2131, 2235); sec. 102, Pub. L. 91–190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.23, 50.35, 50.55, 50.56 also issued under sec. 185, 68 Stat. 955 (45 U.S.C. 2235). Sections 50.33a, 50.55a, and Appendix Q also issued under sec. 102, Pub. L. 91–190, 83 Stat. 853 (42 U.S.C. 4322). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97–415, 96 Stat. 2073, (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80–50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Section 50.103 also issued under sec. 108, 68 stat. 939, as amended (42 U.S.C. 2138). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273), §§ 50.10(a), (b), and (c), 50.44, 50.46, 50.48, 50.54, and 50.80(a) are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); § 50.10(b) and (c) and 50.54 are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and

§§ 50.55(e), 50.59(b), 50.70, 50.71, 50.72, 50.73, and 50.78 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

§§ 50.2, 50.10, 50.21, 50.22, 50.23, 50.30, 50.33, 50.33a, 50.34, 50.35, 50.37, 50.38, 50.41, 50.42, 50.43, 50.44, 50.47, 50.53, 50.54, 50.55, 50.55a, 50.56, 50.70, 50.80, 50.103, and Appendices A, E, F, L, and Q [Amended]

2. The authority citations following §§ 50.2, 50.10, 50.21, 50.22, 50.23, 50.30, 50.33, 50.33a, 50.34, 50.35, 50.37, 50.38, 50.41, 50.42, 50.43, 50.44, 50.47, 50.53, 50.54, 50.55, 50.55a, 50.56, 50.70, 50.80, 50.103, and Appendices A, E, F, L, and Q are removed.

3. In § 50.46, paragraph(a) is revised to read as follows:

§ 50.46 Acceptance criteria for emergency core cooling systems for light water nuclear power reactors.

(a)(1)(i) Each boiling and pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Ziracloy cladding shall be provided with an emergency core cooling system (ECCS) which shall be designed such that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance shall be calculated in accordance with an acceptable evaluation model and shall be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model shall include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data shall be made and uncertainties in the analysis method and inputs shall be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty shall be accounted for so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II, Required Documentation, sets forth the documentation requirements for each evaluation model.

(ii) Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of Appendix K, ECCS Evaluation Models.

(2) Restrictions on reactor operation may be imposed by the Director of Nuclear Reactor Regulation if it is found that the evaluations of ECCS cooling performance submitted are not consistent with paragraphs (a)(1)(i) and (ii) of this section.

(3)(i) Each applicant for or holder of an operating license or construction permit shall estimate the effect of any change to or error in an acceptable evaluation model or in the application of such a model to determine if the change or error is significant. For this purpose, a significant change or error is one which results in a calculated peak fuel cladding temperature different by more than 50°F from the temperature calculated for the limiting transient using the last acceptable model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50°F.

(ii) For each change to or error discovered in an acceptable evalution model or in the application of such a model which affects the temperature calculation, the applicant or licensee shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in § 50.4. If the change or error is significant, the applicant or licensee shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking such other action as may be needed to show compliance with § 50.46 requirements. This schedule may be developed using an integrated scheduling system previously approved for the facility by the NRC. For those facilities not using an NRC approved integrated scheduling system, a schedule will be established by the NRC staff within 60 days of receipt of the proposed schedule. Any change or error correction that results in a calculated ECCS performance that does not conform to the criteria set forth in paragraph (b) of this section is a reportable event as described in § 50.55(e), § 50.72 and § 50.73. The affected applicant or licensee shall propose immediate steps to demonstrate compliance or bring plant design or operation into compliance with § 50.46 requirements.

* * * * *

4. In 10 CFR Part 50 Appendix K, paragraph II.1.b is removed, paragraph II.1.c is redesignated II.1.b, the text of paragraph I.C.5.b and paragraph II.1.b and II.5 are revised, and a new section I.C.5.c added to read as follows:

Appendix K—ECCS Evalution Models

* * * * *

I. Required and Acceptable Features of the Evalution Models* * *

C. *Blowdown Phenomena* * **

5. *Post-CHF Heat Transfer Correlations.* * **

b. The Groeneveld flow film boiling correlation (equation 5.7 of D. C. Groeneveld, "An Investigation of Heat Transfer in the Liquid Deficient Regime's," AECL-3281, revised December 1969) and the Westinghouse correlation of steady-state transition boiling ("Proprietary Redirect/Rebuttal Testimony of Westinghouse Electric Corporation," USNRC Docket RM-50-1, page 25-1, October 28, 1972) are acceptable for use in the post-CHF boiling regimes. In addition the transition boiling correletion of McDonough, Milich, and King (J. B. McDonough, W. Milich, E.C. King, "An Experimental Study of Partial Film Boiling Region with Water at Elevated Pressures in a Round Vertical Tube," Chemical Engineering Progress Symposium Series, Vol 57, No. 32, pages 197-206, (1961) is suitable for use between nucleate and film boilding. Use of all these correlations shall be restricted as follows:

* * * * *

c. Evaluation models approved after (effective date of rule) which make use of the Dougall-Rohsenow flow film boiling correlation (R.S. Dougall and W.M. Rohsenow, "Film Boiling on the Inside of Vertical Tubes with Upward Flow of Fluid at Low Qualities, MIT Report Number 9079 26, Cambridge, Massachusetts, September 1963) shall not use this correlation under conditions where nonconservative predictions of heat transfer result. Evaluation models which make use of the Dougall-Rohsenow correlation and were approved prior to (effective date of rule) continue to be acceptable until such time that a change is made to, or an error is corrected in, the evaluation model that results in a significant reduction in the overall conservatism in the evaluation model. At that time continued use of the Dougall-Rohsenow correlation under conditions where nonconservative predictions of heat transfer result would no longer be acceptable. For this purpose, a significant reduction in the calculated peak fuel cladding temperature of at least 50°F from that which would have been calculated on (effective date of rule) due either to individual changes or error corrections or the net effect of an accumulation of changes or error corrections.

II. Required Documentation

1.a. * * *

b. A complete listing of each computer program, in the same form as used in the evalutation model, shall be furnished to the Nuclear Regulatory Commission upon request.

* * * * *

5. General Standards for Acceptability—Elements of evaluation models reviewed will include the technical adequacy of the calculational methods, including for models

covered by § 50.46(a)(1)(ii), compliance with required features of Section I of this Appendix K; and, for models covered by § 50.46(a)(1)(i), assurance of a high level of probability that the performance criteria of § 50.46(b) would not be exceeded.

Dated at Washington, DC this 26th day of February, 1987.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 87-4431 Filed 3-2-87; 8:45 am]

BILLING CODE 7590-01-M

SECURITIES AND EXCHANGE COMMISSION

17 CFR Part 240

[Rel. No. 34-24135; File No. S7-5-87]

Request for Comments on Proposed Rules 3a43-1 and 3a44-1

AGENCY: Securities and Exchange Commission.

ACTION: Proposed Rulemaking.

SUMMARY: The Commission is publishing for comment proposed rules to implement provisions of the Government Securities Act of 1986 that authorize the Commission, after consultation with the Commodity Futures Trading Commission, to except from the definitions of government securities broker and government securities dealer certain persons directly or indirectly regulated by the Commodity Futures Trading Commission whose government securities activities are incidental to their futures business. The first proposed rule defines as incidental certain transactions for customers by futures commission merchants, primarily as agent, subject to conditions designed to assure that customer funds and securities are safeguarded and that such customer transactions are not advertised or solicited. The second proposed rule defines as incidental certain principal transactions by CFTC-regulated persons including transactions pursuant to futures contracts, exchange for physicals transactions with other CFTC-regulated persons, and certain investment transactions and proprietary hedging and arbitrage transactions with government securities brokers, government securities dealers, and, in some cases, banks.

DATE: Comments should be submitted by April 2, 1987.

ADDRESSES: All comments should be submitted in triplicate to Jonathan G. Katz, Secretary, Securities and Exchange Commission, Washington, DC 20549, and should refer to File No. S7-5-87. All submissions will be available for public inspection at the Commission's Public Reference Section, 450 Fifth Street, N.W., Washington, DC 20549.

FOR FURTHER INFORMATION CONTACT: Lynne G. Masters, Esq., at (202) 272-2848, Division of Market Regulation, Securities and Exchange Commission, 450 Fifth Street, NW., Washington, DC 20549.

SUPPLEMENTARY INFORMATION:

Introduction

On October 28, 1986, the Securities Exchange Act of 1934 (the "Exchange Act") was amended by Pub. L. No. 99-571, the Government Securities Act of 1986 (the "Government Securities Act"). The Government Securities Act provides for the regulation of government securities brokers and government securities dealers. The regulatory system to be established under the Government Securities Act is a limited one in that it does not provide for regulation that would affect particular transactions in government securities, *e.g.*, margin or suitability regulation. Instead, the Government Securities Act requires the Secretary of the Treasury (the "Treasury") to adopt rules concerning the financial responsibility, protection of securities and funds, recordkeeping, reporting, and audit of government securities brokers and government securities dealers. The Commission is to provide for the registration of government securities brokers and government securities dealers that are not financial institutions or registered broker-dealers. Financial institutions and registered broker-dealers are required to file notice of their government securities broker or government securities dealer status with their appropriate regulatory agency.

Under the Government Securities Act, the terms government securities broker [1] and government securities dealer [2] do not include any person registered with the Commodity Futures Trading Commission ("CFTC"), any contract market designated by the CFTC, such a contract market's affiliated clearing organization [3] or any floor trader on such a contract market (hereinafter collectively referred to as "CFTC-regulated persons") solely because such person effects transactions in government securities that the Commission, after consultation with the CFTC, has determined to be incidental to such person's futures-related business.

The Government Securities Act does not set forth standards the Commission is to apply in making its determination. Accordingly, the standards for rulemaking and definitions set forth in sections 23(a) and 3(b) of the Exchange Act apply. These sections authorize rulemaking necessary and appropriate to implement the provisions of the Exchange Act, including the Government Securities Act, and definitions of terms consistent with the provisions and purposes of the Exchange Act. The purposes of the Government Securities Act are set out in findings in section 1(b) of the Government Securities Act. They include "to impose adequate regulation" and "appropriate financial responsibility

[1] New section 3(a)(43) of the Exchange Act defines a government securities broker as:

any person regularly engaged in the business of effecting transactions in government securities for the account of others, but does not include—

(A) any corporation the securities of which are government securities under subparagraph (B) or (C) of paragraph (42) of this subsection; or

(B) any person registered with the Commodity Futures Trading Commission, any contract market designated by the Commodity Futures Trading Commission, such contract market's affiliated clearing organization, or any floor trader on such contract market, solely because such person effects transactions in government securities that the Commission, after consultation with the Commodity Futures Trading Commission, has determined by rule or order to be incidental to such person's futures-related business.

[2] New section 3(a)(44) of the Exchange Act defines a government securities dealer as:

any person engaged in the business of buying and selling government securities for his own account, through a broker or otherwise, but does not include—

(A) any person insofar as he buys or sells such securities for his own account, either individually or in some fiduciary capacity, but not as a part of a regular business;

(B) any corporation the securities of which are government securities under subparagraph (B) or (C) of paragraph (42) of this subsection;

(C) any bank, unless the bank is engaged in the business of buying and selling government securities for its own account other than in a fiduciary capacity, through a broker or otherwise; or

(D) any person registered with the Commodity Futures Trading Commission, any contract market designated by the Commodity Futures Trading Commission, such contract market's affiliated clearing organization, or any floor trader on such contract market, solely because such person effects transactions in government securities that the Commission, after consultation with the Commodity Futures Trading Commission, has determined by rule or order to be incidental to such person's futures-related business.

[3] The CFTC Division of Trading and Markets has advised the Commission's staff that many commodities clearing corporations are not legally controlling, controlled by, or under common control with designated contract markets ("exchanges"). Nevertheless, in this context, the Commission proposes to interpret the term "affiliated clearing organization" to include those clearing organizations that have a contractual or customary arrangement to clear futures contracts traded on a particular futures exchange.

ENCLOSURE B

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE SECRETARY

ACTION - Beckjord

Cys: Stello
Roe
Rehm
Sniezek
Denton
Murray
LShotkin
Shelton
Lesar

January 9, 1987

MEMORANDUM FOR: Victor Stello, Jr.
Executive Director for Operations

FROM: Samuel J. Chilk, Secretary

SUBJECT: SECY-86-318 - REVISION OF THE ECCS RULE CONTAINED IN APPENDIX K AND SECTION 50.46 OF 10 CFR PART 50

This is to advise you that the Commission (with all Commissioners agreeing) has approved publication of the proposed revision to the ECCS rule in 10 CFR 50.46 and Appendix K to 10CFR 50 subject to the following:

a. the methodology used to evaluate uncertainty should be subject to peer and ACRS review.

b. the ACRS question of whether the current ECCS rule should be grandfathered indefinitely should be submitted to the public for comments; and

c. the FRN should note that Commissioner Asselstine requests public comments on the following:

(1) Should this rule change include an explicit degree of conservatism that must be applied to the evaluation models?

(2) This rule change would allow a 5 to 10 percent increase in the fission product inventory that could be released from any core meltdown scenario. Should this rule change explicitly prohibit any increase in approved power levels until all severe accident issues and unresolved safety issues are resolved?

(3) Should the technical basis for this proposed rule change be reviewed by an independent group such as the American Physical Society?

Rec'd Off. 1-12-87
9:30 AM

The proposed rule should be modified to incorporate the above items and forwarded for signature and publication in the Federal Register.

~~(EDO)~~ - (SECY SUSPENSE: 1/30/87)
(RES)

cc: Chairman Zech
Commissioner Roberts
Commissioner Asselstine
Commissioner Bernthal
Commissioner Carr
OGC - H Street

ENCLOSURE C

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 16, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACRS COMMENTS ON CODE SCALING, APPLICABILITY AND UNCERTAINTY METHODOLOGY FOR DETERMINATION OF UNCERTAINTY ASSOCIATED WITH THE USE OF REALISTIC ECCS EVALUATION MODELS

During the 329th meeting of the Advisory Committee on Reactor Safeguards, September 10-12, 1987, we reviewed the methodology developed by the NRC Office of Nuclear Regulatory Research for determination of the overall uncertainty associated with the use of realistic models, including related computer codes, for the calculation of thermal-hydraulic phenomena associated with loss of coolant accidents (LOCAs). In our review, we had the benefit of discussions with representatives of the Office of Nuclear Regulatory Research (RES) and the Office of Nuclear Reactor Regulation (NRR). Subcommittee meetings during which this topic was discussed were held on April 29-30, 1986, August 28, 1986, April 29-30, 1987, and August 4, 1987. We also had the benefit of the documents referenced.

A recently proposed revision to the ECCS Rule (10 CFR 50.46 and Appendix K) will permit use of realistic or "best estimate" methods in demonstrating that a peak cladding temperature (PCT) of 2200°F will not be exceeded during a LOCA. This is in contrast to the original version of the rule which insisted on the use of a number of conservative assumptions which were believed to provide an overestimate of PCT large enough to account for uncertainties. With the new rule change, a licensee may demonstrate that the calculated PCT, when adjusted with an appropriate allowance for overall uncertainty, has an estimated 95% probability of not exceeding 2200°F. In our September 16, 1986 letter to you commenting on the proposed ECCS Rule, we noted the following:

> "The acceptability of realistic evaluation models rests on the development of satisfactory methodology for determination of the overall uncertainty. Most of the development work needed here is either ongoing or planned by the Office of Nuclear Regulatory Research. We recommend that the methodology used to evaluate uncertainty be subjected to peer review. We also wish to review this work."

RES has developed a method for quantifying uncertainty in PCT which it refers to as the Code Scaling, Applicability and Uncertainty (CSAU) evaluation methodology. CSAU is designed to address uncertainties in the capability of a code to extrapolate small-scale test data to full scale, to correctly assess a particular sequence of events, and to account for variability in important parameters. The focus of CSAU is on the important thermal-hydraulic processes with detailed attention given only to those processes which contribute importantly to overall uncertainty. The end product of the CSAU method is an estimate of the total uncertainty associated with the calculation of a key parameter (e.g., PCT) by a given realistic thermal-hydraulic code for a particular plant and a particular accident transient.

It must be recognized that absent an abundance of full-scale LWR plant transient data, it is necessary to rely substantially on engineering judgement in lieu of a rigorous statistical analysis. The CSAU methodology systematizes the application of this judgment for the derivation of a quantitative allowance for uncertainty.

We believe that the CSAU method proposed by RES offers an acceptable means to estimate uncertainty associated with the use of realistic codes. However, we wish to note the following:

- The CSAU methodology has not yet been tested over a wide range of applications. Currently, RES is in the process of demonstrating the applicability of the method by using it to determine the uncertainties resulting from a large break LOCA calculation using the TRAC PF1/MOD-1 code. While it appears that CSAU will be successfully applied to TRAC, we recommend that RES complete an adequate evaluation before the methodology is judged acceptable for use in regulatory actions.

- Before CSAU can be applied to a given code, complete documentation (e.g., code manual, model and correlation quality assurance document, and assessment reports) is necessary. In the past, such thorough documentation has not always been available for licensing codes. We recommend that steps be taken to ensure that future development of codes for licensing activities be performed in a manner that ensures completion and availability of needed documentation before the code is released.

- The codes used to analyze thermal-hydraulic behavior are very large and complex. Validity of calculated results is dependent on the competence of the code user and the way in which the code is used. For CSAU to be effective, the code developers, assessors and users must use the code consistently. We recommend the NRC Staff take the necessary steps to ensure that proper controls are established.

- In order to ensure the ultimate success of the method, we believe it is necessary for RES to direct its experimental thermal-hydraulic programs appropriately to the needs of CSAU. These experimental programs include the MIST, 2D/3D, and ROSA-IV cooperative efforts.

- We wish to caution that use of the CSAU method for regulatory applications will require the maintenance of an ongoing high level of competence and experience on the part of the NRC Staff members. We suggest that the NRR call upon RES for such support as necessary.

We are encouraged by the move toward the use of realistic calculations for ECCS/LOCA phenomena. We intend to follow the progress of this effort closely, and we wish to be kept informed.

Additional comments by ACRS Member Harold W. Lewis are presented below.

Sincerely,

WKerr

William Kerr
Chairman

Additional Comments by ACRS Member Harold W. Lewis

I support the Committee's letter, but do wish to add some cautionary notes about the misuse of some familiar words, which can lead to potential misuse of the CSAU (so-called) methodology.

To begin with, I support the move to "realistic" evaluations, since I believe that all evaluations should be made as honestly and realistically as possible, after which regulatory conservatism can be applied cleanly and openly. That is the thrust of this effort, and is fine. Unfortunately, however, the words "best estimate" are often used interchangeably with "realistic" to describe calculational techniques, and that is an error. To a statistician, a best estimate is an estimate taken from the top of a probability distribution, and that is simply a different idea. This is not sophistry, since the misunderstanding of words that have established technical meanings can lead to incorrect calculations. To call an apple an orange does not make it one.

We were also briefed about a set of calculations in which parameters and assumptions were varied to provide a feel for the sensitivity of the

results to the specific assumptions made. That is a reasonable way to learn about the sensitivity, but is not a way to learn about the "uncertainty" in the result, as any statistician would understand the word uncertainty. Statistical uncertainty in its simplest form is based on the concept of random sampling from a population of known characteristics but unknown parameters. In that case, one can learn the uncertainty in an estimate of a parameter by studying the variance in a set of measurements, but that is not the situation here, where the variance in the results bears no relation whatever to any uncertainty, in any credible statistical sense. The only reason for saying this is that in the familiar case of a normal distribution of sample measurements, one can estimate the uncertainty from the variance, and thereby estimate the probability that the mean of the measurements differs from the true value by any ratio. One can also estimate "confidence levels," but that is another saga.

None of that is true here, and this is again not sophistry. In particular, the draft Regulatory Guide supporting the proposed rule has statements about the "95% probability limit," "confidence level," and such things, and even states that the "use of two standard deviations for evaluating the 95% probability level is acceptable." None of this is possible within the framework described, and simply reflects confusion on the part of the Staff about fundamental statistical concepts.

I still support the letter and the program, since it is a major step forward, but repeat a recommendation I have made many times: the NRC would benefit greatly by hiring a few good statisticians. One cannot do competent safety analysis in the presence of uncertainty (popular use of the word) without doing the statistics carefully.

References:

1. U.S. Nuclear Regulatory Commission, Proposed Rule, "Emergency Core Cooling Systems, Revisions to Acceptance Criteria," February 26, 1987.
2. U.S. Nuclear Regulatory Commission, "Request for Comments on Draft Regulatory Guide, 'Best Estimate Calculations of Emergency Core Cooling System Performance,'" March 1987.
3. U.S. Nuclear Regulatory Commission, NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," April 1987.

(2)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 10, 1988

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: PROPOSED REVISION OF THE ECCS RULE CONTAINED IN 10 CFR 50.46 AND APPENDIX K

During the 337th meeting of the Advisory Committee on Reactor Safeguards, May 5-7, 1988, we met with members of the NRC Staff and reviewed the final version of the proposed revision to the emergency core cooling system (ECCS) rule contained in 10 CFR 50.46 and Appendix K. Our Subcommittee on Thermal Hydraulic Phenomena met on April 20, 1988 to discuss this matter. We also had the benefit of discussions with the NRC Staff and of the documents referenced. The ACRS previously commented on the proposal to issue this rule for public comment in a letter dated September 16, 1986.

The proposed revision to the ECCS rule will eliminate the requirement to use the models specified in Appendix K and allow use of realistic models combined with an uncertainty analysis of the overall calculation. Certain criteria in 10 CFR 50.46, such as 2200°F peak cladding temperature and 17% cladding oxidation, would be maintained. The regulatory guide which will accompany the revised rule describes features of a realistic evaluation model acceptable to the NRC Staff and contains guidance on performing the necessary associated uncertainty evaluation.

No changes have been proposed to the final rule version as a result of the public comments received. The regulatory guide has been modified somewhat to clarify the NRC Staff's intent in certain areas.

The ACRS has long advocated use of best estimate or realistic evaluations for safety analysis. We believe the proposed rule is a major step forward in this effort, and we support its adoption. We wish to note the following points:

- Work to demonstrate the Code Scaling, Applicability, and Uncertainty (CSAU) method for the peak cladding temperature calculated to occur in the reflood phase of a large break LOCA has not been completed. This will be needed to establish guidelines for Staff review of future licensee submittals under the new rule. While the CSAU method has been reasonably demonstrated for the so-called

blowdown peak, application to the reflood demonstration will be more difficult. We do not object to plans to proceed with promulgation of the rule change, but we would like to be kept informed about the development of and allowance for uncertainty in the reflood peak temperature.

* We note that the draft Federal Register notice provided to support the rule change has eliminated reference to any claimed safety advantages for the rule. We believe the safety advantages are substantial.

Additional comments by ACRS Member Harold W. Lewis are presented below.

Sincerely,

W Kerr

W. Kerr
Chairman

Additional Comments by ACRS Member Harold W. Lewis

I have no quarrel with the Committee's letter, but want to seize the opportunity to reinforce a point that has been made before. It is stimulated by unsatisfactory answers to questions at the presentation to the Committee.

The CSAU "methodology" purports to be a systematic procedure for estimating the uncertainty in code calculations. That is a laudable objective, and its achievement would be even more laudable. It would be helpful if, in so doing, there were less confusion between the concepts of uncertainty and a probability distribution, and less misuse of the term "confidence limits." These objectives will not be reached unless some professional statisticians become involved. In this case, it is of more than usual importance, since the uncertainty is directly related to the acceptable level of conservatism which must be added to the realistic calculations.

References:

1. U.S. Nuclear Regulatory Commission, Draft SECY paper for the Commissioners from V. Stello, EDO, "Revision to the ECCS Rule Contained in Appendix K and Section 50.46 of 10 CFR Part 50," provided to the ACRS, April 20, 1988.
2. U.S. Nuclear Regulatory Commission, Draft NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," Office of Nuclear Regulatory Research, dated April 1987.

89 Windsor Road
Waban, MA 02168
617/253-5296
February 24, 1988

Mr. Eric S. Beckjord
Director
Office of Nuclear Regulatory Research
United States Nuclear Regulatory
Commission
Washington, DC 20555

Dear Mr. Beckjord:

This letter presents the report of the peer review group convened by your office to evaluate the Code Scaling, Applicability and Uncertainty (CSAU) Evaluation Methodology being developed under the technical direction of Dr. N. Zuber. This Methodology and a Demonstration calculation were presented to the peer review group (attachment A lists membership) on January 12 and 13, 1988 by the team which is developing this Methodology. The peer review group was asked to answer the following questions:

1. Is the Methodology systematic, logical and practical?

2. Has the NRC demonstrated a Methodology which shows that best-estimate methods can be used to meet the requirements of the modified ECCS rule? (i.e. a proposed rule which would allow the utilization of a fully best-estimate approach for calculating ECCS performance when accompanied by an estimate of the uncertainty)

In addition, and separately, detailed technical comments on the application of the Methodology were solicited.

The task undertaken by the NRC in preparing and demonstrating this Methodology is large and complex. Additionally, the work is still in its early stages of development. Nevertheless, a reasonable coincidence of views exists among the peer group members regarding the answers to these overall questions. However, there are significant viewpoints and insights presented in the individual member's letters which should be recognized. Examination of these letters, which comprise attachments B-K, should be an integral part of your review of the conclusions of this peer review group.

The Methodology has been developed to establish the uncertainty associated with a realistic calculation of peak clad temperature (PCT) in a LOCA. The realistic value is to be calculated using a numerical code suitable to this task. All suitable codes embody a very large number of parameters of order 100 to 200. The task is to devise a means to identify that subset of these parameters which, together with their uncertainties and a procedure for combining their effects, will allow the desired PCT uncertainty to be evaluated. The Methodology and the Demonstration of this Methodology can be summarized as follows. This summary is presented to provide the context of those items upon which we have subsequent comments.

The Methodology centers first on using expert opinion to limit the phenomena which are to be treated in detail to those considered key to the calculation of the goal parameter, the PCT. Next, by examining the selected code, the supporting data base and specifically, the potential effects of nodalization and scaling, consideration of these key phenomena is focused into the consideration of a limited number of specific parameters. Finally, the uncertainties introduced by each specific parameter are combined to evaluate the total uncertainty in the goal parameter.

The Demonstration applies this Methodology to calculate the PCT for the blowdown phase of a large break LOCA in a 4-loop Westinghouse PWR. The TRAC-PF1/-MOD1/Version 14.3 code, which incorporates about 175 parameters to describe plant parameters and models/correlations, was utilized. The key blowdown phenomena influencing the PCT were identified by a structured process for creating phenomena identification and ranking tables (PIRT) as fuel pin stored energy, 2-phase pump performance and critical flow out the primary system break location. Assessment of the uncertainties associated with the applicabilty and utilization of the code, particularly with respect to code models and scaling and nodilization effects, identified 7 parameters by which to quantify the influence of these 3 key phenomena. Of these 7 parameters, 3 related to fuel pin stored energy (peaking factor, gap conductance, fuel conductivity), 1 related to 2-phase pump performance (a flow degradation parameter), 1 related to break critical flow (a discharge parameter) and 2 related to the heat transfer calculation procedure for PCT (the convective heat transfer coefficient and the minimum temperature for film boiling which separates the transition and film boiling heat transfer regimes and is also used to establish the transition boiling heat flux). The estimated uncertainties of each of these 7 parameters were combined by construction of a response surface from only 140 data points obtained from 7 TRAC runs by utilizing 20 supplemental rods. The overall uncertainty in blowdown PCT, established by this calculational method, was compared with two independent estimates using experimental data (LOFT L2-6 test and other tests in LOFT and four other scaled facilities) and found consistent. The reported overall calculational uncertainty for the Blowdown Peak Clad Temperature is bounded, with 95% probability, by 1379°F.

This Metholodogy can be viewed as comprised of three main components: Problem Definition, Code Selection and Accuracy Assessment, and Execution and Results Interpretation. It is convenient to first present comments specific to these components and then present our response to the two more general questions posed.

1. Problem Definition (includes steps 1, 2, 3 and plant input data of step 11 of the CSAU flow chart).

   The identification of key phenomena, although accomplished through a disciplined procedure, essentially rests on subjective individual assessments. For well-studied problems, the results probably are satisfactory. It is important, however, not to apply this procedure, or more importantly, to trust its results for insufficiently studied

problems. Nevertheless, the Methodology offers no criteria for establishing when expert opinion is so immature that the proposed PIRT approach is inapplicable.

A more subtle difficulty, apparently not well addressed, is how to recognize significant influences from low ranking phenomena arising because their effects are known with only low accuracy. Apparently the procedure does not include this effect of considering the interaction of such low ranking phenomena with higher ranking phenomena.

2. Code Selection and Accuracy Assessment (items 4 through 10).
Topics of relevance here involve data assessment matrix, nodalization and scaling.

The data assessment matrix contains a limited number of tests in reactor-like geometries at various scales. Additionally, many other data have been used to qualify the models and correlations of TRAC. Extensive single effect data in the literature from universities exists which have not been utilitized. A fundamental problem exists regarding use of data in later stages of code assessment which might have earlier been used in code tuning. Therefore, it is beneficial to the integrity of this process of methodology development to emphasize the need to locate and utilize test data which has not been already used in code model assessment. Further, contact with peer group members who emphasized this point could lead the NRC team to unexplored sources of useful data.

Nodalization has been long recognized as a source of variability in numerical calculations. Distortions in calculated results from nodalization is controlled in this Methodology by (1) utilizing the extensive experience available to establish a nodalization scheme on a component by component basis and (2) utilizing the same nodalization for calculating the separate and integral effects tests and the selected nuclear power plant or rationalizing the differences. We observe, however, that specific studies have not been done as part of the development of this Methodology to examine the sensitivity of the selected noding scheme to further refinements. The selected nodalization scheme is therefore a matter of judgement, not strictly mathematically defensible, which might still be sensitive to modest node alterations. An applicant desiring an alternative arrangement will need to make significant demonstration calculations in its support. However, it will be difficult for the NRC to establish the amount of demonstration calculations needed since the sensitivity of results to nodalization changes is not quantified.

The Demonstration contends that no scale effect exists for a blowdown PCT uncertainty of $\pm$ 361^{0}F for data from five test facilities. For the blowdown PCT, the Demonstration calculation subsumes scale effects if any, within the $\pm$ 361^{0}F bands. Therefore, in practice the claim that no scale effects must be considered is correct. However, it is likely that these data physically reflect a scale effect which has not yet been detected. This distinction between the practical Demonstration calculation and the potential physical reality should not be submerged but reflected in the

the description of the Demonstration. Further, it should be recognized that only the blowdown PCT has been explored in detail to date.

The above discussion deals with scale effect in the data. An additional question which does not appear to have been explored is whether the code itself shows a scale effect. It is possible that the code will show an effect of scale even when the data does not. For example, the code may tend to over-predict the peak clad temperature in large scale experiments while it does a good job of predicting peak clad temperatures in small scale tests even though the tests themselves indicate no scale effect. This particular example is a safe scenario because the code indicates a positive bias. However, a code with a negative scale bias is a cause for concern because it can be expected to underpredict the peak clad temperature for a full scale plant. The scale bias of the code can be determined by performing data comparisons with tests at two or three different scales. If there is a negative bias, it should be combined with the other uncertainties to obtain the total code uncertainty.

3. Execution and Results Interpretation (steps 11(part), 12 and 13).

The Demonstration developed a response surface from 140 points of which 60% were double cross products (i.e. two of the 7 variables perturbed from their nominal values) and 25% were triple cross products. While it is recognized that judgement must be applied to bound the number of TRAC calculations, it must be equally recognized that the fundamental foundation for the response surface created in the Demonstration suffers from the truncated number of runs. We point out that (1) no double cross products exist for non-core by hydrodynamic variables (i.e. breakflow and pump degradation, and (2) no run was done to assess whether in the period before DNB the rod uncertainties coupled to the hydrodynamics would give rise to different peak clad temperatures).

Finally, we observe that the Demonstration calculation presented focused solely on the calculation of PCT. While this is the relevant parameter for the modified ECCS rule, the development of confidence in the Demonstration requires that realistic results are obtained for other dependent variables and for the whole temperature response. While the NRC team stated that attention had been given to checking this full range of variables during the development of the Methodology and the Demonstration, little such evidence was presented to the peer review group. General acceptance of such LOCA Demonstration calculations will require some degree of demonstration of consistency of prediction versus data for physically significant variables other than the peak clad temperature.

Our response to the two questions posed is as follows:

(1) Is the Methodology systematic, logical and practical? The Methodology is judged to be systematic and logical. Its practicality is not uniformly accepted because of differing views of the breadth of demonstration calculations needed to reach a judgement. When judged only in terms of the LBLOCA blowdown calculation completed to date, the Methodology appears practical.

However, an equally relevant question which should be addressed here is that of the completeness of the Methodology. We observe that the Methodology does not include quantitative criteria to establish the completeness of many of its steps. We cite as unaddressed examples the required breadth of the phenomena screening process, the required evidence of code maturity, the scope of the assessment matrix, and the amount of data necessary to construct the response surface. Questions of this type which were raised throughout the peer review group meeting were answered by the citation of engineering judgement. At this stage in the development of the Methodology and particularly when the experience and skill of the NRC team is considered, such a response is appropriate. Nevertheless, the objective of establishing quantitative criteria of completeness for relevent steps of the Methodology should be maintained. Additionally, the representation of the Methodology as an objective, near quantiative method for establishing uncertainty should give way to its description as a formal method for combining quantitative analysis and expert opinion in a controlled way to minimize the subjectivity of the experts in arriving at computed values of uncertainty.

(2) Has the Methodology been demonstrated?

At the minimum, a promising start has been made in demonstrating the Methodology for the blowdown PCT in a PWR. While a minority holds that the Demonstration is not yet conclusive, no unresolvable difficulties are identified which would prevent achievement of this goal as work progresses. In the practical sense then, it can be assumed that an acceptable Demonstration of the Methodology to the blowdown PCT is or soon could be in hand.

Significantly, however, this does not assure that demonstrations of the Methodology for the LBLOCA reflood phase and small break LOCA's are achievable. The work to date has demonstrated the importance of prior experience of the experts with the phenomena and of physical simplicity of the phenomena in developing and demonstrating the Methodology. Since LBLOCA blowdown PCT is the easiest parameter to evaluate, success in extending the Methodology to these other LOCA cases cannot be assured.

CLOSING OBSERVATIONS

The peer group did not have sufficient time to review, in detail, the TRAC-PF1/-MOD1 Models and Correlations Report nor Section 4.4 of NUREG-1230 which describes

the CSAU Methodology. Revision of Section 4.4 to produce an updated coherent description of the Methodology is desirable because of the broad impact this Methodology will have on LOCA and probably on other accident parameter acceptance criteria. Other means for comprehensive review of the TRAC Report are recommended because of the central role that this TRAC code has in the Demonstration of the blowdown PCT calculation and will presumably have in other NRC LOCA demonstration calculations. The central role which software quality assurance (SQA) will play in application of this Methodology should be recognized now and steps taken to incorporate and appropriate formal SQA plans into the Methodology. Evidence of the need for formal consideration of SQA was the persistent reference in our meeting to the TRAC Report as a QA document. While the report is a good starting description of the models and correlations utilized, it is far short of a QA document.

The peer group strongly encourages continued work on the development and demonstration of this Methodology. The pursuit of such a program of comprehensive utilization of research results, not only has potential benefits for operating reactors, but offers a focus for planners and researchers in the conduct of future investigations. The benefits also include the comprehensive documentation of computational codes and data bases.

The NRC team is to be complimented for their competence and dedication in developing this Methodology thus far. It is not clear, however, that a significantly less skilled team can evaluate industry submitted Methodologies which differ much from the Methodology proposed. This may likely pose a difficulty in eventually managing the process of evaluating applicant responses to the proposed modified ECCS rule.

Sincerely,

Neil Todreas

Neil E. Todreas
Chairman

NET:ewp

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Subject: **Comments on Peer Review Group Meeting on CSAU Methodology**

Date: **January 29, 1988**

From: **Marvin J. Thurgood**

To:
Dr. Neil E. Todreas, Chairman
Dept. of Nuclear Engineering
Massachusetts Institute of Technology
Room 24-102
Cambridge, Massachusetts 02139

The NRC has proposed a revised ECCS rule to allow licensees to utilize a fully realistic approach for calculating ECCS performance. This approach requires an estimate of the uncertainty in the calculations. The combination of the calculation plus the uncertainty would be compared with an acceptance criteria (i.e. , PCT < 2200 F). The commission directed the NRC staff to develop a methodology to determine calculation uncertainty and to subject this methodology to peer review. The members of the Peer Review Group met in Bethesda, Maryland on January 12 and 13, 1988 where members of the NRC staff and various consultants and national laboratory personnel presented to the group the Code Scaling, Applicability and Uncertainty (CSAU) methodology that has been developed by the NRC staff to meet these requirements. Members of the Peer Review Group were requested to evaluate the methodology and to answer the following questions:

1. Is the methodology systematic, logical and practical?
2. Has the NRC demonstrated a methodology that shows that best estimate methods can be used to meet the requirements of the modified ECCS rule?

In addition, and separately, the CSAU Technical Program Group has invited the PRG members to provide any detailed technical comments on the application of the methodology. It is the purpose of this letter to document my response to these questions as a member of the PRG.

First, I will summarize my review of the methodology by giving my response to the two questions that the PRG has been asked to answer. It is my best technical judgement that the CSAU methodology does provide a systematic, logical and practical means for evaluating the calculation uncertainty for Large Break LOCA transients. I reach this conclusion, in part, because the methodology is defined in fairly general terms that allow a great deal of flexibility in the actual application of the

methodology to any particular scenario. The methodology requires that the code be fully documented, including a users manual, programmers guide, developmental assessment manual, and a model and correlations QA manual that describes the correlations, numerical methods and interpolation and flow regime schemes used in the code. These are very important steps and the NRC should require that these items be provided for a frozen version of the code. It is my experience that it is difficult to predict the effect that minor code changes may have on a calculation or data comparison. For this reason, I do not believe that the NRC should accept statements from licensees that code changes do not effect calculated results unless an adequate number of data comparisons have been repeated with the frozen version of the code to clearly demonstrate that the effect of the code changes is indeed minimal. Much of this methodology, and indeed the nature of the problem, requires that the regulator use a great deal of engineering judgement to evaluate a license application. I believe that presenting the results of data comparisons that were done with a large number of different code versions greatly clouds the regulators vision of the code's actual predictive capability and should be discouraged. If the code has retained its predictive capability through all of its revisions it is generally not prohibitively expensive to repeat the calculations on the final version. The large expense comes in trying to improve the code's predictive capabilities by changing or tunning correlations or correlation selection logic.

The methodology requires that the code be evaluated as to its applicability to a given transient and reactor system geometry. This is appropriate and basically requires that the code has been assessed against applicable experimental data and has the required geometrical and operational modeling features. Once this has been done, the methodology requires that an assessment matrix be defined that will adequately determine the code's ability to predict the data (uncertainty) and the ability of the noding selected to adequately predict the physical phenomena. In order to accomplish this, the method requires that the same noding be used for the data comparisons as will be used for the plant calculation. This approach is acceptable provided that experimental data is available in facilities that approach the size of the plant and little or no noding sensitivity is indicated in code predictions of data taken from facilities at different scales using the same noding. The NRC should require that noding sensitivity studies be performed to assure that the results are not overly sensitive to noding in cases were no large scale data is available. I believe that using the same noding for both the experiments and the plant eliminates uncertainties associated with noding by putting this uncertainty in the overall uncertainty of the code prediction of the experimental data. However, it is my judgement that this requires that the code be assessed against data from a variety of facilities at different scales.

The methodology then requires that code uncertainty and bias be evaluated and that the effects of scale be determined. These steps are essential and I believe require that the code uncertainty be based on the assessment of the code against a variety of facilities at different scale. The method then requires that NPP operating parameter ranges and uncertainties be determined and that the uncertainties in the data, code predictions, noding, operating and plant conditions be combined in

an appropriate way. The methodology specifically states that the resultant uncertainty is applicable only to a specific scenario in a specific plant using a specific code. I agree with these steps.

For me, the entire methodology rests on the assumption that there is a fairly large base of experimental data at a variety of different scales for the particular transient to which the methodology is to be applied and that the particular frozen code used in the analysis has been assessed against the majority of this data and has done an acceptable job in predicting the major phenomena in the data base in both a qualitative and quantitative way. Because the methodology is defined in rather general terms, the application of the methodology will require that the NRR staff responsible for evaluating licensee applications under the revised rule demand that the licensee demonstrate that an adequate data base has been used to determine the code uncertainty and that the important parameters have been correctly identified and that sensitivity studies have been performed such that the maximum variation in the calculated variables (i.e. PCT) has been obtained. I believe that this requires that the double and even triple cross products of both fluid and thermal variational parameters must be calculated or that the use of bounding values of the most important parameters will result in the prediction of the maximum value of the acceptance criteria variable (PCT).

This leads to my response to the second question, "Has the NRC demonstrated a methodology that shows that best estimate methods can be used to meet the requirements of the modified ECCS rule?" I congratulate the NRC staff on the significant progress that they have made towards demonstrating the methodology that shows that best estimate methods can be used to license nuclear power plants. I believe that the work done thus far provides a fair indication that this can be done. However, I do not believe that the job is complete. The work done thus far indicates that the methodology will work for determining the blowdown peak. It remains to be demonstrated how well it works for the reflood peak. It is also unclear how applicable this method may be to other types of transients, such as small breaks, since this has not been demonstrated at all. Therefore, I cannot say that the methodology has been demonstrated for these cases. I believe that the particulars of the approach will be quite different for these scenarios because of the lack of data and the variety of possible scenarios. The methodology will have to be demonstrated for each new scenario.

I have the following observations on the application of the methodology:

1. The demonstration of the methodology relied heavily on a TRAC calculation of one LOFT test for determining the code uncertainty. I believe that this is inadequate. Comparing the the code against a variety of tests of different scales gives a better feeling for code uncertainty and the ability of the code itself to scale up. I think that the NRC should insist that licensees include simulations of several different facilities in their uncertainty calculation. Having compared the code to different tests facilities during developmental assessment does not incorporate the effect of these data comparisons into the code uncertainty and leaves one feeling uncertain about the codes ability to predict

data in different scale facilities.

2. It was not clear to me from the presentation that all of the primary variables were tested in performing the uncertainty analysis. For example, post chf heat transfer was indicated to be an important parameter in the ranking for blowdown heat transfer yet I saw little to persuade me that the uncertainty in this parameter was adequately investigated and cross correlated with the other parameters. The development of the response surface should include all of the parameters that were identified to be important and it should include a sufficient number of double and triple cross products to assure that the response surface has been correctly determined.

3. The uncertainty for various parameters was determined by looking at specific correlations. For example, the uncertainty in the minimum film boiling temperature was determined. This is necessary but not sufficient. It has been my experience that the greatest uncertainty is not in the values given by a given correlation but in the ability of the code to select the correct correlation at the correct time. In other words, does the code predict inverted annular film boiling when the data indicates dispersed phase film boiling or slug flow when it should be film/mist flow. Errors in flow and heat transfer regime selection logic result in large magnitude uncertainties in the calculation. I think the best way to demonstrate that the code can correctly select the correct regimes is to base the uncertainty analysis on data comparisons with data from a variety of different facilities. If economics becomes a limiting factor then it is better to compare with a single test from five facilities rather than five tests from one facility.

4. Some of the data comparisons from older code versions seemed to be accepted based on a statement from the code developer that the code changes made subsequent to that version would not effect the result. I have found this difficult to do as it is easy to overlook subtle interactions between various thermal hydraulic models. The NRC should require a reasonable number of the calculations to be repeated on the frozen version of the code.

5. The issue of scaling is difficult to address. I think the argument presented at the meeting that the data indicated no scale effect when plotted against linear heat rate and then showing that the TRAC calculation had a certain deviation from the mean of this data and assigning this deviation as the uncertaining due to noding, code scaling, the effect of tunning and "home made correlations" ect. is good supportive information but is not adequate by itself. This method of correlating the data may only say that the power/volume scaling used to design the test facilities is a good scaling method when linear heat rate is the main correlating variable. It says nothing about the codes ability to scale to larger geometries and may not even say much about scale effect of the facility itself on phenomena that may occur at full scale. Here again, I feel it is important to compare the code against facilities of different scale, calculate the code uncertainty at each scale and then see if the code uncertainty has a scale effect. Does the calculated code uncertainty get larger or smaller

as scale increases. This data can then be used to extrapolate the code uncertainty at full scale. Also it is possible to evaluate if the code has a bias to over predict or unpredict the data as scale increases. I don't see that the codes ability to scale-up has been demonstrated at all. This issue becomes even more important for the reflood period were I feel that scale effects are indeed important. Code comparisons and uncertainty for tests from the FLECHT, NRU, CCTF, PKL, LOFT and other reflood facilities should be used rather than several tests from a single facility, CCTF. Doing this allows the code to be tested against a variety of test scales, power ranges and fuel types (nuclear versus non-nuclear). TRAC, in particular, has heavily emphasised LOFT and CCTF in its developmental assessment. Comparisons with other tests would give a good indication if the code has been overly tuned to a restricted set of data.

6. I agree with Westinghouse that a hot bundle should be included in the analysis.

It is my conclusion that the work done thus far demonstrates that the methodology can be applied to the revised appendix k analysis but it is my hope that the NRC will require more of licensees than has been demonstrated thus far.

Dr. Neil Todreas, Chairman
Department of Nuclear Engineering
Massachusetts Institute of Technology
Room 24-102
Cambridge, Massachusetts 02139

January 20, 1988.

Re.: Peer Review Group Meeting,
CSAU Evaluation

Dear Neil:

This letter is in response to your request instructing the Peer Review Group members to provide an individual written assessment of the results of the group meeting that was held in Washington on January 11 & 12, 1988 with the NRC consultants and staff members.

I will begin with *a general statement. The proposed revision to the ECCS Rule is justified.* The experience accumulated over several decades (experience with full scale reactor operations in the field, the availability of a broad experimental data base dealing with the critical phenomena and covering a wide range of equipment scales, the maturation of computational tools, the availability of very well trained and competent professional manpower), at one end, and the economic penalty associated with continuation of the present rule's methodology (which is based on the use of 'conservative' estimates), at the other - provides a strong rational for the effort directed toward the rule change. The procedure behind the revised licensing rule (formulated around the concept of 'best estimate') is based on the wealth of knowledge in the NPP related engineering sciences, and the state-of-the-art computational methodologies. It is for these reasons that *I strongly support the proposed changes.*

Concerning the specifics, we are asked to evaluate Code Scaling, Applicability and Uncertainty (CSAU) methodology. The methodology was presented to us through a particular example

involving a single reactor (PWR) and a single type of accident (LBLOCA). The methodology has three main components: Problem Statement; Code Selection, and Execution and Results Interpretation. This simplified classification was adopted here so that I could group my comments by addressing a block of common issues together. Briefly, the breakdown of the revised classification and their relation to the NRC original organization are shown below.

A. The Problem Statement is comprised of items 1,2,3 (from the CSAU flow chart), plus specification of the plant input data (including all uncertainty in operational parameters and material properties (these items are contained in part in box 11 of the master chart).

B. The Code Selection: items 4 to 10.

C. The Execution and Result Interpretation: part of item 11, and items 12 and 13.

The methodology associated with block A *(Problem Statement) is well formulated and justified. I do not see any deficiencies in the methodology as it pertains to the specific example presented at the meeting, nor do I see any inherent difficulties in extending the methodology to the other types of reactors or different accident scenarios.*

Block B (Code Selection) involves elaborate step-by-step verification that includes: check of the code ability to simulate the main phenomena for a chosen scenario, validation of the closure relations and an overall verification of the code accuracy (by requiring an acceptable agreement between the code prediction and the available experimental data). For a code that uses 'tuning' to force agreement with experimental data obtained from subscaled systems (either through nodalization selection or adjustments in the closure relations), a procedure for the code up-scaling should be provided. The final product is comprised of a working code (with full documentations), specified bounds of uncertainty for the closure relations (which should include effects of the scatter in

experimental information used in their development), and the uncertainty in the predicted value of the critical parameter(s) due to the code up-scaling.

The proposed methodology applied to the main items in this section, as shown through the specific example, is quite satisfactory as far as it goes; the methodology concerning the up-scaling uncertainty has not been presented in necessary details for us to judge its validity; for the case demonstrated at the meeting, (LGLOCA), no scaling effect was detected and therefore there was no need to deal with this issue within the narrow confines of the example. However, this absence of the scale effect might not, and probably would not, carry on to all accident scenarios, and consequently there is a need for a procedure development that would be able to handle the code up-scaling uncertainties. Until this is done the methodology for the code acceptance could not be considered entirely complete.

Block C (Execution and Result Interpretation) requires evaluation of the 'best estimate' and the total uncertainty in the calculated value of the controlling parameter (here PCT). The input parameters necessary for the execution include the plant input data (with uncertainty in operational parameters and material properties, from Block A), the uncertainty in the closure relations, and the uncertainty due to up-scaling (the last two from Block B). Due to the nonlinear nature of the problem, the total uncertainty in the PCT could not be obtained by addition of the individual effects. At the meeting a combined effect due to simultaneous changes in two parameters was demonstrated. Perhaps in an actual licensing situation more then two combined effects need to be considered. Nevertheless, the expansion to multiparametric analysis for the combined uncertainty is constrained only by the lack of additional computer time (rather than by the absence of a correct methodology). Of some concern however is the fact that *the shown example did not indicate how the code up-scaling uncertainty, when*

present, would be included in the total uncertainty. The methodology for this situation remains to be demonstrated.

In conclusion, *the entire program appears to be very well directed; the overall methodology is structured well but it is still not quite finished. I do not see any reason why the continuing effort would not yield shortly a solid and complete methodology which could be confidently employed as a 'blue print' for the 'best estimate' procedure.*

Sincerely,

Bora Mikic

Bora Mikic,
Professor of Mechanical Engineering,
Massachusetts Institute of Technology

P.S. A final comment. I was very impressed with the presentation of the NRC staff and consultants. Most of them displayed a strong affection and respect for the work they were doing. I believe that this, together with the demonstrated competences in their respective areas, are the best guaranties that they will continue to work aggressively and successfully until completion of the project.

B. M.

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue, Argonne, Illinois 60439

January 22, 1988

Professor Neil E. Todreas
Chairman, Peer Review Group
Department of Nuclear Engineering
Massachusetts Institute of Technology
Room 24-102
Cambridge, MA 02139

Dear Professor Todreas:

Subject: Comments on the Code Scaling, Applicability and Uncertainty (CSAU) Evaluation Methodology Developed by RES, NRC

During the Peer Review Group Meeting on the CSAU methodology, January 12-13, 1988, the Technical Program Team presented the basic methodology and its demonstration. Accordingly, we were asked to comment on the following two questions.

I) Is the methodology systematic, logical and practical?

II) Has the NRC demonstrated the methodology which shows that the best estimate code approach can be used to meet the requirement for the proposed revision of the ECCS rule?

Below, please find my comments concerning these two questions.

I) On CSAU Methodology

(1) The most difficult step in the CSAU methodology is the extrapolation of a predictive tool (code) based on scaled down experiments to untested prototype conditions. Reliable extrapolation of the models or evaluation of uncertainty of a code significantly depends on the type of accidents, available data base and physical understanding of scale effects. Although this methodology is very systematic, the practicality of the methodology may not be determined independently of these three items.

(2) In a step to evaluate the total uncertainty, the CSAU methodology appears not to make a clear logical distinction between the variations in code calculations over possible ranges of parameters and statistical uncertainty of the code prediction for a NPP. In order to bridge these two, it may be necessary to assume that the thermal-hydraulic models are basically correct and all important physical phenomena encountered in a NPP can be modeled by the code. This is a significant step which should be taken carefully with sufficient supporting evidence.

(3) Several steps, for example, 1, 3, 6-8 and 10 in the CSAU methodology require considerable expertise and good understanding of the accident phenomenology. In other words, the reliability of this methodology depends critically on the technical level and experiences of a

user. Therefore, in order to evaluate its proper use for licensing purposes, it is necessary for the NRC to sustain a high level of technical capability in the thermal-hydraulic area.

II) On NRC Demonstration of the Methodology

(1) I think that, in general, the NRC has clearly demonstrated the applicability of the CSAU methodology to estimate uncertainty associated with the use of "best estimate" computer codes. This demonstration has been carried out by the Technical Program Team with a high level of technical competence and considerable past experience in this area. I consider that the ability of the team has been one of the key elements in the successful application of the CSAU methodology.

(2) In Step 3, process screening, the engineering judgment by experts has been used to identify the important processes and to rank them according to their relative importance. For the particular case of the blowdown PCT, this may be quite sufficient because of the extensive experimental data and good physical understanding of the phenomena. However, for broader applications of this methodology, systematic considerations of the following two points may be beneficial.

i) An evaluation of the level of the understanding of the scale effect for each phenomena.

ii) An evaluation of each phenomena from the degree of non-linear effects involved. In particular, bifrucation phenomena should be carefully evaluated in ranking the phenomena as well as in calculating the code uncertainty. This is because a bifrucation phenomenon can lead to considerably different results starting from only slightly different initial conditions. This implies that the uncertainty may not be handled as a linear additive quantity.

(3) In the NRC demonstration of the CSAU methodology, cross products of uncertainty associated with several thermal-hydraulic parameters are not examined or not considered to be important. However, it is necessary to demonstrate that this is a varied approach by making a few sample calculations. If these sample calculations produce surprise effects (highly non-linear nature of uncertainty), some modifications of the methodology may be needed.

Sincerely,

M. Ishii
Reactor Analysis and Safety Division
Argonne National Laboratory

MI:kj

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25 January 1988

Dr. Neil E. Todreas, Chairman
Department of Nuclear Engineering
Massachusetts Institute of Technology
Room 24-102
Cambridge, Massachusetts 02139

Subject: YBR-517-88: Transmittal of CSAU Review Meeting Report

Dear Dr. Todreas:

Attachment 1 to this letter is the report which you requested after our 12, 13 January meeting on the CSAU methodology. If you have any questions on this report or if I can assist you in any other manner in the preparation of the final report, please contact me.

I believe that you did an excellent job, as chairman, managing the diverse contributions and viewpoints of the committee members.

If there are further evaluations of the CSAU method, I will be pleased to participate.

Best Regards,

Larry

L. J. Ybarrondo
President

LJY:cbc

Attachment

File: DN.9

REPORT ON CSAU METHODOLOGY

by

L. J. Ybarrondo, SCIENTECH, Inc.

1.0 INTRODUCTION

On 12, 13 January the author participated as a member of a peer review group appointed by the Office of Nuclear Regulatory Research. The function of the peer review group was related by NRC personnel as follows:

" The NRC has proposed a revised ECCS rule to allow licensees to utilize a fully best estimate approach for calculating ECCS performance. This approach also requires an estimate of the uncertainty in the calculations. The combination of calculation plus uncertainty would then be compared with acceptance criteria (i.e., PCT < 2200 °F). The Commission directed the NRC staff to develop a methodology to determine calculation uncertainty and to subject this methodology to peer review. The peer review should answer the following questions:

1. Is the methodology systematic, logical, and practical?

2. Has the NRC demonstrated a methodology which shows that Best Estimate Methods can be used to meet the requirements of the modified ECCS rule?

In addition, and separately, the CSAU Technical Program Group (TPG) would be interested in any detailed technical comments on the application of the methodology."

2.0 GENERAL COMMENTS

A tremendous amount of work has been done, in the last few months, by the NRC and NRC-Contractor personnel who made the presentations over two days. The material was organized well. The presentations were excellent. The information presented represents over two decades of experimental and analytical research into the Loss-of-Coolant Accident costing over a billion dollars. The piece-by-piece, systematic approach being followed is excellent. I believe the results of this research can and should be used in the NRC licensing process to further improve reactor safety.

2.1 RESPONSES TO QUESTIONS POSED BY NRC

Each of the questions above are repeated below for convenience.

1. Is the methodology systematic, logical, and practical?

RESPONSE. I believe that the the methodology presented appears to be systematic and logical for the example demonstrated. Based upon the example presented, I cannot say whether the methodology is practical because the example used to demonstrate the methodology is too limited in scope to allow a definitive yes or no answer. The example used was confined to the Large Break Loss-of-Coolant Accident(LBLOCA) blowdown. The Refill and Reflood portion of the LBLOCA will be a more complex and demanding test of the practicality of the CSAU method for a LBLOCA. An encouraging first start has been made in a partial portion of one type of accident, to one type of plant (A four loop Westinghouse Pressurized Water Reactor), to a specific code (TRAC) to which the CSAU method is intended to be applied. I recommend that it should be continued. I do not believe that the demonstration offered is complete enough in scope to provide a definitive test of the NRC desired characteristics of systematic, logical, and practical at this time. In my opinion, a complete demonstration of the proposed methodology should encompass the full range of expected applicability of the methodology which includes at least the complete LBLOCA and SBLOCA. Also, the issue of applicability of the CSAU to other codes, PWR's, and the Boiling Water Reactor (BWR) was not addressed, explicitly, in the presentations. If it is judged not practical or necessary to demonstrate the CSAU method for each proposed application, at least such situations should be specifically addressed in order to answer the question posed properly.

2. Has the NRC demonstrated a methodology which shows that Best Estimate Methods can be used to meet the requirements of the modified ECCS rule?

RESPONSE. An initial, limited-scope approach which shows excellent promise has been demonstrated. A methodology which shows that Best Estimate Methods can be used to meet the requirements of the modified ECCS rule has not been demonstrated because the example used is too limited in scope compared to the wide range over which such a proposed methodology must be applied.

2.2 SPECIFIC COMMENTS

This section contains a discussion of individual technical and management areas which came up during the two days of presentations.

2.2.1 Definitions

A set of definitions for key words and words which have relative meanings needs to be developed in order to avoid unnecessary debate and to make the method explicable. For example:

* The use of the the word "scaling" caused substantial nonproductive discussion. Scaling is a central issue for CSAU (see Section 2.2.4). It must be properly defined and used if the CSAU method is to be explicable to technical peers.

* Important parameter, frozen code, uncertainty, capture important phenomena, peak clad temperature, maximum temperature, code uncertainty, response surface (see Section 2.2.5), minimum/maximum variations, etc. are other words which created some confusion.

2.2.2 Quality of Analysis

A limited but good analysis was conducted. In some areas it was elegant in the manner in which the basic physical parameters responsible for observed behavior were isolated and evaluated.

2.2.3 Nodalization

The effects of nodalizaton on computer code calculations (analytically and experimentally) are well known. The CSAU method overemphasizes the attention which needs to be devoted to the area of nodalization. It is suggested that the issue be addressed in a more balanced manner. Currently, nodalization represents almost one third of the diagram which presents the CSAU method.

2.2.4 Scaling

This is a central issue for CSAU. Several times statements which I believe to be incorrect were made with respect to scaling in general and in particular for the experimental blowdown data. For example, " No effect of scale could be observed in peak clad temperature measurements from several scaled experimental facilities." Of course there are scale effects - they have been subsumed within the several hundred degree temperature (±360F) band selected. Those of us who helped design and conduct some of the referenced tests know there are scale effects: it is a matter of degree not existence. Sweeping statements such as the quoted one need to be more carefully stated in order to avoid unnecessary technical argument which detracts from the general credibility of the CSAU method.

2.2.5 Response Surface

The response surface approach described by the NRC and their contractors is completely different from the response surface approach described by the Westinghouse representative. For example, the independent variables used in the NRC approach were physical parameters. But the approach described by Westinghouse used plant actions such as pumps on/off as the independent variables. I strongly suggest that the effect of such differences be evaluated to determine whether the CSAU method can be applied effectively and consistently with such differences.

2.2.6 Overview of Methodology

The description of the CSAU method as such is excellent. But the relation of the method to the "big picture" is missing. A MORT-type chart is needed that explicitly presents the overall relationship of the method to the objective-ECCS RULE REVISION. Then, the applicability to other plants, codes, transients (e.g. are operational transients and ATWS excluded?), assumptions, constraints, regulatory guides, need for software quality assurance, etc would be clear. Without such an overview, it is not practical to answer the two questions posed in Section 1.0.

2.2.7 User Interface

Adaption of the CSAU method will require much more attention to the code user interface. The user interface becomes significantly more important because the extreme bounding approach mandated by Appendix K will be replaced by a realistic best estimate approach with sufficient but less margin. Therefore more NRC control is needed in the area of software quality assurance than currently exists.

2.2.8 Software Quality Assurance (SQA)

Section 2.2.7 indicates why the author believes this area is significantly more important for the CSAU method. In addition, there seems to be some confusion over what constitutes SQA. I recommend it be resolved. A "TRAC-PF1/MOD1 Models and Correlations" report was continually referred to as a QA document verbally and in slides (e.g.,see step 5 in the CSAU diagram); it is not a QA document. It is a good reference document that describes and provides a basis for the models and correlations used. It is an impressive document in size and with respect to the amount of information presented (see comments in Section 3.0).

I believe that it is imperative that a formal SQA plan be required of all organizations that will be associated with the CSAU method. The eleven elements of a SQAP as defined by ANSI/IEEE Std. 730-1984 include requirements in the areas of:

Management; Documentation; Standards; Practices and Conventions; Reviews and Audits; Software Configuration Management; Problem Reporting and Corrective Action; Tools, Techniques, and Methodologies; Code Control; Media Control; Supplier Control; Records Collection, Maintenance, and Retention.

A methodology without these type of defined controls will not be systematic, logical, or practical for the proposed CSAU method.

3.0 Comments on Additional Documents

In addition to the comments on the CSAU method, I reviewed the "TRAC-PF1/MOD1 Models and Correlations" draft report. Comments follow.

The short time available for review of the TRAC document allowed me only a glimpse through the report. Therefore, my review cannot be aimed at review of the quality of the presented models and their rationale or even on the quality of their description and discussion but on rather superficial judgement of overall completeness and quality of the report.

1. The report is impressive with regard to its size and the amount of information presented.

2. It seems that the presented material is basically sufficient to judge the implemented models and provides a good basis for understanding and a good reference for further model development and assessment.

3. The executive summary needs improvement. Lots of the material presented in the Introduction should be included in the Summary. Short forms of the conclusions of each section should be included in the executive summary. Regarding the extent of the document and its subject, the executive summary can be several pages long.

4. Logic diagrams are used in this report only in the section discussing wall heat transfer. Such diagrams should be used for clarity and understanding of implementation of many other models particularly for interfacial heat and momentum transfer.

5. Section 3 on Flow Regimes is missing the subsection on Scaling Considerations and Summary and Conclusions. Scalability of flow regimes is most important with regard to evaluation of code quality for NPP calculations; only small diameter pipes are discussed.

6. Sometimes, some additional explanation of implementation would be of advantage. For example, in the Flow Regime Section it is not stated why void fractions are obtained differently for interfacial drag and interfacial heat transfer calculations (adjoining cell averages vs. cell centers).

7. The role of the liquid velocity in the stratification model is not explained sufficiently.

8. In the Section on Critical Flow (the conclusions) nothing is mentioned about the known problem of superheated, single-phase steam flow (accurate estimation of the entropy).

9. Scaling is not addressed in the pump section.

10. TRAC does not calculate the pump energy source term. In the pump section, this problem is not discussed.

11. Section on Special Component Models should include also a discussion of the accumulator model. It is known that TRAC sometimes calculates unphysical behavior of the accumulator flow.

UNIVERSITY OF CALIFORNIA, SANTA BARBARA

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DEPARTMENT OF CHEMICAL AND NUCLEAR ENGINEERING.

SANTA BARBARA, CALIFORNIA 93106

January 22, 1988

Prof. N. Todreas, Chairman
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Massachusetts Institute of Technology
77 Massachusetts Avenue Bldg. 24-102
Cambridge, MA 02139

Dear Neil,

I am writing to you to document my views regarding the CSAU methodology developed by the NRC in response to the requirements of the modified ECCS rule. The questions that were posed to us are:

1. Is the methodology systematic, logical and practical?

2. Has the NRC demonstrated a methodology which shows that best-estimate methods can be used to meet the requirements of the modified ECCS rule?

In giving you my views on these questions I will deal with the first as if we were asked to consider the overall CSAU method without reference to any particular application, i.e. the comments will be about its applicability to different transients, and its strengths and weaknesses in this context. With regard to the second question, I will focus on the LBLOCA material presented to us. Clearly the two questions are somewhat interrelated and this division is arbitrary.

Comments Related to Question 1

1. The CSAU methodology has as its strong point (and its weak point) the fact that engineering judgement is used to reduce a very large computational problem to practical dimensions. This also allows many different transients to be considered and makes the method quite general. In this process something is lost in that handwaving arguments have to be resorted to in order to capture uncertainties arising out of numerous input parameters and boundary conditions, variations of which are not studied.

The step in the process that leads to a practical problem is Step 3, i.e. phenomena screening. I have reservations about this step because it is only as good as the judgement of the experts who are responsible. In general, their judgement is based only on what has gone before -- perhaps small-scale experiments and corresponding calculations -- and they are unlikely to identify new phenomena that may only arise on a large-scale system. To give a concrete example, over many years it had been the common wisdom that emergency venting of chemical reactors undergoing runaway reactions would lead to all-vapor discharge. This was confirmed by extensive sets of

experiments done on reduced-scale reactors. However, accidents continued to occur on full-scale systems where reaction vessels were severely damaged, apparently due to the vent sizes being too small. It has only been recognized recently that this was due to two-phase discharge which occurred in the large system but not in the power-to-volume scaled small systems. In retrospect, this is because level swell does not scale linearly with height. I can think of several other examples where important phenomena have been missed in deliberations by expert groups.

I feel more stringent safeguards are required in the methodology to ensure that a consensus is not required of the screening group in order for a phenomenon to be ranked highly. This may result in a somewhat enlarged group of parameters for which sensitivity analysis must be done, but this will be in the interests of credibility. Finally, the effects of phenomena in SBLOCA are much less understood than for LBLOCA, so there is some question as to whether screening and ranking can be done with a high degree of confidence.

2. I have reservations about Step 8 in the methodology, which relates to defining the critical nodalization for NPP calculations. The method proposed is not defensible on scientific grounds unless the computer codes being used for the calculations have been explicitly developed to handle lumped components. The present generation of computer codes difference the governing equations and, as such, it is essential that convergence with regard to node size and time step be demonstrated if we are to believe the results. If the nodalization is developed in the way proposed, then it too becomes a set of input parameters for which uncertainty analysis must be done. I have seen no evidence that this uncertainty is properly accounted for (see also Comment 6 in the next section).

3. The methodology does not require that the results of codes be compared with experimental data over a range of dependent variable responses. As discussed in the next section, the application of the methodology to LBLOCA was focused almost entirely on the comparison of peak clad temperature during blowdown with experiment. If flows, pressures and the whole temperature transient had been compared, much greater confidence with regard to the veracity of the code would have been developed. I feel it is a weakness of the methodology that it does not try to develop quantitative measures for uncertainty in several of the important dependent variables.

4. The methodology is focused on use of response surfaces for which it is well known that a very large number of calculations have to be done to develop good results. The minimum (using Latin hypercubes) is $\sim 2n$ where n is the number of parameters, and even for this situation there can be large errors in the results. This forces the whole methodology into selecting only a small number of parameters for sensitivity analysis in generating the response surface. As I mentioned during the presentations, there are alternatives which are much more economical in terms of computer time. One such methodology (see references 1, 2 and 3) is based on adjoint calculations. Once an adjoint code has been developed, only one run is required per response per independent variable. (In addition, one run is required for the nominal response by the main code.) The development of the adjoint code does require significant effort (e.g., for the general circulation model of the atmosphere, it took four man-years to develop the adjoint code). However once the adjoint code was developed, sensitivities could be obtained to several thousand parameters and this could be done very economically. Note that the general circulation model took several hundred hours of CRAY time to run. Documentation of this can be found in

reference 4. I am simply offering this to demonstrate that it is not a practical necessity to use a methodology with a high degree of parameter screening to do large problems. Alternatives are available and as the work on the atmospheric model demonstrated, the sensitivity results can go completely counter to the intuition of experts.

In summary, my answer to question 1 is that the method proposed by the NRC is practical -- maybe too practical. I would say that it is also systematic and logical, but I would have to qualify this by indicating reservations as to the phenomena screening step and the nodalization step. I also have reservations with regard to Steps 12 and 13, though these are less fundamental and arise from the demonstration for the LBLOCA case. More about this in the next section.

Comments Related to Question 2

1. Step 3, which is the phenomena screening step, was conducted very thoroughly by two independent groups. Nonetheless it appears to have missed the importance of nitrogen in effecting the fluid mechanics of refill and reflood -- an effect which calculations done in the U.K. indicate is of primary importance. This is significant because a great deal is known about LBLOCAs as compared to other transients that would be considered by the methodology. Furthermore, the effect of DNB delay time was considered unimportant by the expert groups. This is not borne out by the calculations done by Wulff (see Figures 8 and 9 in his handout) where a DNB delay time of 1 second made 100 K temperature difference with regard to the peak clad temperature. This is significant in comparison to some of the other uncertainties considered.

2. As pointed out in the previous section, the nodalization appears to be tuned to experiments, particularly with regard to LOFT. No arguments were presented justifying the so-called "lumped parameter" aspects of the closure relationships, so this approach is at present without foundation. Much could be made of this problem by intervenors as the procedure flies in the face of established mathematical practice.

3. Far too much emphasis was put in all the presentations on PCT alone. No results were presented to indicate that the code was giving realistic results with regard to the other dependent variables and for whole temperature response..

4. With regard to scaling, the peak clad temperature was plotted against the linear heat rating. This in itself is incorrect. The peak clad temperature should be plotted against the stored energy. At first sight, this is not a serious deficiency, because for nuclear rods, the linear heat rating is directly related to the stored energy. However, in the plots presented, there are a considerable number of non-nuclear experiments -- e.g., Semiscale and THTF. In these experiments the rods were indirectly heated and for a given linear heat rating the amounts of stored energy are quite different. Therefore, if the plot is redone with stored energy as the abscissa, one finds that THTF and Semiscale have considerably lower stored energy than would be indicated by the position on the linear heat rating abscissa. We then begin to see a distinct effect of rod length, e.g., Semiscale and THTF are 12 ft. rods and lie well above the LOFT and PBF data. Note that the Semiscale and THTF results would be shifted to the left compared to the present plot. Note also that LOBI should be deleted from this set because it presupposes a knowledge of the heat flux transient and there is no way that it can be put (with any credibility) on a stored energy abscissa.

In any case, I am not convinced that there are no scale effects associated with rod length from the presentation made. The abscissa should be changed to stored energy which, after all, is the correct quantity based on the arguments presented. (See also reference 5.)

5. I feel that too few runs were done varying the hydrodynamic parameters to quantify the uncertainties associated with cross products between hydrodynamic effects such as uncertainties in the pump model and in the discharge rate model. Furthermore, in order to reduce computer usage, the runs varying fuel rod parameters were decoupled from the hydrodynamics. This in itself is only a valid procedure after DNB has occurred and not a single run was done to see whether the period before DNB with the rod uncertainties coupled to the hydrodynamics would give rise to different peak clad temperatures. I feel that the codes should have been run, at least for the initial period, with coupling between the fuel rods and the thermalhydraulics.

6. The lumping of all uncertainties by comparison between the calculation and the so-called experimental curve for PCT was unconvincing. The experimental curve, as discussed previously under point 4, is heavily dependent on the correlation for peak clad temperature which was incorrectly plotted against the linear heat rating. I have no confidence in the physical basis for such a curve and therefore no confidence in the claimed uncertainties provided by the comparison between the NPP calculations and the linear regression line e.g. in Lellouche's presentation. In general it is incorrect to extrapolate experimental data from relatively small scales to full scales, and use this to provide a datum against which NPP calculations are compared leading to quantification of the remaining uncertainties. It would be much better to do this by honest calculations against new experiments, i.e. experiments which the code has not been tuned against and on the scale that experiments were done at. The key here is to do blind predictions to keep things honest and quantify the total uncertainty.

7. There are a number of miscellaneous points which need to be mentioned, though they may not be of the same significance as those discussed above. First, the assessment matrix should probably contain PKL. Second, the models and correlations document produced for TRAC should be given a thorough examination by experts not involved in the development of TRAC. It is in no sense a QA document at present. Third, blind runs should be done against some LBLOCA experiments which have not been used to develop TRAC with nodalization as proposed for the NPP calculations.

In summary, I feel that while the NRC has made a promising start at developing a methodology to meet the requirements of the modified ECCS rule, it has not conclusively demonstrated its application even for the relatively simple case of the blowdown peak in an LBLOCA. This is not to say that with more work the methodology could not be used. However, in the quest to minimize computer usage, the work presented perhaps took too many shortcuts and I, for one, would like to see a more deliberate and thorough approach to the demonstration. Also I would like to see the methodology applied to a somewhat more complex problem than the blowdown phase of LBLOCA before giving a definitive opinion. The reflood peak for an LBLOCA may be such a problem or a particular scenario for a SBLOCA may be even more convincing.

To conclude, the CSAU methodology has the potential to meet NRC's requirements with the qualifications noted above. None of the points are difficulties that cannot be resolved with further work involving more computer usage. The only one that perhaps falls in the "unresolvable" category relates to phenomena screening and even there, if the net is thrown sufficiently wide and the expert group is made to represent a larger cross-section of the community, then there is a good chance that nothing really important will be omitted. As an aside I would like it noted that alternatives to response surface methodologies (e.g. adjoint methods) exist that do not require phenomena screening to reduce computer usage. Such methods are mucy more "foolproof."

My impression in reviewing the work was that it was done quickly and use was made of whatever was available to shore up the case. For example, the code assessments, nodalization studies, etc. were often done with earlier versions of TRAC-PF1 or even with TRAC-PD2. It is hard to feel reassured by the glib statement that there are no significant differences in the calculated responses between all these versions. Nonetheless, the presentations were sufficient for me to give qualified positive answers to both questions.

On another subject, I have not had the time to review the TRAC QA document in any depth. I strongly recommend that a separate review be conducted.

Please contact me if you need further information.

Yours sincerely,

S. Banerjee
Professor and Chairman

Enclosures

SB:bcf

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January 20, 1988

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Dear Neil:

I have now reviewed my notes and the vast array of documents we were given at the NRC peer review group meeting on CSAU methodology and can give you my comments. In your report I suggest you commend Zuber and his group for the superior organization, planning and documentation of the two day meeting. I have been a consultant to NRC and ACRS for over ten years on and off and have never before participated in a review meeting in which all the information needed was available and which was presented in so logical a manner. This took a major job of organization and planning.

A. GENERAL COMMENTS:

1. The claim that the methodology quantifies the uncertainty in safety of a nuclear power plant is still to be fully confirmed for reasons discussed below. Even so the CSAU methodology, if adopted, can made a major contribution to safety by (a) forcing a documentation of the code and a systematic review of that documentation to reveal code errors, limitations and innacuracies and (b) forcing a systematic review of all the integral effects data to evaluate their validity, uncertainty and applicability to the full scale nuclear power plant. If it did nothing else this would be a historic accomplishment.

2. The CSAU methodology is represented as an objective, near quantitative method for measuring uncertainty in performance of ECCS. As a result of this description it is immediately subject to criticism by those well informed in quantitative analysis and statistics. That happened repeatedly during our meeting and is reflected in the ACRS letter on the subject. Instead, it should be described as <u>a formal method for combining quantitative analysis and expert opinion in a controlled way to minimize subjectivity of the experts in arriving at computed values of uncertainty.</u>

3. The methodology was demonstrated by application to the peak clad temperature during the blowdown of a Westinghouse 4 loop reactor after a large break. We are told that CSAU methodology is now being applied to the reflood peak temperature for which there is a large body of integral effects tests at very large scale. Because the use of these integral effects tests are so central to the method (for deciding uncertainty and for decisions on noding for the full scale calculations) there remains the

question of whether the method can really be used for all other accident scenarios and plant types. Before this can be decided we have to be sure there are enough of the right kind of integral effects tests available. It seems to me this cannot be determined until the effort is actually undertaken and step 7 in the process, "Establish the Assessment Matrix", is reached for each of these other scenarios or plant types or both.

4. It is represented that the method is independent of the code being evaluated and the scenario selected. It is true that once the scenario is selected then step 3, the identification and ranking of specific phenomena expected to occur in the course of the ECCS by the experts, is independent of the code. However many of the other steps are significantly code dependent and will require an expert's familiarity with the code. The impression should not be left that carrying out this process or evaluating how well it is has been carried out by an applicant company is a routine matter.

B. SPECIFIC COMMENTS:

Step3: Phenomena Identication and Ranking: The method makes good use of expert opinion in a controlled way which can be crosschecked and evaluated to arrive at a ranking table. In the demonstration only phenomena which were ranked 9 were subsequently evaluated. In discussion it was indicated that lower ranked ones could also be evaluated. However, an important judgement call is how far down on the list are the phenomena which are to be evaluated. Just preparing the list is not enough. There should be a basis proposed for deciding how far down the list one must go before the phenomena which follow can be ignored.

Step 6: Code Applicability: I am uncomfortable with some aspects of this step. The process looks at the code at several levels. First is it capable of describing the overall reactor type selected, say the PWR?, Second does it have equations or models for the individual phenomena listed in the ranking table? Third are there any defects? The first two are easy to execute. The third is not and depends critically on the technical quality of the investigator. The person(s) doing this job must be familiar with recent literature on a wide range of subjects, must know the difference between an empirical correlation and a physical model, must understand where there are gaps in the technical understanding that simply are not filled. Furthermore the range of technical expertise required is very large (two phase flow, heat transfer, drop mechanics, boiling, flow patterns,wetting phenomena and much more). It seems to me this cannot be done with any confidence until the code has been independently reviewed by a group of experts for soundness in its technical detail. This idea is addressed at the end of this letter.

Step 7: The Assessment Matrix: This experimental results used in this assessment matrix focus largely on data from integral effects tests (Loft, semi scale etc). Almost without exception these were used in various aspects of code development and tuning. Of the 40+ assessment studies of TRAC only two were made against university single effects data as indicated in the report by Boyack.There exist extensive single effect data in the literature from universities which have not been evaluated . Part of this process should emphasize the need to locate and evaluate test data not in used in code tuning.

Step 10: Scaling Effects: The weakest point in the methodology is displayed in this step, at least as applied to peak clad temperature for the blowdown. The PCT data from integral tests taken on equipment at five scales are plotted vs linear heat generation rate. The 95% confidence curves show a scatter in the data of +/- 361 F. Because there is no discernable effects of scale in such a plot (by some simple statistical tests) it is assumed there is none. It is a weak argument.

For any narrow range of values of linear heat generation rate the Loft data itself varies in PCT by about +/- 350 F. It is unlikely that this is simply experimental error. More likely this comes from the fact that during these tests certain initial or operating conditions were systematically varied between runs. (if this was not the case one would have to question the technical judgement of the whole Loft project.) These factors were not taken into consideration in step 10. If the variation in PCT in the plot of the Loft data, say, was truly experimental error then the argument might be partially supportable. But it will be necessary to carefully study the data for any one test series to establish this fact before the conclusion is acceptable.

C. THE TWO QUESTIONS:

Here is my response to the two questions posed by Lou Shotkin.

#1 The methodology appears to be systematic, logical and practical.

#2 The NRC has <u>in part</u> demonstrated a methodology which shows that best estimate methods can be used to meet requirements of the modified ECCS rules. Some weak spots in the method have been identified and its general applicability to a wide variety of scenarios and plant types remains to be fully established.

D. CONCERNS AND RECOMMENDATIONS:

1. The very formulation and acceptance of a plan for estimating uncertaintly in loss of coolant accidents will probably suggest to the Nuclear Community that there is little need any more for basic and applied research on the physical phenomena underlying ECCS performance. One can expect that contractors, utilities and NRC will find it more difficult to justify research. The fact that appalling ignorance and error appear repeatedly in the code will become unimportant - until such time as a strange and unanticipated sequence of events causes an accident or near accident and then the inadeqacy of the existing knowledge will become apparent once again.

2. The members of this group who are available should be constituted once again to review the results on reflood when completed. We have a considerable investment in time and effort getting to our present knowledge about the methodology and can probably make a more expeditious and reasoned review than a new group.

3. As mentioned in B-Step6 above, there is a difficult problem in evaluating a complex code such as TRAC. NRC has invested vast sums of money in developing the code. We finally have a document (erroneously misnamed Q &A document) which explains in great detail the modelling

used by the code for physical processes. This document should be examined in detail by an independent group of experts including wide ranging disciplines to judge whether the code includes the most recent research results for the many technical area it covers, where it is deficient because we simply do not have the fundamental research results needed, whether its methods of approximation are reasonable and where it is in error. I suggest this be undertaken by a committee of the National Academy of Engineering under sponsorship of NRC. The the results would then be of unquestioned integrity. This is quite different in nature and importance than the many so called assessment studies of the code by national laboratories.

Neil, I trust this helps you in preparing the summary statements. I presume you will attach the individual responses to your summary and send each of us the entire package. I, for one, would like to read all of the comments. It was a pleasure working with you.

Cordially,

J. E. Dukler

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Peer Review Group Meeting on
code scaling applicability and uncertainty

Dear Professor Todreas,

before starting with my comments on the best-estimate approach for calculating ECCS performance as proposed by NRC I have to beg your pardon for being late with this letter. Returned from my short trip to Washington I was confronted with some new duties in my university which I had to accomplish immediately. So, I lost a few days before I could start thinking about my comments on the methodology proposed by NRC for calculating ECCS performance with the best-estimate approach.

For many years calculting ECCS performance was done by utilizing a conservative approach and by giving an upper limit for the cladding temperature in the most unfavourable position of the core in case of a Loss of Coolant Accident. Not only before the TMI accident but already when starting risk studies we realized that accident management may play an important role in avoiding catastrophic situations which may go far beyond the large break LOCA-conditions. Conservative rules aiming mainly and almost exclusively on the large break LOCA may handicap or even deteriorate the effectiveness of accident management procedures. Therefore, there is a great demand to judge the LOCA by best-estimate rules instead of conservative ones. In addition we have to realize that an incident resulting from a transient is much more likely than a LOCA. Therefore, I welcome the activities of the NRC developing a methodology for calculating the ECCS performance by using a fully best-estimate appraoch.

Before going in details I would like to state that the methodology presented and documented by the NRC and the members of the Technical Program Committee on the Peer Review Group Meeting in Washington seems to me systematic logical and practicle. It was

methods can be used to meet the requirements of the modified ECCS rules. This, however, in the first step is restricted to large break LOCAS.

The CSAU-methodology is and must not be restricted to or based on a specific computer code. It proposes a general procedure how to quantify core cooling effectiveness and to predict reliably cladding temperatures to a good accuracy.

After at least 15 years experimental and theoretical worldwide research in loss of cooling accidents we should and must have gained enough experience to treat the ECC procedure in a best-estimate way. Experimental facilities like SEMISCALE, LOBI, PKL, SCTF, CCTF, LOFT, and UPTF cover a wide range of geometrical scaling up to the real size of a PWR-nuclear power plant. With UPTF a full-scale mock-up of a PWR-plant is available to produce transient data on the thermohydraulic behaviour of the primary system of a PWR and a number of very informative experiments were already performed in this facility. So, with code scalling it is not necessasry to rely fully and substantially on engineering judgement only. The CSAU methodology must systematizes the application of judging best-estimate ECC-data by using the informations of all these facilities in the international nuclear community including the large scale facilities in Japan and in the Federal Republic of Germany.

Certainly, the CSAU methodology has not yet been tested over a wide range of applications, however, it is a systematic way to appraoch the questions related to core coolability during a LOCA.

There are many thermohydraulic phenomena connected with the emergency core cooling procedure. These phenomena have to be carefully identified and selected to meet the requirements which meet a high status of experience. This experience, however, is available in the international community after doing so many years experiments and computer code development for ECC. Inspite of this experience one may ask whether we do know all important phenomena and whether we are fully aware how these phenomena interact with each other producing unexpected situations. The CSAU-procedure in general is certainly applicable to all LOCA'S but it would be wise to restrict it in a first step to LBLOCA'S until we have more experience with this procedure.

Nodalisation may be a key point for success in applying this new methodology. However, there are new and more sophisticated approaches available by adapting nodal zones to a best-estimate fit in comparing the results with separate effect tests and integral effect tests performed in facilities of different scale up to the full-scale UPTF-facility. Scaling can be checked against UPTF and can be improved in the methodology by doing it.

Codes like TRAC or RELAP 5 were developed and improved in the physical models implemented in, by checking the correlated results against experimental data gained in the SEMISCALE and LOFT facilities mainly. One may argue that no independent checking is possible for the application. However, there are many

example, from PKL, from LOBI, and also from other facilities which can be used to assess the best-estimate methodology. So users should and can take independent data available in the international literature to check it against the code.

One also may argue that new phenomena, unknown up to now, may arise which may make it dangerously to apply this CSAU methodology. According to my opinion and experience the procedure with LOCA is so thoroughly researched and clearly defined - at least as we restrict our effort to large break LOCA'S that one must not be afraid to be confronted with completely new phenomena in the future.

During the presentations on the Peer Review Group Meeting it became evident that ranking the phenomena for doing best-estimate calculations is of great importance. Also phenomena of low ranking may have an unexpected influence, if their effects are known with extremely low accuracy, and if by interacting with a phenomenon of higher ranking order they can change the fluiddynamic situation. Therefore sensitivity studies should be recommended to check carefully possibly unexpected effects.

Applying CSAU successfully and in a correct way needs a certain competence of the user, expecially the code user, therefore regulatory institutions have to take care whether the user is competent enough. In general good enough experience can be expected with users who are active in licensing calculations on the side of the utilities and of the vendors as well as on the regulatory side. These institutions are concerned with ECC-calculations since many years.

It must be free to the experienced user to apply a code which has, according to his experience, the highest status of development. Applying such a code regions of uncertainty have to be pointed out and the applicability of the CSAU-method together with this code has to be justified and documented.

When the methodology is finally matured the NRC will have an effective and independent tool to audit industry submittals. In the presentations during the Peer Review Group Meeting good methods and first results were shown to proof at principle that the CSAU methodology will allow to elaborate reliable data for LOCA-analysis, but more work in this direction should be recommended.

Yours sincerely,

F. Mayinger

Prof. Dr.-Ing. F. Mayinger

THAYER SCHOOL OF ENGINEERING
DARTMOUTH COLLEGE

January 22, 1988

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Dear Neil,

I have reviewed my notes from the Peer Review Group meeting in Bethesda, January 12-13, 1988, as well as documentation supplied before, during and after the meeting. This letter summarizes my reactions to what I saw and heard. I will first discuss the CSAU methodology in general, then comment on its pieces.

The overall CSAU methodology is reasonable and, in principle, is a logical approach to using Best Estimate predictions and experimental data for estimating the probability that key parameters, such as peak clad temperatures, will lie within some required range, with specified uncertainty.

The main questions concerning CSAU involve the adequacy and practicality of completing each step in the process. As each "box" in the diagram is considered, not entirely sequentially but with some iteration and cross-influence, judgments have to be made about "how much is enough?" For example, the codes for reactor analysis have over 100 empirical or tunable coefficients, some of which appear (as in the TRAC Q/A document) to have been chosen on a rather weakly-justified basis. They have been steadily upgraded over many years by tuning the coefficients or the forms of correlations to fit data. An assessment has to be made somehow when a certain code is "mature" and ready to be "frozen."

Similarly, the assessment matrix contains a limited number of tests in reactor-like geometries at various scales. The decision that this provides enough evidence to assess the applicability of a certain code can only be made after exhaustive comparisons, including criteria for acceptable error. Likewise, nodalization is based on 10 to 20 years of experience and there is a general belief that it has converged to an acceptable level of detail to "capture the important phenomena," but it is hard to justify this in a definite, concise way.

Uncertainty analysis seems to be most rationally based on arrays of predictions obtained from a code by varying a set of chosen parameters over a credible range. One still has to address the difficult questions of how many parameters to use, how many runs are needed to define a response surface, how to assure that the combination of all "other" uncertainties has been conservatively estimated and so on.

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The various presentations in Bethesda showed that long steps have been taken toward demonstrating practicality. However, the PRG's questioning indicated a broad range of doubt about whether the evidence presented was adequate to put the various key issues to rest. Therefore, consideration has to be given by NRC not only to the acceptability of the overall methodology, but also to whether the applicants for licenses and the NRC staff are yet in a position to use the approach in all its details. Particularly, my own view is that the numerous references to the use of "judgment" in making key decisions needs to be changed, as much as possible, to well-defined, quantitative criteria for which a documented and Q/A'd rationale is clearly spelled out.

Detailed Comments

In the introductory talk it was stated that the accuracy requirements should be commensurate with the degree of risk. Presumably this means that the 95% level has been logically derived in some way by considering the probability of an accident and its consequences.

Step 3 involved PIRT. Eventually the selection of important parameters must surely be made on a more deductive basis than "asking people." We were told that hellouche/Wulff would later confirm the expert evaluations, but they did not make separate evaluations of the "neglected" variables. Showing that the statistical spread covered the data could be a result of tweaking the major variables over too large a range, rather than being evidence that all else was unimportant.

Lots of "experience" went into the choice of major variables; does this mean that a huge new research base will be needed for new designs?

Zuber's overview of the NRC-sponsored efforts over many years was realistic. He came up against the "good enough?" question at many points.

To an outsider several of the assumptions in TRAC seem strange. They would not be particularly good for predicting simple flows in straight pipes, but appear to be successfully tunable to reactor geometries. Having 175 parameters in a code seems excessive and makes it hard to keep track of rational comparisons. In retrospect it would have been better to have far fewer, but physically-based, parameters that could be checked against many phenomena in a more methodical way in Steps 4 and 5.

In Steps 6 and 7, where the code is being assessed, I'm not clear what the criterion is for "passing" and insuring that faults in the code itself do not get through to influence later uncertainty assessments. I'm not sure about gathering all sorts of "uncertainty" under the same umbrella.

The comparisons we saw with LBLOCA may be good because the flow regimes are mostly close to homogeneous and therefore simple. In the SBLOCA, more phenomena are likely to be important and there is a greater need for a realistic code.

Nodalization was discussed in Step 8. There's a lot of experience behind this. However, the statement that Sandia doubled the number of nodes and changed PCT by 75°K is not reassuring; this is comparable with the uncertainties due to other causes.

BNL's work on assessment of uncertainties associated with the fuel, Step 9, was logical and methodical, the type of thing that will have to be required to get a good uncertainty prediction.

The presentation on scaling, Step 10, was a little odd in that a very simplistic approach was taken, compared with a much more detailed approach in the other analyses. The range of PCT is 1050°KF ± 750°F for all data points, so plotting them versus the main influence of heating rate to reduce the uncertainty by half is not surprising. Surely many patterns in the data are hidden and there are specific known reasons, such as experimental set points or initial conditions that account for some of the trends. I'm nervous about taking such a grand view, treating all scatter as "uncertainty," and deducing "no clear trend with scaling."

The calculations in Step 11 were very limited, involving only seven runs in which thermohydraulic variables were changed. The changes were large and the effects were large, making linear or quadratic fits to the results less secure. I believe we saw a data point for which the predicted change in PCT was much larger when three variables were perturbed than the sum of the changes when each was changed individually, indicating some "cross-product" effects that need more investigation (this did not appear in the tables supplied to us, so I may be mistaken).

Since the temperature versus time curve is double-humped, with either hump being higher under different circumstances, the response surface may not be smooth.

The uncertainty analysis, Steps 12 and 13, is a key element, and one of the few new ones, in the methodology. The overall approach is pertinent and reasonable. The various statistical comparisons are appropriate and encouraging. My major criticism is the attitude that the overall product was "good enough." Surely more tests are needed by running the code to evaluate

- cross-products of the thermohydraulic variables
- the impact of the neglected variables
- trends in data that are not due to randomness but to known differences in the experiments.

There are more significant variables for the SBLOCA and ways will have to be found to make many more computer runs to cover the whole spectrum.

I am not familiar with all of the integrated effects tests but my impression is that TRAC predicts an NPP like LOFT because that is the main basis for its tuning. Were LOFT details duplicated in the other tests?

I assume you will pass these comments on to the appropriate people at NRC.

Sincerely,

Graham

Graham B. Wallis
Professor of Engineering

GBW/dec

D R A F T

PEER REVIEW GROUP MEETING ON CSAU METHODOLOGY

BETHESDA, WASHINGTON DC, 12-13 JANUARY 1988

The following presents my assessment of the material presented at this Review and addresses the questions put to the Review Panel by Dr L Shotkin.

1. THE PROBLEM

The USNRC proposes the use of best estimate codes to calculate ECCS performance as part of a modified approach to licensing. This approach requires an estimate of the uncertainty in the calculation of a small set of key parameters. The USNRC has developed a methodology for determining this calculation uncertainty and has subjected this methodology to peer review. The members of the Peer Review Group are asked to answer four questions which I interpret as follows:

(i) Has the USNRC demonstrated a 'proof of principle'?

(ii) Is the methodology systematic, logical and practical and adequate?

(iii) Is this methodology adequate to provide an independent audit of industry submittals, which are likely to use a somewhat different and independent methodology?

(iv) Does the USNRC methodology provide useful guidance to industry in producing their own methodology and submittals?

This note sets out my own conclusions, after examination of the documents presented to us at the meeting on 12 and 13 January 1988. It has benefited from points made in these presentations and from discussion with other PRG members.

I have also made use of UK experience, as follows:

(i) Use of various versions of TRAC, over a number of years in the analysis of a range of experiments and of plant LB/LOCA transients.

(ii) Interaction with other TRAC users, particularly in the US but also through international contacts, including ICAP.

(iii) Independent UK analysis of the problems of code uncertainty estimation.

(iv) Collaboration with other organisations in the CSNI in an attempt to reach an international consensus on code uncertainty analysis.

2. SCOPE OF THE REVIEW

It is important to recognise that at all stages in this demonstration, we need to exercise 'engineering judgement'. For this reason I believe that a 'proof of principle' is only possible by developing a specific implementation and showing that it works i.e. that it is systematic, logical, practical and adequate. I make three points:

(i) I am impressed by the work carried out by the USNRC team. They have genuinely got to grips with the problem and have produced quantitative results. Most of the material presented is undoubtedly of permanent value.

(ii) There are areas where I disagree with some of the numbers and assumptions. In most cases I am satisfied that this does not affect the adequacy of a demonstration of 'principle;' and that the team would accept that the last word has not yet been said. I have therefore not allowed them to influence my overall conclusions.

(iii) The evidence presented allows conclusions on the uncertainty of calculation of the blowdown PCT. There is only limited analysis for the reflood PCT and no discussion of SB/LOCA problems. It is clear that the analysis for the blowdown PCT is much more straightforward than for reflood and SB/LOCA. I conclude that, at this stage, the demonstration is confined to the estimation of blowdown PCT uncertainty and its success can not be extrapolated to cover the other cases.

3. TRAC-PF1/MOD1 STATUS

It is central to the whole approach that the best estimate code used must have reached a satisfactory state of maturity. To me this implies four conditions.

(i) The code modelling is adequate to capture the major thermal-hydraulic phenomena which influence the transient.

(ii) Enough assessment studies have been carried out for representative SETS and IETS to show that the code achieves satisfactory quantitative accuracy for major variables. Only if this can be claimed is it then acceptable to concentrate on a particular variable such as blowdown PCT. For example I would

not be satisfied to use a code which gives an accurate PCT prediction but got the hydraulics quite wrong. This is an issue of engineering judgement which it is difficult to quantify. On the other hand if it is not met I have doubts about scaling and about robustness to small changes in transient definition.

(iii) The SETS and IETS analysis have been sufficiently extensive to support the noding chosen for the NPP. I accept that, in general, one cannot defend the noding schemes used in TRAC by showing that they are, in some sense, saturated against further node refinement. In practice, at the noding levels used, the codes are a compromise between a finite difference and a lumped parameter approach. The choice of noding for the NPP is therefore a matter of judgement.

(iv) At the stage where the code is to be used for an uncertainty assessment it must have been stable for a period (frozen?) and properly written up (QA).

I am personally satisfied that the version of TRAC-PF1/MOD1 used for this exercise meets all these criteria, certainly to the level needed for a demonstration and possibly for an independent audit. For the first two items in my list, I rely on

independent evidence from our own work in the UK. For the second two my comments are:

(i) The case presented by the USNRC to support their noding choice has been the subject of a thorough review and is well documented.

(ii) The TRAC QA document has been awaited for some time and has only just arrived. On first review I regard it as an excellent production but it will take time to examine it in detail.

4. TUNING AND SCALING

These two issues led to considerable controversy at the PRG discussion. I can only state my own position.

The central role of closure relationships implies that the codes are always fitted to data. This presents problems only when this is not adequately separated from evaluation and the database is too narrow. It is also only objectionable if the data fitting processing is seen or suspected not be scale independent. One approach is to fit only the correlations; the other is to use integral data and separate effects but to claim to model phenomena. There is also the philosophical difficulty that as codes become more advanced (more fundamental?) the correlations they use become more numerous and more complex. I

do not identify any deliberate tuning in the code and (now we have UPTF data) I believe the database is too wide to make such tuning a practical proposition.

The presentation and discussion has considered the issue of scaling. In my view the real issue is whether scaling is properly included in the codes. If it is, then it is not of direct importance that the experimental facilities also scale. The requirement on the experimental database is then that it should be wide enough to capture phenomena relevant to the NPP transient and that analysing this database with the code can demonstrate that these phenomena are adequately modelled. On the evidence available to me I am not satisfied that here is a satisfactory procedure available that can be used to scale NPP calculations by adding a bias or by increasing the uncertainty. However for the case presented, the blowdown PCT, I am satisfied that the NPP calculation scales correctly (or that there is no additional scale effect).

5. OVERALL STRATEGY OF THE METHOD

The issues raised in the previous discussion are important because the quantitative analysis that follows is only meaningful if these issues are dealt with properly. I also note that they are almost entirely matters of opinion (or engineering judgement if you like) and are difficult to quantify. The

quantitative part of the exercise then comes in five stages. I identify these as follows:

(i) There is a NPP calculation line in which a small number of major uncertainties are identified and these are given an uncertainty range (and probability distribution). A sensitivity calculation then provides a response surface (PCT against seven variables). By straightforward techniques this then gives a probability distribution for the PCT and the mean value.

(ii) A separate analysis (W Wulff BNL) provides a direct back-up to part of the NPP calculation.

(iii) A bias and uncertainty range is obtained by computing a number of fuel pins for LOFT L2-6.

(iv) A set of five facilities each claimed to give a good simulation of blowdown PCT for an LB/LOCA is used to provide a mean and an SD for this ensemble.

(v) A final analysis is presented which attempts to pull this material together to give a bias and SD for the NPP calculation.

I give my comments below, against these headings:

6. THE NPP CALCULATION

I regard this as the heart of the matter. I believe the other analyses should be regarding as supporting this main line. I rely on the other calculations to assess bias and to persuade me that essential uncertainties are not omitted by using only the NPP calculation route, or to make some allowance for the missing items.

A key issue is the PIRT analysis. I am impressed by the care and detail that went into this. I think it was valuable to have had two teams to carry out the quantitative PIRT analysis though clearly they are not independent in any statistical sense. I see no reason to disagree with the PIRT rankings but accept that they may change in detail with experience. It was not entirely clear but I assume that items with ranking 9 were automatically included and those from 7 to 9 were given a detailed review. I accept that there are good grounds for regarding the results as code independent.

The selection of the seven PIRT variable and the determination of their uncertainty range is the key to the method and both these can be claimed to be code independent. The techniques used for uncertainty estimation are objective and could be repeated by another team but they are clearly neither mandatory or unique. For some problems, e.g. fuel pin stored

energy, the approach used can be defended in detail. For others, e.g. pump modelling, it is very much a matter of subjective judgement. I conclude that what has been done is sensible and defensible but there is a need for other evidence or arguments to provide some assurance that nothing essential has been missed and that the uncertainty inputs have not been underestimated.

I am not worried about the choice of the probability distributions for the PIRT variables. These are enough variables to be able to rely on the Central Limit Theorem of Statistics. In this case I believe it is adequate to allow us to treat the PCT distribution as gaussian so long as we do not move more than two standard deviations from the peak.

I am also somewhat influenced by my own personal judgement that the quoted standard deviation of 172°F is not unreasonable. I would expect a value between 100°F and 300°F.

7. ANALYSIS BY W WULFF, BNL

I found this analysis very supportive of the main calculation line. It provides valuable back-up on scaling, noding arguments etc. It gives a value of 80°F for the SD of the PCT due to fuel stored energy.

8. LOFT L2-6 ANALYSIS

This is an analysis of calculations of PCT for a number of pins in L2-6 which were at different points in the core and had a range of linear ratings. The difference ($T_{TRAC} - T_{LOFT}$) had a bias of 23°F and a standard deviation of 120°F.

It is very difficult to determine the reason for the scatter in ($T_{TRAC} - T_{LOFT}$) and to relate this to other uncertainties in the NPP calculation. The USNRC analysis presents no guidance except to argue that instrumentation uncertainties are small. My own view is that relatively few of the uncertainties demonstrated in this analysis appear in the NPP analysis. If I accept the argument that instrumentation (and test) errors are small then I am forced to conclude that these errors are an estimate of at least some of the uncertainties 'missed' in the NPP analysis.

I would like to see this part of the exercise extended, possibly by analysing other LOFT Tests. It would also help to analyse the uncertainty background and compare it with what was included in the NPP analysis. In essence I am not satisfied that the way to use this LOFT data, via this type of analysis, has been fully thought through. I conclude however that to treat it is an independent uncertainty component is pessimistic even if not fully argued.

9. EXPERIMENTAL FACILITY ANALYSIS

The blowdown PCT is plotted against linear heat generation rate for five facilities and a 95% confidence band of 360°F is computed which I take to be consistent with an SD of 180°F. It is possible to regard this ensemble of data as a predictor of the blowdown PCT for the NPP. This assumes that all these tests are well designed and scaled. Some of the uncertainties in the NPP analysis are due to scaling up effects and some of those for the experiments are from instruments and test conditions. If I ignore these two items there is an argument that both this that and the NPP calculations have the same uncertainty input.

10. FINAL ANALYSIS

I see the conclusions of the exercise as follows:

(i) NPP sensitivity analysis:

$$T_{mean} = 1106^{\circ}F$$

$$SD = 172^{\circ}F$$

Augmented by L2-6 analysis the SD increases to:

$$SD = 208^{\circ}F$$

(ii) Experimental results:

$$T_{mean} = 1200^{\circ}F$$

$$SD = 180^{\circ}F$$

I conclude that, on the evidence presented, it is possible to take the NPP calculation with a bias of -100°F (i.e. the calculation is optimistic) as the best estimate PCT with a standard deviation of 200°F. I felt that this final reconciliation exercise was in a preliminary stage and required further thought and a better presentation. I believe however that this can be done.

CONCLUSIONS

I have purposely made my conclusions brief and they are as follows:

1. The material presented by the USNRC is adequate for me to assess the method used. There are places where I believe the argument could be strengthened by additional work. In particular, the final section which pulls all the material together should be clarified. This has not prevented me from coming to conclusions.

2. A method has been presented for assessing the blowdown PCT. I have reservations on how well it will work for the reflood PCT and accept that this part of the argument is not yet available. I am not able to form a judgement as to how it could be applied to SB/LOCA transients.

3. The method implies a substantial exercise of engineering judgement, most of which cannot be quantified. I note in particular:

   (i) Evidence that the best estimate code is suitable and mature (QA and validation).

   (ii) The selection of variables and their uncertainty range.

   (iii) Evidence that the analysis is complete and no major factors have been omitted.

   I note however the techniques used to make these judgements quantitative when possible and believe these approaches are sound and a useful model.

4. I accept that the proposed method is systematic, logical and practical. I believe it is a useful model for other approaches and can be used for independent audit.

J FELL
Deputy Director
Water Reactors Programme
UKAEA
AEE Winfrith
Dorchester
Dorset DT2 8DH
UK

21 January 1988

END

MEMO

To: Peer Review Group
Code Scaling Applicability and Uncertainty (CSAU) Methodology

From: N. Todreas

Date: January 22, 1988

This letter summarizes my views of the CSAU Methodology presented to us by the NRC on January 12–13, 1988. These views, together with those submitted by each of you, will be submitted with my summary letter of our deliberations to the NRC Office of Nuclear Regulatory Research (RES).

1. Value of the Process

The NRC should be strongly encouraged to continue this effort not only because of the inherent physical logic of a best–estimate and uncertainty approach but also because of the attendant benefits to the structuring of its research program. The creation of the Methodology and Demonstration have already reinforced the value applying a consistent code version throughout an evaluation process, produced overdue documentation of a key US LOCA analysis code and stimulated a reevaluation of a broad range of valuable and costly experimental data. Further, future research needs should be able to be identified more readily and justified more convincingly if they can be shown to contribute to reducing the uncertainty or evaluating a proposed alternative Methodology.

Inevitably, however, the successful licensing application of this Methodology will probably maintain large break LOCA (LBLOCA) limits in the position of dictating the real operating limits. This is not a problem if the uncertainty can be established with sufficient confidence. If, however, this process of establishing uncertainty evolves to one requiring statements with statistically rigorous bounds, difficulty will follow because the complex nature of the phenomena has not admitted this level of statistical rigor into the establishment of the Methodology and the Demonstration. Rather, while elements exist where statistical methodology has been applied, the establishment of the Methodology and the Demonstration are replete with steps based principally on engineering judgement albeit of experts in the field.

2. Is the Methodology systematic, logical and practical?

I find the Methodology systematic and logical. Its practicality really can only be confirmed by its Demonstration. As discussed under 3. its Demonstration for the blowdown peak clad temperature (PCT), while not fully in hand, is forseeable. For the reflood PCT and small break LOCA's, the Demonstration will be much more difficult to achieve because of the complex nature of processes.

There remains important comments on the Methodology with regard to the assessment matrix, scaling and nodalization.

- It would be desirable to broaden the assessment matrix to include more separate effects tests since relevant ones appear available and there is a strong incentive to insure that the code models and subsequent code assessments are not effectively based on the same data base.

- The Methodology does not present a method for scaling supplemental to the presumption that the code models themselves contain sufficient physics to reproduce scale effects. A key element of the Demonstration was to conclude, by considering a sufficiently broad scatter in the data, that scale effects did not exist. I do not object to the Demonstration that within such a scatter band, effects of scale even if they exist, can be accommodated. However, the accompanying message which the team conveyed that scaling effects in a physical sense do not exist, is inappropriate and seemingly unnecessary.

- The nodalization strategy employed is to require consideration and subsequent establishment of noding for each component and then utilize the same noding for the nuclear power plant calculation as the data assessments. However the number of data assessments made with the NPP noding is limited so that the use of previous experience in investigating sensitivity to nodalization has been critical. The sensitivity of results to the number of nodes in the cold leg where ECC mixing occurs is an area which probably should be investigated now for the selected NPP node array.

3. Has the Methodology been demonstrated?

A distinction between the blowdown PCT and the reflood PCT and SBLOCA cases with regard to demonstration calculations has been made under 2 since issues of practicality and demonstrability seem intertwined.

The key issue on the Demonstration was the adequacy of the calculations performed to define the response surface. The number of cross-products considered were low and, particularly the nonlinear effects between hydrodynamic parameters were not explored. The number of required computer runs must be controlled but the justification for the choice made to date is not physically satisfying. Additionally, the Demonstration presented orally and in written form to the peer group was not complete with respect to evidence in the following areas. In all cases, the following oral assurances were given but they eventually need to be documented.

- The data matrix used to assess the core models and the code assessments was broader than that presented. The issue is one of avoiding the tuning of the code resulting in satisfactory performance over a limited range or for special circumstances.

- Assessments, using different versions of the code (from the frozen one), were utilized as pieces of the Demonstration. Differences between code versions were examined and found inconsequential to the individual assessment being made.

- The same nodalization was utilized in all key data assessments and the NPP calculations.

- The clad temperature thermal history, as well as other hydrodynamic parameters, were examined in addition to PCT in key assessments of data. The overall prediction trends were found reasonable.

4. Closing Comments

- The NRC team is excellent technically, but worked under very tight time constraints to prepare for our meeting. Responses regarding content (I do not refer to opinion where we should welcome expression of differences at this stage) of the Methodology and the Demonstration were not always consistent and led one to question whether these are "frozen" even for the blowdown PCT. Additionally, the description of the Methodology prepared for NUREG-1230 of April 1987 should be brought up-to-date.

ENCLOSURE D

COMMENT LETTERS FOR ECCS RULE

LETTER NO.	DATE	ORGANIZATION	COMMENTER	NO. OF COMMENTS	REACTOR TYPES B&W	CE	W	GE
1	4/20/87	Representing Self	D. L. Johnson	3				
1a*	8/17/87	Representing Self	D. L. Johnson					
2	6/23/87	Westinghouse Owners Group	R. A. Newton	9				
3	6/25/87	Georgia Power	L. T. Gucwa	4			X	
4	6/22/87	Westinghouse Electric	W. J. Johnson	16			X	
5	6/21/87	Representing Self	J. D. Harris	12				
6	6/30/87	Representing Self	Anonymous	2				
7	6/30/87	Advanced Nuclear Fuels	G. N. Ward	9				
8	6/27/87	Ohio Citizens for Responsible Energy	S. L. Hiatt	1				
9	6/25/87	Florida Power Corp	C. E. Simpson	1	X			
10	6/24/87	Arkansas Power	J. T. Enos	8	X	X		
11	6/30/87	B&W Owners Group	C. H. Turk	4	X			
12	6/30/87	Yankee Atomic	D. W. Edwards	8			X	
13	7/1/87	Northeast Utilities	E. J. Mroczka	5				
14	6/29/87	Combustion Engineering	A. E. Scherer	8		X		
15	6/29/87	Penn Power & Light	H. W. Keiser	4				X
16	7/1/87	Flordia Power & Light	C. O. Woody	2		X	X	
17	7/1/87	Wisconsin Electric	C. W. Fay	7			X	
18	7/1/87	Washington Public Power	G. C. Sorensen	5				X
19	7/1/87	Carolina P & L	S. R. Zimmerman	6			X	
20	6/24/87	Duquesne Light	J. D. Sieber	9			X	
21	6/30/87	Virginia Electric	W. L. Stewart	7			X	
22	6/1/87	Portland Gen. Electric	D. W. Cockfield	9			X	
23	7/1/87	GPU Nuclear	J. R. Thorpe	5	X			
24	6/30/87	TU Electric	W. G. Counsil	5			X	
25	7/7/87	BG&E	J. A. Tienan	6		X		
26	6/30/87	New York Power Ath.	J. C. Brons	7			X	X
27	7/15/87	Southern Company Services	L. B. Long	9		X	X	X
28	7/20/87	S. California Edison	M. O. Medford	7				
29	7/27/87	Commonwealth Edison	L. D. Butterfield	2			X	X
30	7/31/87	Ohio Citizens for Responsible Energy	S. L. Hiatt	7				
31	8/8/87	Ohio Citizens for Responsible Energy	S. L. Hiatt	1				
32	6/5/87	General Electric	R. Artigas	6				X

*Note: Mr. Johnson transmitted additional background material to be included with his comments.

PARAPHRASED SUMMARY OF COMMENTS ON ECCS RULE

1. Dan L. Johnson, 1535 Meade Ave., San Diego, Ca., 92116, an employee of Southern California Edison Company commenting on his own behalf.

COMMENTS:

1.1 Mr. Johnson is concerned about steam generator tube integrity in general and specifically concerned about hydraulic loads during a LOCA causing concurrent tube rupture, as expressed in a number of letters to the NRC staff and in formal comments on the proposed ECCS rule revision. He cites tests conducted in Semiscale and incomplete EPRI sponsored work at BNL which showed that under certain conditions (ie., rupture of a specific number of tubes), tube ruptures during a LOCA can result in significantly increased peak cladding temperatures.

1.2 The NRC should resolve the steam generator integrity safety issues left unresolved and review in detail the adequacy of existing Plant Technical Specifications pertaining to primary to secondary coolant system deterioration in light of the 95% confidence level; ensuring a PCT of less than 2200°F during the hypothetical LBLOCA.

1.3 The NRC should impose the requirement of a design basis accident safety evaluation, for both LBLOCA and MSLB, as a 10CFR50.59 safety evaluation of any problem or condition resulting from normal operation or design error, which may threaten primary to secondary coolant system integrity.

2. Roger A. Newton, Chairman, Westinghouse Owners Group.

COMMENTS:

2.1 The proposed changes to 10CFR50.46 and Appendix K, which allow the use of realistic calculational models while retaining the current Appendix K rule, is a preferred approach.

2.2 The uncertainty analysis should not be prescriptive but should remain flexible.

2.3 The 95% probability level is adequate for the large break LOCA and is consistent with good engineering practice.

2.4 The current peak cladding temperature limits of 2200°F, the current oxidation limits, the core configuration for coolability requirement, and the requirements on long-term cooling should be retained.

2.5 The current conservative ECCS evaluation method of Appendix K should be permitted indefinitely for those utilities which currently have acceptable analyses and are within compliance (Q1).

2.6 A predetermined explicit degree of conservatism should not be applied to the proposed evaluation model, rather, the amount of conservatism that is applied and retained will be determined through the uncertainty analysis using the proposed model (Q2).

2.7 The proposed amendments to 10CFR50.46 and Appendix K should not be tied to the resolution of all outstanding safety issues (Q3).

2.8 The Advisory Committee on Reactor Safety and their consultants should be the peer review body for the proposed rule change (Q4).

2.9 The language in the proposed 10CFR50,46(a)(1)(i) should be broadened to permit the use of a range of zirconium based alloys for cladding materials.

3. L. T. Gucwa, Manager, Nuclear Safety and Licensing, Georgia Power Company, 333 Piedmont Avenue, Atlanta Georgia, 30308.

COMMENTS:

3.1 Georgia Power Company strongly supports the proposed rule.

3.2 The language of the proposed 10CFR50.46 (a)(3)(i) should be modified to require that only changes or errors greater than 50^{oF} in the non-conservative direction be considered significant. Changes or errors in the conservative direction should not have the same reporting requirements as significant changes or errors in the non-conservative direction.

3.3 The requirement in the proposed 10CFR50.46(a)(3)(ii) should be modified to require that changes or errors be reported in the Annual Report as opposed to a special report.

3.4 The definition of "significant reduction" listed in Appendix K, Section I.C.5.c should be deleted since it is not necessary if comment 3.2 is adopted.

3.4 The Appendix K methods should be allowed indefinitely because they are conservative (Q1).

4. W. J. Johnson, Manager, Nuclear Safety Department, Westinghouse Electric Corporation, Box 355, Pittsburgh, Pennsylvania, 15230-0355.

COMMENTS:

4.1 The commenter supports the current notice and comment process for rule changes in the Federal Register. The rule promulgation process of soliciting public comment should provide sufficient public input such that a time consuming open adjudicative hearing process is not required.

4.2 The proposed rule change would allow application of safety research results in the licensing process and is supported by Westinghouse.

4.3 The uncertainty calculation at the 95% probability level is judged to be adequate for the types of calculations that would be performed, considering the low probability of a large break LOCA.

4.4 The uncertainty analysis should be flexible and not prescriptive so that different approaches can be suggested by licensees and evaluated independently by the NRC staff.

4.5 The current peak cladding temperature limit of 2200°F, the current oxidation limits, the core configuration for coolability requirement, and the requirement on long-term cooling should be retained.

4.6 The proposed reporting procedure for code errors or changes and reanalysis requirements are reasonable.

4.7 The commenter agrees that the 20°F limit as the demarcation between a small and a significant code change effect be changed to 50°F.

4.8 The current conservative ECCS evaluation method of Appendix K should be permitted indefinitely, for those utilities which currently have acceptable analysis and are within compliance (Q1).

4.9 No predetermined explicit degree of conservatism should be applied to the evaluation model, rather, the amount of conservatism that is applied and retained will be determined through the uncertainty analysis using the proposed model (Q2).

4.10 The proposed rule should not explicitly prohibit core power increases (Q3).

4.11 Unresolved safety issues which are directly related to core power should be reviewed to ensure that there is no additional adverse risk to the public or utility due to the proposed plant uprating (Q3).

4.12 A rule on LOCA methodology is not the appropriate vehicle to address requirements for plant upratings. Independent of considerations of acceptable Appendix K models, NRC approval of a plant uprating should focus on safety issues which are impacted by the proposed increase in power level (Q3).

4.13 The peer review of the proposed Appendix K changes is best accomplished by the public review and comment procedure together with review comments and resolutions by the Advisory Committee on Reactor Safety (Q4).

4.14 The designation of the fuel rod cladding such as that given in the proposed 10CFR50.46(a)(1)(i) should be broadened to permit use of a range of zirconium based alloys for cladding material.

4.15 There is no technical basis for the continued use of the Dougall-Rohsenow correlation without further verification. A reasonable time limit, perhaps 3 years, should be placed on those licensees who use this correlation, to either show additional

justification for its use over the full range of conditions for safety analysis, or to replace this correlation with another suitably acceptable and verified correlation.

4.16 The word "immediate" should be deleted from the statement "The affected applicant or licensee shall propose immediate steps to demonstrate compliance or bring the plant design or operation into compliance with 10CFR50.46 requirements" in proposed 10CFR50.46(a)(3)(ii).

5. John. D. Harris, 781 Walker Springs Rd. N.W., Knoxville TN.87963.

COMMENTS:

5.1 The focus then (original ECCS hearings in 1973-1974) and now is on the short term (less than one hour) effects without sufficient and due regard for what must be done to achieve the long term recovery of the plant. Hence the proposed rule is inadequate to assure appropriate coverage of the long term plant response.

5.2 To modify the existing regulations without understanding the long term effects or without having a long term recovery plan in place is not appropriate. A short/long term integrative analysis approach is required but is not currently done except at a few Combustion Engineering designed units.

5.3 The long term decay heat removal issue (NRC USI A-45) has not yet been resolved and to modify the ECCS rules prior to resolution is not a good design practice. The "excessive" safety margins in current ECCS design criteria may be providing the protection needed to overcome some long term system degraded performance.

5.4 The statement that the results are highly conservative are only applicable to the short term LOCA impacts on the core. Consideration of A-45 is questioned.

5.6 A number of technical manuscripts (unpublished) are provided which highlight a number of long term decay heat removal concerns.

5.7 The current evaluation methods can be grandfathered provided they are made to include a long term decay heat removal analysis and a description of how the plant will be recovered (Q1).

5.8 An explicit degree of conservatism is appropriate; however, a different safety factor may be needed for different types of events (Q2).

5.9 The rule implementation should be delayed until the key unresolved safety issues have been resolved. It is very difficult to recognize if the solution to one problem has created another elsewhere (Q3).

5.10 A review by qualified people would not hurt. If the proposed changes are viable, it would not be affected. If problems are found, it is best to find them now and not at a much later time (Q4).

5.11 Existing and proposed ECCS rules and procedures should cover decay heat removal and residual heat removal systems. The ability to go to cold shutdown and ultimately fuel removal should be demonstrated using only safety grade equipment. Hot shutdown is not an acceptable end state for a DBA (including both LBLOCA and other transients).

5.12 The development of a specific post-accident safety philosophy is of paramount importance. PRA and cost-benefit analyses are not the only tools that backfit decisions should be made. Defense-in-depth, experience, judgement also should play dominant roles.

6. "Alert Expert Insider", anonymous letter received June 30, 1987.

COMMENTS:

6.1 "EMERGENCY CORE COOLING SYSTEMS ARE VERY TENDER (SEE ENCLOSURE ONE FOR A DESCRIPTION OF TENDERNESS) AND TENDERIZER SHOULD NOT BE APPLIED AS RECOMMENDED BY THE NUCLEAR REGULATORY COMMISSION ...." The enclosure is "Maestros of Technology," an interview with Arthur Squires by Hal Bowser, Invention & Technology, Summer, 1987, p.24. The article discusses problems in the way that bureaucracies deal with technology and cites the space shuttle Challenger, Chernobyl, the Swedish battleship Vasa, and the British dirigible R101 as examples.

6.2 "PUBLIC COMMENT IS NOT A FAIR PUBLIC HEARING BECAUSE SOME EXPERT INSIDERS WOULD BE FIRED IF THEY WERE PERCEIVED TO SQUEAL."

7. Gary N. Ward, Manager, Reload Licensing, Advanced Nuclear Fuels Corp., 2101 Horn Rapids Rd., P.O. Box 130, Richland, Wa. 99352-0130.

COMMENTS:

7.1 Current models are very conservative and the rule as currently implemented often results in substantial expenditures of resources to install and maintain complex mitigating systems which do not impact true safety and to analyze changes with no significant impact on plant safety. ANF would implement the revised rule to the extent ANF is encouraged by reactor licensees.

7.1 The revised rule should not require reporting of changes or errors that result in PCT changes of greater than 50°F since this is not a safety hazard unless the criteria (ie., 2200°F) is exceeded.

7.3 Full LOCA/ECCS analyses should not be required for reload fuel suppliers since the hydraulic aspects of the LOCA are unchanged by different fuel. Only a specific analyses of the fuel heatup is required.

7.4 Continued use of Dougall-Rohsenow should be allowed since Appendix K contains a larger total conservatism than the non-conservatism in this correlation.

7.5 Equal opportunity to obtain approval of models using either Appendix K or the new best estimate approach should be assured since the industry has invested great expense in Appendix K models and modifications to these models within the limits of Appendix K would be less costly than shifting to the new method (Q1).

7.6 The method of Appendix K has proven to be conservative and therefore should be permitted indefinitely (Q1).

7.7 The rule as written along with the criteria of 10CFR50.46 provides for sufficient conservatism without additional conservatisms added (Q2).

7.8 Current regulations adequately address the safety issues with regard to power upgrades without promulgating additional regulations in this area (Q3).

7.9 The review of this proposed rule change that has occurred as a result of the research efforts over the past several years has provided a broad based technical community review. There is no obvious benefit of additional review (Q4).

8. Susan L. Hiatt, Ohio Citizens for Responsible Energy, Inc., 8275 Munson Rd., Mentor, Oh 44060.

COMMENTS:

8.1 An extension of the comment period is requested in order to allow review of NUREG-1230 which provides the technical basis for the ECCS rule change.

9. E.C. Simpson, Director Nuclear Site Support, Florida Power Corp., P.O. Box 219, Crystal River, Fl. 32629.

COMMENTS:

9.1 Increased power levels increase the rate of fission product formation, but not necessarily the total amount available for release to the public in a core melt scenario. The amount for release depends on the megawatt-days, with or without the new rule. A five percent power increase would not result in five percent more fission products at the end of life (Q3).

10. J. Ted Enos, Manager, Nuclear Engineering and Licensing, Arkansas Power & Light Company, P.O. Box 551, Little Rock, Arkansas 72203.

COMMENTS:

10.1 The ECCS methods of Appendix K should be permitted indefinitely since the methods are conservative and would allow use of existing models if relief from operating restrictions is not needed or desired (Q1).

10.2 The rule should not contain an explicit degree of conservatism. The conservatism consistent with the uncertainty of the calculation and guidance in the form of a regulatory guide should suffice (Q2).

10.3 The rule should not explicitly prohibit any increase in power if it is demonstrated that the requirements of 10CFR50.46 are met, the uncertainties are adequately quantified and there is negligible effect on the environmental impact (Q3).

10.4 Review of the technical basis for the proposed rule change by an independent group is not warranted. ECCS performance information obtained by NRC, DOE, industry and foreign researchers is considered an adequate basis to proceed with the rule change (Q4).

10.5 The commenter agrees with redefinition of a "significant change" from 20°F to 50°F, but considers other proposed reporting requirements unnecessary or even undesirable. The existing rule does not require reporting per se, rather only that a description of the evaluation model be revised. The proposed requirement to provide a 30 day report is not appropriate because a 50°F change is not directly related to a safety concern requiring immediate

reporting, considering that the change could be in the conservative direction or could still remain well within the safety limit of 2200°F. An annual reporting of significant changes appears reasonable. Also, the requirements of 50.72 and 50.73 provide adequate controls for ensuring proper notification and reporting; consequently, the additional provisions in the proposed rule are unnecessary.

10.6 The reporting of non significant changes is considered unnecessary until such time that the cumulative effect of such changes is greater than 50°F. As stated before, annual reporting is sufficient in this regard.

10.7 Reporting of "applications of such model" that results in peak clad temperature changes is an unnecessary expansion of existing regulation. Changes to the facility, such as fuel property changes, are already subject to 50.59 and thus are duplicative.

10.8 The backfit analysis is incomplete since the proposed rule does require new reporting requirements. The current analysis does not address the additional burden of reporting minor changes to the model and each application of the ECCS model.

11. Charles Turk, Chairman, B&WOG Analysis Committee, B&W Owners Group, Suite 220, 7910 Woodmont Ave., Bethesda Md. 20814.

COMMENTS:

11.1 The commenter agrees that use of best estimate (BE) codes should offer an improved basis for refining our understanding of the margins calculated to be available in nuclear power operation.

11.2 The commenter also feels strongly that use of Appendix K should be permitted indefinitely for those utilities who do not intend to use the new BE methodology (Q1).

11.3 The commenter hopes that the regulatory review process will focus on the relationship of the BE codes to the data, rather than comparison of BE results to EM results. An affirmation of the focus of the review process would be a desirable part of the final rule and would preclude any unnecessary and unproductive reevaluations using EM codes.

11.4 Higher power levels do not necessarily result in a higher fission product content hypothesized for release during a severe accident (Q3).

12. D.W. Edwards, Director of Industry Affairs,Yankee Atomic Electric Co., 1671 Worchester Rd., Framingham, Ma. 01701.

COMMENTS:

12.1 The proposal to allow more realistic and accurate calculations presents numerous advantages including:

1. Improvements in plant operation such as refinement of reactor simulator benchmarking, operator training, and emergency response guidelines.

2. Integrated improvements in plant performance from enhanced fuel design, fuel management, and reactor operation, such as longer fuel cycles, flexible in-core fuel management, reduced vessel fluence, and realistic ECCS surveillance requirements.

3. Improved utilization of manpower and computational resources to support plant operational safety and performance.

12.2 The proposed approach requiring licences to quantify and justify the uncertainty in model predictions, while not prescribing the uncertainty evaluation techniques to be used, is technically sound. This allows licencees to develop models consistent with the scope and type of application (generic vs. plant specific). It also provides flexibility to incorporate future improvements in uncertainty analysis methods and results.

12.3 Research has shown that the five ECCS acceptance criteria provide a conservative bound for significant fuel damage and should be retained.

12.4 The commenter endorses the proposal to define a significant error as a deviation in peak clad temperature of more than 50°F. Such definition balances the safety significance with resources needed for reanalysis. The commenter agrees that the new reporting requirements will allow NRC to monitor changes as they occur, will not impose undue burden on licensees, and will remove ambiguities that exist with current reporting practices.

12.5 The method of Appendix K should be permitted indefinitely since research has proven that Appendix K methods provide conservative results (Q1).

12.6 An explicit degree of conservatism would counter the process discussed in the proposed rule change. The ECCS acceptance criteria contain additional conservatisms to preclude significant core damage from a LOCA (Q2).

12.7 The rule should not explicitly prohibit any increase in power. Any proposed increase in power undergoes a thorough reanalysis and review by the licensees and NRC. All regulations must be met, thereby providing for the health and safety of the public and environment (Q3).

12.8 Government and industry research has been available for and subjected to technical review throughout this time frame. Public and industry comments from both a previous proposed rule and this proposed rule will be incorporated in the rule change. In addition, the ACRS and its consultants will provide review of the rule.

13. E.J. Mroczka, Senior Vice President, Northeast Utilities, P.O. Box 270, Hartford, Connecticut 06141-0270.

COMMENTS:

13.1 The commenter supports the intent of the proposed revisions in that they provide for greater operational flexibility and potentially substantial economic benefits, while at the same time preserving appropriate safety margins.

13.2 The current Appendix K methods should be permitted indefinitely. For plants with large LOCA margins, there should be no need to spend resources to reevaluate the plant. The requirement for many reviews could significantly slow the NRC review process for utilities needing a timely review to resolve a particular safety issue (Q1).

13.3 The proposed rule should not prohibit increases in power until all severe accident and unresolved safety issues are resolved. To include severe accident issues into the rule change is inappropriate since the capacity and cooling capability of an ECC system is not influenced by fission product release following a severe accident scenario (Q3).

13.4 The proposed rule is the culmination of the results of 20 years of research from a number of independent organizations. It is expected that an independent review of the proposed rule would not be of significant value (Q4).

13.5 The proposed reporting requirements represent an improvement over the current Appendix K. However, the commenter believes 100°F is a more appropriate threshold for a significant change in PCT. Changes in PCT of 50°F to 75°F frequently result from normal system and equipment degradation. Such changes are insignificant in terms of plant safety provided there is sufficient margin between the calculated PCT and the 2200°F limit.

14. A.E. Scherer, Director, Nuclear Licensing, Combustion Engineering, 1000 Prospect Hill Rd., P.O. Box 500, Windsor Connecticut 06095-0500.

COMMENTS:

14.1 Use of the amassed body of research data to remove excess conservatism is long overdue.

14.2 Past experience has shown that minor code changes are made and errors do occur. Most do not effect the safety of the plant. Preparing a report to the NRC and having it reviewed would not be an inconsequential task. Having to report each change or error, even on an annual basis, might be counter-productive.

14.3 Based upon experience with potential sensitivity of LOCA models, a change of 100°F is a more appropriate screening value to employ as a measure of a significant change in peak clad temperature.

14.4 Omission of 10CFR21 from the reporting requirements raises a question as to whether or not NRC has made a determination that significant changes or errors, or exceeding the 10CFR50.46(b) criteria does not represent a substantial safety hazard as defined in 21.3(b).

14.5 There does not appear to be a safety related need to phase out Appendix K models. Economic incentives are most likely sufficient in the long term to bring about adoption of the more realistic model approach (Q1).

14.6 Inclusion of an additional explicit degree of conservatism would have the same effect as lowering the acceptance criteria. The acceptance criteria are already conservatively low (Q2).

14.7 A prohibition on an increase in power until all severe accident issues and unresolved safety issues are resolved appears unwarranted. These power increases are the same order of magnitude of power increases already approved in the past (Q3).

14.8 Given the accumulation of data, a review of the technical basis for the proposed rule change by an independent group is not justified. Further, review by an independent group may be impractical due to difficulty finding a sufficiently technically qualified, yet independent and unbiased group (Q4).

15. H.W. Keiser, Vice President-Nuclear Operations, Pennsylvania Power & Light Co., 2 North Ninth St., Allentown, Pa. 18101.

COMMENTS:

15.1 The proposed rule is a very thorough and beneficial approach to endorsing the use of realistic evaluation models.

15.2 The proposed new reporting requirements for changes and error corrections are reasonable and not burdensome to licensees and NRC staff.

15.3 The commenter agrees that grandfathering use of Dougall-Rohsenow is appropriate on the basis that there exists adequate conservatism in Appendix K.

15.4 The commenter also agrees that changes which result in reduction of overall model conservatism deteriorate the basis for continued use of Dougall-Rohsenow. However, since all error corrections are improvements, regardless of the effects on calculated temperature, we disagree that a significant reduction in calculated peak cladding temperature should be justification for disallowing use of Dougall-Rohsenow. The conservatism in Appendix K is not changed by error correction.

16. C.O. Woody, Group Vice President, Nuclear Energy, Florida Power and Light Co.,P.O. Box 14000, Juno Beach, Fl. 33408-0420.

COMMENTS:

16.1 There is no need for the annual reporting of minor changes and errors.

16.2 The commenter suggests that the proposed 50°F definition of a significant error or change be modified to be 100°F or 50% of the margin to the PCT limit, whichever is smaller.

17. C.W. Fay, Vice President, Nuclear Power, Wisconsin Electric Power Co., 231 W. Michigan, P.O. Box 2046, Milwaukee, Wi. 53201.

COMMENTS:

17.1 The Appendix K methods should be permitted indefinitely. Current Appendix K methods are conservative and a requirement to reanalyze would place an unfair burden on utilities with acceptable operating margin. Not all plants are inhibited by the current rule (Q1).

17.2 Conservatism is already included in the uncertainty analysis and the acceptance criteria. The objective of the rule change is to eliminate the excessive and unnecessary conservatism of certain Appendix K requirements (Q2).

17.3 The rule should not explicitly prohibit any increase in power until all severe accident issues and unresolved safety issues are resolved. A utility applying for a plant uprating should be required to satisfy regulatory requirements including severe accident and unresolved safety issues impacted by the application. Explicit prohibiting a power increase may prevent an increase even if health and safety of the public is not affected (Q3).

17.4 The technical basis of the proposed rule should be reviewed by an independent group as provided for by the NRC proposed rule process. The ACRS is an independent group and should review the proposed rule change (Q4).

17.5 The commenter is currently performing a BE calculation of Point Beach and looks forward to taking advantage of the significant investment in analysis methods already developed. Please do not delay in approving the proposed rule.

17.6 The commenter is pleased that the proposed rule does not prescribe the uncertainty techniques. A realistic technique must be allowed to change as the state-of-the-art changes. Ensure that the regulatory guide is not interpreted as a requirement.

17.7 The proposed procedure for reporting code errors and changes appears to be reasonable. The commenter recognizes that the proposal is a relaxation of current practice and may eliminate costs associated with reporting insignificant changes.

18. G.C. Sorensen, Manager, Regulatory Programs, Washington Public Power Supply System, P.O. Box 968, 3000 George Washington Way, Richland, Washington 99352.

COMMENTS:

18.1 The commenter believes that the use of more realistic or "Best Estimate" methods should be encouraged whenever research results in new information which replaces previous uncertainties, thus reducing the need to maintain unnecessary conservative requirements.

18.2 Since changing to the new methods is not necessary for safety reasons, it could be difficult for licensees to justify the not inconsiderable cost involved to their respective State regulatory bodies (Q1).

18.3 No change is being proposed to the Peak Clad Temperature limit of 2200°F. Thus the "explicit degree of conservatism" referred to does exist (Q2).

18.4 Power increases of 5-10% will have an insignificant effect on risk, since the incremental increase in fission product inventories available for release during a worst case DBA would be no more than 5-10%. The 2200°F limit remains, thus there will be a small decrease in margin between the temperature at which significant fuel damage is predicted to begin and the calculated PCT. The increase in risk due to this small decrease in margin is not significant, especially when compared to the margin remaining (Q3).

18.5 The costs incurred by an independent peer review would likely not be justified by the results. The scientific basis for the rule change is well understood, and has been subjected to a very robust review process which extended over a period of many years (Q4).

19. S.R. Zimmerman, Manager, Nuclear Licensing Section, Carolina Power & Light Co., 411 Fayetteville St., P.O. Box 1551, Raleigh, NC 27602.

COMMENTS:

19.1 The revision makes appropriate use of the decade of research on ECCS performance and will provide for improvements to plant safety through better understanding of plant behavior.

19.2 The conservatisms of Appendix K as demonstrated by previous research will ensure that a plant's margin of safety is no less than that provided by the proposed revision. For these reasons, the requirement to implement the proposed rule without the option at plants not restricted by LOCA limits would be a substantial and unnecessary burden (Q1).

19.3 Adequate conservatism will be ensured by the continued use of the 10CFR50.46(b) limits and the requirement to account for uncertainties in the evaluation model. Additional conservatisms would limit the value of the proposed revision(Q2).

19.4 The changes to power level or power distribution are not expected to significantly impact severe accident scenarios or unresolved safety issues (Q3).

19.5 The proposed revision is the product of 13 years of research, review, and improvement in the practical understanding of ECCS performance. We believe that it is highly unlikely that additional review would discover issues which have not already been addressed. There is, however, reason to believe that additional delays in the issuance of a final rule would result (Q4).

19.6 The commenter encourages the Commission to implement the proposed rule revision in a timely manner without significant revision. The revision will prove of benefit to utilities without a reduction in the health and safety of the public.

20. J.D. Sieber, Vice President, Nuclear, Duquesne Light, P.O. Box 4, Shippingport, Pa 15077-0004.

COMMENTS:

20.1 The more accurate calculations will be useful for our operator training, accident management, and simulator response.

20.2 The current limits should be retained. These current limits preclude core temperatures which could result in significant core damage, and as such, provide additional conservatism for the design basis accident as well as economic protection for the reactor. The limits also act to protect the health and safety of the general public (Q2).

20.3 The reporting procedure for code errors or changes and reanalysis requirements appear to be reasonable and reflects a balanced consideration of the financial burden to the utility and plant safety. The proposed 50°F value, while larger than the current 20°F value is more consistent with recognition that a large break LOCA is a very improbable event and that additional margin above 2200°F does exist.

20.4 The uncertainty analysis, which will be required for the proposed rule change, should not be prescriptive but should remain flexible. The evaluation of the uncertainties at a 95% probability level is adequate for the large break LOCA and is consistent with good engineering practice.

20.5 The current Appendix K method has been shown to be conservative, and the requirement of reanalysis using the methods of the proposed rule change would place an unfair financial burden on utilities which already have acceptable operating and ECC margins to the licensing limits and would not result in any improvements in safety (Q1).

20.6 The amount of conservatism that is applied and retained should be determined through the uncertainty analysis using the proposed model (Q2).

20.7 The 5 to 10 percent power uprating is well within the additional margin identified in the source term studies such that there is no calculated increase in risk to the public (Q3).

20.8 It is unrealistic to believe that all unresolved safety issues will be resolved quickly. A requirement of having to resolve all unresolved safety issues before a plant could uprate would unfairly penalize those utilities and their customers who desire to uprate for economic reasons. However, those unresolved safety issues which are directly related to core power should be reviewed to ensure that there is no additional adverse risk to the public or utility due to the proposed plant uprating. The proposed rule should not prevent justified uprating until all unresolved safety issues are completely resolved. The merits of the change should be judged by the research results (Q3).

20.9 A peer review, if properly performed, can be useful. However, we believe that this is best accomplished by a public review and comment procedure together with review of the comments and resolution by the ACRS. The ACRS by law, has the proper advisory role, historical perspective and technical basis. While the APS could perhaps address more philosophical questions as to the impact of the rule change on society; this organization would have to rely on experts from the national laboratories which have been involved in the nuclear program. We believe that this is the proper role for the ACRS and another independent review is not warranted (Q4).

21. W.L. Stewart, Vice President, Nuclear Affairs, Virginia Electric and Power Co., Richmond, Va. 23261.

COMMENTS:

21.1 The new rule should specify cladding as "zirconium base alloy" to permit fuel vendors to develop advanced cladding materials. The fuel vendor would then be required to demonstrate that the acceptance criteria for Zircaloy are conservative for the zirconium base alloy.

21.2 Proposed section 50.46(a)(3)(i) should state "...cumulation of changes and errors such that the net sum of the respective temperature changes is greater than 50°F." For example, a case in which two separate errors caused changes in peak clad temperature of +35°F and -20°F, respectively, would be reported in the annual report, since the net change is only 15°F. As written, the proposed rule would classify this case as a significant change, requiring a report to be submitted within 30 days.

21.3 The requirement for annual reporting of non significant changes and errors) explicitly include a threshold temperature value above which this part of the rule is applicable. Use of a threshold value would preclude reporting of trivial changes which clearly do not impact safety.

21.4 Many nuclear units have existing analyses which do not impose operational restrictions, and therefore may prefer to have the option of continued use of Appendix K for future reanalyses. If Appendix K requirements are to be phased out, consideration should be given to the impact this may have on existing or future small break LOCA analyses. Since small break LOCA analysis results usually do not impose restrictions on plant operations, retention of existing analysis methods, versus imposing the alternative analysis methods in the proposed rule, would be preferable for the analysis of these events (Q1).

21.5 Any degree of conservatism required to meet the acceptance criteria should be specified in the regulatory guide rather than in the rule. This will allow some amount of discretion on the part of the NRC and licensees in evaluating the high level of probability. Such flexibility is preferable when using best estimate calculational techniques since multiple approaches incorporating differing degrees of evaluation model conservatism could provide comparable levels of probability (Q2).

21.6 The proposed rule need not prohibit approval of such applications until severe accident issues and unresolved safety issues are resolved. Including this restriction in the proposed rule provides no additional regulatory value, since existing regulations require the NRC to determine whether any such issues must be resolved prior to approval of a specific license application (Q3).

21.7 This effort involved obtaining new data, developing improved analysis models, and revising analysis methodologies over a 13 year period. The technical basis for this proposed rule should not be reviewed by any independent group, primarily because such a review at this late stage would add no new understanding on which to base the rule.

22. David W. Cockfield, Vice President, Nuclear, Portland General Electric Co., 121 S.W. Salmon St., Portland Oregon 97204.

COMMENTS:

22.1 PGE supports the proposed rule change. The more accurate calculations will also be useful for our operating training, accident management, and simulator response.

22.2 PGE agrees that the current limits should be retained. These current limits preclude core temperatures which could result in significant core damage, and as such, provide additional conservatism for the design basis accident as well as economic protection for the reactor (Q2).

22.3 The reporting procedure for code errors or changes and reanalysis requirements using a 50°F limit between small and larger changes is reasonable and does reflect a consideration of the financial burden to the utility that small changes in the evaluation model can cause.

22.4 The uncertainty analysis, which will be required for the proposed rule change, should not be prescriptive but should remain flexible. The evaluation of the uncertainties at a 95% probability level is adequate for the large break LOCA and is consistent with good engineering practice.

22.5 The current Appendix K method has been shown to be conservative, and the requirement of reanalysis using the methods of the proposed rule change would place an unfair financial burden on utilities which already have acceptable operating and ECC margins to the licensing limits and would not result in any improvements in safety (Q1).

22.6 The amount of conservatism that is applied and retained should be determined through the uncertainty analysis using the proposed model (Q2).

22.7 The 5 to 10 percent power uprating is well within the additional margin identified in the source term studies such that there is no calculated increase in risk to the public (Q3).

22.8 PGE believes that a proposal for a rule change should not be tied to the resolution of all outstanding safety issues. The basis for the rule change is the significant amount of research and development which has been performed on ECCS performance and merits that the change should be judged by the research results (Q3).

22.9 PGE recommends that the ACRS and their consultants be the peer review body for the proposed rule change.

23. J.R. Thorpe, Director, Licensing & Regulatory Affairs, GPU Nulcear Corp., 100 Interpace Parkway, Parsippany, NJ 07054-1149.

COMMENTS:

23.1 This proposed rule should be adopted without delay.

23.2 The non-routine reporting criteria should be included in 10CFR50.72 and 50.73 and not in Appendix K.

23.3 All ECCS evaluation models (EM) which were reviewed and approved by the NRC since the issuance of the final acceptance criteria in 1974, and including those found to be inconsistent with SECY-83472 by the NRC, should be considered acceptable for satisfying the requirements of the new paragraph 50.46(a). Although calculational results using previously approved EM may provide very conservative and inappropriate results for operation and engineering purposes, there should not be new requirements to recalculate in compliance with this rule change. The recalculation should be left to the individual owners of the facilities (Q1).

23.4 Power level increase, if found to be justifiable, should not be prohibited. This 5-10% power increase is not expected to cause difficulty in meeting the existing environmental limits (Q3).

23.5 Through GPU's experience with GE's SAFER-CRECOOL, no additional review by an independent group such as the APS is necessary. The proposed alternative method was reviewed by several different engineering groups within GE and knowledgeable and experienced utility engineers from sponsoring utilities before topical reports were submitted to the NRC. Consultants were brought in by the NRC staff for numerous occasions during the review period. The ACRS staff also reviewed the material. Therefore, additional independent group review is not necessary.

24. W.G. Counsil, TU Electric, 400 North Olive St., LB 81, Dallas, Texas 75201.

COMMENTS:

24.1 The commenter encourages the Commission to implement the proposed revisions without alteration.

24.2 Because the current Appendix K conservatisms ensure margins equal to or greater than those provided by the proposed rule, no safety issue would be created if those plants are allowed to employ the current methodology, thereby avoiding substantial and unwarranted financial burdens involved with reanalysis (Q1).

24.3 Sufficient conservatism is assured under the proposed rule by (1) continued use of conservative 10CFR50.46 limits, and consideration of uncertainties in the PCT calculated by realistic models. Any additional degree of conservatism would be an arbitrary and unwarranted burden which would minimize the incentives for a utility to use the proposed rule (Q2).

24.4 The proposed rule continues to assure adequate conservatism in ECCS models. An arbitrary prohibition of power increase based on yet unquantified and, perhaps unquantifiable impacts, if any, from severe accident and unresolved safety issues would not be based on a cognizable safety concern. This results in economic burdens without a demonstrated and commensurate increase in safety (Q3).

24.5 These results have been reviewed by numerous groups, including government agencies, the ACRS, industry organizations, and independent professional societies. Additional review is unlikely to result in any new developments. Additional reviews could, however, result in further delay in issuing the already long overdue proposed amendments.

25. J.A. Tiernan, Vice President, Nuclear Energy, Baltimore Gas and Electric, Charles Center, P.O. Box 1475, Baltimore, MD. 21203.

COMMENTS:

25.1 We continue to support the Commission's efforts to improve existing regulations through use of improved analysis methodology and recent technical developments. The proposed rule is an acceptable approach to ensure the safety of the public without unnecessarily restricting applicants and licensees.

25.2 The proposed changes to the reporting and reanalysis requirements are realistic and require immediate reporting only if the change or error is significant. However, the proposed rule does provide for a new annual reporting requirement for all changes or errors that affect peak fuel cladding temperature no matter how minor. The licensee is required by 10CFR50.59 and 50.71 to annually submit summaries of completed 10CFR50.59 safety evaluations. Therefore, since most changes or errors are already reported to the staff under existing requirements, the need for a new annual reporting requirement for changes and errors which are not significant is not necessary and this requirement should be deleted from the proposed rule.

25.3 Those utilities which are not LOCA limited should be allowed to continue to use their present ECCS evaluation methodology and thus be provided the option of expending the resources to update to the proposed methodology. The Regulatory and Backfit Analysis did not assess a mandatory requirement to shift to the proposed methodology and any mandatory requirements should be assessed in terms of the value (e.g., public benefits such as safety) and impact (e.g., consequences and costs)(Q1).

25.4 The proposed amendment would allow the use of more realistic calculations of ECCS performance along with reasonable estimates of uncertainty. This, combined with the fact that 10CFR50.46 limits of 2200°F and 17% cladding oxidation are believed to already be appropriate and conservative limits below which substantial core damage would occur, make this application of additional conservatism unnecessary (Q2).

25.5 Current regulations adequately account for any changes in the licensed power. A number of plants have uprated power level and addressed all regulations including increased radioactive source terms. Since the precedent of increasing power has been repeatedly established, prohibiting such an action based on a realistic LOCA analysis is unwarranted (Q3).

25.6 The technical basis for the proposed rule change is supported by over 13 years of LOCA/ECCS tests and experiments, and computer code development research in laboratories around the world. NUREG-1230 presents a detailed and comprehensive discussion of the increase in knowledge acquired by extensive research. Every aspect of the research has already been independently reviewed and the international nuclear technical community has reached a consensus that sufficient information now exists to justify more realistic LOCA calculations.

26. J.C. Brons, Executive Vice President, Nuclear Generation, New York Power Authority, 123 Main St., White Plains, NY 10601.

COMMENTS:

26.1 The commenter strongly support the promulgation of these proposed amendments. NYPA's Fitzpatrick facility has already benefited from the implementation of best-estimate ECCS calculations. Further improvements at Fitzpatrick and equivalent improvements at Indian point are expected if these amendments are adopted.

26.2 Proposed section I.C.5.c of Appendix K should be clarified. Suggested language could be, "For this purpose, a significant reduction in the overall conservatism in the evaluation model is defined as a reduction in the calculated peak fuel cladding temperature of at least 50°F from that which would have been calculated on (effective date of rule) due either to individual changes or error corrections or the net effect of an accumulation of changes or error corrections."

26.3 Utilities should not be required to change analysis methodology unless it is economically favorable to do so (Q1).

26.4 The purpose of performing best estimate calculations, taking into account uncertainties, is to avoid the application of an arbitrary degree of conservatism. There is no reason that an explicit degree of conservatism should be applied to the models (Q2).

26.5 The NRC will require an evaluation to demonstrate that a plant will continue to operate in accordance with all applicable regulations prior to an increase in licensed power. There is no reason that a prohibition of such power increase should be included in this regulation. If the rules are currently sufficient to protect the health and safety of the public, they will continue to be sufficient (Q3).

26.6 Power gained from an increase in licensed power may displace power which would otherwise be generated using a new power plant using nuclear fission as a heat source, so that the overall available source term would be the same, although the public would suffer from inefficient use of capital to build a new plant (Q3).

26.7 This rule has been prepared by the NRC staff based on research which clearly demonstrates that current evaluation models are extremely conservative. An opportunity for industry and public comment has been provided by this rulemaking proceeding (Q4).

27. L.B. Long, General Manager, Nuclear Safety and Fuel, Southern Company Services, Inc., P.O. Box 2625, Birmingham, Alabama 35202.

COMMENTS:

27.1 The commenter supports the proposed rule change which ... will also be useful for operator training, accident management, and simulator response.

27.2 The current peak cladding temperature limit of 2200°F (etc.) ... should be retained. These current limits ... provide additional conservatism for the design basis accident as well as economic protection for the reactor.

27.3 The revised reporting procedure ... is reasonable and does reflect consideration of the financial burden to the utility that small changes in the evaluation model can cause.

27.4 The uncertainty analysis ... should not be prescriptive, but should remain flexible. ... a 95% probability level is adequate for the large break LOCA and is consistent with good engineering practice.

27.5 The current Appendix K method has been shown to be conservative, and the requirement of reanalysis using the methods of the proposed rule would place an unfair financial burden on those utilities which already have acceptable operating and ECC margins ... and would not result in any improvement in safety (Q1).

27.6 The amount of conservatism that is applied and retained should be determined through the uncertainty analysis using the proposed model (Q2).

27.7 The research on fission product source term indicates that significantly lower amounts of fission products would be released than currently used. The 5 to 10 percent power uprating mentioned in the proposed rule change is well within the additional margin identified in the source term studies such that there is no calculated increase in risk to the public (Q3).

27.8 The basis for the rule change is the significant amount of research ... and the merits of the change should be judged by the research results (Q3).

27.9 The ACRS and their consultants should be the peer review body for the proposed rule change (Q4).

28. M.O. Medford, Manager of Nuclear Engineering and Licensing, Southern California Edison Co., P.O. Box 800, 2244 Walnut Grove Ave., Rosemead, California 91770.

COMMENTS:

28.1 The proposed rule recognizes the degree of conservatism in the present requirements ... (and) the knowledge acquired over the past 13 years ....

28.2 Specific potential advantages ... are an increase in load maneuverability, less restrictive fuel assembly power peaking limits and greater margin when performing ECCS equipment surveillances ... reduced neutron fluence at the vessel wall and improved fuel economy.

28.3 Utilizing this more realistic method ... is a major undertaking .... However, based on the potential for these advantages, the Edison Company strongly supports the proposed rule.

28.4 The results of a cost benefit analysis may or may not favor a change in policy in the distant future (Q1).

28.5 The explicit degree of conservatism is adequately addressed in the proposed rule and environmental impact analysis (Q2).

28.6 Increases in approved power levels are adequately addressed in the proposed rule and environmental impact analysis (Q3).

28.7 The proposed rule is currently receiving adequate review by all interested parties (Q4).

29. L.D. Butterfield, Nuclear Licensing Manager, Commonwealth Edison, P.O. Box 767, One First National Plaza, Chicago, Illinois 60690.

COMMENTS:

29.1 The proposed rule would remove needless conservatisms from the regulations while not requiring reanalysis where the analysis supporting the current licensing basis is adequate.

29.2 Proposed section 50.46(a)(2) ... appears to eliminate the need for the Director, NRR to issue an Order to impose restrictions on reactor operation. Thus, this provision expands the Director's discretion by allowing him to impose immediate effective restrictions on reactor operation without a finding that such immediately effective action is required to protect the public health, safety, or interest.

30. Susan L. Hiatt, Ohio Citizens for Responsible Energy, Inc., 8275 Munson Rd., Mentor, Oh 44060.

COMMENTS:

30.1 OCRE believes the proposed revision to the ECCS rule is unwarranted, primarily because it is based on illegal consideration of cost saving to licensees (several court cases and statements by the Director of NRR are cited) and has questionable safety benefits.

30.2 It is complained that the present Appendix K models are unduly conservative. What's wrong with being conservative? When the public health and safety are at stake, it is certainly better to err on the side of caution.

30.3 The commenter quotes from the 1975 GE Nuclear Reactor Study (Reed Report) and highlights statements concerning lack of understanding of phenomena and small safety margins. She also notes that the rule would decrease margins rather than increase them as recommended by GE.

30.4 The fission product inventories will likely increase .... This is an increase in risk, which MAY be offset by safety benefits for SOME plants. As this is plant-specific, generic rulemaking does not seem to be the appropriate avenue. It would make more sense to include taking credit for ECCS margins to gain risk reduction from PTS and diesel failures as an option in the IPE to be conducted as part of the Severe Accident Policy Statement implementation.

30.5 ECC bypass data for a full-scale PWR vessel will be made available from forthcoming experiments in the 2D/3D program. It thus does not appear that sufficient experimental basis exists for rule revision at this time.

30.6 The technical basis for the rule revision should be reviewed by an independent group such as the APS. Increases in power limits should not be permitted until all severe accident issues and unresolved safety issues are resolved.

30.7 OCRE may submit further comments after reviewing NUREG-1230.

31. Susan L. Hiatt, Ohio Citizens for Responsible Energy, 8275 Munson Road, Mentor, OH 44060

COMMENTS:

31.1 The primary motive for the ECCS rule change is cost savings to licensees and this is illegal.

32. (Redesignated as Commenter 1a) Dan L. Johnson, 1535 Meade Ave., San Diego, Ca., 92116, an employee of Southern California Edison Company commenting on his own behalf.

COMMENTS:

Commenter provided no new comments but rather supplied further background documentation supporting his original comment letter.

33. R. Artigas, Manager of Licensing and Consulting Services, General Electric Company, 175 Curtner Ave, San Jose, California.

COMMENTS:

33.1 General Electric Company agrees that the proposed amendment is needed to take advantage of research performed since the current rule was written.

33.2 The question of permitting conservative ECCS evaluation methods of Appendix K indefinitely is premature. It should be considered when significant experience has been gained with implementation of the new features of the proposed rule.

33.3 The research cited in the Federal Register Notice(Vol. 52, No. 41 Pg. 6337) demonstrates that significant fuel damage will not occur below 2600°F This supports an existing 400°F conservatism relative to the 2200°F LOCA limit.

Additional conservatism is introduced when the uncertainty is accounted for such that there is a high level of probability (as defined in SECY 83-472) that the LOCA limit is not exceeded. Further conservatism is not justified.

33.4 Increases in approved power level have been justified under the existing rule and with existing ECCS models. While the proposed rule change may facilitate such increases in the future it does not, by itself, allow high power operation. Such operation must be justified by consideration of many subjects beyond ECCS performance. Furthermore, the small increase in fission product inventory resulting from a 5-10% power increase will have a negligible impact on the absolute magnitude of overall risk. Severe accident issues should be addressed in their own forum and not be used to confound other activities which could offer the opportunity for improved plant performance and reduced power generation costs. Because of this, the suggested prohibition in power level increase is totally inappropriate.

33.5 The technical basis for the proposed rule has been scrutinized for years by NRC, ACRS, National Laboratories, Vendors, et.al.. Further review by yet another party will serve no useful purpose.

33.6 GE has obtained approval of an improved LOCA model and application methodology for the BWR which fully implements the requirements established in SECY 83-472. This fact is acknowledged in the backfit analysis provided in the Federal Register notice (Vol. 52, NO.41 Pg. 6337). It was clearly our intention and the intention of the NRC reviewers, that the approved model and methodology not be impacted by the proposed new LOCA Rule. In order to avoid any unforeseen complications, clarifying language should be added to the amendment of 10CFR50.46 which indicates that prior approved ECCS evaluation models which are consistent with SECY 83-472 are acceptable for satisfying the requirements of paragraph 50.46(a).

ENCLOSURE E

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Emergency Core Cooling Systems; Revisions to Acceptance Criteria

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its regulations to allow the use of alternative methods to demonstrate that the emergency core cooling system (ECCS) would protect the nuclear reactor core during a postulated design basis loss-of-coolant accident (LOCA). The Commission is taking this action because research, performed since the current rule was written, has shown that calculations performed using current methods and in accordance with the current requirements result in estimates of cooling system performance that are significantly more conservative than estimates based on the improved knowledge gained from this research. While the existing methods are conservative, they do not result in accurate calculation of what would actually occur in a nuclear power plant during a LOCA and may result in less than optimal ECCS design and operating procedures. In addition, the operation of some nuclear reactors is being unnecessarily restricted by the rule, resulting in increased costs of electricity generation. This rule, while continuing to allow the use of current methods and requirements, also allows the use of more recent information and knowledge to demonstrate that the ECCS would protect the reactor during a LOCA. This amendment, which applies to all applicants for and holders of construction permits or operating licenses for light water reactors, also relaxes requirements for certain reporting and reanalyses which do not contribute to safety.

EFFECTIVE DATE: ( )

FOR FURTHER INFORMATION CONTACT: L. M. Shotkin, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 492-3530.

SUPPLEMENTARY INFORMATION:

BACKGROUND

On March 3, 1987, the Nuclear Regulatory Commission published in the Federal Register proposed amendments (52 FR 6334) to 10 CFR Part 50 and Appendix K. These proposed amendments were motivated by the fact that since the promulgation of Section 50.46 of 10 CFR Part 50, "Acceptance Criteria for Emergency Core Cooling Systems (ECCS) in Light Water Power Reactors," and the acceptable and required features and models specified in Appendix K to 10 CFR Part 50, considerable research has been performed that has greatly increased the understanding of ECCS performance during a LOCA. It is now confirmed that the methods specified in Appendix K, combined with other analysis methods currently in use, are highly conservative and that the actual cladding temperatures which would occur during a LOCA would be much lower than those calculated using Appendix K methods. In soliciting the public's comments on the proposed rule, the NRC specifically requested its views on questions posed by Commissioner Asselstine and the Advisory Committee on Reactor Safeguards(ACRS). The ACRS requested that the Commission solicit the public's comments on whether the existing rule should be "grandfathered" indefinitely. That is:

1. Should the conservative ECCS evaluation method of Appendix K be permitted indefinitely or should this aspect of the ECCS rule be phased out after some period of time?

Commissioner Asselstine requested the public's comments on the following:

2. Should this rule change include an explicit degree of conservatism that must be applied to the evaluation models?

3. This rule change would allow a 5 to 10 percent increase in the fission product inventory that could be released from any core meltdown scenario. Should this rule change explicitly prohibit any increase in approved power levels until all severe accident issues and unresolved safety issues are resolved?

4. Should the technical basis for this proposed rule change be reviewed by an independent group such as the American Physical Society?

SUMMARY OF PUBLIC COMMENTS

The comment period for the proposed rule revision and the draft regulatory guide (52 FR 11385) expired on July 1, 1987. Twenty-seven letters addressing the proposed rule were received by the expiration date, as well as nine responses to the request for comments on questions in the regulatory guide. A number of late comments were also received. These were also considered to the extent that new and substantial comments were provided.

The public comment on the proposed rule revisions have been divided into thirteen categories and are summarized in the following paragraphs. Catagories one through four represent the responses to the specific questions posed by the ACRS and Commissioner Asselstine. In general, consideration of the public comments resulted in no substantive revision to the proposed rule.

1. <u>Grandfathering of Conservative ECCS Methods of Appendix K (Question 1).</u>

Twenty-one of the commenters specifically addressed the ARCS question concerning the grandfathering of the current Appendix K

approach. Seventeen of these commenters recommended indenfinite grandfathering of the existing Appendix K evaluation models. Most cited the known conservatism as the basis of their recommendation. In addition, several commenters stated that in light of the known conservatisms not allowing continued use of existing Appendix K evaluation models would be unfairly burdensome to licensees who determine that they would not derive an economic benefit by performing realistic analysis of ECCS performance. The position of an additional commenter is unclear concerning grandfathering. The remaining commenter was not opposed to grandfathering but thought the question is premature. This commenter believes that indefinite use of existing ECCS evaluation methods should be considered when significant experience has been gained with the implementation of the new features of the rule but makes no recommendation as to what policy the Commission should pursue in the meantime.

The Commission agrees with the majority of the commenters that existing Appendix K evaluation models should be permitted indefinitely. The Commission also believes that the decision to permit continued use of such models can and should be made at this time because it believes that both methods provide adequate protection of the public health and safety. As described in the regulatory analysis, the probability of a large break is so low, that the choice of best estimate versus Appendix K has little effect on public risk. The TMI action plan calls for industry to improve their small break LOCA evaluation models to be more realistic when evaluating the more probable small break accident scenario. This has been done within the context of § 50.46 and Appendix K compliance and was entirely appropriate since small breaks are not limiting in design basis performance and a better understanding of small break behavior is a desirable safety goal from a risk perspective. Therefore, the grandfathering provision has been retained in the final rule.

2. Specification of Explicit Degree of Conservatism (Question 2). The majority of the responses to this question indicated that the proposed rule already contains conservatism in the required uncertainty evaluation.

The use of additional conservatism would be inconsistent with the objective of the rule which is to provide a realistic evaluation of plant response during a LOCA. The NRC has not included an additional explicit degree of conservatism in this rule.

3. Resolution of all Safety Issues Prior to Allowing Power Level Increases (Question 3). Some commenters pointed out that fission product inventory is not a direct function of total power, but rather it is the rate of fission product formation that is a direct function of power. Fission product inventory available for release during a core meltdown would be a function of burnup, not total power.

Actually, the inventory of fission products is a complex function of both time and power and not as simple as described by the commenters. Short lived isotopes, such as xenon and iodine, quickly reach an equilibrium inventory and total steady state inventory of these fission products is a direct function of power. Inventories of long-lived isotopes, such as strontium and cesium, are functions of total fuel burnup, as described by the commenters. Intermediate-lived isotopic inventories are complex functions of time, power, and integrated power. In an independent study, documented in chapter XII of NUREG 1230, the staff determined that the change in risk due to a 5% power increase is negligible. The arguments above do not alter the Commission's position that the increase in fission products available for release during a core meltdown caused by a 5% power increase is negligible compared to the uncertainty in fission product release. The Commission has decided not to delay the

proposed rule revision pending resolution of all unresolved safety issues or severe accident issues and therefore will proceed with this final rulemaking, as planned.

4. Independent review of Technical Basis (Question 4). Several commenters indicated that the technical basis for the proposed rule has had adequate review as the research was being performed. A number of commenters stated that it was the role of the ACRS to perform any review of the proposed rule revision because it is uniquely qualified due to its familiarity with the research.

The Commission agrees that the technical basis has had adequate review, except for the uncertainty methodology which is new and untried except for the General Electric Company's use of an uncertainty evaluation of their SAFER code. As a proof of principle and demonstration of feasibility, the ACRS and a second independent peer group has reviewed the uncertainty methodology developed by the NRC for use in quantifying the uncertainty of NRC developed thermal hydraulic transient codes. Both the ACRS and the peer group made generally favorable comments concerning the methodology; however, both groups recognized that a complete demonstration (i.e., application to small break LOCA and the reflood portion of large break LOCA) has not yet been accomplished and certain reviewers questioned whether such a demonstration could be performed successfully. The only objectives of the NRC methodology demonstration are to demonstrate feasibility, to develop an audit tool, and to provide the necessary experience to audit licensee submittals. The staff does not believe that NRC demonstration of the methodology is a prerequisite to this rulemaking. Licensees wishing to adopt the best-estimate approach permitted as a result of this rule are neither required to use this methodology nor to model their own methodologies on it. This methodology will play an important part in the best-estimate model review process. The NRC has determined through twenty years of experience that independent analysis with independent methodologies is the most effective way

to intelligently review new vendor or licensee methodologies. It is therefore appropriate that this new methodology be subjected to stringent technical scrutiny, as directed by the Commission. The NRC staff is committed to completing this demonstration by the time that it will be needed to review licensee submittals and is confident that such a demonstration will be successful. Based on the paucity of negative response concerning the technical basis for the proposed rule revision and generally favorable review of the NRC uncertainty methodology, the Commission plans no further review of the technical basis.

5. General Comments on Proposed Rule. Twenty-one commenters made comments of this nature. The majority of the comments came from the nuclear industry of which 19 expressed support of the proposed rule. The industry also strongly supports the specific ECCS rule approach proposed by the NRC. One commenter neither supported nor opposed the proposed approach. One negative comment was received from an anonymous individual within the nuclear industry who implied, without specifics, that the ECCS rule is not sound and that public comment is not a fair hearing because expert insiders would be afraid to comment.

Based on the absence of any supporting justification for the negative response and the unprecedented amount of research supporting the rule revision, the NRC does not consider this comment to be valid and has proceeded with this rulemaking with no major revisions.

One commenter suggested that fuel reload suppliers should not be required to complete full LOCA/ECCS analyses because the hydraulics are not changed by a fuel change.

Although this point is valid, the Commission believes that it is an unworkable situation to allow fuel suppliers to make use of previous analyses performed by others. It is believed that serious questions of accountability would arise in cases where

errors are discovered in evaluation models, requests are made to revise plant technical specifications, or some other questions regarding the analyses are raised. The NRC believes that shared responsibility for evaluation models would not be in the best interest of the public health and safety and therefore has not implemented the suggestion of this commenter.

The NRC received two requests for an extension of the comment period to allow time for review of NUREG-1230, which describes the research supporting the proposed rule revision.

The NRC believes the comment period was sufficient since most of the research is not new and has been extensively reviewed in the past. Both commenters were contacted and told that comments received after the comment period would be considered if time permitted. Comments from both parties were received late and were indeed considered by the NRC.

6. Reporting Requirements. Some commenters viewed the proposed reporting procedures as new requirements needing consideration in the backfit analysis while others stated that they are a major relaxation and clarification of existing reporting requirements.

The NRC position is that the reporting requirements are new in the sense that they will now appear in the Code of Federal Regulations and therefore have been considered in the backfit analysis. In practice, these reporting requirements are indeed a clarification and relaxation over the current interpretation of the existing requirements and therefore the net effect of these requirements will be to reduce the frequency for reporting and reanalysis.

A number of commenters requested that only significant errors or changes in the non-conservative direction or only those that result in exceeding the 2200°F limit be required to be reported. In addition, a number of commenters suggested that the NRC require only annual reporting of significant errors or changes.

The NRC considers a major error or change in any direction a cause for concern because it raises potential questions about the adequacy of the evaluation model as a whole. Therefore, the NRC requires the reporting of significant errors or changes, in either direction, on a timely basis so that the Commission may make a determination of the safety significance. Thus, the final rule contains no change in this requirement.

One commenter recommended that the word "immediate" be deleted from the requirement to propose steps to be taken to demonstrate compliance in the event that the criteria in § 50.46(b) are exceeded.

The Commission considers this a very serious condition in which the plant is not in compliance with the regulations and may be operating in an unsafe manner. The word "immediate" reflects this seriousness and is further defined by reference in other sections of Part 50.

Several commenters questioned the need to report minor or inconsequential errors or changes, even on an annual basis, as required in the proposed rule.

While errors or changes which result in changes in calculated peak clad temperatures of less than 50°F are not considered to be of immediate concern, the NRC requires cognizance of such changes or corrections since they constitute a deviation from what previously has been reviewed and accepted. The proposed annual reporting is believed to be a fair compromise between the burden of reporting and the Commission's need to be aware of changes and error corrections being made to evaluation models. Therefore, the annual reporting of minor errors remains in the final rule.

One commenter interpreted the use of the words "or in the application of such a model" as requiring reporting when facility

changes (already reportable under § 50.59), resulting in model input changes, occur.

The regulatory language referred to is intended to ensure that applications of models to areas not contemplated during initial review of the model do not result in errors by extending a model beyond the range that it was intended. The Commission does not believe that further clarification of this requirement is necessary and has not done so in the final rule.

Several commenters requested a further relaxation of the reporting requirement by changing the definition of significant code errors from 50°F to 100°F.

While justification for the 50°F criteria is largely judgmental, the NRC believes that it is sufficiently large to screen the code error corrections and changes which have little safety significance while providing a mechanism for timely reporting of more serious errors and changes. Since 50°F is a threshold for reporting and no further action is required pending NRC determination of safety significance, the Commission has retained this criteria in the final rule.

One commenter requested consideration for allowing that the cumulative effect of several errors and corrections be applied towards the 50°F threshold.

The requirement, which states that the 50°F criteria applies to the sum of the absolute magnitudes of temperature changes from numerous error corrections or model changes was formulated specifically because the Commission requires knowledge of serious deficiencies in evaluation models in use by licensees. Allowing errors or corrections which offset one another to relieve a licensee of the thirty-day reporting requirement, would be counter to this objective. If this recommendation were accepted, two errors or changes, having a large impact on the

calculated peak cladding temperature but in the opposite direction, would not be reportable if the net magnitude of their difference was less than 50°F. For this reason, and the fact that no further action (beyond reporting within thirty days) is required, the Commission retained this requirement in the final rule.

7. Continued Use of Dougall-Rohsenow. Five comments that addressed this aspect of the proposed rule were received. One commenter believed that this correlation should not be permitted without further verification and should be phased out. Other commenters supported continued use of the correlation subject to the provisions of the proposed rule.

   The NRC position is that no safety concern is created by continued use of the correlation, as long as the evaluation model is overall conservative. Therefore, the Commission can not justify the burden of requiring licensees to modify their evaluation models and to perform reanalysis. As discussed in SECY 83-472, current evaluation models contain more conservatisms than just those required by Appendix K. However, error corrections or changes could alter the conservatism of the model. Therefore, the Commission believes that it is necessary to ensure continued overall conservatism in the evaluation models as a basis for continued use of the correlation. Therefore, the final rule does not modify this requirement except for the correction of a typographical error identified by one commenter.

8. Uncertainty Evaluation. The comments received on the uncertainty evaluation support the proposed rule, particularly the flexibility provided by a non-prescriptive requirement. Therefore, the Commission is publishing the final rule without modification of this requirement.

9. Acceptance Criteria. The three comments received on this topic were all supportive of the existing criteria, as contained in § 50.46(b), and thus the Commission did not give consideration to altering them in the final rule.

10. Cladding Materials. Three commenters requested that the Commission consider broadening the language of the rule to allow the use of a range of zirconium based alloys for cladding material.

The Commission believes that this modification is beyond the scope of the current rule revision and should be considered in a separate rulemaking action in which it would receive appropriate public review and comment prior to implementation. In addition, zircaloy cladding material is specified in other portions of the Code of Federal Regulation, such as § 50.44. Making a change of this type is more suitable in a broader regulatory context. Therefore, the Commission is not broadening the definition of cladding materials within this rulemaking.

11. Other Suggested Expansions to Rule Scope. One commenter believes that hydraulic loads occurring during a LOCA could cause steam generator tubes to rupture and that the NRC should resolve steam generator tube integrity safety issues prior to publishing this rule.

Steam generator tubes are designed to withstand LOCA loads at allowed thinning, and there is no evidence to contradict this. If anything, the problem would be with inspection techniques to detect the actual tube thinning and whether there is an unacceptably high probability that a tube rupture during a LOCA due to tube thinning is in excess of the design basis. However, the risk from LOCA with concurrent tube rupture will not be greatly affected by the proposed rule change. As a result of the commenter's concerns, this issue has been assigned as a generic

issue (GI-141) to be prioritized by the NRC staff. The results of the prioritization process will determine if further action is required.

A second commenter believes that the ECCS rule does not adequately address a plant's long term decay heat removal capability, and recommends a "short/long term integrative analysis approach." Both the existing requirements and the proposed rule contain the requirement to provide for long term cooling subsequent to a LOCA. Small increases in power that may result from the proposed rule should not greatly change decay heat removal requirements following a LOCA or any other accident or transient. Thus, the issue of decay heat removal is not materially impacted by this rulemaking. Moreover, any proposed increase in power resulting from this rule promulgation would be approved only after the licensee demonstrates that decay heat removal capacities remain adequate. The Commission is planning no further action with regard to this issue.

12. Acceptability of Models Approved Under SECY-83-472. One commenter requests that the rule language be modified to state explicitly that ECCS evaluation models that have been previously approved under SECY-83-472 continue to be acceptable under this rule.

SECY 83-472 provides an alternative, acceptable method for developing ECCS evaluation models. Licensees were still required, however, to demonstrate that evaluation models developed using the SECY-83-472 approach complied with the requirements of Appendix K to Part 50. This final rule explicitly finds that ECCS evaluation models, which have been previously approved as satisfying the requirements of Appendix K, remain acceptable. Therefore, the Commission sees no need for further clarification of this issue.

13. Comments Received After Comment Period. Six letters commenting on the proposed rule were received subsequent to the end of the comment period. The Commission considered these comments to the extent that the comments provided substantive information not previously considered.

One commenter believes that the proposed § 50.46(a)(2) expands the discretion of the Director of the Office of Nuclear Reactor Regulation (NRR) by allowing imposition of immediate effective restrictions on reactor operation without a prior determination that such action is required to protect the public health, safety, or interest. NRC's intent is not to alter the responsibilities of the Director of NRR but to simply retain the description of the scope of the authority that is currently found in § 50.46(a)(1)(v). Furthermore, the provisions of § 50.46(a)(2) do not specify the procedure to be followed by the Director of NRR. These procedures are set out in Part 2 and remain unchanged by this rulemaking.

One commenter believes that the rule is illegal because it is based solely on cost savings considerations and that there is nothing wrong with large conservatisms.

The Commission disagrees with this assessment. Safety factors are required to protect the health and safety of the public when uncertainties in plant response exist. As these uncertainties are reduced, it is appropriate to modify these safety factors to provide more realistic evaluation of actual plant response. The large conservatisms of Appendix K served the public well in 1974 when there was great uncertainty in ECCS performance. However, these conservatisms are now known to be very large, and there is no need to "over regulate" by maintaining this unnecessary margin. This type of activity can often result in the expenditure of resources that would be better spent improving safety in other areas. The benefits to safety, while difficult to quantify, are believed to be substantial. While cost savings may have been one factor resulting in the rule change, the

Commission believes that the conservatisms contained in the acceptance criteria themselves, as well as those required in the uncertainty evaluation required in this rule, are adequate to protect the health and safety of the public.

This commenter also cites portions of the 1975 General Electric Company's Nuclear Reactor Study (Reed Report), which claims that there is a lack of understanding of phenomena and small safety margins.

Many of the conclusions of the "Reed Report" were valid in 1975 when it was written and due to this fact it was difficult to show that sufficient safety margins existed. Most of the research discussed in NUREG-1230 has been conducted since the "Reed Report" was written and has resulted in significant improvement in understanding LOCA phenomena. We now know that significant margin to the ECCS acceptance criteria exists, particularly for the BWR/6 which was of concern in the "Reed Report." The contents of this report have been reviewed by the Commission on several occasions, most recently in NUREG-1285, and the finding has been made that no new significant safety issues are identified. For these reasons, the NRC is proceeding with this rulemaking, as proposed.

The same commenter also recommends that credit for ECCS margins be taken in the Individual Plant Examinations (IPE) and not through generic rulemaking.

The Commission agrees that plant specific differences may justify the application of different margins and that these may be addressed through Individual Plant Examinations. However, the requirement for licensees to evaluate ECCS performance and meet the acceptance criteria specified in 10 CFR 50.46(b) is generic. The Commission believes that margins that may be reduced due to a better understanding of a reactor's response to a LOCA should be applied through a generic rulemaking action because it allows a broad range of technical review of the issues, enhances public

participation in the process, and provides a complete public record. Therefore, the Commission has decided to proceed with the rulemaking as planned.

Finally, this commenter questions the experimental basis for this rule because full-scale ECCS bypass data is not yet available.

The 2D/3D tests which will provide this important data represent a small portion of the total research upon which this rule relies. Significant research on ECCS bypass has already been completed in small scale vessels and the full-scale work is required only to confirm the smaller scale results and quantify any uncertainty due to scale effects. One full-scale ECCS bypass test has already been completed under the 2D/3D program which showed that more margin exists than expected from the small scale tests. Completion of the full-scale tests only affects the uncertainties in the calculations, and reduces them. Uncertainties must be addressed by licensees in any analysis under the revised rule whether 2D/3D results are available or not. The Commission concludes that there is no need to delay the final rule, while awaiting these data.

SUMMARY OF RULE CHANGES

Section 50.46 Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors:

Section 50.46(a)(1) is amended and redesignated § 50.46(a)(1)(i) to delete the requirement that the features of Section I of Appendix K to Part 50 be used to develop the evaluation model. This section now requires that an acceptable evaluation model have sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a LOCA. The NRC expects that the analytical technique will, to the extent practicable, utilize realistic methods and be based upon applicable experimental data. The amended rule also requires that the

uncertainty of the calculation be estimated and accounted for when comparing the results of the calculation to the temperature limits and other criteria of § 50.46(b) so that there is a high probability that the criteria would not be exceeded. The Commission expects the realistic evaluation model to retain a degree of conservatism consistent with the uncertainty of the calculation. The final rule does not specifically prescribe the analytical methods or uncertainty evaluation techniques to be used. However, guidance has been provided in the form of a Regulatory Guide[1]. As discussed in SECY-83-472, the NRC has, in the past, found acceptable a method for estimating the uncertainty that was judged to be at least at the 95% probability level. This probability level of 95% is considered adequate to meet the high level of probability required by the rule. It is also recognized that the probability cannot be determined using totally rigorous mathematical methods due to the complexity of the calculations. However, the NRC requires that any simplifying assumptions be stated so that the Commission may evaluate them to ensure that they are reasonable. The NRC has independently developed and exercised a methodology to estimate the uncertainty associated with its own thermal-hydraulic safety codes. This methodology is described in the "Compendium of ECCS Research."[2] This document also provides reference to the large body of relevant thermal-hydraulic research, documents NRC studies on the effects of reactor power increases on risk, and provides background information on the ECCS rule. While this method has not been reviewed for acceptability from the standpoint of safety licensing, it may provide additional guidance on how the uncertainty may be quantified. In addition to providing guidance to industry, this work was undertaken to provide a proof of principle and a tool to independently audit industry submittals. Appendix K, Section II, "Required Documentation," remains generally applicable, with only minor revisions made to be consistent with the amended rule.

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[1]Regulatory Guide, "Best Estimate Calculations of Emergency Core Cooling Systems Performance," was issued to all licensees.

[2]"Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, TBP.

A new paragraph(ii) has been added to § 50.46(a)(1) to allow the features of Section I of Appendix K to be used in evaluation models as an alternative to performing the uncertainty evaluation specified in the amended § 50.46(a)(1)(i). This method would remain acceptable because Appendix K is conservative with respect to the realistic method proposed in the amended § 50.46(a)(1)(i). This would allow both current and future applicants and licensees to use existing evaluation models if they did not need or desire relief from current operating restrictions.

In § 50.46, paragraphs (a)(2) and (3) have been revised to eliminate portions of those paragraphs concerned with historical implementation of the current rule. These provisions have been replaced as described in the following paragraphs:

Section 50.46(a)(2) has been revised to indicate that restrictions on reactor operation may be imposed by the Director of Nuclear Reactor Regulation, if the ECCS cooling performance evaluations are not consistent with the requirements of §§ 50.46(a)(1)(i) and (ii). This section has been added to retain similar requirements that have been deleted from § 50.46(a)(1)(i) by this rule revision. This section does not specify the procedures to be followed by the Director. These procedures are found in Part 2 and are unchanged by this rulemaking.

The current rule contains no explicit requirements concerning reporting and reanalysis when errors in evaluation models are discovered or changes are made to evaluation models. However, current practice has required reporting of errors and changes and reanalyses with the revised evaluation models. This final rule explicitly sets forth requirements to be followed in the event of errors or changes. The definition of a significant change is currently taken from Appendix K, Section II.1.b which defines a significant change as one which changes calculated cladding temperature by more than 20°F.

The revised § 50.46(a)(3) states specific requirements for reporting and reanalyses when errors in evaluation models are discovered or changes are made to evaluation models. It requires that all changes or errors in approved evaluation models be reported at least annually and does not require any further action by the licensee until the error is reported. Thereafter, although reanalysis is not required solely because of such minor error, any subsequent calculated evaluation of ECCS performance requires use of a model with such error, and any prior errors, corrected. The NRC needs to be apprised of even minor errors or changes in order to ensure that they agree with the applicant's or licensee's assessment of the significance of the error or change and to maintain cognizance of modifications made subsequent to NRC review of the evaluation model. Past experience has shown that many errors or changes to evaluation models are very minor and the burden of immediate reporting cannot be justified for these minor errors because they do not affect the immediate safety or operation of the plant. The NRC therefore requires periodic reporting to satisfy NRC's need to be apprised of changes or errors without imposing an unnecessary burden on the applicant or licensee. This report is to be filed within one year of discovery of the error and must be reported each year thereafter until a revised evaluation model or a revised evaluation correcting minor errors is approved by the NRC staff.

Significant errors require more timely attention since they may be important to the safe operation of the plant and raise questions as to the adequacy of the overall evaluation model. This final rule defines a significant error or change as one which results in a calculated peak fuel cladding temperature different by more than 50°F, or an accumulation of errors and changes such that the sum of the absolute magnitude of the temperature changes is greater than 50°F. More timely reporting (30 days) is required for significant errors or changes. This definition of a significant change is based on NRC's judgement concerning the importance of errors and changes typically reported to the NRC in the past. This final rule revision also allows the NRC to determine the schedule for reanalysis based on the

importance to safety relative to other applicant or licensee requirements. Errors or changes that result in the calculated plant performance exceeding any of the criteria of § 50.46(b) mean that the plant is not operating within the requirements of the regulations and require immediate reporting as required by § 50.55(e), § 50.72 and § 50.73 and immediate steps to bring the plant into compliance with § 50.46.

Appendix K ECCS Evaluation Models:

Amendments have been made to Appendix K, Section I.C.5.b, to modify the post-CHF heat transfer correlations listed as acceptable. The "McDonough" reference has been replaced with a more recent paper by the same authors entitled "An Experimental Study of Partial Film Boiling Region With Water at Elevated Pressures in a Round Vertical Tube" which is more generally available and which includes additional data.

The heat transfer correlation of Dougall and Rohsenow, listed as an acceptable heat transfer correlation in Appendix K, paragraph I.C.5.b, has been removed, since research performed since Appendix K was written has shown that this correlation overpredicts heat transfer coefficients under certain conditions and therefore can produce nonconservative results. A number of applicants and licensees currently use the Dougall-Rohsenow correlation in approved evaluation models. The NRC has concluded that the continued use of this correlation can be allowed. This is appropriate (even though parts of the approved evaluation model, Dougall-Rohsenow, are known to be nonconservative) because the existing evaluation models are known to contain a large degree of overall conservatism even while using the Dougall-Rohsenow correlation. This large overall conservatism has been demonstrated through comparisons between evaluation model calculations and calculations using NRC's best estimate computer codes. Thus requiring that the applicants and licensees remove the Dougall-Rohsenow correlation from their current evaluation models cannot be justified as necessary to maintain safety. The stipulation

that the Dougall-Rohsenow correlation will cease to be acceptable for previously approved evaluation models applies only when changes to the model are made which reduce the calculated peak clad temperature by 50°F or more. However, the requirement to report any changes or culmination of changes, such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50°F, still applies.

A new Section I.C.5.c has been added to Appendix K to state the Commission's requirements regarding continued use of the Dougall-Rohsenow correlation in existing evaluation models. Evaluation models which make use of the Dougall-Rohsenow correlation and have been approved prior to the effective date of this rule may continue to use this correlation as long as no changes are made to the evaluation model which significantly reduce the current overall conservatism of the evaluation model. If the applicant or licensee submits proposed changes to an approved evaluation model, or submits corrections to errors in the evaluation model which significantly reduce the existing overall conservatism of the model, continued use of the Dougall-Rohsenow correlation under conditions where nonconservative heat transfer coefficients result would no longer be acceptable. For this purpose, a significant reduction in overall conservatism has been defined as a "net" reduction in calculated peak clad temperature of at least 50°F from that which would have been calculated using existing evaluation models. A reduction in calculated peak clad temperature could potentially result in an increase in the actual allowed peak power in the plant. An increase in allowed plant peak power with a known nonconservatism in the analysis would be unacceptable. This definition of a significant reduction in overall conservatism is based on a judgement regarding the size of the existing overall conservatism in evaluation model calculations relative to the conservatism required to account for overall uncertainties in the calculations.

Appendix K, Section II.1.b, has been removed since this requirement has been clarified in the amended § 50.46(a)(3).

Likewise, Appendix K, Section II.5, has been amended to account for the fact that not all evaluation models will be required to use the features of Appendix K, Section I. These minor changes to Appendix K do not affect any existing approved evaluation models since the changes are either "housekeeping" in nature or are changes to "acceptable features," not "required features."

AVAILABILITY OF DOCUMENTS

1. Copies of NUREGs 1230 and 1285 may be purchased from the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, D.C. 20013-7082. Copies are also available from the National Technical Information Service 5285 Port Royal Road, Springfield, VA 22161. A copy is also available for public inspection and/or copying at the NRC Public Document Room, 1717 H Street NW., Washington, DC 20555.

2. Copies of SECY-83-472, an information report entitled "Emergency Core Cooling Systems Analysis Methods," dated November 17, 1983, is available for inspection and copying at the NRC Public Documents Room 1717 H Street NW., Washington, DC 20555. Single copies of this report may be obtained by writing (give name and address of the contact).

3. Regulatory Guide, "Best Estimate Calculations of Emergency Core Cooling Systems Performance" may be obtained by writing to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

4. The Paraphrased Summary of Public Comments on ECCS Rule is available for public inspection at the NRC Public Documents Room, 1717 H Street NW., Washington, DC 20555.

FINDING OF NO SIGNIFICANT ENVIRONMENTAL IMPACT: AVAILABILITY

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in

Subpart A of 10 CFR Part 51, that this rule is not a major Federal action significantly affecting the quality of the human environment and therefore an environmental impact statement is not required. The primary effect of the rule is to allow an increase in the peak local power in the reactor. This could be used either to tailor the power shape within the reactor or to increase the total power. Changing the power shape without changing the total power has a negligible effect on the environmental impact. The total power could also be increased, but is expected to be increased by no more than about 5% due to hardware limitations in existing plants. This 5% power increase is not expected to cause difficulty in meeting the existing environmental limits. The only change in non-radiological waste will be an increase in waste heat rejection commensurate with any increase in power. For stations operating with an open (once through) cooling system, this additional heat will be directed to a surface water body. Discharge of this heat is regulated under the Clean Water Act administered by the U.S. Environmental Protection Agency (EPA) or designated state agencies. It is not intended that NRC approval of increased power level affects in any way the responsibility of the licensee to comply with the requirements of the Clean Water Act. The environmental assessment and finding of no significant impact on which this determination is based are available for inspection at the NRC Public Document Room, 1717 H Street NW, Washington, DC. Single copies of the environmental assessment and the finding of no significant impact are available from L. M. Shotkin, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington DC. 20555, telephone (301) 492-3530.

PAPERWORK REDUCTION ACT STATEMENT

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). These reporting requirements were approved by the Office of Management and Budget (Approval Number 3150-0011).

REGULATORY ANALYSIS

The Commission has prepared a regulatory analysis for this final regulation which examines the costs and benefits of the alternatives considered and is available for inspection and copying at the NRC Public Document Room, 1717 H Street NW, Washington, DC. Single copies of the analysis may be obtained from L. M. Shotkin, Office of Nuclear Regulatory Research, Washington, DC. 20555, telephone (301) 492-3530.

REGULATORY FLEXIBILITY CERTIFICATION

As required by the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission certifies that this rule will not have a significant economic impact upon a substantial number of small entities. This rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration in 13 CFR Part 121. Since these companies are dominant in their service areas, this rule does not fall within the purview of the Act.

BACKFIT ANALYSIS

Except for the reporting requirements, a backfit analysis is not required by 10 CFR 50.109 because the rule does not require applicants or licensees to make a change but only offers additional options. Nonetheless, the factors in 10 CFR 50.109(c) have been analyzed for the entire rule.

1. <u>Statement of the specific objectives that the backfit is designed to achieve</u>.

   The objective of the rule is to modify 10 CFR 50.46 and Appendix K to permit the use of realistic ECCS evaluation models. More realistic estimates of ECCS performance, based on the improved knowledge gained from recent research on ECCS

performance, may remove unnecessary operating restrictions. Also experience with the previous version of § 50.46 has demonstrated that a clearer definition of reporting requirements for changes and errors is very desirable.

2. General description of the activity that would be required by the licensee or applicant in order to complete the backfit.

The amendment allows alternative methods to be used to demonstrate that the ECCS would protect the nuclear reactor core during a postulated design basis loss-of-coolant accident (LOCA). While continuing to allow the use of current Appendix K methods and requirements, the rule also allows the use of more recent information and knowledge currently available to demonstrate that the ECCS would perform its safety function during a LOCA. If an applicant or licensee elects to use a new realistic model they will be required to provide sufficient supporting justification to validate the model and include comparisons to experimental data and estimates of uncertainty. In accounting for the uncertainty, the analysis would have to show, with a high level of probability, that the ECCS performance criteria are not exceeded. Whether or not a licensee or applicant chooses to use realistic analysis, complete with an uncertainty analysis, each licensee must comply with the requirement to report changes to their evaluation models (i.e., less than 50°F change in calculated peak cladding temperature) annually to the NRC. In addition, significant changes (those which have a greater than 50°F change in calculated peak cladding temperature) have to be reported within 30 days.

3. Potential change in risk to the public from the accidental offsite release of radioactive materials.

The rule could result in increased local power within the reactor core and possibly increases in total power. Power increases on the order of 5 will have an insignificant effect on

risk. One effect of increased power could be to increase the fission product inventory. A five percent power increase would result in a less than five percent increase in fission products. Thus, less than five percent more fission products might be released during core melt scenarios and potentially released to the environment during severe accidents.

The rule still requires that fuel rod peak cladding temperature (PCT) remain below 2200°F. Reactors choosing to increase power by about five percent will be operating with less margin between the PCT and the 2200°F limit than previously. The increased risk represented by this decrease in margin and increase in fission product inventory is negligible and falls within the uncertainties of PRA risk estimates. In addition, other safety limits, such as departure from nucleate boiling (DNB), and operational limits, such as turbine design, will limit the amount of margin reduction permitted under the rule. The rule could also potentially reduce the risk from pressurized thermal shock by allowing the reactor to be operated in a manner which reduces the neutron fluence to the vessel.

4. <u>Potential impact on radiological exposure to facility employees</u>.

Since the primary effect of the rule involves the calculational methods to be used in determining the ECCS cooling performance, it is expected that there will be an insignificant impact on the radiological exposure to facility employees. Because of the reduced LOCA restrictions resulting from the new calculations it is possible for the plant to achieve more efficient operation and improved fuel utilization with improved maneuvering capabilities. As a result, it is conceivable that there could be a reduction in radiological exposure if the fuel reloads can be reduced. This effect is not expected to be very significant.

5. Installation and continuing costs associated with the backfit, including the cost of facility down times or the cost of construction delay.

LOCA considerations resulting from the present rule are restricting the optimum production of nuclear electric power in some plants. These restrictions can be placed into the following three categories:

(1) Maximum plant operating power,
(2) Operational flexibility and operational efficiency of the plant, and
(3) Availability of manpower to work on other activities.

The effect of the rule will vary from plant to plant. Some plants may realize savings of several million dollars per year in fuel and operating costs. Significantly greater economic benefit would be realized by plants able to increase total power as a result of this final rule. The regulatory analysis cited above indicates that the total present value of the energy replacement cost savings for a five percent power upgrade would vary between 18 and 127 million dollars depending on the plant. Additional information concerning these potential cost savings are included in the regulatory analysis.

The costs associated with the new reporting requirements are deemed to be minimal. Although the existing Appendix K has no official reporting requirements, paragraph II.1.b was interpreted by the staff to require a reanalysis and report to NRC when significant changes are made which change the peak cladding temperature by more then 20°F. Therefore, this rule change, by changing the definition of significant changes to 50°F, is actually a relaxation of current practices. The annual reporting of changes that are not significant is not viewed by the NRC as a major burden since no other action is required.

6. The potential safety impact of changes in plant or operational complexity including the effect on other proposed and existing regulatory requirements.

There are safety benefits derivable from alternative fuel management schemes that could be utilized. The higher power peaking factors that would be allowed with the final rule provide greater flexibility for fuel designers when attempting to reduce neutron flux at the vessel wall. This can result in a corresponding reduction in risk from pressurized thermal shock.

The reduced cladding temperatures that would be calculated under the revised rule offers the possibility of other design and operational changes that could result from the lower calculated temperatures. ECCS equipment numbers, sizes or surveillance requirements might be reduced and still meet the ECCS design criteria (if not required to meet other licensing requirements). Another option may be to increase the diesel/generator start time duration.

In summary, the effect of this rule on safety would have both potential positive and negative aspects. The potential for reduction of ECC system capability in existing or new plants is present. However, several positive aspects may also be realized under the final rule. The net effect on safety would be plant specific. However, the probability of a large break LOCA is so low that the choice of best estimate versus Appendix K would have little effect on public risk.

7. The estimated resource burden on the NRC associated with the proposed backfit; and the availability of such resources.

The major staff resources required under the final rule are to review the realistic models and uncertainty analysis required by the revised ECCS Rule. Based on previous experience with the

General Electric Company's SAFER model and the learning that has resulted from these efforts, it is estimated that approximately one staff year would be required to review each generic model submitted. There are four major reactor vendors (GE already has a revised evaluation model approved under the existing Appendix K for both jet pump and non-jet pump plants and may update their methodology under this new rule) and several fuel suppliers and utilities which perform their own analyses and potentially might submit generic models for review. However, it is expected that only 3 or 4 generic models would be submitted since not all plants would benefit from this rule. Thus, about 3-4 staff years would be required to review the expected generic models. Once a generic model is approved, the plant specific review is very short. In addition, several vendors are currently planning to submit realistic models in conjunction with the use of SECY-83-472. Therefore, staff resources would be expended to review these models in any event. Since these models would not change as a result of the revised ECCS rule, there should be no net increase in resources required over that already planned to be expended. In summary, while it is difficult to estimate accurately, it is expected that the rule change will have a small overall impact on NRC resources.

8. <u>The potential impact of differences in facility type, design or age on the relevancy and practicality of the backfit</u>.

The degree to which the rule would affect a particular plant depends on how limited the plant is by the LOCA restrictions. General Electric Company (GE) plants do tend to be limited in operation by LOCA restrictions and would benefit from relief from LOCA restrictions. However, this relief is already available for most GE plants through the recently approved SAFER evaluation model. Any additional relief due to a rule change would be of little further benefit. Westinghouse (<u>W</u>) plants would appear to directly benefit from relaxation of LOCA limits. <u>W</u> plants represent the largest number of plants, with 47 plants operating

and 10 additional plants being constructed. W indicates that most of these plants are limited by LOCA considerations. The potential benefit for plants of B&W and CE design is uncertain at this time.

9. Whether the proposed backfit is interim or final and if interim, the justification for imposing the proposed backfit on an interim basis.

The rule, when made effective, will be in final form and not interim form. It will continue to permit the performance of ECCS cooling calculations using either realistic models or models in accord with Appendix K.

LIST OF SUBJECTS IN 10 CFR PART 50

Antitrust, Classified information, Fire prevention, Incorporation by reference, Intergovernmental relations, Nuclear power plants and reactors, Penalty, Radiation protection, Reactor siting criteria, Reporting and Recordkeeping requirements.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C 552 and 553, the NRC is adopting the following amendments to 10 CFR Part 50.

PART 50-DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for Part 50 continues to read as follows:

AUTHORITY: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246, (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185, 68 Stat. 936, 955, as amended (42 U.S.C. 2131, 2235); sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.23, 50.35, 50.55, 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a, and Appendix Q also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073, (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80-50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Section 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273), §§ 50.10(a), (b), and (c), 50.44, 50.46, 50.48, 50.54, and 50.80(a) are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); §§ 50.10(b) and (c) and 50.54 are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and §§ 50.9, 50.55(e), 50.59(b), 50.70, 50.71, 50.72, 50.73, and 50.78 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

2. In § 50.46, paragraph(a) is revised to read as follows:

§ 50.46 Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.

(a)(1)(i) [~~Except as provided in paragraph (a)(2) and (3) of this section,~~] Each boiling and pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Zircaloy cladding must be provided with an emergency core cooling system (ECCS) that must be designed such that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling

performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. [~~Appendix K, ECCS Evaluation Models, sets forth certain required and acceptable features of evaluation models.~~] Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II, Required Documentation, sets forth the documentation requirements for each evaluation model. [~~Conformance with the criteria set forth in paragraph (b) of this section with ECCS cooling performance calculated in accordance with an acceptable evaluation model, may require that restrictions be imposed on reactor operation.~~]

(ii) Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of Appendix K ECCS Evaluation Models.

(2) The Director of Nuclear Reactor Regulation may impose restrictions on reactor operation if it is found that the evaluations of ECCS cooling performance submitted are not consistent with paragraphs (a)(1)(i) and (ii) of this section.

(3)(i) Each applicant for or holder of an operating license or construction permit shall estimate the effect of any change to or error in an acceptable evaluation model or in the application of such

a model to determine if the change or error is significant. For this purpose, a significant change or error is one which results in a calculated peak fuel cladding temperature different by more than 50°F from the temperature calculated for the limiting transient using the last acceptable model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50°F.

(ii) For each change to or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation, the applicant or licensee shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in § 50.4. If the change or error is significant, the applicant or licensee shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with § 50.46 requirements. This schedule may be developed using an integrated scheduling system previously approved for the facility by the NRC. For those facilities not using an NRC approved integrated scheduling system, a schedule will be established by the NRC staff within 60 days of receipt of the proposed schedule. Any change or error correction that results in a calculated ECCS performance that does not conform to the criteria set forth in paragraph (b) of this section is a reportable event as described in §§ 50.55(e), 50.72 and § 50.73. The affected applicant or licensee shall propose immediate steps to demonstrate compliance or bring plant design or operation into compliance with § 50.46 requirements.

* * * * *

3. In 10 CFR Part 50 Appendix K, paragraph II.1.b is deleted, paragraph II.1.c is redesignated II.1.b, the text of paragraph I.C.5.b and paragraphs II.1.b and II.5 are revised, and a new section I.C.5.c is added to read as follows:

* * * * *

I. REQUIRED AND ACCEPTABLE FEATURES OF THE EVALUATION MODELS***

C. Blowdown Phenomena***

5. Post-CHF Heat Transfer Correlations.***

b. The Groeneveld flow film boiling correlation (equation 5.7 of D. C. Groeneveld, "An Investigation of Heat Transfer in the Liquid Deficient Regime," AECL-3281, revised December 1969)[~~,-the-Dougall-Rohsenow-flow-film-boiling-correlation-(R.-S.-Dougall-and-W.-M.-Rohsenow,-"Film-Boiling-on-the-Inside-of-Vertical-Tubes-with-Upward Flow-of-Fluid-at-Low-Qualities,"-MIT-Report-Number-9079-26,-Cambridge, Massachusetts,-September-1963),~~] and the Westinghouse correlation of steady-state transition boiling ("Proprietary Redirect/Rebuttal Testimony of Westinghouse Electric Corporation," USNRC Docket RM-50-1, page 25-1, October 26, 1972) are acceptable for use in the post-CHF boiling regimes. In addition, the transition boiling correlation of McDonough, Milich, and King (J. B. McDonough, W. Milich, E. C. King, [~~"Partial-Film-Boiling-with-Water-at-2000-psig-in-a-Round-Vertical Tube,"-MSA-Research-Corp.,-Technical-Report-62-(NP-6976),-(1958)~~] "An Experimental Study of Partial Film Boiling Region with Water at Elevated Pressures in a Round Vertical Tube," Chemical Engineering Progress Symposium Series, Vol. 57, No. 32, pages 197-208, (1961) is suitable for use between nucleate and film boiling. Use of all these correlations is [~~shall-be~~] restricted as follows:

* * * * *

c. Evaluation models approved after ( ) which make use of the Dougall-Rohsenow flow film boiling correlation (R. S. Dougall and W. M. Rohsenow, "Film Boiling on the Inside of Vertical Tubes with Upward Flow of Fluid at Low Qualities, MIT Report Number 9079 26, Cambridge, Massachusetts, September 1963) may not use this correlation under conditions where nonconservative predictions of heat transfer result. Evaluation models that make use of the Dougall-Rohsenow correlation and were approved prior to ( ) continue to be acceptable until a change is made to, or an error is

corrected in, the evaluation model that results in a significant reduction in the overall conservatism in the evaluation model. At that time continued use of the Dougall-Rohsenow correlation under conditions where nonconservative predictions of heat transfer result will no longer be acceptable. For this purpose, a significant reduction in the overall conservatism in the evaluation model would be a reduction in the calculated peak fuel cladding temperature of at least 50°F from that which would have been calculated on ( ) due either to individual changes or error corrections or the net effect of an accumulation of changes or error corrections.

II. REQUIRED DOCUMENTATION

1.a. * * *

b. A complete listing of each computer program, in the same form as used in the evaluation model, must be furnished to the Nuclear Regulatory Commission upon request.

* * * * *

5. General Standards for Acceptability - Elements of evaluation models reviewed will include technical adequacy of the calculational methods, including : for models covered by § 50.46(a)(1)(ii), compliance with required features of Section I of this Appendix K [~~and provision of a level of safety and margin of conservatism comparable to other acceptable evaluation models, taking into account significant differences in the reactors to which they apply.~~] ; and, for models covered by § 50.46(a)(1)(i), assurance of a high level of probability that the performance criteria of § 50.46(b) would not be exceeded.

Dated at Rockville, MD this _____ day of _____, 1988.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,
Secretary of the Commission.

ENCLOSURE F

REGULATORY GUIDE

BEST ESTIMATE CALCULATIONS OF EMERGENCY CORE COOLING SYSTEMS PERFORMANCE

BEST ESTIMATE CALCULATIONS
OF EMERGENCY CORE COOLING SYSTEMS PERFORMANCE

A. INTRODUCTION

Paragraph 50.46(a)(1) of 10 CFR Part 50, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," requires that light water nuclear reactors fueled with uranium oxide pellets within cylindrical zircaloy cladding be provided with emergency core cooling systems (ECCS) that are designed in such a way that their calculated core cooling performance after a postulated loss-of-coolant accident (LOCA) conforms to certain criteria specified in Paragraph (b) § 50.46. Paragraph (b)(1) requires that the calculated maximum temperature of fuel element cladding not be greater than 2200°F. In addition, Paragraphs (b)(2) through (b)(5) of § 50.46, which contain required limits for calculated maximum cladding oxidation and maximum hydrogen generation, require that calculated changes in core geometry remain amenable to cooling and that long term decay heat removal be provided.

In 1987, the NRC staff proposed to amend the requirements of § 50.46 and Appendix K, "ECCS Evaluation Models," so that the regulations would reflect the improved understanding of ECCS performance during reactor transients that had been obtained through the extensive research performed since the promulgation of the original requirements in January of 1974. By this revision, paragraph § 50.46(a)(1) now permits licensees or applicants to use either Appendix K features or a realistic[1] evaluation model. These realistic evaluation models[2] must

[1] For the purpose of this guide, the terms "best-estimate" and "realistic" have the same meaning. Both terms are used to indicate that the techniques attempt to predict realistic reactor system thermal-hydraulic response. Best estimate is not used in a statistical sense in this guide.

[2] The term "evaluation model" refers to a nuclear plant system computer code or any other analysis tool designed to predict the aggregate behavior of a loss-of coolant accident. It can be either best-estimate or conservative and may contain many correlations or models.

include sufficient supporting justification to demonstrate that the analytic techniques realistically describe the behavior of the reactor system during a postulated loss-of-coolant accident. 10 CFR Part 50.46(a)(1) also requires that the uncertainty in the realistic evaluation model be quantified and considered when comparing the results of the calculations with the applicable limits in § 50.46(b) so that there is a high probability that the criteria will not be exceeded.

This regulatory guide describes models,[3] correlations,[4] data, model evaluation procedures, and methods that are acceptable to the NRC staff for meeting the requirements for a realistic or best-estimate calculation of ECCS performance during a loss-of-coolant accident and for estimating the uncertainty in that calculation. Methods for including the uncertainty in the comparisons of the calculational results to the criteria of paragraph 50.46(b), in order to meet the requirement that there be a high probability that the criteria would not be exceeded, are also described in this regulatory guide. Paragraph (a) of § 50.46 also permits licensees to use evaluation models developed in conformance with Appendix K.

Other models, data, model evaluation procedures, and methods will be considered if they are supported by appropriate experimental data and technical justification. Any models, data, model evaluation procedures, and methods listed as acceptable in this regulatory guide are acceptable in a generic sense only and would still have to be justified to the NRC staff as being appropriately applied and applicable for particular plant applications.

The Appendix to this regulatory guide lists models, correlations, data, and model evaluation procedures that the NRC staff considers

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[3]The term "model" refers to a set of equations derived from fundamental physical laws and designed to predict the details of a specific phenomenon.

[4]The term "correlation" refers to an equation having empirically determined constants such that it can predict some details of a specific phenomena for a limited range of conditions.

acceptable for realistic calculations of ECCS performance. It also provides a description of the acceptable features of best-estimate computer codes and acceptable methods for determining the uncertainty in the calculations.

This regulatory guide contains no new information collection requirements and, therefore, is not subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.).

Any information collection activities discussed in this draft regulatory guide are contained as requirements in 10 CFR Part 50, which provides the regulatory basis for this guide. The information collection requirements in 10 CFR Part 50 have been cleared under OMB Clearance No. 3150-0011.

B. DISCUSSION

The criteria set forth in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems in Light Water Nuclear Power Reactors," and the calculational methods specified in Appendix K were promulgated in January 1974 after extensive rulemaking hearings and were based on the understanding of ECCS performance available at that time. In the years following the promulgation of those rules, the NRC, the nuclear industry, and several foreign institutions have conducted an extensive program of research that has greatly improved the understanding of ECCS performance during a postulated LOCA. The methods specified in Appendix K were found to be highly conservative; that is, the fuel cladding temperatures expected during a loss-of-coolant accident would be much less than the temperatures calculated using Appendix K methods. In addition to showing that Appendix K is conservative, the ECCS research provided information that allows for quantification of that conservatism. The results of experiments, computer code development,

and code assessment allow more accurate calculations of ECCS performance, along with reasonable estimates of uncertainty, during a postulated loss-of-coolant accident than is possible using the Appendix K procedures.

It was also found that some plants are being restricted in operating flexibility by limits resulting from conservative Appendix K requirements; thus, preventing optimal operation. Based on the research performed, it was determined that these restrictions could be relaxed through the use of more realistic calculations without adversely affecting safety. The Appendix K requirements tended to divert both NRC and industry resources from matters more relevant to reactor safety to analyses with known nonphysical assumptions.

In recognition of the known conservatisms in Appendix K, the NRC adopted an interim approach, described in SECY-83-472[5], to accommodate industry requests for improved evaluation models for the purpose of reducing reactor operating restrictions. This interim approach is a step in the direction of basing licensing decisions on realistic calculations of plant behavior. Although the approach permits many "best estimate" methods and models to be used for licensee submittals, it retains those features of Appendix K that are legal requirements. The current revision of 10 CFR 50.46 now permits ECCS evaluation models to be fully "best estimate" and removes the arbitrary conservatisms contained in the required features of Appendix K for those licensees wishing to use these improved methods. The NRC staff believes that safety is best served when decisions concerning the limits within which nuclear reactors are permitted to operate are based upon

[5] Information Report from William J. Dircks to the Commissioners, dated November 17, 1983, "Emergency Core Cooling System Analysis Methods," SECY-83-472.

realistic calculations. This approach is currently being used in the resolution of almost all reactor safety issues (e.g., anticipated transients without scram, pressurized thermal shock, and operator guidelines) and is now available for one of the last remaining major issues still treated in a prescriptive manner, the loss-of-coolant accident.

The NRC staff amended § 50.46 of 10 CFR Part 50 to allow realistic methods to be used for the ECCS performance calculations in place of the evaluation models that use the required Appendix K features. This rule change also requires analysis of the uncertainty of the best-estimate calculation and requires that this uncertainty be considered when comparing the results of the calculations to the limits of § 50.46(b) so that there is a high probability that the criteria will not be exceeded. In this manner, more realistic calculations are available for regulatory decisions, yet appropriate conservatism would be maintained consistent with the accuracy of the calculation.

Many of the methods and models needed for a best-estimate calculation are the same as those used previously for evaluation model analyses. Although licensees and applicants are well acquainted with them, explicit guidance on acceptable methods and models (based on NRC experience with its own best-estimate advanced codes such as TRAC-PWR, TRAC-BWR, RELAP5, COBRA and FRAP) would be useful. Further, the NRC has not made acceptable methods for uncertainty analyses widely available. Therefore, the NRC staff decided that guidance in the form of a regulatory guide would be useful in order to document procedures acceptable to the regulatory staff.

C. REGULATORY POSITION

The features presented in the Appendix to this regulatory guide, "Description of Best Estimate Calculations of Emergency Core Cooling

Systems Performance and Estimate of the Calculational Uncertainty," are acceptable to the NRC staff for demonstrating compliance with Paragraph (a)(1)(i) of 10 CFR 50.46.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide.

Licensees and applicants may propose means other than those specified by the provisions of Section C of this guide for meeting applicable regulations. The guidance provided in Section C has been approved for use by the staff in the evaluation of submittals in the following categories and as an acceptable means of complying with the Commission's regulations described in Section A:

1. Construction Permit applicants that choose to make use of the provisions of the rule that allows the use of realistic models as an alternative to the features of Appendix K of 10 CFR Part 50.

2. Operating License applicants that choose to make use of the provisions of the rule that allows the use of realistic models as an alternative to the features of Appendix K of 10 CFR Part 50.

3. Operating Reactor Licensees will not be evaluated against the provisions of this guide except for new submittals which make use of the provisions of the rule that allow the use of realistic models as an alternative to the features of Appendix K of 10 CFR Part 50.

E. VALUE/IMPACT ANALYSIS

A value/impact analysis has not been prepared in support of this regulatory guide. Such a study was performed as part of the regulatory analysis which supports the rulemaking effort. This study is available through the NRC Public Document Room.

APPENDIX

DESCRIPTION OF BEST ESTIMATE CALCULATIONS OF EMERGENCY CORE COOLING SYSTEMS PERFORMANCE AND ESTIMATION OF THE CALCULATIONAL UNCERTAINTY

1. BEST ESTIMATE CALCULATIONS

1.1 General

A best-estimate calculation uses modeling that attempts to realistically describe the physical processes in a nuclear reactor. There is no unique approach to the extremely complex modeling of the processes occurring during a loss-of-coolant accident (LOCA). The NRC has developed and assessed several best-estimate, advanced thermal-hydraulic transient codes. These include TRAC-PWR, TRAC-BWR, RELAP5, COBRA, and the FRAP series of codes (References 1, 2, 3, 4, 5, and 6). These codes predict the major phenomena observed for a broad range of thermal-hydraulic and fuel tests reasonably well. Licensees and/or applicants may use, but are not limited to, these codes and/or specific models within them to perform best-estimate calculations of emergency core cooling system (ECCS) performance. However, since the NRC staff has not performed the plant-specific uncertainty analysis required by the proposed revision to 10 CFR 50.46, the licensee must demonstrate that the code and/or models are acceptable and applicable to the specific facility over the intended operating range and must quantify the uncertainty in the specific application. General features expected in a best-estimate calculation are described in this section, and specific examples of features that are considered acceptable best-estimate models are given in section 1.2. Other models and/or correlations will be considered acceptable if their technical basis is demonstrated with appropriate data and analysis.

A best-estimate model should provide a realistic calculation of a particular phenomenon to the degree practical with the currently available data and knowledge of the phenomenon. The model should be compared with applicable experimental data and should predict the mean of the data, rather than providing a bound to the data. The effects of all important variables should be considered. If it is not possible or practical to consider a particular phenomenon, the neglect of this phenomenon should not normally be treated by including a bias in the analysis directly, but should be included as part of the model uncertainty. The importance of neglecting a particular phenomenon should be considered within the overall calculational uncertainty.

Careful consideration should be given to the range of applicability of the model when used in a best-estimate code. When comparing the model to data, judgments of the applicability of the data to the situation that would actually occur in a reactor should be made. Correlations generally should not be extrapolated beyond the range over which they were developed or assessed. If the model is to be extrapolated beyond the conditions where valid data comparisons have been made, judgments should be made as to the effect of this extrapolation and should be considered in the uncertainty evaluation. The use of fundamental laws of physics, well established data bases (e.g., steam tables), and sensitivity studies should be used to assist in the estimation of uncertainty that results from extrapolation.

A best-estimate code contains all the models necessary to predict the important phenomena that might occur during a loss-of-coolant accident. Best-estimate code calculations should be compared with applicable experimental data (e.g., separate effects tests and integral simulations of loss-of-coolant accidents) to determine the overall uncertainty and biases of the calculation. In addition to providing input to the uncertainty evaluation, integral simulation data comparisons should be used to ensure that important phenomena that are

expected to occur during a loss-of-coolant accident are adequately calculated. The following paragraphs list some of the primary features that should be included in best-estimate thermal-hydraulic transient codes. In general, these features will have uncertainties associated with their use for predicting reactor system response. These uncertainties should be considered in the overall uncertainty analysis described in Section 2 of this Appendix.

The above discussion is an idealized definition of best-estimate. In practice, best-estimate codes may contain certain models which are simplified and/or contain conservatism to some degree. This conservatism may be introduced for the following reasons:

1. The model simplification or conservatism has little effect on the result and, therefore, the development of a better model is not justified.

2. The uncertainty of a particular model is difficult to determine, and only an upper bound can be determined.

3. The particular application does not require a totally best-estimate calculation, so a bias in the calculation is acceptable.

The introduction of conservative bias or simplification in otherwise best-estimate codes should not, however, result in calculations which are nonphysical, unrealistic, do not include important phenomena, or contain bias and uncertainty that cannot be bounded. Therefore, any calculational procedure determined to be a best-estimate code in the context of this guide or for use under 10 CFR 50.46(a)(i) should be compared with applicable experimental data to ensure that realistic calculations of important phenomena result.

1.2 Features of Best Estimate Codes

Some features that are acceptable for use in best-estimate codes are described in the following paragraphs. Models which address these features may be used with the basic requirement that a specific model is acceptable if it has been compared with applicable experimental data and shown to provide reasonable predictions. Reference 7, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG 1230, provides a summary of the large experimental data base available, upon which best-estimate models may be based. While inclusion in Reference 7 does not guarantee that the data or model will be acceptable, the report represents a large body of data generally applicable to best-estimate models. NUREG 1230 also provides documentation of NRC studies of the effect of reactor power increase on risk, background information on the ECCS rule, and a description of the NRC developed methodology for estimating thermal-hydraulic transient code uncertainty.

For any models or correlations used in a best-estimate code, sufficient justification must be provided to substantiate that the code performs adequately for the classes of transients to which it is applied. In general these features have uncertainties associated with their use for predicting reactor system response. These uncertainties should be considered as part of the overall uncertainty analysis described in Section 2 of this Appendix.

1.2.1 Basic Structure of Code

1.2.1.1 Numerical Methods

A best-estimate code consists of a numerical scheme for solving the equations used to represent the various models. The numerical scheme is, in itself, a complex process which can play an important role in the overall calculation. Careful numerical modeling, sensitivity studies, and evaluations of numerical error should be performed to ensure that the results of the calculations are representative of the models used in the code. Numerical simulations of complex problems, such as those considered here, treat the geometry of the reactor in an approximate

manner, making use of discrete volumes or nodes to represent the system. Sensitivity studies and evaluations of the uncertainty introduced by noding should be performed. Numerical methods treat time in a discrete manner, and the effect of time step size should also be investigated.

1.2.1.2 Computational Models

A best-estimate code typically contains equations for conservation of mass, energy, and momentum of the reactor coolant and noncondensible gases, if important (e.g., air, nitrogen, etc). Energy equations are also used to calculate the temperature distribution in reactor system structures and in the fuel rods. The required complexity of these equations will vary depending on the phenomena that are to be calculated and the required accuracy of the calculation. NRC staff experience with its own best-estimate computer codes has indicated that separate flow fields for different fluid phases, or types, and calculation of nonequilibrium between phases may be required to calculate some important phenomena (e.g., countercurrent flow, reflood heat transfer) to an acceptable accuracy. NRC staff has also determined that certain phenomena require that the equations be solved in multiple dimensions. However, one-dimensional approximations to three-dimensional phenomena will be considered acceptable if those approximations are properly justified. Other basic code features include equations of state and other material properties. Sensitivity studies and comparisons to data should be performed to determine the importance of the simplifications used.

1.2.2 Initial and Boundary Conditions and Equipment Availability

The heat generated by the fuel during a loss-of-coolant accident depends on the power level of the reactor at the time of the loss-of-coolant accident and on the history of operation. The most limiting initial conditions expected over the life of the plant should be based upon sensitivity studies. It is not necessary to assume initial conditions which are impossible to occur in combination. For example, beginning-of-life peaking factors and end-of-life decay heat

may be unrealistic conditions and, therefore, would not require consideration. Given the assumed initial conditions, relevant factors such as the actual total power, actual peaking factors, and actual fuel conditions should be calculated in a best-estimate manner.

Calculations should be performed that are representative of the spectrum of possible break sizes from the full double-ended break of the largest pipe down to a size small enough that it can be shown that smaller breaks are of less consequence than those already considered. The analyses should also include the effects of longitudinal splits in the largest pipes, with the split area equal to twice the cross-sectional area of the pipe. The detail of break sizes considered should be sufficient so that the system response as a function of break size is defined well enough to confidently interpolate between calculations, without unexpected behavior between the break sizes considered.

Other boundary and initial conditions and equipment availability should be based upon plant technical specification limits. These other conditions include, but may not be limited to, availability and performance of equipment, automatic controls, and operator actions. Appendix A to 10 CFR 50 requires that a single failure be considered when analyzing safety system performance and that the analysis consider the effect of using only onsite power and only offsite power.

1.2.3 Sources of Heat During a Loss-of-Coolant Accident

Models should account for the sources of heat discussed below and the distribution of heat production.

1.2.3.1 Initial Stored Energy of the Fuel

The steady state temperature distribution and stored energy in the fuel before the postulated accident should be calculated in a best-estimate manner for the assumed initial conditions, fuel conditions, and

operating history. To accomplish this, the thermal conductivity of the fuel pellets and the thermal conductance of the gap between the fuel pellet and the cladding should be evaluated. Thermal conductivity of fuel is a function of temperature and is degraded by the presence of gases in crack voids between fuel fragments. An acceptable model for thermal conductivity should be developed from the in-pile test results for fuel centerline and off-center temperatures, taking into account the conductivity of gases in crack voids.

Thermal conductance of the fuel-cladding gap is a strong function of hot gap size and of the composition and pressure of the gases in the fuel rod. The calculation of hot gap size should take into account UO_2 or mixed oxide fuel swelling, densification, creep, thermal expansion and relocation, and cladding creep. Fuel swelling is a function of temperature and burn-up. Fuel densification is a function of burn-up, temperature, and initial density. Densification can result from hydrostatic stresses imposed on fuel during pellet-cladding mechanical interaction and should be considered. Fuel creep is a function of time, temperature, grain size, density, fission rate, oxygen-to-metal ratio and external stress. Fuel thermal expansion represents dimensional changes in unirradiated fuel pellets caused by changes in temperature. An acceptable model for fuel swelling should be based on in-pile and out-of-pile test data. Cladding creep introduces compressive creep strain in cladding during steady-state operation, reducing the gap between the fuel pellet and cladding. Cladding creep is a function of fast neutron flux (>1 Mev), cladding temperature, hoop stress, and material. Cladding materials may be cold-worked and stress-relieved or fully recrystallized, and there is a significant difference in the magnitude of creepdown between these materials. During pellet cladding mechanical interaction, cladding experiences deformation due to tensile creep, which is significantly different from that due to compressive creep. An acceptable model for cladding tensile creep should be based on in-reactor tensile creep data.

Best-estimate fuel models will be considered acceptable provided that the models include essential phenomena identified above and their technical basis is demonstrated with appropriate data and analysis.

The model evaluation procedure and data presented in this section would be considered acceptable for assessing a model used to calculate stored energy and heat transfer in fuel rods.

1.2.3.1.1 Model Evaluation Procedure

A model to be used in ECCS evaluations to calculate internal fuel rod heat transfer should:

a. be checked against several sets of relevant data
b. recognize the effects of fuel burnup, fuel pellet cracking, fuel pellet relocation, cladding creep, and gas mixture conductivity.

The model described by Lanning (Ref. 8) compared well to inpile fuel temperature data. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analysis.

1.2.3.1.2 Experimental Data

The correlations and data of Reference 9 would be considered acceptable for calculating the initial stored energy of the fuel and subsequent heat transfer.

1.2.3.2 Fission Heat

Fission heat should be included in the calculation and should be calculated using best-estimate reactivity and reactor kinetics calculations. Shutdown reactivities resulting from temperatures and voids should also be calculated in a best-estimate manner. The point kinetics formulation is considered an acceptable best-estimate method for determining fission heat in loss-of-coolant accident safety calculations. Other best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses. Control rod assembly insertion may be assumed if it is expected to occur.

1.2.3.3 Decay of Actinides

The heat from radioactive decay of actinides, including neptunium and plutonium generated during operation, as well as isotopes of uranium, should be calculated in accordance with fuel cycle calculations and known radioactive properties. The actinide decay heat chosen should be that appropriate for the operating history. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analysis.

1.2.3.4 Fission Product Decay Heat

The heat generation rates from radioactive decay of fission products, including the effects of neutron capture, should be included in the calculation and should be calculated in a best-estimate manner. The energy release per fission (Q value) should also be calculated in a best-estimate manner. Best-estimate methods will be considered acceptable provided that their technical basis is demonstrated with appropriate data and analysis. The model of Reference 10 would be considered acceptable for calculating fission product decay heat.

1.2.3.4.1 Model Evaluation Procedure

The values of mean energy per fission (Q) and the models for actinide decay heat should be checked against a set of relevant data.

1.2.3.5 Metal-Water Reaction Rate

The rate of energy release, hydrogen generation, and cladding oxidation from the reaction of the zircaloy cladding with steam should be included in the calculation in a best-estimate manner. Best-estimate models will be considered acceptable provided that their technical basis is demonstrated with appropriate data and analysis. For rods for which cladding is calculated to rupture during the loss-of-coolant accident, the oxidation of the inside of the cladding should be included in the

calculation in a best-estimate manner. The following model evaluation procedures and data would be considered acceptable for assessing a model used to calculate hydrogen generation and cladding oxidation.

1.2.3.5.1 Model Evaluation Procedure

Correlations to be used to calculate metal-water reaction rates for the temperature range which are less than or equal to 1900°F temperature range should:

a. be checked against a set of relevant data
b. recognize the effects of steam pressure, preoxidation of the cladding, deformation during oxidation, and internal oxidation from both steam and UO_2 fuel.

The data of Reference 11 would be considered acceptable for calculating the rates of energy release, hydrogen generation, and cladding oxidation for cladding temperatures greater than 1900°F.

1.2.3.6 Reactor Internals Heat Transfer

Heat transfer from piping, vessel walls, and internal hardware should be included in the calculation and should be calculated in a best-estimate manner. Heat transfer to channel boxes, control rods, guide tubes, and other in-core hardware should also be considered. Models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analysis.

1.2.3.7 Primary To Secondary Heat Transfer (not applicable to boiling water reactors)

Heat transferred between the primary and secondary systems through the steam generators should be considered in the calculation and should be calculated in a best-estimate manner. Models will be considered

acceptable provided that their technical basis is demonstrated with appropriate data and analysis.

1.2.4 Reactor Core Thermal/Physical Parameters

1.2.4.1 Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters

A calculation of the swelling and rupture of the cladding resulting from the temperature distribution in the cladding and from the pressure difference between the inside and outside of the cladding, both as a function of time, should be included in the analysis and should be performed in a best-estimate manner. The degree of swelling and rupture should be taken into account in calculations of gap conductance, cladding oxidation and embrittlement, and hydrogen generation and in calculating heat transfer and fluid flow outside of the cladding. The calculations of fuel and cladding temperatures as a function of time should use values of gap conductance and other thermal parameters as functions of temperature and time. Acceptable best-estimate methods to calculate the swelling of the cladding should take into account spatially varying cladding temperatures, heating rates, anisotropic material properties, asymmetric deformation of cladding, and fuel rod thermal and mechanical parameters. Best-estimate methods will be considered acceptable provided that their technical basis is demonstrated with appropriate data and analysis.

1.2.4.2 Other Core Thermal Parameters

As necessary and appropriate, physical/chemical changes in in-core materials (e.g., eutectic formation, phase change, or other phenomena caused by material interaction) should be accounted for in the reactor core thermal analysis. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analysis.

1.2.5 Blowdown Phenomena

1.2.5.1 Break Characteristics and Flow

In analyses of hypothetical loss-of-coolant accidents, a spectrum of possible break sizes should be considered, as indicated in Paragraph 1.2.2 above. The discharge flow rate should be calculated with a critical flow rate model that considers the fluid conditions at the break location, upstream and downstream pressures, and break geometry. The critical flow model should be justified by comparison to applicable experimental data over a range of conditions for which the model is applied. The model should be a best-estimate calculation, with uncertainty in the critical flow rate included as part of the uncertainty evaluation. Best-estimate models will be considered acceptable provided that their technical basis is demonstrated with appropriate data and analyses.

The model evaluation procedure and data discussed below would be considered acceptable for assessing a model or correlation used to calculate the discharge flow rate during a loss-of-coolant accident.

1.2.5.1.1. Model Evaluation Procedure

Critical flow models to be employed in ECCS evaluations should:

a. be checked against an acceptable set of relevant data
b. recognize thermal nonequilibrium conditions when the fluid is subcooled
c. provide a means of transition from nonequilibrium to equilibrium conditions.

The uncertainties and bias of a correlation or model used to calculate critical flow should be stated, as well as their range of applicability.

The mechanistic thermal nonequilibrium and slip model of Richter (Ref. 23) compares well to small- and large-scale test data (Ref. 24).

1.2.5.1.2 Experimental Data

An acceptable set of relevant critical flow data should cover the fluid conditions, geometries, and types of breaks pertinent to light water reactor loss-of-coolant accidents. The following tests should be considered in establishing an acceptable set of relevant data:

- o Marviken tests (Ref. 12)
- o Moby Dick experiments (Ref. 13)
- o BNL critical flashing flows in nozzles (Ref. 14)
- o Sozzi-Sutherland tests (Ref. 15)
- o Edwards experiments (Ref. 16)
- o Super Moby Dick experiments (Ref. 17 and 18)

For critical flow from small breaks under stratified conditions, currently acceptable test data for assessing models and codes include those reported by:

- o Anderson and Owca (Ref. 19)
- o Reimann and Khan (Ref. 20)
- o Schrock et al (Ref. 21 and 22)

1.2.5.2 ECCS Bypass

The best-estimate code should contain a calculation of the amount of injected cooling water that bypasses the vessel during the blowdown phase of the loss-of-coolant accident. The calculation of ECCS bypass should be a best-estimate calculation using analyses and comparisons with applicable experimental data. Although it is clear that the dominant processes governing ECC bypass are multidimensional, single dimensional approximations justified through sufficient analysis and data may be acceptable. Best-estimate methods will be considered

acceptable provided that their technical basis is demonstrated with appropriate data and analysis. Cooling water that is not expelled, but remains in piping or is stored in parts of the vessel, should be calculated in a best-estimate manner based on applicable experimental data.

The model evaluation procedure and data discussed below would be considered acceptable for assessing a model or correlation used to calculate ECC bypass during the blowdown phase of a loss-of-coolant accident.

1.2.5.2.1 Model Evaluation Procedure

A correlation or model to be used in ECCS evaluations to calculate ECC bypass should:

a. be checked against an acceptable set of relevant data
b. recognize the effects of pressure, liquid subcooling, fluid conditions, hot walls, and system geometry.

Uncertainties and bias in the correlations or models used to calculate ECC bypass should be stated, as well as the range of their applicability.

For scaled-down PWR downcomers, correlations by Beckner and Reyes (Ref. 30) compared well to the bypass data of References 25 and 26. Correlations of Sun (Ref. 31) and Jones (Ref. 32) compare well to counter-current flow limiting (CCFL) test data of interest to BRWs.

1.2.5.2.2 Experimental Data

The following test should be considered in establishing a set of data for scaled-down PWR downcomers:

BCL test (Ref. 25)
CREARE test (Refs. 25 and 26)

For a full-scale PWR vessel, ECC bypass data will become available from the forthcoming upper plenum test facility (UPTF) experiments performed as part of the 2D/3D program sponsored by the Federal Republic of Germany, Japan, and the United States.

For BWRs, the following test should be considered in establishing an acceptable set of relevant data:

SSTF test data (Refs. 27 through 29)

1.2.6 Noding Near the Break and ECCS Injection Point

The break location and ECCS injection point are areas of high fluid velocity and complex fluid flow and contain phenomena that are often difficult to calculate. The results of these calculations are often highly dependent on the noding. Sufficient sensitivity studies should be performed on the noding and other important parameters to ensure that the calculations provide realistic results.

1.2.7 Frictional Pressure Drop

The frictional losses in pipes and other components should be calculated using models that include variation of friction factor with Reynolds number and account for two-phase flow effects on friction. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analysis.

1.2.7.1 Model Evaluation Procedure

A model for frictional pressure drop to be used in ECCS evaluation should:

a. be checked against a set of relevant data

b. be consistent with models used for calculating gravitational and acceleration pressure drops. If void fraction models or correlations used to calculate the three components of the total pressure drop differ one from another, a quantitative justification must be provided.

Uncertainties and bias of a correlation or model should be stated as well as the range of applicability.

1.2.7.2 Experimental Data

An acceptable set of relevant data should cover, as far as possible, the ranges of parameters (mass flux, quality, pressure, fluid physical properties, roughness, and geometries) that are found in actual plant applications. The following tests should be considered in establishing an acceptable set of relevant data:

Vertical tubes
- Cambridge tests (Ref. 33)
- CISE test (Ref. 34 and 35)
- HTFS data bank (Ref. 36)

Horizontal tubes
- Cambridge test (Ref. 33)
- HTFS data bank (Ref. 36)
- GE tests (Ref. 37 and 38)

Rod bundles
- HTFS data bank (Ref. 36)
- GE tests (Ref. 39)

1.2.8 Momentum Equation

The following effects should be taken into account in the two-phase conservation of momentum equation: (1) temporal change in momentum, (2) momentum convection, (3) area change momentum flux, (4) momentum change

due to compressibility, (5) pressure loss resulting from wall friction, (6) pressure loss resulting from area change, and (7) gravitational acceleration. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

1.2.9 Critical Heat Flux

Best-estimate models developed from appropriate steady-state or transient experimental data should be used in calculating critical heat flux (CHF) during loss-of-coolant accidents. The codes in which these models are used should contain suitable checks to assure that the range of conditions over which these correlations are used are within those intended. Research has shown that CHF is highly dependent on the fuel rod geometry, local heat flux, and fluid conditions. After CHF is predicted at an axial fuel rod location, the calculation may use nucleate boiling heat transfer correlations, if the calculated local fluid and surface conditions justify the reestablishment of nucleate boiling. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

1.2.10 Post-CHF Blowdown Heat Transfer

Models of heat transfer from the fuel to the surrounding fluid in the post-CHF regimes of transition and film boiling should be best-estimate models based on comparison to applicable steady-state or transient data. Any model should be evaluated to demonstrate that it provides acceptable results over the applicable ranges. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analysis.

The model evaluation procedure and data discussed below would be considered acceptable for assessing a model or correlation used to calculate the heat transfer from the fuel to the surrounding fluid while in the post-CHF regime.

1.2.10.1 Model Evaluation Procedure

A model to be used in ECCS evaluation to calculate post-CHF heat transfer from rod bundles should:

a. be checked against an acceptable set of relevant data
b. recognize effects of liquid entrainment, thermal radiation, thermal nonequilibrium, low and high mass flow rates, low and high power densities, and saturated and subcooled inlet conditions.

The uncertainties and bias of models or correlations used to calculate post-CHF heat transfer should be stated as well as the range of their applicability.

1.2.10.2 Experimental Data

The acceptable set of relevant data should cover power densities, mass flow rates, fluid conditions, and rod bundle geometries pertinent to light-water reactor designs and applications. The following tests should be considered in establishing an acceptable set of relevant data:

- ORNL tests (Ref. 40 and 41)
- INEL tests (Ref. 42)
- ORNL data bank (Ref. 43)

1.2.10.3 Post-CHF Heat Transfer from Uncovered Bundles

During some time periods of small break LOCAs and during portions of large breaks prior to reflood, partial or complete core uncovering may be calculated to occur. During these time periods, very little liquid and low flow usually exist in the uncovered regions. Special considerations are therefore appropriate.

1.2.10.3.1 Model Evaluation Procedures

A correlation to be used in ECCS evaluations to calculate heat transfer from uncovered rod bundles should:

a. be checked against against an acceptable set of relevant data
b. recognize the effects of radiation and laminar, transition, and turbulent flows.

Uncertainties and bias in the models and/or correlations used to calculate post-CHF heat transfer should be stated as should the range of their applicability.

The correlation derived should include a stated procedure for correcting for radiative heat transfer and for estimating the vapor temperatures. The Hottel procedure, cited in Reference 9 is a satisfactory example.

The turbulent correlation may be of the general form:

$$Nu = A\ Re^m\ Pr^n$$

for higher Reynolds numbers, where the coefficients A, m and n are modifications from the basic Dittus-Boelter form and may be functions of other variables. The physical properties may be defined as wall, film, or vapor values.

A distinction from, and transition to, laminar convection (i.e., $Re < 2000$) should be made, with a value of the laminar heat transfer for rod bundles that is appropriate for the applicable bundle geometry and flow conditions.

Other forms and values, depending on the bundle geometry and flow conditions are also appropriate.

1.2.10.3.2 Experimental Data

An acceptable set of relevant data for post-CHF heat transfer from uncovered rod bundles should cover power densities, fluid conditions, and rod bundle geometries pertinent to light-water reactor design and application. The following tests should be considered in establishing an acceptable set of relevant data:

- o ORNL-THTF tests (Ref. 44 and 45)
- o ORNL Data Base (Ref. 46)

1.2.11 Pump Modeling

The characteristics of rotating primary system pumps should be derived from a best-estimate dynamic model that includes momentum transfer between the fluid and the rotating member, with variable pump speed as a function of time. The pump model resistance and other empirical terms should be justified through comparisons with applicable data. The pump model for the two-phase region should be verified by comparison to applicable two-phase performance data. Pump coastdown following loss of power should be treated in a best-estimate manner. A locked rotor following a large break loss-of-coolant accident need not be assumed unless it is calculated to occur. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analysis.

1.2.12 Core Flow Distribution During Blowdown

The core flow through the hottest region of the core during the blowdown should be calculated as a function of time. For the purpose of these calculations, the hottest region of the core should not be greater than the size of one fuel assembly. Calculations of the flow in the hot

region should take into account any crossflow between regions and any flow blockage calculated to occur during the blowdown as a result of cladding swelling or rupture. The numerical scheme should ensure that unrealistic oscillations of the calculated flow do not result. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analysis.

1.2.13 Post-Blowdown Phenomena

1.2.13.1 Containment Pressure

The containment pressure used for evaluating cooling effectiveness during the post-blowdown phase of a loss-of-coolant accident should be calculated in a best-estimate manner and should include the effects of containment heat sinks. The calculation should include the effects of operation of all pressure reducing equipment assumed to be available, as discussed in Paragraph 1.2.2 of this Appendix. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analysis.

1.2.13.2 Calculation of Post-Blowdown Thermal-Hydraulics for Pressurized Water Reactors

The refilling of the reactor vessel and the ultimate reflooding of the core should be calculated by a best-estimate model that takes into consideration the thermal and hydraulic characteristics of the core, the emergency core cooling systems, and the primary and secondary reactor system. The models should be capable of calculating the two-phase level in the reactor during the postulated transient. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analysis.

1.2.13.2.1 Model Evaluation Procedures

A correlation or model to be used in ECCS evaluation to calculate level swell should be checked against an acceptable set of relevant

data, and should recognize the effects of depressurization, boil-off, power level, fluid conditions and system geometry.

The correlation proposed by Chexal, Horowitz, and Lellouche (Ref. 52) provides acceptable results when compared to experimental data reported in References 44, 48, 49, and 51.

Uncertainties and bias of a correlation or model used to calculate level swell should be stated, as should the range of applicability.

The primary coolant pumps should be assumed to be operating in the expected manner, based on the assumptions of Paragraph 1.2.11, when calculating the resistance offered by the pumps to fluid flow. Models will be considered acceptable provided that their technical basis is demonstrated through comparison with appropriate data and analysis.

The total fluid flow leaving the core exit (carryover) should be calculated using a best-estimate model which includes the effect of crossflow on carryover and core fluid distribution. Thermal-hydraulic phenomena associated with unique emergency core cooling systems, such as upper plenum injection and upper head injection, should be accounted for. The effects of the compressed gas in the accumulator, which is discharged, following accumulator water discharge should be included in the calculation. Any model or code used for this calculation should be assessed against applicable experimental data. Reference 7 describes a large body of refill/reflood thermal hydraulic data obtained from the 2D/3D program which is appropriate for consideration.

1.2.13.2.2 Experimental Data

The following test should be considered when establishing an acceptable set of relevant data:

- GE tests (Ref. 47 and 48)
- ORNL tests (Ref. 44 and 49)
- FLECHT-SEASET test (Ref. 50)
- THETIS tests (Ref. 51)

1.2.13.3 Steam Interaction With Emergency Core Cooling Water in Pressurized Water Reactors

The thermal-hydraulic interaction between the steam or two-phase fluid and the emergency core cooling water should be taken into account in calculating the core thermal-hydraulics and the steam flow through the reactor coolant pipes during the time that the accumulators are discharging water. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analysis.

1.2.13.4 Post-Blowdown Heat Transfer for Pressurized Water Reactors

During refilling of the reactor vessel and ultimate reflooding of the core, the heat transfer should be based on a best-estimate calculation of the fluid flow through the core, accounting for unique emergency core cooling systems. The calculations should also include the effects of any flow blockage calculated to occur as a result of cladding swelling or rupture. Heat transfer calculations, which account for two-phase conditions in the core during refilling of the reactor vessel, should be justified through comparisons with experimental data. Best-estimate models will be considered acceptable provided that their technical basis is demonstrated through comparison with appropriate data and analysis.

The FLECHT-SEASET tests (Ref. 53, 54, 55) should be considered when establishing an acceptable set of relevant data. Reference 7 contains extensive information regarding a large amount of experimental reflood heat transfer data. This information should also be considered when developing and assessing models. In particular, the results from the 2D/3D are particularly relevant.

1.2.14 Convective Heat Transfer Coefficients for Boiling Water Reactor Rods Under Spray Cooling

Models will be considered provided their technical basis can be justified with appropriate data and analysis.

Following the blowdown period, convective heat transfer coefficients should be determined based on the calculated fluid conditions and heat transfer modes within the bundle and on the calculated rod temperatures.

During the period following the flashing of the lower plenum fluid, but prior to ECCS initiation, heat transfer models should include cooling by steam flow or by a two-phase mixture, if calculated to occur.

Following initiation of ECCS flow, but prior to reflooding, heat transfer should be based on the actual calculated bundle fluid conditions and best-estimate heat transfer models which take into account rod-to-rod variations in heat transfer.

After the two-phase reflood level reaches the level under consideration, a best-estimate heat transfer model should be used. This model should include the effects of any flow blockage calculated to occur as a result of cladding swelling or rupture.

Thermal-hydraulic models which do not calculate multiple channel effects should be compared with applicable experimental data or more detailed calculations to ensure that all important phenomena are adequately calculated.

1.2.15 The Boiling Water Reactor Channel Box Under Spray Cooling

Following the blowdown period, heat transfer from the channel box and wetting of the channel box should be based on the calculated fluid conditions on both sides of the channel box and should make use of best-estimate heat transfer and rewetting models that have been compared with applicable experimental data. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analysis.

1.2.16 Special Considerations for a Small-Break Loss-of-Coolant Accident in Pressurized Water Reactors

The slower small-break loss-of-coolant accident leads to fluid conditions characterized by separation of the fluid phases versus the more homogeneous fluid conditions that would result from rapid large-break loss-of-coolant accident transients. Phenomena that would occur in a PWR during a small-break loss-of-coolant accident would, therefore, be significantly different from those phenomena that would occur during a large-break loss-of-coolant accident. The distribution of liquid throughout the reactor system, in addition to the total liquid inventory, is of increased importance for the small-break loss-of-coolant accident. A number of special factors must be given increased consideration in small-break loss-of-coolant accident calculations to correctly predict phenomena influenced by the liquid inventory distribution.

Break flow may be greatly influenced by the location and specific geometry of the break. For a break in a horizontal pipe containing stratified flow, the quality of the break flow will be a strong function of the assumed location of the break on the pipe (e.g., top or bottom). Small-break loss-of-coolant accident calculations should, therefore, include various assumed break locations in the spectrum of breaks analyzed. The assumed operating state of the reactor coolant pump will also influence the distribution of liquid throughout the system and the amount of liquid lost through the break.

The pump operation assumptions used in the calculations should be the most likely, based on operating procedures, with appropriate consideration of the uncertainty of the pump operation during an actual event. Level depression in the core region and subsequent core heatup may be influenced by liquid holdup in the steam generator tubes, manometric effects of liquid in the piping and loop seal region, and

liquid levels relative to vent paths for steam through upper plenum bypass flow paths and vent values. Steam generator heat transfer under "reflux" or "boiler-condensor" modes of operation may also strongly influence core inventory through level depression and the effect on total system pressure and, thus on ECCS flow. These phenomena should be carefully considered in the calculation, as should sensitivity studies of the importance of these effects for use in the uncertainty evaluation.

Heat transfer from an uncovered core under high pressure conditions typical during a small-break loss-of-coolant accident may include contributions from both convective and radiation heat transfer to the steam. Models will be considered acceptable provided that their technical basis is demonstrated through comparison with appropriate data and analysis. Specific guidance regarding uncovered bundle heat transfer is given in Section 1.2.10.3.

1.2.17 Other Features of Best Estimate Codes

No list of best-estimate code features could be all-inclusive, because the important features of a best-estimate code may vary depending on the transient to be calculated and the required accuracy of the calculation. Because of this, no attempt has been made to construct an exhaustive list of best-estimate code features. Rather, features which were identified as important for inclusion in Appendix K were used as a basis for the above list. These features are not necessarily any more or less important than other code features, but were highlighted because it is necessary to give specific examples of how current best-estimate models may vary from methods used traditionally in evaluation model codes using the various Appendix K conservatisms. In addition, models have not been included in areas where it was felt that the best model would be highly dependent on the specific plant design or the specific transient under consideration.

The NRC staff believes that good examples of best-estimate thermal-hydraulic transient codes are those developed by the NRC (e.g., TRAC-PWR, TRAC-BWR, RELAP5, COBRA and FRAP). Although these codes are subject to further improvement, based on their ongoing use and assessment, they currently provide reasonable best-estimate calculations of the loss-of-coolant accident in a full-scale light-water reactor. This is substantiated through the code development and assessment literature generated by the NRC and its contractors over the past several years.

It is possible, however, to generally describe how other features of best-estimate codes should be constructed. Two basic criteria should be applied:

1.2.17.1 Completeness

Best-estimate codes should contain models in sufficient detail to predict phenomena that are important to the desired result of the calculation (e.g., peak cladding temperature). Simplifications are acceptable as long as code uncertainties or bias do not become so large that they cast doubt on the actual behavior that would occur or on the true effect of assumed initial and boundary conditions (e.g., equipment sizing, safety system settings, etc.). Comparisons of the overall calculations to integral experiments should be performed to ensure that important phenomena can be predicted and to help in making judgments on the effect of code simplifications. Consideration should also be given to the uncertainty and validity of the experiment to ensure that meaningful comparisons are being made.

1.2.17.2 Data Comparisons

Individual best-estimate models should be compared to applicable experimental data to ensure that realistic results are predicted and

that relevant experimental variables are included. Uncertainty analyses are required to ensure that a major bias does not exist in the models and that the model uncertainty is small enough to provide a realistic estimate of the effect of important experimental variables. Uncertainty analyses should also consider experimental uncertainty to ensure that meaningful comparisons are being made.

2. Estimation of Overall Calculation Uncertainty

2.1 General

The term "uncertainty," when applied to best-estimate thermal-hydraulic transient codes, is used at two levels. At the lower, or more detailed level, the term refers to the degree to which an individual model, correlation, or method used within the code represents the physical phenomenon which it addresses. These individual uncertainties, when taken together, comprise the "code uncertainty."

The combined uncertainty associated with individual models (i.e., code uncertainty) within the best-estimate codes does not account for all of the uncertainty associated with the model's use. In addition to the code uncertainty, various other sources of uncertainty are introduced when attempting to use best-estimate codes to predict full-scale plant thermal hydraulic response. These include uncertainty associated with the experimental data used in the code assessment process (including applicability of the data to full-scale reactors), the input boundary and initial conditions, and the fuel behavior. Additional sources of uncertainty stem from the use of simplifying assumptions and approximations. A careful statement of these assumptions and approximations should be made, and the uncertainty associated with them should be taken into account. Therefore, the "overall calculational uncertainty" is defined as the uncertainty arrived at when all the contributions from the sources identified above, including the code uncertainty, are taken into account.

A 95% probability level is considered acceptable to the NRC staff for comparison of best-estimate predictions to the applicable limits of § 50.46(b) to meet the requirement of § 50.46(a)(1)(i) to show that there is a high probability that the criteria will not be exceeded. The basis for selecting the 95% probability level is primarily for consistency with standard engineering practice in regulatory matters involving thermal-hydraulics. Many parameters, most notably the departure from nucleate boiling ratio (DNBR), have been found acceptable at the 95% probability level by the staff in the past.

This 95% probability level would also be applied to small-break loss-of-coolant accidents, which have a higher probability than large breaks. The dominant factors influencing risk from small-break loss-of-coolant accidents include equipment availability and operator actions. Calculational uncertainties are much less important than factors such as operator recognition of the event, the availability of equipment, and the correct use of this equipment. The use of a best-estimate calculation with reasonable and quantifiable uncertainty is expected to provide a reduction in the overall risk from a small-break loss-of-coolant accident by providing more realistic calculations with which to evaluate operator guidelines and determine the true effect of equipment availability.

This section provides a description of the features that should be included in the overall code uncertainty evaluation that is called for in 10 CFR 50.46(a)(1). This uncertainty evaluation should make use of probabalistic and statistical methods to determine the code uncertainty bias (if any). For a calculation of this complexity, a completely rigorous mathematical treatment is neither practical nor required. In many cases, approximations and assumptions may be made to make the overall calculational uncertainty evaluation possible. A careful statement of these assumptions and approximations should be made so that the NRC staff may make a judgement as to the validity of

the uncertainty evaluation. The purpose of the uncertainty evaluation is to provide assurance that for postulated loss-of-coolant accidents a given plant will not, with a probability of 95% or more, exceed the applicable limits specified in 10 CFR 50.46(b).

2.2 Code Uncertainty

This regulatory guide makes a distinction between the terms "code uncertainty" and "overall calculational uncertainty." The latter term is defined in Section 2.1 of this Appendix and includes the contributions to the uncertainty described in Sections 2.2 and 2.3. The features of the code uncertainty (i.e., the contribution to the overall uncertainty due to the models and numerical methods used) are described in this section.

Code uncertainty should be evaluated through direct data comparison with relevant integral systems and separate effects experiments at different scales. In this manner, an estimate of the uncertainty attributable to the combined effect of the models and correlations within the code can be obtained for all scales and for different phenomena. Comparison to a sufficient number of integral systems experiments, from different test facilities and different scales, should be made to assure that a reasonable estimate of code uncertainty and bias has been obtained. Where necessary, separate effects experiments should be used to establish code uncertainty for specific phenomena (e.g., comparisons to Cylindrical Core Test Facility data to ascertain code uncertainty in modeling upper plenum injection performance). Code comparisons should account for limitations of the measurements and calibration errors.

These comparisons should be performed for important key parameters to demonstrate the overall best-estimate capability of the code. For large-break loss-of-coolant accidents, the most important key parameter

is peak cladding temperature, which is addressed by one of the criteria of 10 CFR 50.46(b) and has a direct influence on the other criteria. In addition, a code uncertainty evaluation should be performed for other important parameter(s) for the transient of interest to evaluate compensating errors. For small-break loss-of-coolant accidents, the cladding temperature response is the most important parameter; however, the ability of the codes to predict overall system mass and reactor vessel inventory distribution should also be statistically examined.

In evaluating the code uncertainty, it will be necessary to evaluate the code's predictive ability over several time intervals, since different processes and phenomena occur at different intervals. For example, in large-break loss-of-coolant accident evaluations, separate code uncertainties may be required for the peak cladding temperature during the blowdown and post-blowdown phases. Justification for treating these uncertainties individually or methods for combining them should be provided.

The experimental information used to determine code uncertainty will usually be obtained from facilities that are much smaller than nuclear power reactors. Applicability of these results should be justified for larger scales. The effects of scale can be assessed through comparisons to available large-scale separate effects tests and through comparison to integral tests from various sized facilities. If there are scaling problems, particularly if predictions are nonconservative, the code should be improved for large scale plants (i.e., nuclear plants). Codes not having scaling capability will not be acceptable if their predictions are nonconservative.

2.3 Other Sources of Uncertainty

When a best-estimate methodology is used to predict reactor transients, sources of uncertainty other than the limitations in the

individual models and numerical methods (i.e., code uncertainty) are introduced. The following contributors to the overall calculational uncertainty should also be considered in the uncertainty analysis.

2.3.1 Initial and Boundary Conditions and Equipment Availability

When a plant input model is prepared, certain relationships describing the plant boundary and initial conditions and the availability and performance of equipment are defined. These include factors such as initial power level, pump performance, valve activation times, and control systems functioning. Uncertainties associated with the boundary and initial conditions and the characterization and performance of equipment should be accounted for in the uncertainty evaluation. It is also acceptable to limit the variables to be considered by setting their values to conservative bounds.

2.3.2 Fuel Behavior

Variability of the results of plant transient calculations can result from uncertainties associated with fuel behavior that is not included in the comparisons of code results with integral experiments, since most integral tests use electrical heater rods. This uncertainty includes many effects such as fuel conductivity, gap width, gap conductivity, and peaking factors. These uncertainties should be quantified and used in the determination of the overall calculational uncertainty.

2.3.3 Other Variables

There may be individual models within the best-estimate code whose effect may not have been evaluated by the comparison to the integral systems data. For example, since most integral systems experiments use electrically heated rods, uncertainties associated with the prediction

of core decay heat and cladding metal-water reaction have not been evaluated. In addition, to demonstrate the overall adequacy of the predictive ability of the best-estimate code, it may be necessary to use empirical break discharge coefficients to obtain a reasonable break flow. The uncertainty in the individual models, that have not been evaluated by comparison to integral systems data should be quantified and used in the determination of overall code uncertainty.

2.4. Statistical Treatment of Overall Calculational Uncertainty

The methodology used to obtain an estimate of the overall calculational uncertainty at the 95% probability limit should be provided and justified. If linear independence is assumed, suitable justification should be provided. The influence of the individual parameters on code uncertainty should be examined by making comparisons to relevant experimental data. Justification should be provided for the assumed distribution of the parameter and the range considered.

In reality, the true statistical distribution for the key parameters (e.g., peak cladding temperature) is unknown. The choice of a statistical distribution should be verified using applicable engineering data and information. The statistical parameters appropriate for that distribution should be estimated using available data and results of engineering analyses. Supporting documentation should be provided for this selection process. These estimated values are assumed to be the true values of the statistical parameters of the distribution. With these assumptions, an upper one-sided probability limit can be calculated at the 95% level. As the probability limit approaches 2200°F, more care must be taken in the selection and justification of the statistical distribution and in the estimation of its statistical parameters. If a normal distribution is selected and justified, the probability limit can be conservatively calculated using two standard deviations. The added conservatism of the two standard deviations compared to the 95 percentile is used to account

for uncertainty in the probability distribution. Other techniques which account for the uncertainty in a more detailed manner may be used. These techniques may require the use of confidence levels which are not required by the above approach.

The evaluation of the peak cladding temperature at the 95% probability level need only be performed for the worst case break identified by the break spectrum analysis in order to demonstrate conformance with 10 CFR 50.46(b). However, in order to use this approach, justification which demonstrates that the overall calculational uncertainty for the worst case bounds the uncertainty for other breaks within the spectrum must be provided. It may be necessary to perform separate uncertainty evaluations for large- and small-break loss-of-coolant accidents due to the substantial difference in system thermal hydraulic behavior.

The revised § 50.46 (a)(1)(i) requires that it be shown with a high probability that all the criteria of § 50.46 (b) will not be exceeded, not just the peak cladding temperature criterion. However, since the other criteria are strongly dependent on peak cladding temperature, explicit consideration of the probability of exceeding the other criteria may not be required if it can be demonstrated that meeting the temperature criterion at the 95% probability level ensures with a similar or greater probability that the other criteria will not be exceeded.

2.5 NRC Approach to LOCA Uncertainty Evaluation

Chapter 4 of the "Compendium of ECCS Research for Realistic LOCA Analysis" (Ref. 7) presents a methodology that is being considered for evaluating a best-estimate LOCA analysis code and determining the overall calculational uncertainty in peak cladding temperature predictions.

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ENCLOSURE G

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SUMMARY OF PUBLIC COMMENTS

On March 3, 1987, the Nuclear Regulatory Commission published in the Federal Register (52FR6334) proposed amendments to 10 CFR 50 and Appendix K. These proposed amendments were motivated by the fact that since the promulgation of Section 50.46 of 10 CFR 50, "Acceptance Criteria for Emergency Core Cooling Systems (ECCS) in Light-Water Power Reactors," and the acceptable and required features and models specified in Appendix K to 10 CFR 50, considerable research has been performed that has greatly increased the understanding of ECCS performance during a loss-of-coolant accident (LOCA). It is now known that the methods specified in Appendix K, combined with other analysis methods currently in use, are highly conservative and that the actual cladding temperatures which would occur during a LOCA would be much lower than those calculated using Appendix K methods.

In soliciting the public's comments on the regulatory guide, the NRC requested general comments and responses to the following questions:

1. Should the regulatory guide list models, data, and model evaluation procedures that the NRC considers to be acceptable for realistic calculations of ECCS performance?

2. Are the models, data, and model evaluation procedures listed in Appendix A to the draft regulatory guide appropriate?

3. Does Appendix B of the draft regulatory guide appropriately discuss the features of the best-estimate codes and the estimation of overall calculational uncertainty?

The comment period for the proposed rule revision and the draft regulatory guide (52FR11385) expired on July 1, 1987. The NRC received nine responses to the request for comments on the regulatory guide questions. The names and organizations of the nine people who responded to the request for comments are provided in Section A-1. Section A-2 contains the paraphrased summaries of the comments submitted by the nine people listed in Section A-1. The NRC responses to the draft regulatory guide comments are presented in Section A-3. The comments are numbered to facilitate cross-referencing with the appropriate response category. The response categories cover general comments and responses to the three questions listed above.

The NRC staff incorporated Appendix B of the draft regulatory guide into Appendix A after the original draft had been submitted for review. The numbering sequence in the original Appendix B is nearly identical to the sequence in the newly merged Appendix in the current guide. Comments will be referenced to the sections in the new Appendix. If the numbers are different between the draft and the current guide, cross-referencing to the sections in the draft guide will be indicated in square brackets, (e.g., . . . Section 1.2.5.2 [2.1] . . .).

SECTION A-1
LIST OF PUBLIC COMMENTERS

This section lists the names of the people and organizations that responded to the NRC staff's request for public comment on the draft regulatory guide.

1. G. S. Lellouche, S. Levy Incorporated, 3425 S. Bascom Avenue, Campbell, California 95008-7006.

2. Gary N. Ward, Manager, Reload Licensing, Advanced Nuclear Fuels Corporation, 2101 Horn Rapids Road, P.O. Box 130, Richland, Washington 99352-0130.

3. Susan L. Hiatt, OCRE Representative, Ohio Citizens for Responsible Energy, 8275 Munson Road, Mentor, Ohio 44060.

4. W. L. Stewart, Vice President, Nuclear Operations, Virginia Electric and Power Company, Richmond Virginia 23261.

5. C. O. Woody, Group Vice President, Nuclear Energy, Florida Power and Light Company, P.O. Box 14000, Juno Beach, Florida 33408-0420.

6. C. W. Fay, Vice President, Nuclear Power, Wisconsin Electric Power Company, 231 W. Michigan, P.O. Box 2046, Milwaukee, Wisconsin 53201.

7. G. C. Sorensen, Manager, Regulatory Programs, Washington Public Power and Supply System, P.O. Box 968, 3000 George Washington Way, Richland, Washington 99352.

8. A. E. Scherer, Director, Nuclear Licensing, Power Systems, Combustion Engineering, Incorporated, 1000 Prospect Hill Road, P.O. Box 500, Windsor, Connecticut 06095-0500.

9. J. R. Thorpe, Director, Licensing and Regulatory Affairs, GPU Nuclear Corporation, 100 Interpace Parkway, Parsippany, New Jersey 07054-1149.

SECTION A-2

PUBLIC COMMENTS ON THE DRAFT REGULATORY GUIDE

1. G. S. Lellouche, S. Levy Incorporated, 3425 S. Bascom Avenue, Campbell, California 95008-7006.

COMMENT:

1.1 Mr. Lellouche comments that the extant literature does not support a constant Nusselt number (Nu = 8) for laminar flow, independent of bundle geometry. He recommends using the method employed by Weisman (NSE 1959) to take into account the effect of different pitch to diameter ratios on the Nusselt number.

2. Gary N. Ward, Manager, Reload Licensing, Advanced Nuclear Fuels Corporation, 2101 Horn Rapids Road, P.O. Box 130, Richland, Washington 99352-0130.

COMMENTS:

2.1 Mr. Ward indicates that the draft guide is overly prescriptive and that NRC would be limited in its receptiveness to models other than those identified as "acceptable" by the draft guide.

2.2 The regulatory guide implies that complex "best-estimate"[1] LOCA models developed by NRC, or their equivalent, are a necessary and

[1]For the purpose of the regulatory guide supporting the ECCS rule, the terms "best-estimate" and "realistic" have the same meaning. Both terms are used to indicate that the techniques attempt to predict realistic reactor system thermal-hydraulic response.

integral part of quantifying uncertainties of simpler models used in commercial licensing calculations. Development of a best estimate model, comparison of this model to experiments, and quantification of the uncertainties are burdensome for a single reactor vendor, fuel vendor, or utility.

2.3 It would seem appropriate for NRC to provide the industry with best estimate model LOCA reference plant results and quantification of uncertainties by conducting a spectrum of LOCA calculations for each generic plant type. Simpler commercial licensing models could then be compared to the NRC generic reference calculations.

2.4 Appendix A is a reasonably complete listing of models, data and model evaluation procedures.

2.5 Appendix B [Section A.2] is not sufficiently developed for a complete statistical approach with which to calculate uncertainties. It does not add to the information on how to interpret the regulation.

3. Susan L. Hiatt, OCRE Representative, Ohio Citizens for Responsible Energy, 8275 Munson Road, Mentor, Ohio 44060.

COMMENT:

3.1 Ms. Hiatt requests that NRC extend the comment period for the draft regulatory guide to September 1, 1987, because of insufficient time to review NUREG-1230.

4. W. L. Stewart, Vice President, Nuclear Operations, Virginia Electric and Power Company, Richmond, Virginia 23261.

COMMENTS:

4.1 The draft guide should specify cladding as zirconium base alloy rather than Zircaloy, so as not to impede the development of future improvements in cladding.

4.2 In evaluating Section 1.2.2 of Appendix B, it is suggested that conservative combinations of input parameters be allowed in best estimate calculations, provided their impact upon the calculated result can be quantified and the total calculational uncertainty adjusted to take this impact into account.

4.3 Is it intended that the confidence level used in a specific uncertainty evaluation be chosen and justified as part of the licensing application?

5. C. O. Woody, Group Vice President, Nuclear Energy, Florida Power and Light Company, P.O. Box 14000, Juno Beach, Florida 33408-0420.

COMMENTS:

5.1 FPL states that the changes proposed are, for the most part, a very positive modification. They are currently using the guidelines of the proposed regulatory guide to perform a small break LOCA analysis in a statistical manner.

5.2 The guide should note that the goal of the uncertainty analysis is to ensure that the calculated PCT is at least at the 95% probability level, and not to exactly quantify the amount of uncertainty.

6. C. W. Fay, Vice President, Nuclear Power, Wisconsin Electric Power Company, 231 W. Michigan, P.O. Box 2046, Milwaukee, Wisconsin 53201.

COMMENTS:

6.1 The guide should state clearly that the models, data, and model evaluation procedures listed are not the only set acceptable to the NRC.

6.2 The models for calculating the initial stored energy in the fuel should recognize, in addition to the variables listed, the effect of thermal and elastic strain of the fuel and cladding material.

6.3 Correlations for metal-water reaction rate calculations should recognize, in addition to the variables listed, the effect of cladding temperature.

6.4 The regulatory guide should recommend use of the Cathcart correlation for all temperatures of concern (1500°F to 2200°F).

6.5 Section 2.5 should be rewritten to reflect the reality of liquid in the flow field when the bundle is uncovered.

6.6 With respect to Section 1.2.3.1, the fuel model need not consider changes in the fuel properties due to burnup during the course of the LOCA, since they do not change significantly during the course of a LOCA.

6.7 The requirement in Section 1.2.4 that cladding swelling models take into account asymmetric deformation of the cladding is unclear. The direction of asymmetry should be clarified. The guidance would be unreasonable if the asymmetry of the deformation is in the azimuthal direction of one fuel rod.

6.8 There does not appear to be any basis for providing noding studies in areas of large thermal non-equilibrium (Section 1.2.6).

6.9 Research shows that the effect of flow blockage on post-blowdown heat transfer is minimal; therefore, there is no reason to require calculation of that effect (Section 1.2.13.4).

6.10 The best estimate codes cited in Section 1.2.17 should be identified more clearly. The code version should be identified and the list expanded to cover additional aspects of the analysis, such as fuel performance and containment analysis.

6.11 To avoid double accounting of uncertainties, the experimental data should be treated as absolute. No uncertainty should be associated with experimental data.

7. G. C. Sorensen, Manager, Regulatory Programs, Washington Public Power and Supply System, P.O. Box 968, 3000 George Washington Way, Richland, Washington 99352.

COMMENTS:

7.1 The regulatory guide represents a step forward toward a more reasonable approach to regulation.

7.2 Issuance of the regulatory guide should be delayed until NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," is available and reviewed.

7.3 The section on Metal-Water Reaction Rate should be corrected to read "greater than 1500°F and less than 1900°F."

7.4 Finite break opening time should be allowed for ECCS performance as it is for other portions of LOCA analyses. Appropriate evaluation of uncertainties in break opening time would then be included in the overall model uncertainty evaluations.

8. A. E. Scherer, Director, Nuclear Licensing, Power Systems, Combustion Engineering, Incorporated, 1000 Prospect Hill Road, P.O. Box 500, Windsor, Connecticut 06095-0500.

COMMENTS:

8.1 Use of a regulatory guide is an appropriate vehicle for relating those features of a best estimate evaluation model which the NRC staff has found to be acceptable.

8.2 CE believes that the regulatory guide must indicate that alternate models, data, and model evaluation procedures would be equally acceptable to the NRC if they can be shown to be appropriate for the intended purpose.

8.3 CE has reviewed Appendix A and can accept the models, data, and model evaluation procedures listed.

8.4 The regulatory guide should be updated periodically as new data are obtained.

8.5 The sections on ECC Bypass (1.2.5.2), Core Flow Distribution During Blowdown (1.2.12) and Containment Pressure Calculations (1.2.13) should be annotated to indicate that they are either not necessary or do not require the detail specified for small break LOCA evaluation models.

8.6 A statement should be incorporated into Section 2.0 [App A, Section B] indicating that alternate approaches for dealing with the various components of overall calculational uncertainty would be acceptable if it can be shown that these approaches provide a degree of conservatism equivalent to or greater than that provided by strictly adhering to the 95% probability approach described.

9. J. R. Thorpe, Director, Licensing and Regulatory Affairs, GPU Nuclear Corporation, 100 Interpace Parkway, Parsippany, New Jersey 07054-1149.

COMMENTS:

9.1 The guide fails to define what qualifications should actually certify a best-estimate code. The burden of proof that the codes identified in the regulatory guide are really best-estimate is unfairly handed to the licensee. The licensee can only be responsible for identifying sources of uncertainty derived from specific applications, not the code uncertainty. Code uncertainty analysis should be performed by the NRC with a well-planned and committed schedule.

9.2 The NRC should establish an unequivocal standard for qualifying best-estimate codes. With respect to acceptance criteria for a best-estimate code, the following questions need to be clearly addressed in the guide:

a. Does the guide imply that a code can be certified as best-estimate if the code uncertainty analysis results in a relatively small uncertainty?

b. Does the guide imply that industry developed codes cannot be considered as best-estimate because of their relatively larger uncertainty?

c. What is the borderline between best- and next-to-best estimate codes?

d. Should there be application categories so that certain codes are acceptable for a limited category of application?

9.3 Sections 1.2.2, 1.2.13.1, 2.1, and 2.3.1 are unclear regarding what is an acceptable or preferred method of factoring equipment availability into the best estimate calculations and uncertainty analysis.

SECTION A-3

SUMMARY OF COMMENTS ON DRAFT REGULATORY GUIDE

This section consists of a synthesis of the comments from the reviewers of the draft regulatory guide and NRC staff's proposed response to those comments. The comment numbers cross-reference the commenter with the comment number. For example, Comment 9.3 references the ninth commenter's third comment.

GENERAL COMMENTS ON THE DRAFT REGULATORY GUIDE (Refs. 3.1 and 7.2):

Two reviewers requested that the NRC extend the comment period for the draft regulatory guide review because of insufficient time to review NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," which describes the research supporting the proposed ECCS rule revision.

NRC RESPONSE:

The NRC staff believes the comment period was sufficient, since most of the research is not new and has been extensively reviewed in the past. Both commenters were contacted and told that comments received after the comment period would be considered if time permitted. Comments from both parties were received late and were, indeed, considered by the NRC.

COMMENT (Ref 5.1, 7.1 and 8.1):

Three reviewers commented favorably on the draft regulatory guide. One of the reviewers is using the guidelines of the proposed regulatory guide to perform a small break LOCA analysis in a statistical manner.

NRC RESPONSE:

No response required.

QUESTION 1:

Should the regulatory guide list models, data, and model evaluation procedures that the NRC considers to be acceptable for realistic calculations of ECCS performance?

COMMENT (Ref 2.1):

One of the reviewers indicated that the draft guide is overly prescriptive and that NRC would be limited in its receptiveness to models other than those identified as "acceptable" by the draft guide.

NRC RESPONSE:

The regulatory guide, by virtue of references to qualified data, gives the user the option of creating new models. The only requirement is that the new model must be able to reproduce the referenced experimental data acceptably well.

COMMENT (Ref 2.2):

One reviewer stated that development of a best-estimate model, comparison of this model to experiments, and quantification of the uncertainties are burdensome for a single reactor vendor, fuel vendor, or utility.

NRC RESPONSE:

This regulatory guide does not imply that NRC models must be used to benchmark simpler commercial models. Paragraph 2.2.2 clearly states that code uncertainty should be evaluated through direct data comparisons with relevant integral systems and separate effects experiments at different scales. In any event, this regulatory guide is not a requirement. Each licensee must evaluate whether any additional burden in complying with this rule is sufficiently offset by the operating benefits that would result.

COMMENT (Ref 2.3):

The same reviewer suggested that the NRC provide the industry with best-estimate model LOCA reference plant results and quantification of uncertainties by conducting a spectrum of LOCA calculations for each generic plant type. Simpler commercial licensing models could then be compared to the NRC generic reference calculations.

NRC RESPONSE:

The NRC makes available, to any licensee who requests it, any of its thermal-hydraulic safety codes and any non-proprietary assessment calculations. In addition, the NRC has developed a Code Scaling, Applicability, and Uncertainty (CSAU) Methodology, which is generally described in NUREG-1230. Detailed applications of this methodology will be documented in the near future. Although licensees may avail themselves of any of this work, the objective is not to perform uncertainty analyses for licensees, but to provide an independent audit tool for the NRC. Therefore, the NRC plans no further work analyzing "reference plants" as part of the CSAU demonstration.

COMMENT (Ref 6.1, 8.2):

Two reviewers stated that the regulatory guide must indicate that alternate models, data, and model evaluation procedures would be equally acceptable if they can be shown to be appropriate for the intended purpose.

NRC RESPONSE:

The NRC staff agrees with this comment. The introduction to Section 2 of Appendix A clearly states that the basic requirement for acceptability of a specific model is an accurate comparison with applicable data.

QUESTION 2:

Are the models, data, and model evaluation procedures listed in Appendix A to the draft regulatory guide appropriate?

COMMENT (Ref 1.1):

The reviewer commented that the laminar Nusselt number depends on rod pitch to diameter ratio for triangular rod bundle lattices, so that the implied constant laminar Nu number of 8 is not supported by the extant literature.

NRC RESPONSE:

The NRC staff encourages the use of models that accurately calculate the phoneomena. The acceptance of the model will be based on comparison of applicable experimental data. The correlation of Nu = 8 for laminar flow was intended as an example of a distinct transition to the laminar flow region. If such a model were used for evaluations, the model would have to be compared to appropriate experimental data, accounting for the particular geometry and flow condition.

In order to avoid confusion over the applicability of this illustrative example, the laminar flow correlation has been deleted in the final regulatory guide.

COMMENT (Ref 2.4):

One reviewer agrees that Appendix A is a reasonably complete listing of models, data, and model evaluation procedures.

NRC RESPONSE:

No response required.

COMMENT (Ref 8.3 and 8.4):

A reviewer accepted the models, data, and model evaluation procedures and suggested that the regulatory guide should be updated periodically as new data are obtained.

NRC RESPONSE:

The staff agrees that the regulatory guide should be updated periodically to incorporate the results of new research.

COMMENT (Ref 4.1):

One reviewer stated that the regulatory guide should specify cladding as zirconium base alloy rather than zircaloy so as not to impede the development of future improvements in cladding.

NRC RESPONSE:

The staff believes that such a modification is beyond the scope of the current regulatory guide and should be considered in a separate rulemaking action in which it would receive appropriate public review and comment prior to implementation. Additionally, zircaloy cladding material is specified in other portions of the Code of Federal Regulations, such as 10 CFR 50.44, thereby making a change of this type more suitable to a broader regulatory context. Therefore, the staff is not broadening the definition of cladding materials in 10 CFR 50.46, as requested by this commenter.

COMMENT (Ref 6.2, 6.3):

One reviewer commented that the models for calculating the initial stored energy in the fuel should consider the effect of strain of the fuel and cladding.

NRC RESPONSE:

The models for calculating initial stored energy and gap heat transfer during the LOCA should indeed consider thermal and other factors which cause elastic and plastic strain (deformation) of the fuel and cladding.

COMMENT (Ref 6.3, 6.4, 7.3):

The reviewers commented on the temperature range over which the metal-water reaction rate correlations are applicable.

NRC RESPONSE:

The language of the draft regulatory guide could have been better structured. For cladding temperatures above 1900°F, metal-water reaction data are well correlated by the Cathcart-Pawel rate equation (Ref. A-11). For temperatures below 1900°F but above 1500°F, no specific correlation was recommended. The Cathcart-Pawel model is not best-estimate in this range. For temperatures below 1500°F, the Biederman model has been deleted. Therefore, all calculations below 1900°F should be conducted with a model validated by data comparisons in the appropriate temperature range.

QUESTION 3:

Does Appendix B of the draft regulatory guide appropriately discuss the features of the best-estimate codes and the estimation of overall calculational uncertainty?

COMMENT (Ref 2.5):

One reviewer commented that Appendix B is insufficiently developed for a complete statistical approach with which to calculate uncertainties. Furthermore, it does not add to the information on how to interpret the regulation.

NRC RESPONSE:

The regulatory guide is supported by the information in draft NUREG-1230. This information will become formally available when NUREG-1230 is published in final form. The staff desires to keep the description of acceptable uncertainty methodologies general so that the licensee and vendor may develop methods appropriate to their codes. The staff is aware that many licensees and vendors have developed their own uncertainty methodologies.

COMMENT (Ref 4.2):

A reviewer suggested that conservative combinations of input parameters be allowed in best-estimate calculations provided their impact upon the calculated result can be quantified and the total calculational uncertainty can be adjusted to take this impact into account.

NRC RESPONSE:

This approach is acceptable to the staff provided the licensee or vendor can quantitatively show that the conservatisms do not function to compensate errors and that the important phenomena relevant to each transient analyzed are predicted acceptably well. Note that the guide does not prohibit this approach (Section 1.2.2).

COMMENT (Ref 4.3):

One reviewer pointed out that the regulatory guide proposed the use of 95% as the probability of not violating the limits of 10 CFR 50.46(b) for specific applications of best-estimate ECCS analyses; however, the guide did not discuss the confidence level associated with this probability. The reviewer asked whose responsibility it would be to assign the confidence interval and would that interval have to be justified as part of each licensing application.

NRC RESPONSE:

The NRC agrees that the draft regulatory guide is silent in its treatment of confidence limits. Therefore, the final guide has been modified to describe an acceptable statistical technique which does not require the utilization of confidence limits. Other techniques which account for uncertainty in a more detailed manner may also be used. These techniques may require the use of confidence levels which are not required by the NRC suggested approach.

COMMENT (Ref 5.2 and 8.6):

Two reviewers commented that the guide should note that the goal of the uncertainty analysis is to ensure that the calculated PCT is at least at the 95% probability level, rather than to quantify exactly the degree of uncertainty.

NRC RESPONSE:

The purpose of the uncertainty analysis is as stated by these reviewers and was stated in Section 2.1 of the guide. Nevertheless, the analyst must perform a thorough uncertainty analysis before concluding that the calculated PCT is within the desired limits. The staff believes that the licensee and vendor should not be required to conform to a rigorous set of probability values; rather, the licensee should be allowed to work within a set of constraints that assures the safety of the public.

COMMENT (Ref 6.5):

Section 2.5 (Uncovered Bundle Heat Transfer) should be rewritten to reflect the reality of liquid in the flow field when the bundle is uncovered.

NRC RESPONSE:

The Uncovered Bundle Heat Transfer section in the draft guide addresses steam cooling only. The guide addresses liquid entrainment in Section 1.2.9, Post-CHF Heat Transfer.

COMMENT (Ref 6.6):

One reviewer commented that the fuel model need not consider changes in the fuel properties due to burnup during the course of the LOCA, since they do not change significantly during the course of a LOCA.

NRC RESPONSE:

The NRC staff agrees that the effect of fuel burnup during a LOCA is negligible compared to the effects of burnup prior to the postulated accident. One of the purposes of the fuel models is to calculate the steady-state temperature distribution and stored energy in the fuel before the postulated LOCA.

COMMENT (Ref 6.7):

The reviewer states that Section 1.2.4 is unclear regarding the requirement that cladding swelling models take into account asymmetric deformation. The reviewer requests that the direction of asymmetry be clarified. Additionally, the reviewer states that the guidance would be unreasonable if the asymmetry of the deformation is in the azimuthal direction of one fuel rod.

NRC RESPONSE:

The staff believes that the inclusion of asymmetric deformation capabilities in a cladding swell model will result in the closest approach to a true best-estimate calculation. However, the staff also

recognizes the difficulties encountered in developing computer codes that can calculate azimuthal,asymmetric cladding deformation. Since the purpose of this guide is to provide guidelines for performing best estimate analyses of ECCS performance, the staff concedes the use of codes that can calculate symmetric deformation only. As more sophisticated computer codes become available, the staff will decide if the added sophistication needs to be considered in design basis analysis. In the interim, the licensee and vendor should attempt to quantify the uncertainty associated with only calculating symmetric deformation of the cladding only.

COMMENT (Ref 6.8):

The reviewer states that noding studies are most often performed for areas of high fluid velocity or complex fluid flow. Therefore, there is no basis for providing noding studies in areas of large thermal non-equilibrium.

NRC RESPONSE:

The break flow in a large break LOCA begins with high fluid velocities, and ECCS injections generally result in the mixing of highly subcooled fluid with a two-phase, liquid-vapor mixture (which can result in a metastable state). Many studies of break nodalization sensitivities have been performed that show nodalization significantly affects these types of flow configurations. To clarify the intent, the wording of the regulatory guide has been changed to reflect the reviewer's comment.

COMMENT (Ref 6.9):

The reviewer states that detailed modeling of flow blockage significantly reduces or eliminates the heat transfer penalty traditionally associated with flow blockage. Since research shows that the effect is minimal, there is no reason to require calculation of the effect of flow blockage on heat transfer.

NRC RESPONSE:

The staff requires that the effect of flow blockage on total heat transfer be quantified for a best-estimate analysis, because the analysis is no longer bounded by the conservatisms found in Appendix K of 10CFR50. Detailed calculations of flow blockage heat transfer may not be necessary if the licensee or vendor can show that the results of the research are applicable to the specific fuel bundle design.

COMMENT (Ref 6.10):

The reviewer states that the regulatory guide should list the names of specific codes instead of code families. Additionally, the reviewer questions the inclusion of the FRAP family of codes in the list of best estimate codes because the FRAP codes are not considered to be thermal-hydraulic analysis tools. The reviewer also suggests that containment codes should be included in the list.

NRC RESPONSE:

The intent of listing families of codes was to keep the regulatory guide as general as possible to allow the greatest degree of flexibility in the selection of the analysis tools. The fuel behavior codes (e.g., FRAP codes) are included in this list because the licensee and vendor must be able to analyze the effect of periods of reduced core cooling on fuel rod cladding integrity and to estimate fuel initial conditions at the start of the postulated accident. The staff agrees that a list of suggested containment codes might be useful but has decided not to include them because they do not substanially contribute to the calculation of ECCS performance or the uncertainty.

COMMENT (Ref 6.11):

Section 2.2 states that code comparisons should account for limitations of the measurements and calibration errors. The reviewer states that including these items in code uncertainty may result in

double accounting of uncertainties when the uncertainty of the same variables is included in other sources of uncertainty.

NRC RESPONSE:

The limitations of measurement and calibration errors should be considered in the data qualification phase prior to acceptance of the data for comparison with code results. Any uncertainties in data should be considered when comparing code results, especially when those uncertainties could result in an approach to the upper PCT limit. Where it can be shown that double accounting for uncertainties occurs, proper adjustments may be made to correct the estimate of the uncertainty.

COMMENT (Ref 7.4):

The reviewer states that a finite break opening time should be allowed for ECCS performance as it is for other portions of LOCA analyses. Evaluation of uncertainties in break opening time would then be included in the overall model uncertainty evaluations.

NRC RESPONSE:

The commenters approach implies the utilization of the "leak before break" concept. The recent revision to GDC-4 specifically stated that ECCS design would not be affected by the use of "leak before break" as now allowed in other areas by the new version of GDC-4. However, the commission, in a separate rulemaking action, is soliciting comments on expanding this technology to ECCS design application. Until this separate rulemaking is complete, it would be premature to allow this technology to be applied to best estimate ECCS analysis.

COMMENT (Ref 8.5):

The reviewer states that the sections on ECC Bypass (1.2.5.2), Core Flow Distribution During Blowdown (1.2.12), and Containment

Pressure Calculations (1.2.13) should be annotated to indicate that they are either not necessary or do not require the detail specified for small break LOCA evaluation models.

NRC RESPONSE:

The staff does not agree that these models should be given blanket exclusion for small break LOCAs. These models may be required depending on the transient scenario. They should not be ignored until they are determined to be of no consequence for a specific postulated accident.

COMMENT (Ref 9.1):

A reviewer stated that the regulatory guide forces the licensee to assume too much of the burden of proving that the codes in the regulatory guide are really best estimate.

NRC RESPONSE:

In formulating the rule, the staff recognized that this might be a concern for some licensees. It is for this reason that the rule permits the licensee to choose whether it will perform a best-estimate ECCS evaluation or continue to use the existing Appendix K methodology. However, most of the commenters already use best-estimate codes for various analyses, and some are participating in programs to develop uncertainty methodologies for their codes. Many licensees have already assumed some of the burden of proof voluntarily because they recognize the necessity of understanding the limitations of their codes before they commit substantial analysis resources to a project.

COMMENT (Ref 9.2):

The reviewer states that Appendix B does not appropriately discuss the features of a best-estimate code and overall calculational

uncertainty. Specifically, the guide fails to define what qualifications should actually certify a best-estimate code.

The reviewer believes that the following questions need to be clearly addressed in the regulatory guide:

a. Does the guide imply that a code can be certified as best estimate if the code uncertainty analysis results in a relatively small uncertainty?

b. Does the guide imply that industry-developed codes cannot be considered as best-estimate because of their relatively larger uncertainty?

c. What is the borderline between best-and next-to-best estimate codes?

d. Should there be application categories so that certain codes are acceptable for a limited category of application?

NRC RESPONSE:

The staff intended that the regulatory guide should not prescribe specific measures to be taken in certifying a code as best-estimate. This approach is taken because often regulatory guides are interpreted as de facto requirements. The staff desires to keep the description of acceptable uncertainty methodologies general so that the licensee may develop methods appropriate to their codes. The staff is also aware that many licensees and vendors have already developed their own uncertainty methodologies. A suggested methodology for quantifying the uncertainty of several NRC-sponsored codes is presented in NUREG-1230.

COMMENT (Ref 9.3):

The reviewer states that Appendix B Sections 1.2.2, 1.2.13.1, 2.1, and 2.3.1 are unclear regarding what is an acceptable or

preferred method of factoring equipment availability into the best-estimate calculations and uncertainty analysis.

NRC RESPONSE:

Most best-estimate calculations include a series of sensitivity studies to determine the effect of equipment unavailability and degraded/enhanced equipment performance. These sensitivity studies are then used to bound the best-estimate calculation. Often, the sensitivity studies do not require the extensive use of a best-estimate code; rather, an engineering analysis is used to estimate the bounds. The staff believes that the licensees and vendors possess the engineering resources to perform these types of analyses.

ENCLOSURE H

NUCLEAR REGULATORY COMMISSION

[10 CFR PART 50]

Acceptance criteria for Emergency Core Cooling Systems;
Environmental Assessment and Finding of No Significant Impact

The U.S. Nuclear Regulatory Commission (The Commission) is revising § 50.46 and Appendix K of 10 CFR Part 50 which specify requirements of emergency core cooling systems (ECCS) for light water reactors.

ENVIRONMENTAL ASSESSMENT

IDENTIFICATION OF ACTION TO BE TAKEN:

Section 50.46(a)(1) has been revised to eliminate the requirement to use the features of Appendix K when calculating ECCS performance during a loss-of coolant accident (LOCA). Section 50.46(a)(1)(i) of the final rule allows use of realistic analytical techniques and requires that the uncertainty of the calculation be evaluated and considered when comparing the results of the calculation with the temperature limits and other criteria of § 50.46(b). Section 50.46(a)(1)(ii) has been added to allow continued use of the features of Appendix K as an alternative to the uncertainty evaluation required by § 50.46(a)(1)(i). Sections 50.46(a)(2) and 50.46(a)(3) have been revised to eliminate historical implementation sections and to specify requirements for reanalyses and reporting which are excluded from consideration in this environmental assessment per § 51.22 of 10 CFR Part 51. Appendix K of 10 CFR Part 50 has been revised to make minor technical changes to the acceptable features of the calculations.

NEED FOR ACTION:

The revisions of 10 CFR Part 50 and Appendix K are required in order to permit new knowledge of ECCS performance gained through research to be used in the calculations of ECCS performance. The

improved calculations allow relaxation of restrictions which are preventing optimal operation of some reactors and are not necessary to adequately protect the health and safety of the public.

ENVIRONMENTAL IMPACTS OF THIS ACTION:

If power levels remain unchanged, the revisions will reduce the cladding temperatures that are calculated during a LOCA. If peak calculated cladding temperatures are allowed to remain at or below regulatory limits, then the peak local power of the reactor can be increased. An increase in the allowed peak local power can be used in either or both of the following manners:

1. The total maximum allowed power of the reactor remains unchanged, but plant efficiency can be improved by increased flexibility in the allowed power shape. More efficient fuel utilization, more flexibility in changing power shape, and reduced derating of plants due to fuel limits are possible.

2. The total maximum allowed power of the reactor can be increased. The expected maximum increase in total power for existing and currently planned reactors is approximately 5% based on practical limits of plant hardware.

Either of these actions require an amendment to the plant license to change the technical specification limits and, therefore, result in an environmental assessment specific to that particular plant and the specific amendment being considered. This environmental assessment is a generic evaluation considering the typical impact of the rule revision.

A change in the allowed peak local power, without an increase in total power, produces no significant environmental impact. The total fission product inventory, routine releases of radioactive materials and thermal releases to the environment are essentially unchanged. Fuel cycle changes are in the direction of improved use of fuel and should not significantly change the environmental impact of the fuel

cycle unless major new fuel cycle methods (e.g., plutonium recycle) are adopted. Such changes are beyond the scope of this rule revision and are not considered.

For the case of a small (i.e., 5%) increase in total power, there is a correspondingly small increase in fission product inventory, routine releases of radioactivity and fuel use. However, maximum allowed releases of radioactivity during both accident situations and routine operation are specified by technical specifications and other sections of 10 CFR Part 50 which are unchanged by this rule. It is not expected that the small increase in total power that could result from this revision would result in difficulty in meeting the existing release limits. An increase in total power increases the thermal discharge to the environment by an amount approximately proportional to the increase in power. The discharge of heat to surface waters is regulated under the Clean Water Act by the U.S. EPA or designated state agencies. NRC defers to procedures under that Act to establish the acceptability of any increase in waste heat discharge. It is not intended that NRC approval of increased power level affect in any way the responsibility of the licensee to comply with requirements of the Clean Water Act. These being the only potential environmental considerations, the NRC staff believes that site specific environmental impact assessments will not be of help to the decision process.

ALTERNATIVES TO THIS ACTION:

The staff has considered a number of alternatives to revise § 50.46 and Appendix K. However, all the alternatives considered would allow similar increases in local or total power, with the exception of the alternative of making no changes. Since the environmental impact of the proposed revision is considered to be not significant and the revision is required to reduce unwarranted restriction on the operation of some reactors, the Commission has proceeded with the final rule revision.

AGENCIES AND PERSONS CONSULTED:

The NRC staff consulted U.S. manufacturers of nuclear power plants to determine the maximum increase in local or total power that might result from application of the proposed rule revisions. The staff did not consult other agencies or persons.

FINDING OF NO SIGNIFICANT IMPACT

The Commission has decided not to prepare an Environmental Impact Statement for this action. The foregoing environmental assessment of this action has concluded that the proposed action does not significantly effect the quality of the human environment.

Dated at Rockville, Maryland this 1st day of June, 1988.

For The Nuclear Regulatory Commission

Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

ENCLOSURE I

REGULATORY ANALYSIS FOR

REVISION OF THE ECCS RULE

AND SUPPORTING REGULATORY GUIDE

REGULATORY ANALYSIS FOR REVISION OF THE ECCS RULE AND SUPPORTING REGULATORY GUIDE

This regulatory analysis provides the value/impact analysis for both the ECCS Rule and the supporting regulatory guide. The calculations performed in support of this analysis utilized information in the regulatory guide as well as the final rule.

1. STATEMENT OF THE PROBLEM

1.1 Background

Section 50.46 of 10 CFR Part 50 requires that calculations be performed to show that the emergency core cooling systems (ECCS) will adequately cool the reactor in the event of a loss-of-coolant accident (LOCA). Appendix K sets forth certain required and acceptable features that the evaluation models, used to perform these calculations, must contain. The results of these calculations are used to determine the acceptability of the ECCS performance. In many instances, these calculations result in technical specification limits on reactor operation (e.g., peak local power) in order to comply with the 2200°F cladding temperature limit and other limits of § 50.46. These limits restrict the total power output and optimal operation of some reactors in terms of efficient fuel utilization, maneuvering capability and surveillance requirements.

The NRC, DOE (including AEC and ERDA), U.S. nuclear industry and foreign research on ECCS performance since the present ECCS rule was issued provides a technical understanding which shows that the existing ECCS rule restrictions are more stringent than required to protect the health and safety of the public.

1.2 Discussion of the Rulemaking

This final rule and supporting regulatory guide allow alternative methods to be used to demonstrate that the ECCS would protect the nuclear reactor core during a postulated design basis loss-of-coolant accident (LOCA). While continuing to allow the use of current Appendix K methods and requirements, the rule and regulatory guide also allow the use of more recent information and knowledge currently available to demonstrate that the ECCS would perform its safety function during a LOCA. Procedural changes have also been made to relax requirements for certain reanalyses which do not contribute to safety. This regulation applies to all applicants for and holders of construction permits or operating licenses for light water reactors.

The effect of this rule change will be to reduce the peak cladding temperatures that are calculated to occur during a LOCA. This will permit some plants to increase allowed peak local power by increasing allowed peaking factors and/or increasing total power. This regulatory analysis discusses the effect of the rule change in terms of "value" (e.g., public benefits, such as safety) and "impact" (e.g., consequences, such as costs). The intent of the rule and regulatory guide is to reduce the prescriptiveness, which unnecessarily restricts applicants and licensees (negative impact), while continuing to ensure the health and safety of the public.

This final rule and regulatory guide will result in a significant flexibility and cost benefit for some plants. However, application of the rule will be optional, so that each applicant or licensee may perform their own analysis to determine whether or not they should avail themselves of the greater flexibility permitted. The value of this rule may have some negative aspects, since an increase in plant power could increase risk to the public. This increase in risk will be shown to be negligible.

Information used in this regulatory analysis was obtained from a number of sources:

(1) Previous studies sponsored by DOE.[1,2]

(2) Formal responses from the major reactor vendors to a request by NRC for input.[3,4,5]

(3) Informal discussions with reactor vendor, utility, national laboratory, and NRC staff.

In summary, this regulatory analysis will not be the same as a conventional value-impact analysis, because this rule reduces the prescriptiveness of former ECCS regulations in order to remove unnecessary operating restrictions. It is expected that a significant effect will be a cost savings to industry. While there may be a slight negative effect on risk, this effect may be offset for some plants. This rule will continue to provide a more than sufficient safety margin to ensure the health and safety of the public.

2. OBJECTIVES

The objective of this final rule and regulatory guide is to incorporate into the regulations the improved knowledge gained from recent research on ECCS performance, so as to remove unnecessary operating restrictions.

3. ALTERNATIVES

A number of alternative approaches have been considered by the staff, and each approach has been evaluated in terms of safety, impact on the industry, NRC and industry resources required, and the possibility of challenge both during the rulemaking process and during application of the rule. The alternatives that were considered are:

A. Retain the existing rule with its present conservatism (no change).

B. Modify the ECCS rule, as stated in the advance notice of proposed rulemaking published in the Federal Register on December 6, 1978.

C. Modify only certain models contained in Appendix K for which research investigations have been completed and a well-documented data base exists. These changes have been selected in areas for which new experimental data have shown that the existing models contain a larger degree of conservatism than justified by current data uncertainties or are obviously unrealistic.

D. Eliminate the requirement to use Appendix K models and allow realistic models to be used. Reduce the 2200°F and 17% oxidation limits of § 50.46 appropriately to ensure that sufficient conservatism exists to cover uncertainties in the realistic calculation.

E. Eliminate the requirement to use Appendix K models and allow realistic models with an evaluation of the uncertainty in the overall calculation added to the results, similar to that discussed in SECY-83-472,[8] to be used. The § 50.46 limits of 2200°F and 17% oxidation would be unchanged.

These alternatives are discussed in Section 5.

4. CONSEQUENCES

Since licensees can, at their discretion, continue under the existing requirements, one possibility is that a licensee's costs and benefits would be unaffected by this proposed action. It is only under those circumstances where a licensee's own analysis demonstrates a net benefit that the relaxation of these requirements would be adopted. Therefore, this regulatory analysis of the relaxed requirements is not as specific in the identification and quantification of attributes as most regulatory analyses.

4.1 Safety and Risk Effects

The value of this rule must be evaluated in terms of the effect on safety. The rule will probably result in increased local power within the reactor core and possibly in increases in total power. Power increases on the order of 5% will have an insignificant effect on risk. One effect of increased power will be to increase the fission product inventory in the fuel rods. A five percent power increase would result in a no more than a five percent increase in fission products. Therefore, the upper bound on fission products which would be released during core melt scenarios and potentially released to the environment during severe accidents would increase by no more than five percent.

This final rule still requires that the calculated fuel rod peak cladding temperature (PCT) remains below 2200°F. However, those reactors with power increased by five percent could be operating with less margin between the PCT and the 2200°F limit than previously. The increased risk represented by this decrease in margin and increase in fission product inventory is negligible and falls within the uncertainties of probabilistic risk assessment (PRA) risk estimates.[9] In addition, other safety limits, such as departure from nucleate boiling (DNB), and operational hardware limits, such as turbine design, will limit the amount of margin reduction permitted under the revised rule.

There are also safety benefits derivable from alternative fuel management schemes that could be utilized when this rule is implemented. An important safety benefit will be realized due to less stringent restrictions on core power peaking. High neutron leakages at the core outer boundary are used in PWRs to flatten the radial core power profile. This inefficient fuel management procedure is needed to maintain peaking within tight limits. In addition, the resultant high neutron fluence leakage can enhance vessel embrittlement resulting in pressurized thermal shock (PTS) concerns. The higher

power peaking factors that will be allowed will provide greater fuel management flexibility when attempting to reduce neutron flux at the vessel. This can result in a corresponding reduction in risk from PTS.

The reduced cladding temperatures that would be calculated under this rule offer the possibility of other design and operational changes that may result from the lower calculated temperatures. ECCS equipment numbers, sizes, or surveillance requirements might be reduced and still meet the ECCS design criteria (if not required to meet other licensing requirements). The diesel/generator start time duration could be increased from the current technical specification limit of 10 seconds, typical for BWRs, to up to 70 seconds.[7] These two potential modifications could result in offsetting effects on risk. On one hand, elimination of an ECCS component would tend to increase risk due to loss of redundancy (single failure criteria still required); however, increasing the time for diesel/generator start-up may result in improved operational flexibility and diesel reliability with concomitant decrease in risk. The improved diesel reliability could result from decreased stress, wear, and possibly test frequency during surveillance testing, as well as the effect of the relaxed design requirement (70- versus 10-second start-up). In either case, any proposed changes in technical specifications of the facility would be at the option of the licensees and would include the consideration of the financial burden associated with technical specification changes.

In summary, the effect of the rule on safety will have both positive and negative aspects. The potential for reduction of ECC system requirements in existing or new plants is present, but the likelihood of reduction of safety systems in operating plants is extremely low. However, several safety benefits will also be realized under the final rule. The net effect on risk is believed to be plant specific. The probability of a large break is so low, however, that the choice of best estimate versus Appendix K has little effect on public risk as shown in a limited generic analysis of the effect on safety.[9]

4.2 Costs and Benefits

LOCA considerations resulting from the current requirements are restricting the optimum production of nuclear electric power in numerous ways. These restrictions can be placed into the following three categories:

(1) Maximum plant operating power.

(2) Operational flexibility and operational efficiency of the plant.

(3) Availability of manpower to work on other activities.

Maximum plant operating power at some nuclear facilities is limited by the present Appendix K licensing results. However, it can be very difficult to clearly separate these LOCA rule restrictions from other licensing issues and limitations. There are numerous limits that can restrict total plant power, as well as the ability to maneuver the power over a wide range. Typically, this limit is associated with either the PCT calculated to occur during LOCA transients or as a result of DNB restrictions. Additionally, there are limits to plant power because of NRC guidance on total allowable thermal power of 3800 MW and because of physical hardware limits in existing reactors on the balance of the plant (turbine, condenser, pumps, and steam generators).

Regarding the second category of operational flexibility, some plants have very little LOCA margin. Such a limited margin necessitates additional core power surveillance to prevent peaking factor violations. This may also require special supplementary nuclear or safety analyses and restrictive fuel management schemes, resulting in inefficient fuel burnup and no extended burnup cycles.

The third category concerns periodic reanalyses which are required by the current rule. If an error (± 20°F change in PCT for the limiting transient) is found in an accepted evaluation model, a new LOCA analysis must be performed immediately even if the error correction results in a decrease in PCT in the limiting transient. This was a major problem several years ago, and several cases have also occurred recently, with each reanalysis costing the licensee about $150K[1] and diverting both licensee and NRC staff from other, more productive activities. One NRC staff year is normally needed to perform a reanalysis review when required because of errors discovered or because of other moderate changes to existing analyses. A current estimate of the incremental cost corresponding to one staff year of review time is $72K. New models and analyses would require about 3-4 staff years for review. Very often, this reanalysis contributes very little to plant safety. This final rule requires that a complete reanalysis be performed whenever changes to the evaluation model result in changes to the calculated peak clad temperature exceeding the original prediction by ±50°F. If the new limiting transient calculation exceeds the criteria of § 50.46(b), then immediate steps must be taken to achieve compliance. If the criteria of § 50.46(b) are not exceeded, a reanalysis will be performed on a schedule proposed by the licensee and approved by the NRC.

The degree to which the rule will benefit a particular plant depends on how limited the plant is by these LOCA restrictions. The Babcock and Wilcox (B&W) and Combustion Engineering (CE) companies have informally indicated that they do not feel that the plants which they design are limited by LOCA restrictions and, therefore, B&W and CE plants may not benefit from the first two categories. General Electric Co. (GE) plants do tend to be limited in operation by LOCA restrictions and will benefit from relief from LOCA restrictions. However, this relief is already available for most GE plants through the recently approved SAFER evaluation model. Any additional relief

due to a rule change will be of little further benefit.[4] Westinghouse (W) plants are the only plants which appear to directly benefit from relaxation of LOCA limits. Forty-nine operating reactors, as well as ten under construction, are of the Westinghouse design. W indicates that most of these plants are limited by LOCA considerations.[3]

It can, therefore, be estimated that there are at least 49 nuclear plants on line that are limited by LOCA considerations either in total power and/or in flexibility of operation. Up to ten additional plants being limited by LOCA considerations may also come on line. Any rule change that produces a PCT decrease of 100°F can be translated into a total plant power increase of approximately 5%, based on LOCA limit considerations. This 5% increase in power represents a power increase which is within the capabilities of typical W plants based on existing hardware and is still well below other limits (e.g., DNB limits). Calculations show that this final rule change will provide a reduction in PCT of more than 100°F.[3,11,12]

The economic impact of this increase in power can be viewed in terms of energy replacement cost savings.[13] Since Westinghouse plants would be most likely to implement these power upgrades, an analysis has been performed to determine the present values of the energy replacement cost savings which would be derived over the remaining life of each operating Westinghouse plant. The analysis was performed for the 49 plants currently operable and used the following assumptions:

1. Replacement energy cost penalties are assumed to be constant in real terms over the remaining useful life of the reactor. This means that costs are not assumed to increase faster than the rate of general inflation.

2. The commercial operating life of a reactor is assumed to be 30 years. Thus, the remaining useful life of a reactor equals 30 minus the number of years in operation prior to 1987.

3. All costs will be expressed in 1985 constant dollars and discounted back to 1985 assuming a 10% real discount rate. The final cost estimate will represent a 1985 present worth value in 1985 dollars.

4. The average cost to upgrade equipment in order to increase power is $150/kWe.[5]

Table A presents the average daily energy replacement cost for each Westinghouse plant, the yearly energy replacement cost savings resulting from a five percent power increase, and the present value (1985 dollars) of these cost savings over the remaining life of each plant. The total present value of the energy replacement cost savings for a 5 percent power increase in all 49 Westinghouse plants is estimated to be $2.8 billion dollars. Expressed in 1987 dollars, this figure converts to $3.0 billion dollars. As part of a sensitivity analysis, a 5% real discount rate applied to the same plants resulted in a total present value of energy replacement cost savings of $4.2 billion dollars in 1985 dollars. Neither of these values includes the cost to upgrade plant equipment, which is small compared to the present value of the savings for most plants.

TABLE A. PRESENT VALUE OF ENERGY REPLACEMENT COST SAVINGS DUE TO A 5% POWER INCREASE (Millions of 1985 Dollars)

	*Average Daily Energy Replacement Cost	**Yearly Energy Replacement Cost Savings	Present Value of Cost Savings Over Plant Life
Beaver Valley 1	.191	3.48	29.1
Beaver Valley 2	.208	3.80	35.8
D.C. Cook 1	.255	4.64	38.1
D.C. Cook 2	.265	4.83	44.2
Comanche Peak 1***	.736	13.43	126.6
Comanche Peak 2***	.736	13.43	126.6
Salem 1	.482	8.79	74.4
Salem 2	.494	9.02	80.9
Braidwood 1	.574	9.61	90.6
Bryon 1	.556	10.45	97.3
Bryon 2	.494	9.01	85.0
Callaway 1	.313	5.72	53.1
Kewaunee 1	.150	2.73	21.9
Point Beach 1	.145	2.64	18.7
Point Beach 2	.145	2.64	20.1
Zion 1	.526	9.60	75.1
Zion 2	.526	9.60	75.1
Prairie Island 1	.126	2.30	18.0
Prairie Island 2	.129	2.35	18.9
Ginna 1	.271	4.95	35.2
Haddam Neck 1	.281	5.12	29.5
Indian Point 2	.466	8.50	68.2
Indian Point 3	.517	9.43	78.8
Millstone 3	.620	11.32	105.9

* Includes plant-specific capacity factors and power.[10,13]

** Yearly savings associated with a 5% power upgrade.

*** Based on assumed 1987 start-up using 1984 dollar estimates adjusted to 1985.

TABLE A. (cont.)

PRESENT VALUE OF ENERGY REPLACEMENT COSTS

DUE TO A 5% POWER INCREASE (Millions of 1984 Dollars)

	*Average Daily Energy Replacement Cost	**Yearly Energy Replacement Cost Savings	Present Value of Cost Savings Over Plant Life
Catawba 1	.313	5.71	53.0
Catawba 2	.418	5.73	53.6
Seabrook 1	.642	11.71	109.6
Yankee Rowe 1	.090	1.64	5.2
Farley 1	.396	7.22	61.4
Farley 2	.391	7.14	64.1
Harris 1	.340	6.20	58.5
McGuire 1	.440	8.04	72.1
McGuire 2	.440	8.04	74.1
North Anna 1	.343	6.27	52.4
North Anna 2	.344	6.28	35.6
H.B. Robinson 2	.242	4.42	32.6
Sequoyah 1	.196	3.58	32.1
Sequoyah 2	.203	3.70	33.6
Surry 1	.284	5.19	39.5
Surry 2	.282	5.15	40.3
Turkey Point 3	.350	6.39	48.6
Turkey Point 4	.3530	6.44	50.4
V.C. Summer 1	.324	5.92	54.6
Watts Bar 1	.186	3.40	28.7
Wolf Creek 1	.306	5.59	51.9
Diablo Canyon 1	.701	12.78	118.7
Diablo Canyon 2	.722	13.18	123.4
San Onofre 1	.288	5.26	34.2
Trojan 1	.380	6.94	58.0

* Includes plant-specific capacity factors and power.[10,13]

** Yearly savings associated with a 5% power upgrade.

Table B provides the present value cost savings for several different increments of power increase and does not include the average cost of upgrading plant equipment. This table assumes a 10% real discount rate.

TABLE B.

CHANGES IN TOTAL PRESENT VALUE COST SAVINGS DUE TO A PERMANENT POWER INCREASE (Billions of 1985 Dollars)

Power Increase	Total Present Value Cost Savings
1.0 %	0.56
2.0 %	1.12
3.0 %	1.68
4.0 %	2.24
5.0 %	2.80
6.0 %	3.36
7.0 %	3.92
8.0 %	4.48
9.0 %	5.04
10.0 %	5.60

These potential cost savings represent hypothetical maximum savings which do not include plant upgrade costs. It is difficult to estimate how many plants would take advantage of such a rule revision and upgrade power. Factors influencing the decision of an individual utility to upgrade a plant would vary and would depend upon the need for additional capacity and other means of obtaining additional capacity, other limits such as environmental factors (thermal pollution), potential local opposition to plant modification, and plant-specific cost-benefit analyses.

The reduced LOCA restrictions would result in further savings due to more efficient plant operation, irrespective of whether or not the total power of the plant was upgraded. These improvements include improved fuel utilization and improved maneuvering capabilities. Core management and advanced fuel management concepts are complicated subjects, and LOCA limits are only one of many factors to consider.[2] Thus, obtaining precise estimates of potential savings is difficult. However, savings of 3 to 6 million dollars per plant per year would not be unreasonable.[5] Even if a utility did not increase power or change fuel management, the simpler generic reload calculations possible with less restrictive LOCA limits would save $250,000 per plant per year.[1]

4.3 Potential Impacts

In contrast to the economic and safety consequences described in section 4.2, there are some detrimental impacts that might result from this rule. Some believe that a rule change might destabilize the present licensing process which is overly conservative but is well known and predictable. However, others believe that the licensing process will not be fully stable until the rule is revised. Any disruption of the licensing process is prevented by "grandfathering," which will give utilities the option of adopting the new analysis methods or continuing with the old Appendix K procedures.

5. DECISION RATIONALE

The various alternatives of Section 3 were considered and evaluated before finally settling on the recommended regulatory action. The considerations follow:

Alt A: Retain the existing ECCS rule with its present conservatism (no change).

PRO: a. The current well established and stable licensing process would be retained.

b. No staff resources would be required for rulemaking.

CON: a. Many plants would continue to be unnecessarily restricted in operation by the current rule.

b. Many licensees would continue to seek relief from restrictions through requests for exemptions or by using the approach discussed in SECY-83-472. Both these approaches are interim measures and should not substitute for revising the rule to make it consistent with current knowledge and practice.

Alt. B: Modify the ECCS rule as stated in the advanced notice of proposed rulemaking published in the Federal Register on December 6, 1978; 43 FR 57157.

PRO: Consistent with previously stated plans and would allow minor changes to be quickly implemented.

CON: a. Substantial changes would be delayed until a later phase.

b. Does not resolve comments received on the advanced notice recommending more substantial changes.

Alt. C: Modify certain models, discussed in Appendix K for which research investigations have been completed and a well-documented data base exists. These changes would be selected in areas for which new experimental data have shown that existing models contain a larger degree of conservatism than justified by current data uncertainties or are obviously unrealistic.

PRO: a. Plants would no longer be limited in operation by current ECCS rule restrictions.

b. This would be in agreement with Commission policy (NUREG-0885) to incorporate the results of research into the licensing process.

CON: a. The revised ECCS rule might not contain sufficient, quantified conservatism to account for calculational uncertainty. Additional analyses would be required to demonstrate that sufficient conservatism remained in the calculation.

b. The ECCS rule would have to be changed in the future to make use of research results or other information which may become available.

Alt. D: Eliminate the requirement to use Appendix K models and allow realistic models to be used. Reduce the 2200°F and 17% oxidation limits of § 50.46 appropriately to ensure that sufficient conservatism exists to cover uncertainties in the realistic calculation.

PRO: a. Maximum use of completed research could be made in licensing to relax unnecessary operating restrictions. This would be in agreement with Commission policy (NUREG-0885) to incorporate the results of research into the licensing process.

b. Licensing models would provide more realistic calculations to allow more accurate determination of the effect of equipment changes or failures and operating procedures.

CON: a. The conservatism used to account for uncertainties would be fixed through the revised § 50.46 limits

and could not be varied to account for more accurate calculations of uncertainty which may be available in the future (i.e., little incentive for further improvement).

b. Additional staff resources would be required to establish fixed conservatisms applicable to all plant types.

c. The introduction of a less prescriptive rule would provide a greater opportunity to challenge licensing amendments.

Alt. E: Eliminate the requirement to use Appendix K models and allow realistic models combined with an evaluation of the uncertainty in the overall calculation, similar to that discussed in SECY-83-472. The § 50.46 limits of 2200°F and 17% oxidation would be unchanged.

PRO: a. Maximum use of completed and future research could be made in licensing to relax unnecessary operating restrictions. This would be in agreement with Commission policy (NUREG-0885) to incorporate the results of research into the licensing process.

b. Licensing models would provide more realistic calculations to allow more accurate determination of the effect of equipment changes or failures and operating procedures.

c. The uncertainty evaluation would quantify the conservatism in the calculations which could change as the accuracy of the calculations improved.

d. The industry and NRC staff are already investing effort to follow this approach.

CON: The introduction of a less prescriptive rule would provide a greater opportunity to challenge licensing amendments.

In all alternatives considered, the current Appendix K would remain available for those applicants or licensees not desiring to use a revised evaluation model.

The staff believes that Alternatives A and B, which would provide little or no change in the ECCS rule, are unacceptable. The ECCS rule should be changed because:

(1) A data base now exists that supports relaxation of the ECCS rule.

(2) A revised ECCS rule would remove unnecessary operating restrictions on plants.

(3) Almost all U.S. research on LOCAs has been completed. The remaining portions of the MIST and 2D/3D programs are expected to provide valuable information for assessment of models, but should not affect the proposed rule.

(4) Nuclear reactor vendors are currently working on future plant designs which would be influenced by the revised ECCS rule.

The staff has also considered Alternative C, which would modify certain models in Appendix K for which research investigations are completed. The revisions considered under Alternative C include:

a. Reanalysis requirements

b. Post-critical heat flux heat transfer

c. Return to nucleate boiling

d. Refill and reflood transfer (steam cooling below reflood rate of 1 inch per second)

e. Fission product decay

f. Metal-water reaction

g. Discharge model

Based on recent supporting analyses performed by vendors and national laboratories, the staff has determined that if Appendix K were to be revised according to Alternative C, it is possible that the remaining overall conservatism in the evaluation models would be on the same order or less than the uncertainty of the calculation.[4] This would be unacceptable, since one could no longer assume that Appendix K contained sufficient conservatism to account for the total uncertainty in the calculation. Thus, use of Alternative C without supporting uncertainty analysis was dropped from consideration.

The staff also considered revising Appendix K in a manner similar to Alternative C, but requiring an additional uncertainty analysis to ensure that the evaluation model contained sufficient conservatism. This option would require two calculations, a realistic calculation with uncertainty analysis and an evaluation model calculation. This option was also rejected, since the licensee would be required to perform two calculations, of which one (i.e., the evaluation model) would provide little benefit to safety.

Based on the current understanding of ECCS performance, the approach of a prescriptive Appendix K is no longer believed to be appropriate. A realistic calculation, taking into account the overall uncertainty in the analyses, is believed to be the correct approach to ensure the safety of the public without unnecessarily restricting applicants and licensees. Thus, the ECCS rule should be revised accordingly and the requirement to use Appendix K eliminated. Therefore, Alternatives D and E were considered, both of which use realistic calculations. The difference between the alternatives is in the treatment of uncertainties. Alternative D would reduce the § 50.46 limits of 2200°F and 17% cladding oxidation to cover uncertainties in the calculation and uncertainties in the point at which substantial core damage would occur. Alternative E would require an uncertainty factor to be added to the best estimate calculation. Alternative D was not selected because (1) the § 50.46 limits of 2200°F and 17% cladding oxidation are believed to be appropriate and conservative limits below which substantial core damage will not occur,[14] (2) the conservatism used to account for uncertainties in the calculation would be fixed and could not be varied to account for more accurate calculations, and (3) further staff analyses would be required to support establishing these limits.

It was decided that Alternative E be adopted. This alternative will require that the licensee show that the criteria of § 50.46 are met using a realistic calculation combined with an evaluation of the uncertainty of the overall calculation. This uncertainty evaluation, combined with the additional conservatism in the 2200°F peak clad temperature and the 17% cladding oxidation criteria, should ensure a negligible risk to the public. This approach to licensing is consistent with the interim method discussed in SECY-83-472, except that the additional Appendix K calculation, which contributes little to safety, would not be required. As discussed in SECY-83-472, the combined conservatism in Appendix K methods resulted in calculated peak clad temperatures of nearly 2200°F. Extensive research has shown

that the Appendix K models provide excess conservatism in the calculated PCT, and thereby afford greater margin than deemed necessary. Realistic calculation methods indicate that the peak cladding temperature during a LOCA ranges from 1400°F to 1700°F. Therefore, the margin of excess conservatism in the current Appendix K appears to be large. Thus, increases in the operating linear heat generation rate can be obtained while continuing to meet the criteria of 50.46(b). Further, Appendix K would remain available (with minor modifications) as an alternative. Therefore, licensees and applicants who neither need nor desire relief from current operating restrictions would have no new requirements and could continue to meet existing Appendix K requirements. The burden of performing new calculations will be placed only on those applicants and licensees who elect to gain relief from LOCA restrictions.

This rule will also provide relief from the reanalysis requirements that do not contribute substantially to safety and will allow use of research data that has been obtained since the current rule was written. The rule will allow applicants and licensees relief from unnecessary operational restrictions resulting from loss-of-coolant accident (LOCA) analyses and still result in an adequate level of conservatism in the ECCS analyses. The net effect will be to allow increased operational flexibility in the form of increased linear heat generation rates and more optimum fuel utilization while retaining the conservative margins set forth in 50.46(b).

6. IMPLEMENTATION

6.1 Schedule

No implementation problems are now anticipated. Each applicant or licensee may, at its discretion, continue to utilize the existing Appendix K criteria for development of the ECCS evaluation model. With regard to the reporting of significant changes to the accepted

evaluation models, the rule provides for the establishment of a mutually agreed upon schedule for completing required actions.

6.2 Relationship to the Existing or Proposed Requirements

In view of the fact that an integrated schedule is to be used for prescribing necessary actions, it is not expected that actions resulting from other requirements will be seriously affected. A backfit analysis, developed primarily from this regulatory analysis has been included in the Notice of Rulemaking.

7. VALUE/IMPACT OF REGULATORY GUIDE

Having decided to proceed with Alternative E for the promulgation of the ECCS rule, the staff considered the question of whether or not a regulatory guide should be prepared and what form such a guide might take.

7.1 ALTERNATIVES CONSIDERED

The staff identified three viable alternatives concerning the preparation of a regulatory guide. Each alternative has been evaluated using the same criteria as those applied to the rule. The alternatives considered are:

A. Do not prepare a regulatory guide and rely on licensees existing expertise with thermal-hydraulic and interactions with NRC licensing staff to ascertain compliance.

B. Prepare a regulatory guide which describes technical areas to be considered, gives examples of acceptable features and data bases of best-estimate models, and generally describes uncertainty estimation methods.

C. Prepare a prescriptive regulatory guide which identifies the best-estimate models, data, and

correlations which should be used and the uncertainty methodology which should be applied.

7.2 DECISION RATIONALE

The various alternatives of section 7.1 were considered and evaluated before deciding upon the approach taken. The considerations follow:

Alt A: Do not prepare a regulatory guide and rely on licensees existing licensee expertise with thermal-hydraulic and interactions with NRC licensing authorities to ascertain compliance.

PRO: a. Great latitude would be afforded the licensee in complying with the rule.

b. NRC staff time to prepare such a guide would be saved.

CON: a. May be more costly to both the NRC and licensee because more potentially unacceptable or questionable licensee submittals would be expected.

b. There is limited experience in performing uncertainty analysis and licensees would be precluded from having NRC's views on how it might be done.

Alt B: Prepare a regulatory guide which describes technical areas to be considered, gives examples of acceptable features and data bases of best-estimate models, and generally describes uncertainty estimation methods.

PRO: a. Retains the flexibility of Alt A but allows for some features to be preapproved as acceptable should licencees want to use them.

b. Less costly for NRC review because many features will be familiar and preapproved.

CON: a. Regulatory guides are sometimes treated as "defacto" regulations by the licensee.

b. Limits NRC flexibility to determine acceptability of models on a case-by-case basis.

Alt C: Prepare a prescriptive regulatory guide which identifies the best-estimate models, data, and correlations which should be used and the uncertainty methodology which should be applied.

PRO: a. Less costly to both NRC and licensee because prescriptiveness will result in less interaction.

b. Compliance should be straightforward because of similarity to existing Appendix K approach.

CON: a. Prescriptiveness will prevent licensees from utilizing current or future safety research in their best-estimate models so that they will not be able to take optimum advantage of the rule.

b. Prescriptivenss could result in specification of features in excess of what is required for safety without giving licensees the opportunity to justify alternatives.

The staff concluded that the disadvantages of Alternative C far outweighed the advantages particularly in light of the problems with the current Appendix K approach. In addition, this approach afforded licensees the least amount of flexibility in complying with the revised rule's features. Therefore, Alternative C was eliminated from consideration. Alternative A was attractive from the standpoint of allowing maximum flexibility but was thought to be the most burdensome to both licensee and NRC because no information about acceptable ways of meeting the requirements would be available. This is seen as a serious deficiency particularly since the uncertainty quantification is relatively new to most licensees. This alternative would be the most disruptive to the licensing process and would most likely cause delays in obtaining NRC approval. Therefore, the staff concluded that by adopting Alternative B optimum flexibility and guidance are provided.

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ENCLOSURE J

Margin Inherent in the 2200°F Limit

The Commission paper (SECY-86-318) which transmitted the notice of proposed rulemaking (Enclosure A of this package) also contained an enclosure (Enclosure C to SECY-86-318) which addressed the conservatism in Appendix K and in 50.46. That document stated that the criteria "of 17% maximum allowed cladding oxidation, combined with the 2200°F limit, is an appropriate limit, ----, which will ensure survival of the cladding following reflood." This conclusion was based on a large body of research mandated by the Commission when 50.46 was originally promulgated. This conclusion regarding the performance limits of 2200°F and 17% oxidation as they relate to cladding embrittlement is well founded and equally valid today.

In SECY-86-318 the staff also addressed the question of the 2200°F limit as it related to the contribution of the metal-water reaction to the burden for heat removal. The staff stated at that time that additional conservatism existed in the 2200°F peak cladding temperature (PCT) limit in Section 50.46 for ECCS analysis. It was further stated that this conservatism was about 400°F based on data from the Power Burst Facility (PBF) severe accident experiments which showed that "at a temperature of 2600°F, the steam-zircaloy reaction becomes sufficiently rapid to produce an auto- catalytic temperature excursion (i.e., cladding oxidation is a self-sustaining process). Thus 2600°F is a real limit, above which significant fuel damage would occur." The 400°F margin between 2200°F and 2600°F was also referred to in the rulemaking notice (Enclosure A of this package). On this basis it was recommended that the 2200°F limit not be modified. Additional experimental and analytical information from PBF and other sources have been examined which show that some rapid temperature rises comparable to the original PBF data begin below and

some above 2600°F. Thus, the original basis for the 400°F margin proposed in SECY-86-318 was questioned. A careful examination was performed for cladding temperature transients in both design basis LOCA analysis and in severe accident experiments and calculations. It was observed that even more rapid temperature rises occur in design basis LOCA analysis beginning at temperatures as low as 1200°F during refill, but these rapid rises are easily arrested with nominal reflood heat transfer. The rapid temperature rise in design basis analyses is due to the high decay heat which occurs early in time, and other conservative assumptions used regarding power peaking factors. The rapid temperature rises observed in severe accident experiments are not terminated because the fluid flowing past the fuel rods and hence the heat transfer is markedly less than that observed in reflood LOCA analysis and experiments. This condition exists because severe accident experiments and analyses are performed with no intentional core cooling in order to achieve and study degraded core conditions. Since severe accident thermal behavior is so different from LOCA behavior there is very little relevance to the issue of performance limit margins. Therefore, the staff does not believe that 2600°F should be characterized as a "real" limit above which phenomena would occur which make significant core damage inevitable. Other recent experiments show that control rod failure can occur below 2200°F in such a way that fuel cladding damage may result. Some of these experiments are discussed in the Compendium of ECCS Research (NUREG-1230). Because control rods are not a heat source, control rod temperatures would be substantially below fuel rod temperatures during a large break LOCA. Only if fuel cladding temperatures were substantially above 2200°F would thermal radiation from fuel to control rods cause the control rods to reach temperatures at which they would fail. To achieve this condition would require a

significant degradation in reflood heat transfer or increase in rod power. Therefore, even though some severe accident research appears to show lower thresholds for temperature excursion or cladding failure than previously believed, when design basis heat transfer and decay heat are considered, some margin above 2200°F exists, but it is too uncertain to quantify at this time. We do not believe any information suggests that the 2200°F limit, and most certainly the 17% reaction limit are the edge of a cliff. Nevertheless, the viability of this rule does not depend on the existence of any margin above 2200°F, only that quantifiable margin exist between the calculated results and the 2200°F limit.