DOCKET NUMBER
PROPOSED RULE RM-50-1

CONCLUDING STATEMENT OF POSITION OF THE REGULATORY STAFF

DOCKETED
USAEC
APR 17 1973
Office of the Secretary
Public Proceedings
Branch

PUBLIC RULEMAKING HEARING ON:

ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER COOLED NUCLEAR POWER REACTORS

DOCKET NO. RM-50-1

U. S. ATOMIC ENERGY COMMISSION
WASHINGTON, D. C. 20545

4096

CONCLUDING STATEMENT

OF THE

REGULATORY STAFF

Acceptance Criteria
for Emergency Core Cooling Systems
for Light-Water-Cooled Power Reactors

April 16, 1973

Docket RM-50-1

U. S. Atomic Energy Commission
Washington, D. C. 20545

4971?

TABLE OF CONTENTS

LJ071F .

CONCLUDING STATEMENT

OF THE

REGULATORY STAFF

Acceptance Criteria
for Emergency Core Cooling Systems
for Light-Water-Cooled Power Reactors

Docket RM-50-1

April 16, 1973

I. INTRODUCTION

A. Regulatory Staff Conclusion

Pursuant to the Board's Order of March 15, 1973 the Regulatory staff here files its Concluding Statement in the above-captioned proceeding.[1/] The staff in preparing this Statement has considered the entire evidentiary record of the proceeding as well as arguments contained in the Concluding Statements filed by the other participants and the various responses commenting on those Statements.

The proposal of the Regulatory staff for resolution of the issue set down for rule making[2/] is the Proposed Rule set forth in Chapter II of this Statement. This Proposed Rule contains the criteria and evaluation models now proposed by the staff for future use.

1/ Pursuant to Commission Order of December 12, 1972, the staff reserves the right to file a Supplement to this Concluding Statement in light of the NEPA portion of this proceeding.

2/ "[W]hether or not the subject interim policy statement should be retained in its present form or adopted in some other form," Commission order dated November 26, 1971, 36 F.R. 22774.

B. Information Basis

In keeping with the decisional premises established by the Commission for the proceeding[3/] the staff has relied in this Statement exclusively on the evidentiary record. Additional information lately proposed[4/] for inclusion in the record has not been used by the staff as a basis for its conclusions, because such information is not in the record and because the information is not complete enough to permit an adequate staff evaluation. Moreover, the staff is of the opinion that ample time was afforded to all participants in this proceeding to submit such information by way of sworn testimony subject to questioning at the times provided for such submission. As the staff has pointed out,[5/] ongoing research and development programs are continuing to develop new analytical and experimental information related to ECCS technology. This is expected to continue; the staff believes it to be important that it does continue.

The staff, correspondingly, will continue its present practice of reviewing all new information as it becomes available.

3/ "Public Rule Making Hearing: Supplemental Notice" dated January 6, 1972 and January 18, 1973.

4/ Concluding Statements of participants Babcock & Wilcox (Appendix B), Westinghouse (pages A-10, A-15), letter Reis to Goodrich, et al. on behalf of Combustion Engineering, February 22, 1973.

5/ Staff Testimony, Sections 1.2 and 1.3; Staff Supplemental Testimony, pages 1-4 and 1-5.

Such changes in ECCS regulations as may be shown to be appropriate as a result of review of new information will, of course, be proposed by the staff, as warranted.

C. Changes with Respect to Interim Policy Statement

Changes in the proposed rule with respect to the Interim Policy Statement (IPS) of 1971 were foreshadowed in the Supplemental Testimony of the Regulatory staff (Exhibit 1113) filed in October 1972. The technical discussions of the Supplemental Testimony have now been related to the entire technical record and the proposals modified, where needed, in Chapter III of this Statement, which gives the staff's reasons for now proposing each item in the Proposed Rule and the staff's consideration of the proposals of the other participants.

The principal changes are the following:

1. Criteria

Peak Cladding Temperature. The new limit of 2200°F replaces the old 2300°F, based on data in the record from zirconium embrittlement experiments.

Maximum Cladding Oxidation. This is a new criterion. It is based on data in the record from zirconium embrittlement experiments.

409

2. Evaluation Models

Cladding Swelling and Rupture. These phenomena are now required to be taken into account explicitly when and where they are calculated to occur. The effects of calculated cladding swelling and rupture are expected to be significant in calculations of gap conductance, oxidation on the outside and inside surfaces of the cladding, core flow during blowdown, and local core flow during the steam-only portion of PWR reflood, and thus to affect significantly the calculated peak cladding temperatures.

Core Flow Distribution During Blowdown (PWR's only). The previously used 0.8 factor relating calculated average core flow to hot channel flow is replaced with a more realistic hot region flow calculation.

Other less significant changes and clarifications have been proposed by the staff to take into account the increase of knowledge of ECCS phenomena, as reflected by the information contained in the record of the proceeding.

D. Application

The staff's proposed application of the Proposed Rule is given in subdivisions (2), (3), and (4) of Paragraph 50.46(a). The proposal requires rapid conformance: Licensees with operating plants are required to make revised ECCS calculations in accordance with the new regulations as soon as possible, but no later than 4 months after publication of the new regulations. In addition, operation of these plants must be modified when the new calculations are submitted to the Commission, if this is necessary to bring them within the new criteria. In this way, their operation will already have been modified as required during the time that the Commission is conducting its review of the new calculations. Of course, if the Commission finds that additional calculations or modifications will be needed, they will be required.

The basic justification for allowing a transition period is the evidence in the record[6/] that the Interim Policy Statement now in force is acceptable on an interim basis to evaluate the performance of ECCS.

The proposed new regulations are believed by the staff to constitute an improvement over the Interim Policy Statement. The only significant change in the acceptance criteria themselves is the replacement of a single temperature limit by a combination of temperature and oxidation limits - a change foreseen in the Interim Policy Statement itself.[7/] The changes in the evaluation

---

6/ Exhibit 1001, sections 1.2.6 and 1.3, Transcript pages 8136-8140; 19,727-9; 19,778-9; 19,801; 19,804-06; 20,400-412.

7/ Section IV.A.1.

models require various aspects of the calculation to be done better, define better the procedures and parameters used, and take better account of the various physical phenomena now known to occur during postulated LOCA's.

It is worth noting that such changes in evaluation models are not to be accomplished overnight, especially in view of the requirements for convergence, noding, and sensitivity studies of the new evaluation models, newly clarified in Section III.A of the proposed Appendix K.

The continued acceptability of the Interim Policy Statement and the need for development and validation of the new evaluation models are the justification and basis of the transition period proposed by the staff.

E. Additional Matters

Procedural Setting

In assessing the technical record developed in this proceeding it is useful to bear in mind the liberal procedural framework for public participation within which this record was made. Over and above the procedural dictates of applicable law in regard to rule making, the Commission, as a matter of discretion, provided for an oral hearing and other procedural features designed to blend "on an experimental basis, limited adjudicatory-type procedures with more traditional rule making processes."[8/] Among the key trial-type procedural rights afforded the participants is "the opportunity for relevant questioning of the witnesses of other participants."[9/] Participants were also required to make "appropriate documents available"[10/] and to "produce on request the documents on which they rely."[11/] Finally, of central importance, is the Commission's own commitment to rely in its rule-making decision on the record of the proceeding.[12/]

The procedures afforded were amply used. Indeed, one of the seven primary participants (Consolidated National Intervenors or "CNI") was accorded over 50 actual hearing days (of the total of 122 thus far) within which to question other participants' witnesses. Moreover, although there was no formal discovery, as such, many hundreds of documents running to tens of thousands of pages were made available. Estimates of the quantity of documents provided the participants run far in excess of 50,000 pages. Nor were the documents made available by the staff selective, in terms of all being supportive of its views. Internal memoranda, properly privileged, were released as a matter of Commission

---

8/ Commission Memorandum of June 16, 1972, page 2.

9/ Supplemental Notice of Hearing, dated January 6, 1972, Rule 1.

10/ *Ibid.*, Rule 3.

11/ *Id.*

12/ Supplemental Notice of Hearing, Rule 2.

discretion.[13] Drafts of drafts and similar materials far beyond the reach of the Freedom of Information Act or normal evidentiary disclosure were voluntarily provided, and copies of the comments of all of the staff's consultants on its draft testimony were made freely available. These comments include reference to areas in which unanimity on technical matters does not exist. The staff's prodigious effort to make any and all documents - however remotely relevant - available, has afforded the participants and the public with an extensive record of a highly complex technical subject. Included within the documentation, as could reasonably be expected, are shadings of technical opinion on various facets of the technical issues involved.

The area of documentary materials was not the only one in which extraordinary effort was expended in order to be certain that the record was as full and complete - with all shades of technical opinion - as possible. Thus, although no subpoena power, as such, was provided for, in addition to presenting thirteen witnesses in support of the staff's position, the staff also presented five witnesses from the Oak Ridge National Laboratory and seven from the Aerojet Nuclear Company. Certain of these witnesses, at least initially, expressed some disagreement with

13/ Commission Order of February 4, 1972.

various technical views of the staff. Finally, the staff made available for the evidentiary record the views of those two members of the involved staff who, in large measure, disagreed with the original staff position.

In connection with the preparation of the staff's Supplemental Testimony, every point of view available was sought. In addition, all of the views expressed during the development of the Supplemental Testimony were carefully documented and then made publicly available. This documentation included not only the detailed comments of the staff's consultants submitted in written form but, in addition, copies of early drafts of the Supplemental Testimony and even over one hundred pages of secretarial notes of oral discussions between the staff and its consultants.[14/] Moreover, during the redirect/rebuttal phase of the proceeding, after having made available ten witnesses for questioning, the staff responded positively to a request that it make available for questioning a group of additional witnesses. Two of these witnesses had nothing whatsoever to do with the staff's Supplemental (redirect/rebuttal) Testimony. However, with respect to the remaining thirteen witnesses requested, the staff stood ready to make such persons available. The party requesting such witnesses did not pursue

14/ Transcript page 20,235.

the matter further when it became apparent that the Hearing Board was, in light of past performances, insisting on reasonable rules concerning relevancy and the scope of questioning.[15]

In all events, the record of this proceeding reflects that, by any objective standards, a full and open hearing has been afforded.

Several blanket allegations of procedural error have been made by one participant.[16] The approach adopted is to assert a "continuing exception to each adverse ruling in this proceeding," but not to identify among the more than 22,000 pages of transcript and hundreds of pages of Commission and Board Orders just where each so-called "adverse ruling" occurs. This posture on asserted errors in the record renders it effectively impossible to meaningfully review the broadside claims of procedural misstep. It is the staff's own view - a view based upon its day to day familiarity with the proceeding and a sensitivity to the possibility of prejudicial error finding its way into the process - that no prejudicial error undermines this record. Obviously, in a record of more than 22,000 pages, a large proportion of which constituted hostile questioning of adverse witnesses by CNI, the possibility

---

15/ Transcript page 21,959. The essence of the Board's ruling was that questioning must be directed to the technical areas on which these witnesses had expressed viewpoints, whether supportive or not.

16/ CNI Concluding Statement, page 25.

of some error in procedural rulings cannot be dismissed. However, nothing which could reasonably be deemed substantial or prejudicial error has been noted. In the absence of particularization of error and substantiation as to prejudicial effect, CNI's generalized assertions in this regard ought not be further considered.

Error in the procedures established for obtaining the views of the Advisory Committee on Reactor Safeguards (ACRS) is also perceived by this participant.[17/] The claim of error is pressed in the face of specific responses to specific interrogatories filed with the ACRS by that participant. Indeed, it is possible that this charge of procedural error emanates from a certain dissatisfaction with one of the ACRS responses to an interrogatory. That ACRS response was to the effect that although certain aspects of ECCS analysis have not yet been "proven to be conservative, the ACRS nevertheless believes that they can be handled in such a manner that there is reasonable assurance that, with the appropriate use of the Interim Acceptance Criteria and other applicable design and evaluation criteria, water reactors of current design can be operated without undue risk to the health and safety of the public."[18/] Be that as it may, the contention that

17/ Concluding Statement of Consolidated National Intervenors, (hereafter "CNI Statement") page 2.14.

18/ Exhibit 1115, page 7.

the Commission should require the 15 members of the ACRS to appear and give testimony in this proceeding was carefully considered by the Commission in its Order of January 26, 1972, prior to the start of the hearing. For reasons grounded in the statutory makeup and functions of the ACRS and set forth in some detail in the Commission Order, the matter of ACRS response to the request for its further views was dealt with in the form of permitting a reasonable number of relevant interrogatories. While we believe this course to be eminently reasonable, given the considerations delineated by the Commission and the manifold avenues open for the development of a full record, declining to order ACRS participation in this proceeding other than by way of interrogatories is purely a discretionary decision which in no way constitutes procedural or substantive legal error.

Rulings on Scope of Proceeding

Of all the more specific claims of procedural error, none is pressed with greater vigor than the claim by CNI that Commission and Board rulings on the scope of the proceeding were wrong. Regrettably this zeal even begets the assertion that the scope rulings of the Board constituted "prejudicial misconduct."[19/] The various rulings as to the scope of this proceeding by the Commission and the Board proceed from reasonable, logical and clearly sound legal bases, as

19/ CNI Statement, page 2.8.

we shall demonstrate below with respect to the individual rulings involved. Apart from the foregoing, however, there is also another important, indeed, almost overriding consideration to be kept in mind with respect to scope rulings in the context of a rule-making proceeding. An issue or subject ruled beyond the scope of this rule-making proceeding is not proper within the scope of any rule that ultimately eventuates. Thus, rulings that have the effect of narrowing or expanding the scope of the rule making merely result in a narrowing or expansion of the potential coverage of any ultimate rule. And the breadth of the rule - be it narrow or broad - cannot be the source of "prejudice" to anyone, since matters ruled within the scope of the proceeding are proper for consideration here, while matters ruled beyond the scope of the present rule making are subject to consideration either in individual licensing proceedings or in the context of a petition for further rule making.

In addition to the fact that scope rulings are "neutral" in the sense of prejudice to any participant, there is an important practical consideration which should be remembered in considering complaints ostensibly engendered by such rulings. There is a practical necessity to place some outer bounds on the matters that can be usefully coped with in a single proceeding. The record of this very proceeding stands as incontrovertible evidence of this need.

40231

The rulings on scope by the Commission and the Board were reasonable and practical and did not prejudice any participant.

Criticism of scope rulings was centered by CNI on some eight general areas.

1. Causes of the loss of coolant accident.[20]

2. Probability that a loss of coolant accident would ever occur.[21]

The first two scope rulings excluded from this hearing "on the intrinsic merits of the ECCS criteria"[22] the peripheral considerations of (i) what mechanism might cause an accident which the emergency core cooling systems would be called upon to control, and (ii) what is the likelihood of such an accident ever happening. These subjects may merit exploration in an appropriate context, but they are wholly beyond the proper scope of a rule-making proceeding which assumes, as its reason for being, that an accident stemming from a specific circumstance has already occurred. This rule making is concerned with the validity of performance criteria for systems designed to control such a postulated accident. In short, relevance is dictated by the fact that an accident with a particular genesis is assumed to occur and the question is whether the ECC system required by the criteria will control it.[23]

20/ CNI Statement, pages 2.7 and 3.18-3.28.

21/ CNI Statement, page 2.7.

22/ Commission Order of February 23, 1972.

23/ Of course, the remote likelihood of such an accident occurring is one necessary pre-condition to the postulation of any radiological impact on the environment, and, as such, it is dealt with in the Staff's Final Environmental Statement.

40732

3. Consequences of a LOCA.[24]

In a category similar to the first two scope rulings, but at the opposite scope boundary, is the subject of consequences of a LOCA. This rule making is focused on the validity of criteria for systems designed to prevent consequences. And again, as fascinating as it may be to dwell on the hypothesized aspects of an uncontrolled core melt down, doing so was ruled[25] beyond the scope of this proceeding because it does not deal with the technical adequacy of the ECCS criteria. If the emergency core cooling system should ever be called upon to cool the core and should it fail to do so, serious consequences could - but not necessarily would - ensue. Nevertheless, the nature of such consequences is not, for the reason just mentioned, the proper focus of this rule making.

4. Defense-in-depth.[26]

The charge[27] that the so-called defense-in-depth concept was excluded from consideration in the proceeding is exaggerated, as reference to the Commission's Order of February 23, 1972 will quickly show. That order, which issued sua sponte because of "the slow pace at which the hearing is moving" and "the relative lack of consideration of substantive issues in the questioning to date," provided "firm

24/ CNI Statement, page 2.7.

25/ See, e.g., Board ruling at Transcript pages 16,595-604.

26/ CNI Statement, pages 2.7, 3.2-3.17.

27/ CNI Statement, page 3.2 et seq.

guidance to the participants as to the purpose and attendant scope of this hearing." With respect to defense-in-depth, the Order says that "[while] a desirable hearing record should contain testimony and questioning regarding these contextual matters [as was in fact the case], the primary focus of record development should be with regard to the technical considerations involved in the ECCS criteria themselves." The basic idea concerning these peripheral issues, however, was "to maintain a sense of proportion and perspective."

5. Steam generator tube failures.[28/]

6. Pressure vessel failures.[29/]

7. Fuel densification.[30/]

In its Order of February 23, 1972 the Commission observed with respect to scope rulings that "the technology and the issues may not present clear-cut answers to questions of inclusion [of matters in the hearing record]." Some cases are clearer than others, however. Thus, where the integrity of steam generator tubes and reactor pressure vessels are covered by other Commission regulations,[31/] such matters are properly excluded from the scope of the present proceeding which

28/ CNI Statement, pages 2.7, 3.1.

29/ CNI Statement, pages 2.8, 3.1.

30/ CNI Statement, pages 2.8, 3.26-3.41.

31/ 10 CFR Part 50 Appendix A, General Design Criteria.

is concerned with emergency core cooling systems' ability to control the consequences of a large pipe rupture, not a break in a pressure vessel or cracks in steam generator tubes, which problems are dealt with elsewhere.

Similarly, the question of fuel densification - a phenomenon which came to light wholly outside the ECCS hearing but while it was still going on - is treated on an ad hoc, case-by-case basis. The staff chapter on this subject in its Supplemental Testimony - included for information purposes - made just this point and was stricken on that basis as beyond the scope of the proceeding. In all events, since the question is not within the scope of this rule making, it is properly a subject for consideration and has been so considered in individual cases that have arisen.

8. ECCS designs and design changes.[32/]

CNI is of the view that limitations imposed during the hearing on exploring individual ECCS designs of particular manufacturers "has no support in applicable law and is, in fact, directly inconsistent with the requirement of the Atomic Energy Act and NEPA."[33/] Just what "the requirement" is which produces the asserted inconsistency remains unspecified. Be that as it may, when "the primary focus of record development should be with regard to the technical

32/ CNI Statement, pages 3.29-3.36.

33/ CNI Statement, page 3.33.

U1935

considerations involved in the ECCS criteria themselves,"[34/] the question of compliance with any particular criteria and the details of the design developed to bring about that compliance are properly excluded from the scope of the proceeding. The criteria do not and would not operate to approve individual designs. Rather, they establish the performance standards which designs, whatever they may be, must live up to. This proceeding is concerned with what the performance standards should be, not with whether an individual design meets the standards, once they are established. This latter question is properly the concern of individual reactor licensing proceedings.

Shades of Technical Opinion

In an Order issued prior to the commencement of the hearing, the Commission observed that "[t]he subject matter of this rule making proceeding is one of technical complexity on which there is variance in technical opinion."[35/] There can be no ignoring the fact that technical opinion on the enormous variety of complex technical subjects involved in ECCS evaluations is not in all cases unanimous. Examples of disagreements with this judgment or that view, both in the direct phase of the hearing and in the redirect/rebuttal phase, can be found in the record. By the same token, however, there can

34/ Commission Order of February 23, 1972, page 6.

35/ Commission Order of January 26, 1972, page 3.

be no ignoring the fact that the vast majority of knowledgeable experts believe the staff's Proposed Rule to be conservative.[36/] Indeed many regard the staff's views as excessively conservative.[37/]

Just as the Commission found it in the public interest to release otherwise privileged internal memoranda, many of which contained criticisms of the position being expressed by the staff, so too the staff deemed it appropriate to present for the record the views of those of its consultants and advisors which differed from the consensus of staff experts. The staff thus sought to present for the record all shades of knowledgeable opinion regardless of whether or not the opinion happened to coincide with the staff consensus. Moreover, when the staff developed its Supplemental Testimony,[38/] it solicited and received views from all of its consultants, spread these views on the record and, where disagreements persisted, explained why the staff holds the view it does. As will be noted from a perusal of Exhibit 1113, there are still some differences among our consultants, even with respect to the more conservative position reflected in our Supplemental Testimony. The remaining differences, however, cannot be fairly characterized as destroying an overall consensus among the staff and its consultants with respect to the conservatism and adequacy

36/ The staff's redirect/rebuttal Testimony, Exhibit 1113, reflects those few instances when anyone at either ANC, ORNL, PNL, or on the staff itself disagrees with a staff technical judgment.

37/ The vendors and the utilities all regard the revised criteria suggested by the staff as conservative. See Final Statements of B&W, CE, GE, Westinghouse and the ECCS Utility Group.

38/ Exhibit 1113.

4093?

of the numerous engineering judgments involved in the staff's Proposed Rule.

Any attempted portrayal of the total picture of informed technical opinion should take appropriate account of the views of the ACRS. Those views, previously adverted to in another connection, are that although certain aspects of ECCS analysis have not yet been proven to be conservative, these aspects can nevertheless be handled in such a manner that there is reasonable assurance that, "with the appropriate use of the Interim Acceptance Criteria and other applicable design and evaluation criteria, water reactors of current design can be operated without undue risk to the health and safety of the public."[39/] Finally, the views of Milton Shaw, Director of the Commission's Division of Reactor Development and Technology, reflect his support for the adequacy of the Interim Acceptance Criteria.[40/]

The Evidence

Although the vast record thus far developed in this proceeding has been marred by excessive focus on peripheral matters[41/] and seemingly interminable arguments among counsel and between counsel and the Board in an atmosphere too closely "akin to a criminal trial

39/ Exhibit 1115, page 7. Identical language also appears in the ACRS letter to Chairman Schlesinger of January 7, 1972 attached to the Commission Order of January 26, 1972.

40/ Transcript page 7183. See also Exhibit 1005, page 2.

41/ See Commission Order of February 23, 1973, page 2.

portrayed in the popular media",[42/] there is, nevertheless, a substantial amount of testimonial and documentary evidence on the central technical issues in the proceeding. Each of the principal participants, including the staff, presented evidence in support of its views. In addition, the staff presented for the evidentiary record certain divergent technical viewpoints on particular technical subject areas.[43/] The open inquiry directed by the Commission in this rule-making proceeding has adduced evidence from the various participants - of diverse qualitative weight - which is designed to support points of view running across the entire decisional spectrum (from support for a peak clad temperature of 2700° at one end to a call for a virtual moratorium on power-reactor licensing at the other). Nevertheless, it is the staff's view that a critical evaluation of all of the evidence in the record of this proceeding as it has developed thus far will show that the reliable, probative and substantial evidence[44/] provides full and firm support for the improvements to the Interim Criteria proposed by the staff.

42/ Commission Order of February 4, 1973.

43/ See, e.g., Exhibits 1043 and 1044.

44/ The evidentiary test of "the preponderance of the evidence", urged by two participants (General Electric Company, Initial Closing Statement, page I-9 and CNI Statement, page 4.47), lacks legal justification. This proceeding, as GE itself has pointed out, is "within the spirit...of rulemaking actions conducted on the record." Such actions are governed by the evidentiary test of 5 U.S.C. 556 which requires that rules growing out of rule makings on a record be supported by "the reliable probative and substantial evidence."

II. PROPOSED RULE

A. The Proposed Statement of Considerations

On June 29, 1971, the Atomic Energy Commission published an Interim Policy Statement: "Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors" (36 F.R. 12247) which was subsequently amended (December 18, 1971, 36 F.R. 24082). The Interim Policy Statement, as amended, includes: (1) general criteria for emergency core cooling systems applicable to all light-water power reactors, (2) requirements for analysis using a suitable evaluation model, (3) provisions for application to various classes of reactors by specified dates, (4) provision for variance under stated conditions, and (5) an appendix delineating acceptable evaluation models.

Following promulgation of the 1971 Interim Policy Statement, AEC reactor licensees and applicants for reactor licenses brought their reactors or designs, as appropriate, into conformance with the published criteria as required by the Statement. (Some "backfitting" of older reactors is still being completed.)

As anticipated in the Statement, and in response to requests, a rule-making hearing was subsequently held "for the purpose of aiding the Commission in its determination as to whether or not the subject interim policy statement should be retained in its present form or adopted in some other form," (36 F.R. 22774, November 26, 1971).

The hearing board consisted of Nathaniel H. Goodrich, Esq., presiding, Dr. Lawrence R. Quarles, and Dr. John H. Buck. The following were participants in the hearing: The Babcock and Wilcox Company (B & W), Combustion Engineering, Incorporated (CE), Consolidated National Intervenors, a group of about 60 organizations and individuals (CNI), The ECCS Utilities Group, comprised of 17 electric utility companies (CU), General Electric Company (GE), Lloyd Harbor Study Group (LHSG), AEC Regulatory Staff (Reg), State of Maine (Maine), State of Minnesota (Minn.), State of Vermont (Vt.), and Westinghouse Electric Corporation (W).

In addition, statements by way of limited appearance were made by the following:

Mrs. Ann Carl, on behalf of Lloyd Harbor Study Group (afterwards a Participant)

Dr. Ralph E. Lapp

Dr. Norman C. Rasmussen

Mr. Harold Reis, Esq., on behalf of Iowa Electric Light and Power Company.

Dr. Richard Wilson

The massive evidentiary record thereafter developed[45/] consists of more than 22,000 pages of verbatim transcript of oral proceedings,

45/ The record of this proceeding is maintained in Docket RM-50-1 in the AEC Public Document Room, 1717 H Street, N.W., Washington, D.C., and is available for reference during normal business hours.

together with additional thousands of pages of written testimony and evidentiary exhibits.

In implementation of the National Environmental Policy Act of 1969, (P.L. 91-190), a Draft Environmental Statement concerning the proposed rule making was forwarded to the Council on Environmental Quality on December 6, 1972, and circulated for comment to participants in the hearing and interested Federal Agencies on December 7, 1972. Notice of public availability of the Statement and an invitation for comment was also published in the Federal Register at that time. Comments on the Draft Statement were received and a Final Environmental Statement was published on ________________.

The Commission provided[46/][47/] an opportunity for public hearings on those aspects of the Final Environmental Statement which were non-duplicative of matters dealt with in the earlier phases of the subject rule-making proceeding.

Protection of the public health and safety from radiological effects is a statutory responsibility of the AEC under the Atomic Energy Act and has always been foremost in its Regulatory program. Protection against a highly unlikely loss-of-coolant accident has long been an essential part of the defense-in-depth concept used by the nuclear power industry and the AEC to assure the safety of nuclear

46/ Commission Order of December 12, 1972.

47/ Commission Order of March 16, 1973.

power plants. In this concept, the primary assurance of safety is accident prevention by correct design, construction and operation of the reactor. Extensive and systematic quality assurance practices are required and applied at every step to achieve this primary assurance of safety. Nevertheless, deviations from expected behavior are postulated to occur, and protective systems are required and provided to take corrective action in such events. Notwithstanding all this, the occurrence of serious accidents is postulated, in spite of the fact that they are highly unlikely, and engineered safety features are provided to mitigate the consequences of these unlikely events. The loss-of-coolant accident (LOCA) is such a postulated improbable accident; the emergency core cooling system (ECCS) is one of the engineered safety features required to mitigate its consequences.

The regulations set forth below, like the 1971 Interim Policy Statement, deal with the effectiveness of the ECCS. In developing such regulations, the assumptions are made that a LOCA has occurred and that certain ECCS equipment functions according to its design (and other equipment does not). Starting from these assumptions, calculations are made of the effectiveness of the ECCS in cooling the core and maintaining the temperature, geometry, and oxidation of the cladding within acceptable limits. These regulations establish acceptance criteria for ECCS performance and set forth certain required and acceptable features of evaluation models as well as descriptions of evaluation models for use in making the calculations of cooling performance.

Other aspects of a postulated LOCA and ECCS acceptability, such as (1) the design, redundancy, and reliability of a particular ECCS design, (2) the effect of LOCA blowdown forces on core internals, (3) the design and reliability of a particular ultimate heat sink design, or (4) the radiological implications outside containment, are not treated in these regulations but are required to be evaluated pursuant to other applicable AEC regulations.

The regulations which are presented below apply only to light-water reactors with cylindrical, Zircaloy-clad, oxide fuel. Emergency core cooling systems for light-water reactors with stainless steel cladding and those with non-cylindrical cladding will continue to be considered on a case-by-case basis.

Fuel densification must be taken into account as appropriate in all ECCS evaluations. For reactors within the scope of these regulations, this is to be accomplished by ensuring that the parameters used in the evaluation models, and the derivation of operating limits from the results of calculations using the evaluation models, include where appropriate the effects of any postulated or observed densification. Because of differences in fuels, this must be considered on a case-by-case basis.

The new regulations effect changes in the criteria and evaluation models set forth in the 1971 Interim Policy Statement (IPS). The technical bases for such changes are set forth in great detail in the rule-making hearing records in the ECCS proceeding; see especially "Concluding Statement of the Regulatory Staff, April 16, 1973" and "Supplemental Testimony of the AEC Regulatory Staff, October 26, 1972," both in Docket RM-50-1.[48] The significant changes are summarized as follows:

1. The peak cladding temperature criterion has been reduced to 2200°F and a new cladding oxidation criterion has been added, based on data from embrittlement experiments.

2. Required and Acceptable Features of Evaluation Models - Many of these considerations are contained in Appendix A to the IPS. In some cases, the IPS language has been clarified. Other considerations were not included in the IPS and have been added based on the record of the ECCS proceeding.

3. Complete Evaluation Models - The origin of this material is in the IPS Appendix A. The descriptions and requirements have been brought up to date as required by increased knowledge obtained since publication of the IPS and contained in the record of the ECCS proceeding.

As provided in the regulations which follow, each reactor falling within the scope of the regulations shall be evaluated for

[48] See footnote 45, supra.

uqpus

ECCS cooling performance by the use of calculations to be performed in accordance with an acceptable evaluation model. An applicant or licensee may propose any suitable evaluation model for a particular reactor. Such a proposed evaluation model will be reviewed either on a case-by-case basis or (if appropriate) as a proposal for rule making.

The rule does, however, set forth certain required and acceptable features of evaluation models. In addition, the rule describes certain revisions needed in order that previously accepted evaluation models may be made acceptable under the new requirement. The AEC expects that the present descriptions (including required changes) will be replaced, after review, finding, and rule making as discussed above, with more definitive descriptions with the required changes integrated therein.

For certain classes of reactors, no evaluation models have been presently accepted for postulated breaks smaller in area than 0.5 square feet. For such breaks in such reactors, it is expected that evaluation models will be evaluated on a case-by-case basis.

Evaluation models based on computer programs developed as part of the AEC Safety Research Program are under active development. Models suitable to both BWR's and PWR's are anticipated, and should be made available about January 1, 1974.

The regulations set out below are not intended to stifle improvements and innovations in performance evaluation or in ECCS design. The record of the rule-making hearing sets forth the results of research programs directed toward increasing the knowledge relevant to ECCS performance. The nuclear industry and the Commission have underway at the present time several such programs, both analytical and experimental. The Commission expects that both governmental and private programs will be pursued diligently, and expects to consider promptly the new knowledge as it becomes available, and to consider such changes in these regulations as they appear appropriate in the light of all information then available.

B. <u>The Proposed Rule</u>

1. A new sentence is added to Section 50.34(a)(4) of 10 CFR Part 50 to read as follows:

§ 50.34 Contents of applications: technical information

(a) **

(4) *** Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of § 50.46 for facilities for which construction permits may be issued after [one year from date of publication of rule].

2. A new sentence is added to Section 50.34(b)(4) 10 CFR Part 50 to read as follows:

49749

§ 50.34 Contents of applications; technical information.

(a) ***

(b) ***

(4) *** Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of § 50.46 for facilities for which a license to operate may be issued after [one year from date of publication of rule]. In the event the facility has previously been determined to comply with the requirements of § 50.46 pursuant to § 50.46(a), additional analysis and evaluation need only be submitted if pertinent information developed since the prior submittal would alter the prior analysis and evaluation.

3. A new § 50.46 is added to 10 CFR Part 50 to read as follows:

§50.46 Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors.

(a)(1) Except as provided in subparagraphs (2), (3) and (4) of this paragraph, each boiling and pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Zircaloy 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). 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 entire spectrum of postulated loss-of-coolant accidents is covered. Appendix K, ECCS Evaluation Models, sets forth certain required and acceptable features of acceptable evaluation models, and describes certain evaluation models. Conformance with the criteria set forth in paragraph (b), with ECCS cooling performance calculated in accordance with an acceptable evaluation model, may require that restrictions be imposed on reactor operation.

(2) With respect to reactors for which operating licenses have been or may be issued after January 1, 1968 but before [one year after publication of rule] the following shall apply:

(i) An evaluation in accordance with subparagraph (1) of this paragraph (a) shall be submitted to the Commission as soon as practicable, but in no event later than [4 months after publication of rule], except for good cause shown. The evaluation shall be accompanied by such proposed changes in technical specifications or license amendments as may be necessary to bring reactor operation in conformity with the subparagraph.

(ii) The facility shall meet the requirements of subparagraph (1) of this paragraph (a) no later than [one year after publication of rule].

(iii) Pending determination by the Commission whether the facility meets the requirements of subparagraph (1) of this paragraph (a) including pendency of any proceedings under Subpart G of Part 2 of this chapter that may be required on this matter, the facility may continue or commence operation provided that:

(A) There has been a prior determination by the Commission that the facility meets the Interim Acceptance Criteria for Emergency Core Cooling Systems published on June 29, 1971 (36 F.R. 12247), as amended (December 18, 1971, 36 F.R. 24082).

(B) Such operation after the date of submittal of the evaluation in accordance with subdivision (2)(i) of this paragraph shall be conducted within the limits of both the proposed technical specifications or license amendments submitted with the evaluation and the technical specifications or license conditions previously imposed by the Commission in accordance with the other regulations in this chapter and the Interim Acceptance Criteria referred to in subdivision (2)(iii)(A) of this paragraph.

(C) Further restrictions on reactor operation will be imposed if the Commission finds that the evaluation submitted pursuant to subdivision (2)(i) of this paragraph is not consistent with subparagraph (1) of this paragraph (a) and as a result such further restrictions are required to protect the public health and safety.

(3) With respect to reactors for which operating licenses have been issued on or before January 1, 1968, the following shall apply:

(i) An evaluation in accordance with subparagraph (1) of this paragraph (a) shall be submitted to the Commission as soon as practicable, but in no event later than [4 months after publication of rule] except for good cause shown. The evaluation shall be accompanied by such changes in technical specifications or license amendments as may be necessary to bring reactor operation in conformity with the subparagraph within the time period specified in subdivision (3)(ii) below.

(ii) The facility shall meet the requirements of subparagraph (1) of this paragraph (a) no later than [July 1, 1974 or 1 year after publication of rule, whichever is later.]

(iii) Pending determination by the Commission whether the facility meets the requirements of subparagraph (1) of this paragraph (a), including pendency of any proceedings under Subpart G of Part 2 of this chapter, that may be required on this matter, the facility may continue operation, provided that interim ECCS improvements, augmented inservice inspection, and augmented detection of primary system leakage instituted in accordance with sections IV.C.(b)(2), (3), and (4) of the Interim Acceptance Criteria for Emergency Core Cooling Systems published on June 29, 1971 (36 F.R. 12248), as amended (December 18, 1971, 36 F.R. 24082) shall be continued.

(4) Construction permits may be issued after [publication of rule] but before [one year after publication of rule] subject to any applicable conditions or restrictions imposed pursuant to other regulations in this chapter and the Interim Acceptance Criteria for Emergency Core Cooling Systems published on June 29, 1971 (36 F.R. 12248) as amended (December 18, 1971, 36 F.R. 24082): <u>Provided, however</u>, that no operating license shall be issued unless the Commission determines, among other things, that the proposed facility meets the requirements of subparagraph (1) of this paragraph.

(b)(1) <u>Peak Cladding Temperature.</u> The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

(2) <u>Maximum Cladding Oxidation.</u> The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to

a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

(3) Maximum Hydrogen Generation. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Coolable Geometry. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) Long-Term Cooling. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat removed for the extended period of time required by the long-lived radioactivity remaining in the core.

(c) As used in this section:

(1) Loss-of-coolant accidents (LOCA's) are accidents that result from the loss of reactor coolant, at a rate in excess of the capability

of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.

(2) An evaluation model is the calculational framework for evaluating the behavior of the reactor system during a postulated LOCA, including one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedures for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

(d) The requirements of this section are in addition to any other requirements applicable to ECCS set forth in this Part. The criteria set forth in paragraph (b), with cooling performance calculated in accordance with an acceptable evaluation model, are in implementation of the general requirements with respect to ECCS cooling performance design set forth in this Part, including in particular Criterion 35 of Appendix A.

4. A new Appendix K is added to 10 CFR Part 50 to read as follows:

Appendix K - ECCS Evaluation Models

I. Definitions.

II. Required and Acceptable Features of Evaluation Models.

III. Complete Evaluation Models.

I. DEFINITIONS

As used in this appendix:

A. Blowdown means the portion of loss-of-coolant accident (LOCA) during which primary system pressure is higher than containment pressure and is decreasing with time, and primary fluid is expelled from the postulated break.

B. Computer Program means the sets of statements, in suitable computer language, necessary and sufficient to perform a calculation on a computer.

C. Conservatism means the property of a calculation, less favorable than "realistic" in order to provide margin for errors or unknowns, and sometimes to take sensitivity into account.

D. Moody Multiplier means the dimensionless multiplicative constant used to modify predictions of Moody's equations (see below) for flow out of pipes.

E. Realism means the property of a calculation that is intended to predict the course of actual or postulated events within some degree of approximation that may or may not be stated.

F. Refill means the portion of a loss-of-coolant accident after blowdown and extending until the level of reactor coolant rises to the bottom of the core. This period does not occur for LOCA's in which the reactor coolant level does not go below the bottom of the core.

G. Reflood means the portion of a loss-of-coolant accident after refill during which the coolant level rises in the core.

H. Sensitivity means the degree to which calculated results vary with a specified change in input information.

I. Loss-of-coolant accidents and evaluation models are as defined in § 50.46(c).

II. REQUIRED FEATURES OF EVALUATION MODELS

A. Single Failure Criterion. The combination of emergency core cooling subsystems assumed to operate in analyses shall be derived from a failure modes and effects analysis, using the single failure criterion.

B. Break Characteristics and Flow

1. The spectrum of LOCA's specified and defined in 10 CFR §§ 50.46(a) and (c) shall be analyzed.

2. Where the fluid reaching the break is calculated to be subcooled or saturated liquid, a discharge model appropriate to these conditions shall be used to calculate break flow.

3. For the period of transition from saturated liquid to low-quality two-phase fluid at the break exit plane, a discharge model appropriate to these conditions shall be used to calculate break flow.

4. Where the fluid reaching the break is calculated to be a two-phase fluid, or saturated vapor, the Moody discharge model (Moody, F.J.,

"Maximum Flow Rate of a Single Component, Two-Phase Mixture," Journal Of Heat Transfer, Trans. American Society of Mechanical Engineers, 87, No. 1., February 1965)49/50/51/ shall be used to calculate break flow.

5. Over the entire spectrum, the postulated break shall be assumed to occur instantaneously and shall be modeled as discharge from a single node as though through an open pipe having the postulated break area, and with a Moody multiplier (MM) of unity where the Moody model is used.

6. For postulated breaks in pressurized water reactor inlet lines and boiling water reactor recirculation lines, analyses shall be made assuming that the pipe fails as a complete instantaneous severance (guillotine). Flow shall be assumed to occur unimpeded from both ends of the open pipe without interaction between the discharging fluid streams. This model shall be used with at least three constant values of MM ranging from 0.6 to 1.0. If the trend is for peak cladding temperatures thus calculated to increase as MM decreases, the range of MM shall be extended to smaller values until the maximum peak clad

---

49/ The incorporation by reference provisions of Section II of this Appendix were approved by the Director of the Federal Register on ____________________.

50/ The staff is filing in this proceeding a complete list of all documents incorporated in the Proposed Rule and addresses of the respective publishers.

51/ Copies may be obtained from [address of publisher]. Copies are available for inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C.

temperature has been reached; that is, until further reduction in MM results in a lower peak clad temperature.

7. Noding detail in the vicinity of the break shall be sufficient to assure that the flow discharge calculation is performed with appropriate local fluid conditions.

C. Chemical Reactions and Heat Sources. The following chemical reactions and sources of heat shall be accounted for as a function of time and other variables as follows:

1. The reactor shall be assumed to have been operating continuously at a power level no lower than 1.02 times maximum licensed power level (to allow for instrumentation error), with the maximum peaking factor allowed by the technical specifications. A range of power distribution shapes and peaking factors representing power distributions over the core lifetime shall be studied and the combination selected that results in the most severe calculated consequences for the spectrum of postulated breaks and single failures analyzed.

2. Fission heat shall be calculated using reactivity and reactor kinetics. Shutdown reactivities due to temperatures and voids shall be given their minimum values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors studied in paragraph 1 of this Section C. Rod trip and insertion may be assumed if they are calculated to occur.

3. Radioactive decay of actinides, including neptunium and plutonium generated during operation, as well as isotopes of uranium, shall be calculated in accordance with fuel cycle calculations and known radioactivity properties. The most unfavorable time in the fuel cycle shall be assumed, independent of whatever such assumption was made in connection with paragraph 1 of this Section C.

4. Radioactive decay of fission products shall be estimated using 1.2 times the values for infinite operating time in the ANS Standard (Proposed American Nuclear Society Standard - "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," Approved by Subcommittee ANS-5, ANS Standards Committee, October 1971).[51/] The fraction of the gamma decay energy generated that is deposited in the fuel (including the cladding) may be equal to or less than 1.0; if the value used is less than 1.0, it shall be justified by a suitable calculation.

5. The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation (Baker, L., Just, L.C., "Studies of Metal Water Reactions at High Temperatures, III. Experimental and Theoretical Studies of the Zirconium-Water Reaction," ANL-6548, May 1962),[51/] with a coefficient of unity. The reaction shall be assumed not to be steam limited. For rods whose cladding is calculated to rupture during the LOCA (see Section II.H), the inside of the cladding shall also be

assumed to react. The calculation of the reaction rate on the inside of the cladding shall also follow the Baker-Just equation with a coefficient of unity, starting at the time when the cladding is calculated to rupture, and extending axially no less than 1.5 inches each way from the axial location of the rupture, with the reaction assumed not to be steam limited.

6. Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account.

7. Heat transferred between primary and secondary systems through heat exchangers shall be taken into account.

D. Frictional Pressure Drops. The frictional losses in pipes and other components including the reactor core shall be calculated using models that include realistic variation of friction factor with Reynolds number, and realistic two-phase friction multipliers that have been adequately verified by comparison with experimental data. The modified Baroczy correlation (Baroczy, C.J., "A Systematic Correlation for Two-Phase Pressure Drop," Chem. Engng. Prog. Symp. Series, No. 64, Vol. 62, 1965)[51] or a combination of the Thom correlation (Thom, J.R.S., "Prediction of Pressure Drop During Forced Circulation Boiling of Water," Int. J. of Heat & Mass Transfer, 7, 709-724, 1964)[51] for pressures equal to or greater than 250 psia and the Martinelli-Nelson correlation (Martinelli, R.C., Nelson, D.B., "Prediction of Pressure Drop During Forced Circulation Boiling of Water," Transactions of ASME, 695-702, 1948)[51/] for pressures lower than 250 psia is acceptable for calculating two-phase friction multipliers.

E. Momentum Equation. The following effects shall be taken into account in the conservation of momentum equation: 1. temporal change of momentum, 2. momentum convection, 3. area change momentum flux, 4. momentum change due to compressibility, 5. pressure loss due to wall friction, 6. pressure loss due to area change, and 7. gravitational acceleration. Any omission of one or more of these terms under stated circumstances shall be justified by comparative analyses or by experimental data.

F. Critical Heat Flux.

1. Correlations developed from appropriate steady-state and transient-state experimental data are acceptable for use in predicting the critical heat flux (CHF) during LOCA transients. The computer programs in which these correlations are used shall contain suitable checks to assure that the physical parameters are within the range of parameters specified for use of the correlations by their respective authors.

2. Steady-state CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following:

(a) W-3. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-uniform Heat Flux Distribution," Journal of Nuclear Energy, vol. 21, 241-248, 1967.[51/]

(b) B&W-2. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," Two-Phase Flow and Heat Transfer in Rod Bundles, ASME, New York, 1969.[51/]

(c) Hench-Levy. J. M. Healzer, J. E. Hench, E. Janssen, S. Levy "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors," APED-5186, GE Company Private report, July 1966.[52/]

(d) Macbeth. R. V. Macbeth, "An Appraisal of Forced Convection Burnout Data," Proceedings of the Institute of Mechanical Engineers, 1965-1966.[51/]

(e) Barnett. P. G. Barnett, "A Correlation of Burnout Data for Uniformly Heated Annuli and Its Uses for Predicting Burnout in Uniformly Heated Rod Bundles," AEEW-R 463, 1966.[51/]

(f) Hughes. E. D. Hughes, "A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia," IN-1412, Idaho Nuclear Corporation, July 1970.[51/]

3. Correlations of appropriate transient CHF data may be accepted for use in LOCA transient analyses if comparisons between the data and the correlations are provided to demonstrate that the correlations

[52/] Copies are on file in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. Since the document contains proprietary information (10 CFR §§ 2.790 and 9.5(a)(4)), its availability is governed by 10 CFR §§ 2.744 or 9.10.

predict values of CHF which allow for uncertainty in the experimental data throughout the range of parameters for which the correlations are to be used. Where appropriate, the comparisons shall use statistical uncertainty analysis of the data to demonstrate the conservatism of the transient correlation.

4. Transient CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following:

(a) GE transient CHF. B. C. Slifer, J. E. Hench, "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," NEDO-10329, General Electric Company, Equation C-32, April 1971.[51]

5. After CHF is first predicted at an axial fuel rod location during blowdown, the calculation shall not use nucleate boiling heat transfer correlations at that location subsequently during the blowdown even if the calculated local fluid and surface conditions would apparently justify the reestablishment of nucleate boiling. Heat transfer assumptions characteristic of return to nucleate boiling (rewetting) shall be permitted when justified by the calculated local fluid and surface conditions during the reflood portion of a LOCA.

G. Post-CHF Heat Transfer Correlations.

1. Correlations of heat transfer from the fuel cladding to the surrounding fluid in the post-CHF regimes of transition and film boiling shall be compared to applicable steady-state and transient-state data

using statistical correlation and uncertainty analyses. Such comparison shall demonstrate that the correlations predict values of heat transfer coefficient equal to or less than the mean value of the applicable experimental heat transfer data throughout the range of parameters for which the correlations are to be used. The comparisons shall quantify the relation of the correlations to the statistical uncertainty of the applicable data.

2. The Groeneveld flow film boiling correlation (equation 5.9 of D. C. Groeneveld, "An Investigation of Heat Transfer in the Liquid Deficient Regime," AECL-3281, revised December 1969),[51/] the modified Dougall-Rohsenow flow film boiling correlation (D. H. Roy, "Direct Testimony on Behalf of Babcock and Wilcox, AEC Docket No. RM-50-1," March 23, 1972, page 7-8[51/]; and R. S. Dougall and W. M. Rohsenow, "Film Boiling on the Inside of Vertical Tubes with Upward Flow of the Fluid at Low Qualities," MIT Report Number 9079-26, Cambridge, Massachusetts, September 1963[51/]), and the Westinghouse correlation of steady-state transition boiling ("Proprietary Redirect/Rebuttal Testimony of Westinghouse Electric Corporation," U.S.A.E.C. Docket RM-50-1, page 25-1, October 26, 1972)[52/] 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)[51/] is suitable

for use between nucleate and film boiling. Use of all these correlations shall be restricted as follows:

(a) the Groeneveld correlation shall not be used in the region near its low-pressure singularity,

(b) the first term (nucleate) of the Westinghouse correlation and the entire McDonough, Milich, and King correlation shall not be used during the blowdown after the temperature difference between the clad and the saturated fluid first exceeds 300°F,

(c) transition boiling heat transfer shall not be reapplied for the remainder of the LOCA blowdown, even if the clad superheat returns below 300°F, except for the reflood portion of the LOCA when justified by the calculated local fluid and surface conditions.

H. Cladding Swelling and Rupture. Calculations of gap conductance, cladding temperature, cladding embrittlement, and hydrogen generation from cladding-water chemical reactions shall take swelling and rupture of the cladding into account wherever the course of the postulated loss-of-coolant accident, calculated in accordance with an accepted evaluation model, leads to predictions of cladding swelling or rupture. Each evaluation model, therefore, shall where required include a model for predicting cladding swelling or rupture from consideration of the cladding axial temperature distribution and the pressure differential, both as functions of time. To be acceptable, a swelling and rupture model shall be based on applicable data in a conservative way.

I. Initial Stored Energy in Fuel. The steady-state temperature distribution and stored energy in the fuel before the accident shall be evaluated as a function of power density, fuel density, cold gap dimension, fuel thermal conductivity, fuel heat capacity, cold-fill gas composition and pressure, and burnup (cracking of fuel, sorbed gas and fission gas release, changes in fuel density, cladding creep). The values used and the burnup chosen (time in core lifetime) shall be such as to maximize the calculated initial stored energy in the fuel. For this calculation, the reactor operating power shall be assumed to be no less than 1.02 times maximum licensed power.

J. Fuel Rod Thermal Parameters During Postulated Accident.

1. The calculations of the fuel and the cladding temperatures as functions of time shall use values for gap conductance and other thermal parameters as functions of temperature and other applicable time dependent variables.

2. If cladding swelling or rupture is calculated to occur, the gap conductance shall be varied in accordance with the change in its dimensions and any other applicable variables.

K. Modeling of Rotating Pumps. The characteristics of rotating primary system circulating pumps (axial flow, turbine, or centrifugal) shall be derived from a dynamic model that includes momentum transfer between the fluid and the rotating member, with variable pump speed as a function of time. The pump resistance used for analysis should be justified. The pump model for the two-phase region shall be verified by applicable two-phase pump performance data.

L. Containment Pressure. The containment pressure used for evaluating cooling effectiveness during reflood and spray cooling shall not exceed a pressure calculated conservatively for this purpose. The calculation shall include the effects of operation of all installed pressure-reducing systems and processes.

M. Spray Cooling Heat Transfer (Applies Only to Boiling Water Reactors). Following the blowdown period, convective heat transfer shall be calculated using coefficients based on appropriate experimental data. For reactors with jet pumps and having fuel rods in a 7 x 7 fuel assembly array, the following convective coefficients are acceptable:

1. During the period following lower plenum flashing but prior to the core spray reaching rated flow, a convective heat transfer coefficient of zero shall be applied to all fuel rods.

2. During the period after core spray reaches rated flow but prior to reflooding (see paragraph 3), convective heat transfer coefficients of 2.0, 3.2, 1.5, and 1.7 $Btu\text{-}hr^{-1}\text{-}ft^{-2}\text{-}°F^{-1}$ shall be applied to the fuel rods in the outer corners, outer row, next to outer row, and to those remaining in the interior, respectively, of the assembly.

3. After the two-phase reflooding fluid reaches the level under consideration, a convective heat transfer coefficient of 25 $Btu\text{-}hr^{-1}\text{-}ft^{-2}\text{-}{}^{\circ}F^{-1}$ shall be applied to all fuel rods.

N. Channel Box Wetting (Applies Only to Boiling Water Reactors). Following the blowdown period, heat transfer from, and wetting of, the channel box shall be based on appropriate experimental data. For reactors with jet pumps and fuel rods in a 7 x 7 fuel assembly array, the following heat transfer coefficients and wetting time correlation are acceptable.

1. During the period after lower plenum flashing, but prior to core spray reaching rated flow, a convective coefficient of zero shall be applied to the fuel assembly channel box.

2. During the period after core spray reaches rated flow, but prior to wetting of the channel, a convective heat transfer coefficient of 5 $Btu\text{-}hr^{-1}\text{-}ft^{-2}\text{-}{}^{\circ}F^{-1}$ shall be applied to both sides of the channel box.

3. Wetting of the channel box shall be assumed to occur 60 seconds after the time determined using the correlation based on the Yamanouchi analysis ("Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," General Electric Company Report NEDO-10329, April 1971).[51/]

O. Core Flow Distribution During Blowdown (Applies Only to Pressurized Water Reactors).

1. The flow rate through the hot region of the core during blowdown shall be calculated as a function of time. For the purpose of these calculations the hot region chosen shall not be greater than the size of one fuel assembly. Calculations of average flow and flow in the hot region shall take into account cross flow between regions and any flow blockage calculated to occur during blowdown as a result of cladding swelling or rupture. The calculated flow shall be smoothed to eliminate any calculated rapid oscillations (period less than 0.1 seconds).

2. If fuel cladding swelling or rupture is calculated to occur in the hot region during blowdown, the hot region flow shall be multiplied by a flow reduction factor of 0.8 to form the flow input data for the hot channel heatup calculation.

3. A method shall be specified for determining the enthalpy to be used as input data to the hot channel heatup analysis from quantities calculated in the blowdown analysis, consistent with the flow distribution calculations.

P. Cooling Water Injected During Blowdown (Applies Only to Pressurized Water Reactors). For postulated cold leg breaks, all emergency cooling water injected into the inlet lines or the reactor vessel during the bypass period shall in the calculations be subtracted from the reactor vessel calculated inventory. This may be executed in the calculation during the bypass period, or as an alternative the amount of emergency core cooling water calculated to be injected during the bypass period may be subtracted later in the calculation from the water remaining in the inlet lines, downcomer, and reactor vessel lower plenum after the bypass period. This bypassing shall end in the calculation at a time designated as the "end of bypass," after which the expulsion or entrainment mechanisms responsible for the bypassing are calculated not to be effective. The end-of-bypass definition used in the calculation shall be justified by a suitable combination of analysis and experimental data. Acceptable methods for defining "end of bypass" include, but are not limited to, the following: 1. Prediction by the blowdown calculation of downward flow in the downcomer for the remainder of the blowdown period; 2. Prediction of a threshold for droplet entrainment in the upward velocity, using local fluid conditions and a conservative critical Weber number.

Q. Reflood Heat Transfer (Applies Only to Pressurized Water Reactors). For the early portion of the reflood period, during which droplet entrainment or fluid oscillations do not transport a two-phase mixture to the core hot spot, heat transfer calculations shall be for steam-only cooling and shall take into account any flow blockage calculated to occur as a result of cladding swelling or rupture as such blockage might affect both local steam flow and heat transfer. A transition to reflood heat transfer coefficients based on applicable experimental data, including FLECHT results ("PWR FLECHT (Full Length

Emergency Cooling Heat Transfer) Final Report," Westinghouse Report WCAP-7665, April 1971)[51/] shall be made when calculated conditions are sufficient to transport a two-phase mixture to the hot spot. The criteria for such transition shall be justified by analysis and/or experimental results. The use of a correlation derived from FLECHT data shall be demonstrated to be conservative for the transient to which it is applied; presently available FLECHT heat transfer correlations ("PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report," Westinghouse Report WCAP-7435, January 1970; "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Group II Test Report," Westinghouse Report WCAP-7544, September 1970; "PWR FLECHT Final Report Supplement," Westinghouse Report WCAP-7931, October 1972)[51/] are not acceptable. New correlations or modifications to the FLECHT heat transfer correlations are acceptable only after they are demonstrated to be conservative, by comparison to FLECHT data, for a range of parameters consistent with the transient to which they are applied.

R. Steam-Liquid Interaction in Pipes (Applies Only to Pressurized Water Reactors). During the refill and reflood periods steam flow through primary coolant pipes is subject to potential interference by injected emergency core cooling water. This effect shall be included as appropriate in the thermal and hydraulic aspects of reflooding rate calculations. During refill and reflood the calculated steam flow in reactor coolant pipes shall be taken to be zero during the time that accumulators are discharging water into those pipes, and emergency cooling water shall be assumed to mix homogeneously with steam, unless experimental evidence is available regarding the realistic thermal-hydraulic interaction between the steam and the liquid. The thermal-hydraulic interaction between steam and all emergency core cooling water shall be taken into account in calculating core reflooding rate.

S. Reflooding Rate Calculations (Applies Only to Pressurized Water Reactors). The refilling and reflooding flow rate shall be calculated as a function of time using an acceptable thermal and hydraulic model. Core reflooding calculations which neglect dynamic effects leading to fluid oscillations in the system are acceptable, as are calculations which include dynamic effects. For both calculational options, core and system thermal-hydraulic phenomena shall be modeled and reactor primary coolant pumps shall be assumed to have locked impellers. The ratio of total fluid flow at the core exit plane to the total liquid flow at the core inlet plane (carryover rate fraction) shall be used to determine the core exit flow and shall be determined in accordance with applicable experimental data (for example, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," Westinghouse Report WCAP-7665, April 1971; "PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report," Westinghouse Report WCAP-7435, January 1970;

"PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Group II Test Report," Westinghouse Report WCAP-7544, September 1970; "PWR FLECHT Final Report Supplement," Westinghouse Report WCAP-7931, October 1972).[51/]

The effects on the reflooding rate of the compressed gas in the accumulator, which is discharged following accumulator water discharge, shall also be taken into account.

III. COMPLETE EVALUATION MODELS

A. Documentation

1.(a) A description of each proposed evaluation model shall be furnished. The description shall be sufficiently complete to permit technical review of the analytical approach including the equations used, the assumptions made, and the values of all parameters or the procedure for their selection, as for example, in accordance with a specified physical law or empirical correlation.

(b) The description shall be sufficiently detailed and specific to require significant changes in the evaluation model to be specified in amendments of the description. For this purpose, a significant change is a change that would result in calculated fuel cladding temperatures different by more than 20°F than the temperatures calculated (as a function of time) previously for a postulated LOCA.

(c) A complete listing of each computer program, in the same form as used in the evaluation model, shall be furnished to the Atomic Energy Commission.

2. For each computer program, solution convergence shall be demonstrated by studies of system modeling or noding and calculational time steps.

3. Appropriate sensitivity studies shall be performed for each evaluation model, to evaluate the effect on the calculated results of variations in noding, phenomena assumed in the calculation to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items shown to be sensitive, the choices made shall be justified.

4. To the extent practicable, predictions of the evaluation model, or portions thereof, shall be compared with applicable experimental information.

B. General Standards For Acceptability - Delineation of all details (in addition to the required features set forth in Section II) of the basis for general acceptance of evaluation models is not possible, because of their complexity. Elements of evaluation models reviewed will include technical adequacy of the calculational methods, including compliance with required features of Section II, 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.

C. Evaluation Models - The following evaluation models, when modified as provided below for each, will be acceptable:

1. AEC Evaluation Model for Pressurized Water Reactors.

(a) This model will be acceptable for application to pressurized-water reactors using 2, 3, or 4 primary coolant loops, and for postulated breaks with areas of 0.5 square feet and larger.

(b) The computer programs, parameters, and other features of this evaluation model are as described in the following references:

(1) "RELAP3-A Computer Program for Reactor Blowdown Analysis" Idaho Nuclear Corporation Report IN-1321, June 1970.[51/]

(2) A suitable refill and reflood model and computer program to determine system behavior from the end of blowdown to completion of core reflooding.

(3) "THETA1-B, A Computer Program for Nuclear Reactor Core Thermal Analyses," Idaho Nuclear Corporation Report IN-1445, February 1971.[51/]

(4) "GAPCON: A Computer Program to Predict Fuel-to-Cladding Heat Transfer Coefficients in Oxide Fuel Pins," Hanford Engineering Development Laboratory Report HEDL-TME-72-128, September 1972.[51/]

(5) Thom, J.R.S., et al., "Boiling in Sub-Cooled Water during Flow Up Heated Tubes or Annuli," Proc. Inst. Mech. Engrs., Vol. 180, part 3C, 1965-1966.[51/]

(6) Dittus, F.W., Boelter, L.M.K., "Heat Transfer in Automobile Radiators of the Tubular Type", Pub. in Eng., Vol. 2, n. 13, Univ. of Calif., pp. 443-461, 1930.[51/]

11075

(c) The analytical methods employed in the above reports and the computer programs should be modified as required to include the following considerations:

(1) Blowdown Analysis

(i) The break flow model and the noding specifications described in Section II.B shall be used.

(ii) The transfer of heat from the sources described in Section II.C shall be included.

(iii) Friction pressure drop calculations shall be treated in accordance with Section II.D.

(iv) The conservation of momentum equation shall be treated in accordance with Section II.E.

(v) Reactor recirculation pump dynamic performance shall be treated in accordance with Section II.K.

(vi) The reactor core heat transfer correlations shall be treated in accordance with Sections II.F and II.G.

(vii) The following fluid heat transfer correlations shall be used for the core region, as appropriate:

Nucleate Boiling-Thom

Single Phase Forced Convection - Dittus-Boelter

(viii) Cooling water injection into the reactor coolant system shall be treated in accordance with Section II.P.

(ix) Nitrogen discharged from accumulator tanks shall be treated in accordance with Section II.S.

(x) Core flow in the hot region during blowdown shall be calculated in accordance with Section II.O.

(2) Fuel Pin Thermal Analysis

(i) The conservation of fluid energy equation shall be solved for one- and two-phase flow conditions.

(ii) The calculation of critical heat flux shall be performed in accordance with Section II.F.

(iii) Fuel element heat transfer shall be treated, as appropriate, in accordance with Sections II.C, II.G, II.H, II.I, and II.J.

(iv) The following fluid heat transfer correlations shall be used, as appropriate:

Nucleate Boiling - Thom

Single Phase Forced Convection - Dittus-Boelter

(v) Core flow during blowdown shall be used as input data to the fuel pin thermal analysis in accordance with Section II.O.

(vi) During blowdown, the core pressure and hot region inlet enthalpy derived from the blowdown analysis shall be used as input to the fuel pin thermal analysis.

(vii) During the reflood portion of a LOCA, clad-to-fluid heat transfer shall be treated in accordance with Section II.Q, and containment pressure shall be treated in accordance with Section II.L.

(3) The reflood analysis.

(i) Calculation of reactor vessel water inventory shall be treated in accordance with Section II.P.

(ii) The containment pressure used in the reflood analysis shall be treated in accordance with Section II.L.

(iii) Fluid pressure drop in the primary side of the steam generators shall be calculated considering heat transfer from the secondary side and with the inlet conditions calculated according to Section II.S.

(iv) Steam flow calculations shall be treated in accordance with Section II.R, and appropriate consideration shall be given to reactor internals vent valves.

(v) Refill and reflood rate calculations shall be treated in accordance with Section II.S.

2. Babcock and Wilcox Evaluation Model.

(a) This model will be acceptable for application to pressurized-water reactors, with and without reactor internals vent valves, for postulated breaks with areas of 0.5 square feet and larger.

(b) The computer programs, parameters, and other features of this evaluation model are as described in the following references:

(1) "CRAFT - Description of Model for Equilibrium LOCA Analysis Program" Report BAW-10030, October, 1971.[51/]

(2) "REFLOOD - Description of Model for Multinode Core Reflood Analysis," Report BAW-10031, October 1971.[51/]

(3) "THETA 1-B, a Computer Program for Nuclear Reactor Core Thermal Analysis," Idaho Nuclear Corporation Report IN-1445, February, 1971.[51/]

(4) "Multinode Analysis of B&W's 2568 MWt Nuclear Plants during a Loss-of-Coolant Accident" Report BAW-10034, October 1971.[51/]

(5) "REFLOOD - Description of Model for Multinode Core Reflood Analysis - Supplement 1," Report BAW-10031, Supp. 1, April 1972.[51/]

(6) "TAFY - Fuel Pin Temperature and Gas Pressure Analysis," Report BAW-10044, April 1972.[51/]

(7) "Multinode Analysis of B&W's 205- Fuel Assembly Nuclear Plants during Loss-of-Coolant Accident," Report BAW-10045, May 1972.[51/]

(8) "Multinode Analysis of B&W's 145-Fuel Assembly Nuclear Plants during Loss-of-Coolant Accident," Report BAW-10048, June 1972.[51/]

(9) Response of B&W to Regulatory questions regarding B&W Report 10048, September 18, 1972.[51/]

(10) "Multinode Analysis of B&W's 145-Fuel Assembly Nuclear Plants during Loss-of-Coolant Accident," Supplement 1, Revision 2 of Report BAW-10048, October, 1972.[51/]

(11) Thom, J.R.S., et al., "Boiling in Sub-Cooled Water during Flow Up Heated Tubes or Annuli," Proc. Inst. Mech. Engrs., Vol. 180, part 3C, 1965-1966.[51/]

(12) Dittus, F.W., Boelter, L.M.K., "Heat Transfer in Automobile Radiators of the Tubular Type", Publ. in Eng., Vol. 2, n. 13, Univ. of Calif., pp. 443-461, 1930.[51/]

(c) The analytical methods employed in the above reports and computer programs should be modified as required to include the following considerations:

(1) CRAFT computer program.

(i) The break flow model and the noding specifications described in Section II.B shall be used.

(ii) The transfer of heat from the sources described in Section II.C shall be included.

(iii) Friction pressure drop calculations shall be treated in accordance with Section II.D.

(iv) The conservation of momentum equation shall be treated in accordance with Section II.E.

(v) Reactor recirculation pump dynamic performance shall be treated in accordance with Section II.K.

(vi) Reactor core heat transfer correlations shall be treated in accordance with Sections II.F and II.G.

(vii) The following fluid heat transfer correlations shall be used for the core region, as appropriate:

Nucleate Boiling - Thom

Single Phase Forced Convection - Dittus-Boelter

(viii) Cooling water injected into the reactor coolant system shall be treated in accordance with Section II.P.

(ix) Nitrogen discharged from accumulator tanks shall be treated in accordance with Section II.S.

(x) Core flow in the hot region during blowdown shall be calculated in accordance with Section II.O.

(2) THETA 1-B computer program.

(i) The conservation of fluid energy equation shall be solved for all one- and two-phase flow conditions.

(ii) The calculation of critical heat flux shall be performed in accordance with Section II.F.

(iii) The fuel rod heat transfer shall be treated, as appropriate, in accordance with Sections II.C, II.G, II.H, II.I, and II.J.

(iv) The following fluid heat transfer correlations shall be used, as appropriate:

Nucleate Boiling - Thom

Single Phase Forced Convection - Dittus-Boelter

(v) Core flow during blowdown shall be input to the THETA 1-B program in accordance with Section II.O.

(vi) During blowdown the core pressure and entering plenum enthalpy derived from the CRAFT computer program shall be used as input to the THETA 1-B computer program.

(vii) During the reflood portion of a LOCA, clad-to-fluid heat transfer shall be treated in accordance with Section II.Q, and containment pressure shall be treated in accordance with Section II.L.

(3) REFLOOD computer program.

(i) Calculation of reactor vessel water inventory shall be treated in accordance with Section II.P.

(ii) The containment pressure used in the reflood analysis shall be treated in accordance with Section II.L.

(iii) Fluid pressure drop in the primary side of the steam generators shall be calculated considering secondary side heat transfer and with the inlet conditions calculated according to Section II.S.

(iv) Steam flow calculations shall be treated in accordance with Section II.R, and appropriate consideration shall be given to reactor internals vent valves.

(v) Refill and reflood rate calculations shall be treated in accordance with Section II.S.

3. Combustion Engineering Evaluation Model.

(a) This model will be acceptable for application to Combustion Engineering pressurized water reactors for postulated breaks with areas of 0.5 square feet and larger.

(b) The computer programs, parameters, and other features of the evaluation model are as described in the following references:

(1) "Description of Loss-of-Coolant Calculational Procedures," CENPD-20, Proprietary Combustion Engineering Report, August 1971.[52/]

(2) "Description of Loss-of-Coolant Calculational Procedures," Proprietary Combustion Engineering Report, Supplement 1 to CENPD-26, October 1971.[52/]

(3) "Steam Venting Experiments and Their Application to CE Evaluation Model," Proprietary Combustion Engineering Report, Supplement 2 to CENPD-26, November 1971.52/

(4) "Moisture Carry-over During PWR Post-LOCA Core Refill," informal proprietary Combustion Engineering submittal, November 1971.52/

(5) Thom, J.R.S., et al., "Boiling in Sub-Cooled Water during Flow Up Heated Tubes or Annuli," Proc. Inst. Mech. Engrs., Vol. 180, part 3C, 1965-1966.51/

(6) Dittus, F.W., Boelter, L.M.K., "Heat Transfer in Automobile Radiators of the Tubular Type, Publ. in Eng., Vol. 2, n. 13, Univ. of Calif., pp. 443-461, 1930.51/

(c) The analytical methods employed in the above reports and computer programs should be modified as required to include the following considerations:

(1) CEFLASH computer program.

(i) The break flow model and noding specifications described in Section II.B shall be used.

(ii) The transfer of heat from the sources described in Section II.C shall be included.

(iii) The friction pressure drop calculations shall be treated in accordance with Section II.D.

(iv) The conservation of momentum equation shall be treated in accordance with Section II.E.

(v) Reactor recirculation pump dynamic performance shall be treated in accordance with Section II.K.

(vi) Reactor core heat transfer correlations shall be treated in accordance with Sections II.F and II.G.

(vii) The following fluid heat transfer correlations shall be used for the core region, as appropriate:

Nucleate Boiling - Thom

Single Phase Forced Convection - Dittus-Boelter

(viii) Cooling water injected into the reactor coolant system shall be treated in accordance with Section II.P.

(ix) Nitrogen discharged from safety injection tanks shall be treated in accordance with Section II.S.

(x) Core flow in the hot region during blowdown shall be calculated in accordance with Section II.O.

(2) STRIKIN computer program.

(i) The fuel rod heat transfer shall be treated, as appropriate, in accordance with Sections II.C, II.G, II.H, II.I, and II.J.

(ii) The following fluid heat transfer correlations shall be used, as appropriate:

Nucleate Boiling - Thom

Single Phase Forced Convection - Dittus-Boelter

(iii) Core flow during blowdown shall be input to the STRIKIN program in accordance with Section II.O.

(iv) During blowdown the core pressure and entering plenum enthalpy derived from the CEFLASH computer program shall be used as input to the STRIKIN computer program.

(v) During the reflood portion of a LOCA clad-to-fluid heat transfer shall be treated in accordance with Section II.Q, and containment pressure shall be treated in accordance with Section II.L.

(3) PERC computer program.

(i) Calculation of reactor vessel water inventory shall be treated in accordance with Section II.P.

(ii) The containment pressure used in the reflood analyses shall be treated in accordance with Section II.L.

(iii) Fluid pressure drop in the primary side of steam generators shall be calculated considering secondary side heat transfer and with the inlet conditions calculated according to Section II.S.

(iv) Steam flow calculations shall be treated in accordance with Section II.R.

(v) Refill and reflood rate calculations shall be treated according to Section II.S.

4. <u>General Electric Evaluation Model.</u>

(a) This model will be acceptable for application to General Electric Company boiling water reactors using jet pumps and a 7 x 7 fuel rod array in each fuel assembly.

(b) The computer programs, parameters and other features of the evaluation model are described in the following reference:

(1) "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors" NEDO-10329, April 1971.[51/]

a. Short-Term Thermal Hydraulic Model, Appendix A; NEDO-10329

b. Long-Term Thermal Hydraulic Model, Appendix B, NEDO-10329

c. Transient Critical Heat Flux Model, Appendix C, NEDO-10329

d. Core Heatup Model, Appendix D, NEDO-10329

(c) The analytical methods employed in the above report and computer programs should be modified as required to include the following considerations:

(1) Short-Term Thermal Hydraulic Model.

(i) The break flow and noding specifications described in Section II.B shall be used.

(ii) The transfer of heat from the sources described in Section II.C shall be included.

(iii) The friction pressure drop calculations shall be treated in accordance with Section II.D.

(iv) The conservation of momentum equation shall be treated in accordance with Section II.E.

(v) Reactor recirculation pump performance shall be treated in accordance with Section II.K.

(2) Transient Critical Heat Flux Model.

(i) The calculation of critical heat flux shall be performed in accordance with Section II.F. For 1967 and 1969 product line reactors with a 7 x 7 fuel rod array in an assembly, the critical heat flux correlations specified in Appendix C of NEDO-10329 are acceptable.

(ii) The fuel rod heat transfer shall be treated in accordance with Sections II.C, II.G, II.H, II.I, and II.J.

(3) Core Heatup Model.

(i) The fuel rod heat transfer shall be treated in accordance with Sections II.C, II.F, II.G, II.H, II.I, II.J, II.L, II.M, and II.N. For 1967 and 1969 product line reactors with a 7 x 7 fuel rod array in an assembly, the critical heat flux correlations specified in Appendix C of NEDO-10329 are acceptable. The emissivity of Zircaloy shall be assumed to be 0.67 prior to wetting and 0.9 after wetting.

(4) Long-Term Thermal Hydraulic Model.

(i) The break flow and noding specifications described in Section II.B shall be used.

(ii) The transfer of heat from the sources described in Section II.C shall be included.

(iii) The friction pressure drop calculations shall be treated in accordance with Section II.D.

(iv) The conservation of momentum equation shall be treated in accordance with Section II.E.

(v) Reactor recirculation pump performance shall be treated in accordance with Section II.K.

5. Westinghouse Evaluation Model.

(a) This model will be acceptable for application to Westinghouse pressurized water reactors using 2, 3, or 4 primary coolant loops, and for postulated breaks with areas of 0.5 square feet and larger.

(b) The computer programs, parameters, and other features of the evaluation model are as described in the following references:

(1) "Westinghouse PWR Core Behavior Following a Loss-of-Coolant Accident," WCAP-7422-L, January 1970 (Proprietary).[52/]

(2) "Emergency Core Cooling Performance," Supplemental Proprietary Westinghouse Report, June 1971.[52/]

(3) "Westinghouse PWR Core Behavior Following a Loss-of-Coolant Accident," WCAP-7422, August 1971.[51/]

(4) "Additional Testimony of Applicant Concerning Emergency Core Cooling System Performance," Indian Point Station, Unit No. 2, USAEC Docket No. 50-247, July 13, 1971.[51/]

(5) "A Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant (SATAN 4 Digital Code)," Westinghouse Report WCAP-7750, August 1971.[51/]

(6) "LOCTA-R2 Program: Loss-of-Coolant Transient Analysis," Westinghouse Report WCAP-7437L, January 1970.[52/]

(7) Thom, J.R.S., et al., "Boiling in Sub-Cooled Water during Flow Up Heated Tubes or Annuli," Proc. Inst. Mech. Engrs., Vol. 180, Part 3C, 1965-1966.[51/]

(8) Dittus, F. W., Boelter, L. M. K., "Heat Transfer in Automobile Radiators of the Tubular Type", Publ. in Eng., Vol. 2, n. 13, Univ. of Calif., pp. 443-461, 1930.[51]

(c) The analytical methods employed in the above reports and computer programs should be modified as required to include the following considerations:

(1) SATAN computer program.

(i) The break flow model and the noding specifications described in Section II.B shall be used.

(ii) The transfer of heat from the sources described in Section II.C shall be included.

(iii) Friction pressure drop calculations shall be treated in accordance with Section II.D.

(iv) The conservation of momentum equation shall be treated in accordance with Section II.E.

(v) Reactor recirculation pump dynamic performance shall be treated in accordance with Section II.K.

(vi) Reactor core heat transfer correlations shall be treated in accordance with Sections II.F and II.G.

(vii) The following fluid heat transfer correlations shall be used for the core region, as appropriate:

Nucleate Boiling - Thom

Single Phase Forced Convection - Dittus-Boelter

(viii) Cooling water injected into the reactor coolant system shall be treated in accordance with Section II.P.

(ix) Nitrogen discharged from accumulator tanks shall be treated in accordance with Section II.S.

(x) Core flow in the hot region during blowdown shall be calculated in accordance with Section II.O.

(2) LOCTA computer program.

(i) The calculation of critical heat flux shall be performed in accordance with Section II.F. In applying the W-3 correlation a simple grid design should be assumed.

(ii) The fuel rod heat transfer shall be treated, in accordance with Sections II.C, II.G, II.H, II.I and II.J.

(iii) The following fluid heat transfer correlations shall be used, as appropriate:

Nucleate Boiling - Thom

Single Phase Forced Convection - Dittus-Boelter

(iv) Core flow during blowdown shall be input to the LOCTA program in accordance with Section II.O.

(v) During blowdown the core pressure and entering plenum enthalpy derived from the SATAN computer program shall be used as input to the LOCTA computer program.

(vi) During the reflood portion of a LOCA, clad-to-fluid heat transfer shall be treated in accordance with Section II.Q, and containment pressure shall be treated in accordance with Section II.L.

(3) Reflood computer program.

(i) Calculation of reactor vessel water inventory shall be treated in accordance with Section II.P.

(ii) The containment pressure used in the reflood analysis shall be treated in accordance with Section II.L.

(iii) Fluid pressure drop in the primary side of the steam generator shall be calculated considering secondary side heat transfer and with the inlet conditions calculated according to Section II.S.

(iv) Steam flow calculations shall be treated in accordance with Section II.R.

(v) Refill and reflood rate calculations shall be treated in accordance with Section II.S.

III. TECHNICAL DISCUSSION

A. Discussion of Criteria

PEAK CLADDING TEMPERATURE AND MAXIMUM OXIDATION

The Proposed Rule, §50.46, Acceptance Criteria (b)(1) and (b)(2)

(b)(1) Peak Cladding Temperature. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

(2) Maximum Cladding Oxidation. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

Discussion of Peak Cladding Temperature and Maximum Oxidation

Criteria (b)(1) and (b)(2) are proposed to replace the single clad temperature limit of 2300°F specified as an Interim Acceptance Criterion in June 1971.

Understanding of the proper limits on cladding behavior has evolved through a number of stages. Each stage represents increased understanding of zirconium embrittlement and other physical phenomena. The stages are all related, but in a complex manner as described in the following paragraphs.

A maximum clad temperature limit is specified in order to preclude clad melting during a LOCA and to limit energy release associated with the zirconium-steam reaction. These reasons were addressed in the staff Direct Testimony (Exhibit 1001, Section 2.2), and they remain as valid justification for specifying a maximum clad temperature limit. However, melting and energy release from zirconium-steam reaction are not the basis for specifying a 2200°F limit; in fact, a 2300°F limit would be sufficient in this regard. A 2300°F limit is also sufficient in the staff's present opinion to limit cladding damage by eutectic formations, even though the staff Supplemental Testimony suggested a 2200°F limit to preclude a damaging amount of zirconium-nickel or zirconium-iron eutectic (Exhibit 1113, Section 19). The staff clarified that earlier suggestion by stating in response to questioning that if the effects of grid spacer flux depression, cladding pre-oxidation, and other factors were considered, a peak cladding temperature of 2300°F would be sufficiently low to limit damage by eutectics (Transcript 20,538-41).

Limits on both the temperature and the degree of reaction were first shown to be necessary to quantitatively limit embrittlement and fragmentation by the quench tests of Hesson (see Exhibit 1113, page 18-1). Several series of quench experiments have since been performed with comparable results (Exhibit 509; Exhibit 1151, page 16-34; Exhibit 1122, Appendix A; Exhibit 1066, Section 2.1 and Transcript 13,374-6; Exhibit 1137, Section 6). Some of the quenched samples from the ORNL experiments (Exhibit 509) were subjected to diametral compression by impact loads. From this information ORNL was able to correlate the degree of cladding embrittlement to the parameter Fw. (Fw is the fraction of material remaining as prior-β phase; therefore a correlation with ξ penetration is a correlation [of different form] with Fw; see Exhibit 1113, Section 18.) Additional metallurgical and slow compression mechanical tests on other quenched samples from the ORNL experiments (Exhibit 1126) indicated that an important consideration was the amount and distribution of oxygen in the nominally ductile prior-β phase. However, these factors could not be correlated as functions of time and temperature in the same manner as the ξ penetration. In particular, the slow compression tests indicated a greater degradation in cladding ductility at higher temperatures than would be expected from considerations of ξ penetration alone. It was on this basis that the staff previously suggested a 2200°F maximum cladding temperature (Exhibit 1113, page 18-18).

To preclude clad fragmentation and to account for effects noted in the tests described above, a limit of $\xi_T/Wo \leq 0.44$ was earlier suggested by the Regulatory staff as an embrittlement criterion (Exhibit 1113, page 18-18). This limit was inferred from quench tests and mechanical tests. Criterion (b)(2) is now proposed as a better method of specifying a similar limit on the extent of cladding oxidation. The bases for proposing this method are described below.

The degradation in cladding ductility as described by the staff (Exhibit 1113, Section 18.0), has been asserted to be an insufficient reason to limit cladding temperature to 2200°F (Westinghouse Concluding Statement, page A-6; B&W Concluding Statement, page 238; GE Concluding Statement, page M-30; CE Concluding Statement, pages 2-8, 9; ECCS Utility Group Concluding Statement, page 39). However, it will be demonstrated below that the phenomenon of degraded ductility of oxidized cladding as a function of temperature is well supported by evidence in the hearing record.

A limit of 2200°F is consistent with the results of the impact tests at ORNL, despite the suggestion that the impact tests showed no inability to correlate ZDT with Fw (B&W Concluding Statement, page 238). In fact, the phenomena suggested by the ORNL slow compression tests could not have been detected by the impact tests. What was observed in the slow compression tests was that 6 samples exposed at 2400°F for only two minutes and with relatively high values of Fw (all greater than 0.65) all fractured with nil ductility. On the other hand, no impact test samples exposed at 2400°F had values of Fw greater than 0.44. With such low values of Fw (large ξ penetrations), brittle failure would be expected. This is consistent with the results observed with samples exposed at lower temperatures which also had low Fw. Only when brittle failure was detected at high Fw in the slow compression tests did the suspicion arise that ductility was a function of both Fw and the exposure temperature.

B&W supports the concept of deviation from linear behavior in correlating the ductility with Fw as the temperature increases (B&W Concluding Statement, page 239). However, B&W asserts that the deviation occurs at a temperature higher than 2200°F. B&W bases their assertion on the belief that the actual temperatures above 2200°F in the ORNL experiments were higher than the reported values (B&W Concluding Statement, page 239). B&W also asserts that the reaction in the ORNL tests was steam limited. The staff believes that the reaction was not stoichiometrically steam limited if the total steam available is considered. However, at the surface of the sample the reaction may have been oxygen limited. The steam flow for both the ORNL and B&W tests was certainly in the laminar regime. In fact this also appears to be the case for both the ANL tests (see Exhibit 1113, page 18-1) and the CE tests (Concluding Statement). If so, then diffusion through the steam boundary layer would tend to decrease the oxygen concentration at the surface.

4471

The staff believes there is some validity to the B&W claims concerning steam-limiting and higher-than-reported temperatures. To answer these questions, future experiments should be performed with steam in the turbulent flow regime. Also, the temperature of the experimental samples should be measured, rather than inferred from the heater temperatures. If the ORNL exposures were indeed at higher temperatures than reported, as claimed by B&W, then any effect of these errors in temperature could not have been serious or the ORNL reaction kinetics data would not correlate so well with other data sets. On the other hand, if laminar flow was a real problem and involved consequent steam limiting, the rate information at high temperatures may be non-conservative.

B&W has expressed concern that several samples from the same experimental tube, nominally isothermal, showed variations in Fw of 30 percent (B&W Concluding Statement, page 247). Such a variation could be caused by imprecision in measurement of the α/β boundary or by axial temperature variations in the tests. The temperature variations are important because alpha incursions into the β phase begin to occur as the oxygen concentration in the β phase approaches saturation and the α/β boundary becomes somewhat irregular. The actual technique used to measure the α/β boundary is very precise and should not be the source of the 30 percent variation. However, the α/β boundary is very irregular, and therefore the reported value of ξ is dependent upon subjective judgment by the experimenter. Although variations in Fw are important in the statistical analysis of the information as applied to rate equation

4971

derivation, these variations are unimportant in correlating the mechanical properties of the clad as a function of Fw since the measured values of Fw and temperature are used.

At the time of the staff Supplemental Testimony (Exhibit 1113, pages 18-14 to 18-16), the ORNL slow compression test results were new. The principles of beta-phase oxygen concentration were acknowledged and discussed (Transcript 21,321-3; 20,935-9; 20,631-8; 21,498-501; 20,627-8), but the principles were not sufficiently developed and supported with calculations and measurement to allow more than the simple limitation of 2200°F. It was stated (Exhibit 1113, page 18-18) that:

> "The staff recognizes the importance of oxygen concentration in the β phase in determining the load bearing ability of Zircaloy cladding, and the implication from the recent compression tests that this may not be satisfactorily characterized above 2200°F by a ZDT as a function of remaining β fraction only."

This principle still holds. Moreover, the considerations of temperature errors and steam limiting serve to reinforce the suggested need for caution in interpreting data taken at high temperatures.

As oxygen saturation is reached in the beta phase, precipitation of alpha zirconium in the grain boundaries and other rapid diffusion paths can occur as shown by the ORNL (Exhibit 1126) and CE (Concluding Statement) experiments for exposures below 2400°F for long exposure times. As the temperature rises above 2200-2300°F, solid solution hardening in the β phase appears to contribute significantly to formation of a brittle structure. That is, brittle failure occurs even though alpha incursions are not observed, and the fraction of remaining β is greater than that observed in lower temperature tests. This is confirmed by examination of the six samples from the ORNL specimen exposed at 2400°F for two minutes (Exhibit 1126). A high fraction of remaining β (Fw >0.65) and no incursion of α into the prior-β region were observed, which observations are consistent with high saturation concentrations at elevated temperatures (See phase diagram, Exhibit 1122, page 2-5). Solid solution hardening is a well known metallurgical phenomenon and does not require saturation concentrations, but just a sufficient concentration, which can occur at the higher temperatures without saturation.

Westinghouse (Exhibit 1151, page 16-34) displays the available quench test data in such a way that the oxygen content required to cause quench failure shows a definite trend to decrease with increasing temperature. This is quite consistent with higher oxygen concentration and possible solid solution hardening in the beta phase at higher exposure temperatures. This is because most of the oxygen goes into the oxide and oxygen-stabilized alpha layers. Therefore, if the relatively small amount of oxygen in the beta layer is controlling, brittleness can occur without the heavy oxide and alpha layers; i.e., with less total oxygen content.

Although CE claims there is no anomaly exhibited by the data for failure loads of specimens exposed at temperatures above 2200°F, they did suggest a trend to lower failure loads with increasing exposure temperature for comparable values of Fw (CE Rebuttal Testimony, Section 2). The hardness measurements presented in Exhibit 1126 also show a trend to increasing hardness, and therefore decreased ductility, with increasing exposure temperature.

From the foregoing there is ample evidence that load bearing ability and ductility decrease with increasing exposure temperatures, even for transients with comparable Fw or ξ_T/Wo. Increased solubility of oxygen in the prior-β phase has been discussed as a contributing factor (Transcript 20,627-59; Exhibit 1113, Section 18). CE has suggested that hardness determinations are a measure of oxygen concentration (Transcript 20,627-59) as well as brittleness. This implies that solid solution hardening may be a contributing factor to the observed degraded behavior at higher temperatures and high solubilities.

GE suggests that quench experiments exhibit transients more severe than BWR LOCA transients in terms of temperature as well as time at temperature (Exhibit 1122, page 2-5; GE Concluding Statement, Section M). However, those discussions do not establish what constitutes a more severe transient. It can be inferred that if the experimental temperature-time curve is outside the LOCA transient curves, GE considers it to be more severe. That is, it is more severe for those times at which the temperature in the experiment is greater than the calculated temperature in the LOCA. It is possible, however, that this is not a sufficient measure of severity. A slower cooldown prior to quench may enhance local oxygen concentration in grain boundaries and thus promote fragmentation. Thus, a specimen which is at a higher temperature for longer time, but is cooled rapidly prior to quench, may actually exhibit greater strength and ductility. In fact, GE cites an example (Exhibit 1122, page 2-5) where a sample, which was slowly cooled, failed. The extent of oxidation in the sample was less than would be expected for failure on the basis of other samples which were rapidly quenched. GE attributes this to long cooldown near the transformation temperature where precipitation of oxide in the grain boundaries is the cause of the brittle behavior. Consideration of the phase diagram as presented by GE in Exhibit 1122 suggests to the staff that in the α zirconium at the boundary between the α phase and the ZrO_2 the oxygen concentration is very high and is relatively independent of temperature. None of the α phase at the α/ZrO_2 boundary has an oxygen concentration as low as the concentration of oxygen at the α/β boundary. Therefore, precipitation of ZrO_2 in the prior-β region is very unlikely. Some small amounts of oxide precipitation might occur near the α/ZrO_2 boundary. Thus a more likely explanation of the GE observation is that α-phase precipitation occurred in the grain boundaries. This could occur during the cooldown until the transformation temperature is reached (about 1550°F).

The slow cooldown and the high temperature phenomena discussed previously can be related. At high temperature, more oxygen is dissolved in the β phase. As cooldown proceeds from high temperature, the concentration of oxygen during phase transformation would produce α phase zirconium with a very high local oxygen concentration, as has been shown experimentally. Thus, important considerations with respect to embrittlement are the cooldown rate and the temperature from which cooldown occurs.

The staff believes that because of high temperature degradation and slow cooldown phenomena (both strongly suggested by the experimental evidence cited) the suggested 2200°F limit should be imposed. After experimental and analytical information becomes available on these phenomena, reconsideration of this limit would be proper.

The staff does not believe that the stress calculations performed by the vendors (Exhibit 1122, Section 4; Exhibit 1151, Section 2; Exhibit 1144, Section 2.2) can be relied upon for making licensing decisions. The calculations have been used primarily to demonstrate that quench loads are so predominant that all other loads can be effectively ignored. The staff believes that quench loads are likely the major loads, but the staff does not believe that the evidence is as yet conclusive enough to ignore all other loads (Exhibit 1113, page 18-9). The oxidation of zirconium proceeds in such a manner that the material can in no way be considered homogeneous. As oxygen saturation is approached in the beta phase, incursions or fingers of alpha zirconium protrude into the beta region. During cooldown and prior to quench "precipitation" can occur along grain boundaries. Even without this irregular formulation of oxygen rich α-zirconium, the stress calculations are of doubtful quantitative value because an oxygen gradient would exist in the β phase. Property data for variously oxidized zirconium samples are simply not available (Exhibit 1113, page 18-9), and the important mechanical properties very likely are strongly related to oxygen concentration and distribution. In short, property data for varying oxide concentrations, distributions, and applicable morphologies must be obtained before a calculational approach to stresses and loads can be relied upon. It is worth noting that two vendors suggested the incorrectness of the calculational approaches taken by each other, and in particular the role of the oxide in these stress calculations (GE Concluding Statement, Section M; Westinghouse Concluding Statement).

The staff recognizes that the failure mode in ring compression tests is not the same as would occur during a LOCA (GE Concluding Statement, Section M). However, such tests do allow investigation and quantification of certain phenomena not readily studied in quench tests. Load deflection data allows assessment of stress and strain relationships. Ductile-brittle transition can be assessed at known temperatures (ZDT), whereas the precise quench temperatures in quench experiments are not known. Maximum cross head travel in an experimental apparatus is not as important a concern as has been suggested (Exhibit 1078, Section 2.2.6; B&W Concluding Statement, page 259). What is of interest is the strain (elastic and plastic) up to failure. In fact, Combustion Engineering (Concluding Statement) has suggested that determination of strain for a sample subjected to ring compression could, in part, form the basis for defining a limit on degradation of cladding ductility. However, the staff does not believe that strain can now be determined from either diametral compression tests or from calculations with sufficient accuracy for licensing evaluations.

Various methods for defining a clad embrittlement limit have been suggested. Several have been compared by GE (Exhibit 1122). All methods define a numerical limit and a calculational procedure for comparing temperature histories to that limit. Any method can be made more conservative by adjusting the limit or the calculational procedure.

Westinghouse suggested that a limit based on a parameter relating to oxygen in the β phase would be better than a limit based on the value of ξ_T/Wo or equivalent zirconium converted to oxide (Concluding Statement, Section III-B). The staff agrees that this would probably be more meaningful than some of the other methods discussed above. However, the state of the art does not now permit assessment of such a β phase parameter (only closed solutions for semiinfinite geometry were presented in the hearing; no finite difference solutions were presented).

Westinghouse (Concluding Statement, Section III-B), GE (Concluding Station, Section M), B&W (Concluding Statement) and the ECCS

Utility Group (Concluding Statement) favor a limit on percent oxidation (defined as the percent of cladding converted to oxide if all the oxygen absorbed by the cladding were converted to stoichiometric ZrO_2). They also favor more realistic equations such as Klepfers or the Westinghouse correlation (Concluding Statement, page A-10) to calculate the percent oxidation. Table 1 presents a summary of recommendations by various participants.

GE (Concluding Statement, Section M), B&W (Concluding Statement), and the Utilities (Concluding Statement) assert that percent reaction, since it includes total oxygen uptake, satisfactorily accounts for the oxygen in the β phase. However, percent oxygen does not in any direct manner account for the concentration distribution throughout the β phase. As an example, a sample at 2200°F, if exposed for a sufficient time, could be expected to have an average β phase oxygen content greater than 5000 ppm. However, a sample exposed at 2100°F could never reach this concentration because the saturation concentration is less than 5000 ppm at 2100°F (see phase diagram presented by GE in Exhibit 1122). Although the amount of oxygen in the β phase may be a small fraction of the total oxygen (about 4-20 percent), it is an important factor in determining the load bearing ability of the cladding.

Although percent reaction is not the most direct method for characterizing degraded cladding, certain features of the 17 percent reaction limit support its use as a limit, at least for the present.

TABLE 1

RECOMMENDATIONS FOR EMBRITTLEMENT CRITERIA AND METHODS

Participant	Source	Temperature Limit °F	Oxidation Limit	Inside Reaction	Clad Thinning Percent Expansion	Zirc-Water Reaction Equation
Utilities	Conclusions	2500	12-17% clad reacted	0-75% of outside	0-40	Klepfer
B & W	Conclusions	2400	19% clad reacted	0	0	Klepfer
G.E.	Conclusions	2300*	None*	0	0	Baker-Just
C.E.	Conclusions	2500	Fw > 0.65	2/3 of outside	50%	C.E.
Westinghouse	Conclusions	2700	$\xi t/wo < 0.47$**	0	0	Westinghouse
Staff	Rebuttal	2200	$\xi t/wo < 0.44$	100% Baker-Just after rupture***	Calculate	Baker-Just
Staff	Conclusions	2200	17% clad reacted	100% Baker-Just after rupture***	Calculate	Baker-Just

* G.E. believes 2700°F and 17 percent reaction are better limits but does not recommend any change from Interim Acceptance Criteria.

** Westinghouse states that this is equivalent to 16 percent clad reaction.

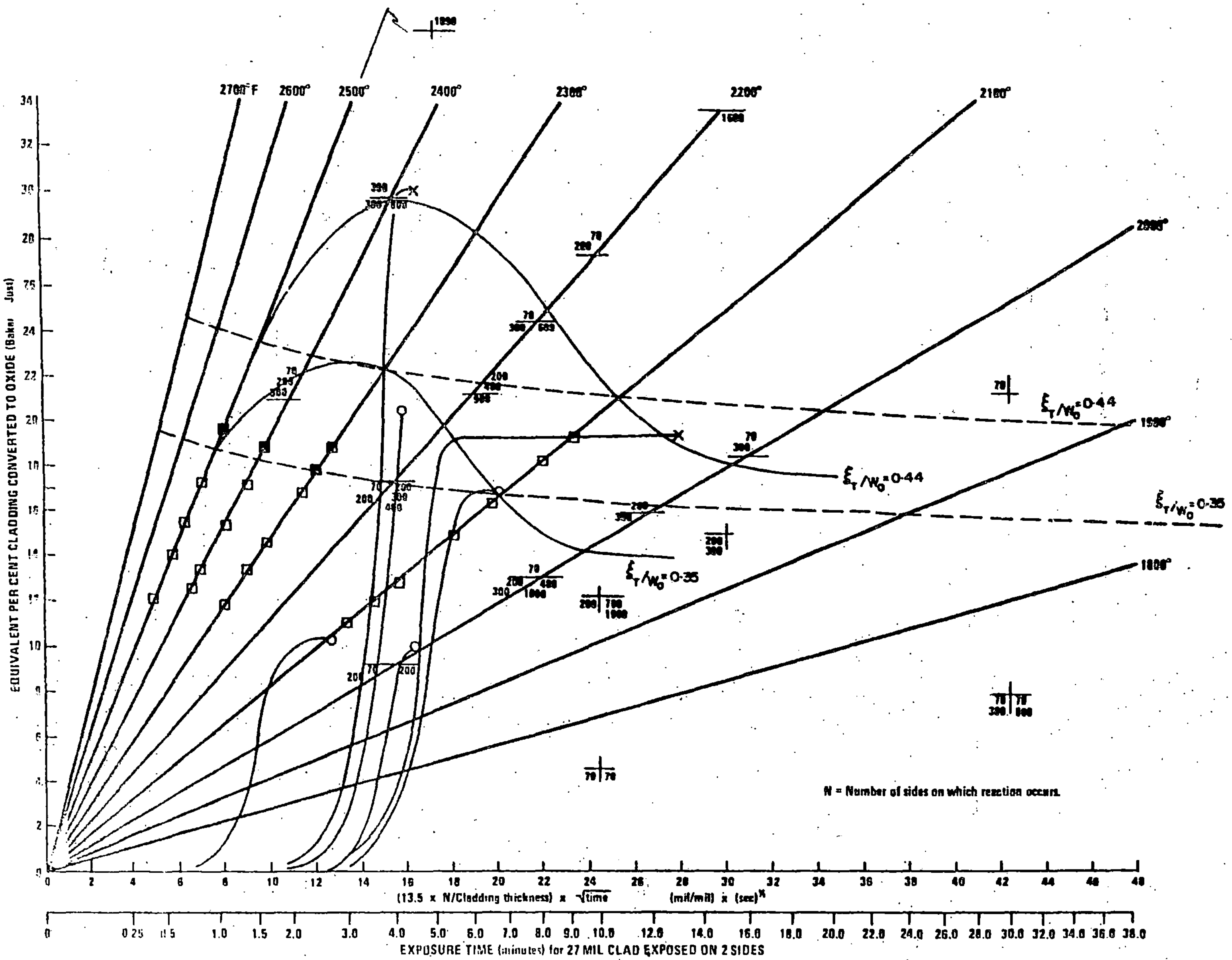
*** Within 1.5 inches of the center of the rupture.

The 17 percent reaction limit has been well tested as a limit for quench experiments (Exhibits 509; 1151, page 16-34; 1122, Appendix A). Its use with the Baker-Just equation is conservative when compared to the previously suggested limits of ξ_T/Wo <0.44. This is shown in Figure 1 for isothermal conditions. Four lines of constant calculated ξ_T/Wo are constructed on the plot of percent reaction versus a parameter proportional to the square root of exposure time. The solid ξ_T/Wo lines are based on Pawels equation (Exhibit 1133), and the dashed lines are based on Exhibit 509, page 9, Figure 5. As can be seen, the ξ_T/Wo = 0.44 lines are both above the 17 percent reaction line. The ξ_T/Wo = 0.35 lines become more conservative than the 17 percent limit for temperatures below about 2100°F for isothermal exposure. This 0.35 value of ξ_T/Wo is nearly equivalent to the Fw = 0.65 recommended by CE (Concluding Statement, Section 2.1.).

Upon reconsideration since the Supplemental Testimony phase of the hearing, the Regulatory staff does not believe that evaluation of several different methods for calculating ξ_T/Wo would be suitable. It should be noted by comparing the solid and dashed ξ_T/Wo curves that the method of calculation is extremely important. Both solid and dashed curves were derived from the same data set (Exhibit 509). The dashed curves were based on an Arrenhius type equation such as Baker-Just; whereas, the solid lines were strictly empirical without particular regard for kinetic theory. It is apparent from the figure that different oxidation correlations have advantages in different regions. In

40601

FIGURE 1 CALCULATED EQUIVALENT PER CENT CLADDING
Converted To Oxide vs. Exposure Parameter



particular the empirical correlation is very beneficial in the regime from 2200°F to 2400°F. That is, the same isothermal time-temperature exposure that results in a calculated $\xi_T/Wo = 0.44$ using the empirical equation results in about 25 to 30 percent clad converted to oxide using Baker-Just in this temperature range. It is in this particular regime that the staff is urging caution. The upward trend of the dashed lines with increasing temperatures indicate that even Pawel's Arrenhius equation tends to be beneficial at higher temperatures. Therefore, Baker-Just with its known increasing conservatism with increasing temperature is preferred (see discussion for Section II.C.5).

Of primary interest is the comparison of the proposed oxidation limit (17 percent equivalent cladding converted to oxide) and the proposed method (Baker-Just) with experimental data. By comparing calculated total oxidation as a function of $\sqrt{t}$ to the oxidation limits for various experiments, isothermal data may be evaluated quite readily. Isothermal calculations for parabolic behavior are represented by straight lines of constant temperature in Figure 1. Variations of thickness and whether or not the reaction was considered to be one-sided or two-sided (N = 1 or 2) can be accommodated in this figure. The ORNL slow compression and impact tests (Exhibit 509) are represented as horizontal intersections of the reaction isotherms. The numbers represent deformation temperatures. Numbers to the right of the isotherms are impact tests, to the left are slow compression tests.

If the number appears above the horizontal line the sample or samples failed with zero ductility as determined by the experiment. By this criterion no samples tested by slow compression above 200°F failed with zero ductility if the calculated reaction was less than 17 percent. All of the failed samples tested under impact had greater than 17 percent calculated reaction. The CE single sided data discussed in the hearing (Transcript 13,374-6) are represented by squares on the isotherms. If the sample fractured on compression by CE's loading standard, it was considered to have failed and is represented by a filled-in square. By this standard only samples above 17 percent calculated reaction failed. To use Figure 1, oxidation may be calculated for transient cases by standard methods (see for example Exhibit 1133). However, the calculated transient lines have no relationship to the calculated isotherms. Also, since the lines of constant ξ_T/Wo are based on isothermal calculation and keyed to Baker-Just isothermal curves, they do not represent true ξ_T/Wo limits for transient oxidation. Five of Hesson's transients (see Exhibit 1113, page 18-1) were presented by GE (Exhibit 1122). These transients were integrated using the Baker-Just equation and are shown in Figure 1. If the sample failed as defined by Hesson, the curve is terminated with an X. A circle indicates termination of a transient for an unfailed specimen. The comparison with the 17 percent limit is good, two transients failed with reactions calculated to be about 19 and 30 percent and three survived with reactions calculated to be 10, 10, and 20 percent. The 17 percent limit satisfactorily bounds the behavior of calculated transient oxidation for the failed specimens.

Two additional experimental data sets are worthy of note. Westinghouse experiments showed that only their samples with a reaction measured to be greater than 17 percent failed upon quench (Exhibit 1151, page 16-34). However, the transients are not amenable to Baker-Just calculation since they were exposed in air. The General Electric TTE experiments (Exhibit 1122, Appendix A) demonstrated approximately the same quench failure behavior. However, the effect of the reaction of alumina with the inside of the cladding makes the comparison with results of calculations somewhat doubtful.

In summary, the very good comparison between the 17 percent limit and a wide variety of experiments calculated with the Baker-Just equation supports adoption of this procedure.

The staff believes that properly designed experiments would be necessary in order to justify reconsideration of the conservative embrittlement criterion we have now proposed. Quench and metallurgical examination should be included in such experiments. Suggestions for examining slow cooldown, diffusion, and reaction rates have already been made (Exhibit 1113).

MAXIMUM HYDROGEN GENERATION

The Proposed Rule, §50.46, Acceptance Criterion (b)(3)

(3) Maximum Hydrogen Generation. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

Discussion of Maximum Hydrogen Generation

The proposed maximum hydrogen generation limit is essentially the same as the previous Interim Acceptance Criterion 2. However, there is now a more detailed specification of the amount of metal to be considered in treating this important source of hydrogen (Staff Testimony, Exhibit 1001, Section 2.3). Combustion Engineering suggests that because all past criteria, and now the recommended criteria of the Proposed Rule, apply to the core hot spot, limits on core-wide zirconium-water reaction are unnecessary (CE Concluding Statement, Section 2.2). The CE suggestion is true with respect to cladding integrity, but it does not address the hydrogen generation aspects of the criterion. That is, the fact that core-wide zirconium-water reaction has not been the limiting criterion for LOCA transients considered to date does not negate the soundness of establishing such a limit. Elimination of this criterion on the basis of past calculations prejudges the nature of calculated transients in the future. For example, reductions in reactor power peaking factors have occurred in the past, and they can be anticipated for the future. Each such reduction results in more cladding being at higher temperatures in LOCA analyses, and, hence, there is a potential for increased hydrogen generation. Therefore, the upper limit of one percent should be maintained for maximum hydrogen generation.

COOLABLE GEOMETRY

The Proposed Rule, §50.46, Acceptance Criterion (b)(4)

(4) Coolable Geometry. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

Discussion of Coolable Geometry

The question of maintaining a substantially intact (and thereby coolable) geometry requires simultaneous consideration of LOCA induced environments and associated fuel cladding response. The previous Interim Acceptance Criterion 3 reads:

"The clad temperature transient is terminated at a time when core geometry is still amenable to cooling, and before the cladding is so embrittled as to fail during or after quenching."

Since definition of coolable geometry is implicitly dependent on calculational procedures, the staff has now proposed the above revised wording for this criterion.

Coolable geometry aspects were discussed throughout the hearing, but in terms of related phenomena, viz: cladding embrittlement (Transcript 9685-8; 12,860-70; 12,144-6); cladding ductility (Transcript 3059; 9555-6); cladding maximum temperature (Transcript 6220-6); coolant blockage (Transcript 4117-4300; 9473-4; 11,226-234); clad swelling (Transcript 9707-8; 9780-4; 14,016-7). Coolable geometry aspects must therefore be reviewed in terms of such considerations.

Embrittlement of the fuel cladding is a key factor in coolable geometry considerations (Exhibit 1001, pages 2-8 to 2-9) since it can result in potential clad failure with attendant coolant channel blockage effects. Embrittlement (which is inseparable from ductility considerations for LOCA evaluations) is discussed in the Staff Testimony (Exhibit 1001, page 2-1 to 2-10) and Supplemental Testimony (Exhibit 1113, Sections 18 and 19). As discussed in preceding sections, the staff's conclusions regarding embrittlement (or loss of ductility) have resulted in recommendations for revised criteria which limit clad temperature to 2200°F and which specify that the calculated total oxidation shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

Clad ductile behavior, characterized as swelling or ballooning during the hearings, can result in coolant channel flow blockage with attendant flow redistribution or altered heat transfer behavior. The staff's review of opinions and concerns regarding flow blockage, and its coolable geometry aspects, is contained in Section 20 of Exhibit 1113. In brief, the staff's conclusions based on this evidence are, for PWR's:

1) Oscillatory reflood behavior (as typified by PWR-FLECHT-SET) may provide an enhanced heat transfer rate. Also, FLECHT tests with blockage plates indicated better cooling at the location of measured peak clad temperatures than did FLECHT tests without blockage.

2) Calculations show that although high core-wide (e.g., 60 percent in the radial plane) flow blockage causes increased core pressure

drops, there are insignificant changes in total core flow during both blowdown and reflood.

3) Once ballooning, or rupture, is predicted to occur during blowdown, a 20 percent reduction in hot channel flow is appropriate to account for flow redistribution between subchannels in the hot region.

4) Heat transfer and steam flow redistribution effects related to calculated blockage should be evaluated for the early portion of the reflood transient for which droplet entrainment has not been established, or for which calculations indicate that oscillations are not significant enough for a two-phase mixture to reach the core midplane.

For BWR's the staff concludes on the basis of Exhibit 1113, Section 20, that the effects on heat transfer of clad swelling without rupture are small and will not increase clad temperature more than 60°F above those currently calculated. For example, the Zr2K tests (which had extensive ballooning) showed no measureable change in convective heat transfer during the spray cooling mode (Exhibit 1113, Section 16).

The following comments are offered in view of expressed concerns regarding flow blockage (e.g., Exhibits 1041, 1044 and 506). First of all, CNI described core blockage as producing an "uncontrollable event" (Exhibit 1041, Chapter 7). This position has been tempered in CNI's Concluding Statement (pages 5.3-5.13) which discusses flow blockage in terms of "knowledge of extent of blockage" and "knowledge of the consequences of flow blockage." The staff concludes that evidence contained in its Direct and Supplemental Testimony (Exhibit 1001, Section 2.4 and Exhibit 1113, Section 20) is a proper basis for rejecting CNI's "uncontrollable event" hypothesis. Furthermore, the staff cautions against evaluating flow blockage phenomena in terms of only partial understanding of selective experimental evidence without suitable interpretation and transposition of this information into credible LOCA circumstances (e.g., see Exhibit 1041 and ensuing questioning of CNI by General Electric at Transcript Volume 96, <u>in toto</u>).

CNI's Concluding Statement, pages 5.3-5.11, depends strongly on statements regarding fuel pin swelling which were made by ORNL (e.g., Exhibit 506 and Transcript 4909-28), and it refers to ORNL calculations of channel blockages in excess of 100 percent (page 5.5). ORNL personnel testified (Transcript 8629-31) that their clad swelling tests and analyses were based on static conditions. They also testified that the expected conditions for a LOCA would require a dynamic analysis. The Westinghouse Concluding Statement (pages C-23 to C-34), summarizing the results of questions addressing ORNL's flow blockage calculations, points out a series of errors and discrepancies in Exhibit 1050 (see Westinghouse Concluding Statement, pages C-33 and C-34).

The staff has been aware of blockage problem areas (e.g., concerns expressed in Exhibits 506 and 1044) and these were reviewed again during preparation of Supplemental Testimony (Exhibit 1113). In addition questions were raised by ORNL concerning flow blockage effects on clad temperatures during reflood (pages 20-12 to 20-15 of Exhibit 1113). The staff has concluded that these ORNL evaluations are not valid for the entire span of reflood heat transfer up to turnaround (page 20-15 of Exhibit 1113), because they are not substantiated by the existing experimental data. However, early reflood heat transfer in the period of steam-only cooling can be influenced by local blockage, as discussed under Section II.Q, below.

CNI's insistence (Concluding Statement, page 5.4) that:

"The extent and distribution of blockage in a core is a prime input to the proper analysis of a LOCA."

is inconsistent with observed physical phenomena. Flow blockage is not a prime input to a LOCA analysis since it would constitute an a priori assumption. Rather, the staff concludes that flow blockage should be treated in concert with evaluations of clad swelling and rupture; fuel thermal parameters; transient gap conductance; core flow distribution; cladding-water reactions; and reflood heat transfer. Therefore, the staff rejects the proposal that has been suggested in the proceedings; namely that a specific penalty be assigned to the peak cladding temperature to account for the effects of flow blockage and redistribution (see page 7-13, Exhibit 1113). Instead, the staff recommends adoption of suitable evaluation models which address the specific physical phenomena, thereby providing a means to assess flow blockage effects and attendant clad temperatures.

Further specific opinions related to coolable geometry are summarized in the following paragraphs and then related to the staff's conclusions.

B&W's Concluding Statement discusses flow blockage and attendant clad swelling during both blowdown (pages 165-184) and reflood (pages 223-231). B&W concludes (page 183) that clad swelling does not have a significant effect on cladding temperature response during the blowdown phase of the LOCA, and is "the product of constraints imposed in the calculational methods." B&W also addresses flow redistribution and fuel element clad swelling as it pertains specifically to the reflood phase and concludes (page 231) that "the effects of fuel element clad swelling and flow redistribution during the reflood stage of the LOCA are inherently small."

CE's Concluding Statement recommends a cladding oxidation criteria (Section 2.1) based on temperature-time exposure limits, and states (page 2-17) "that a coolable geometry will be maintained in the core during a postulated LOCA" if the criterion recommended by CE is adhered to.

The Westinghouse Concluding Statement (Appendix B) discusses assembly flow blockage and summarizes (page B-4) "the technology detailed by Westinghouse in its testimony provides an adequate technical base and clear calculational approach for the evaluation of assembly flow blockage for calculating the generic local effects of a loss-of-coolant accident transient."

GE's Initial Closing Statement, Volume 2, pages 0-1 to 0-4, briefly discusses flow blockages as a "highly localized phenomena of clad bulging" and concludes (page 0-4) "The overwhelming weight of evidence (and all of the reliable evidence) indicates that flow blockage is not a concern for the BWR." and "...flow blockage considerations do not warrant any changes in the GE model."

All participants have relied on calculational techniques related to observed or measured phenomena to establish their views and opinions regarding coolable core geometries. Therefore, they have all implicitly agreed with the staff conclusion that "coolable geometry" is intrinsically related to the use of calculational techniques (evaluation models).

In summary, the staff concludes that the concept of coolable geometry is appropriately addressed via the evaluation-model requirements outlined in Appendix K of the Proposed Rule. Several features of these models, typified by Chemical Reactions and Heat Sources (II.C); Clad Swelling and Rupture (II.H); Initial Stored Energy in Fuel (II.I); Fuel Rod Thermal Parameters During Postulated Accident (II.J); Core Flow Distribution During Blowdown (II.O); and Reflood Heat Transfer (II.Q), when taken together provide an interactive means to evaluate "coolable geometry" aspects. Swelling (or ballooning), steam-water reactions, gap conductance, flow redistribution, and reflooding rates, would therefore be analyzed in detail. Thus, changes in core geometry can be accounted for and evaluated to determine if they are such that the core remains amenable to cooling.

LONG-TERM COOLING

The Proposed Rule, §50.46, Acceptance Criterion (b)(5)

(5) Long-Term Cooling. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Discussion of Long-Term Cooling

The need for long-term cooling of the core results from the long-term nature of the fission-product decay process. Following the termination of the fuel cladding temperature transient there is a substantial amount of decay heat generated in the core (Exhibit 1001, page 2-25). This proposed criterion assures that adequate provisions to remove the decay heat load are incorporated in the design of the facility (Exhibit 1001, page 2-25).

The Interim Acceptance Criteria, and therefore this specific criterion, have been supported by both Babcock and Wilcox (Exhibit 1059, page 8.1) and Westinghouse (Exhibit 1078, page 84). General Electric states that (Exhibit 1069, page 72): "Attainment of this objective [long-term cooling] is an appropriate requirement." Combustion Engineering also supports this specific criterion (Exhibit 1066).

The staff reaffirms its previous conclusion (Exhibit 1001) and recommends that the above criterion be adopted.

B. Discussion of Evaluation Models

BREAK CHARACTERISTICS AND FLOW

The Proposed Rule, Appendix K, Section II.B

B. Break Characteristics and Flow

1. The spectrum of LOCA's specified and defined in 10 CFR §§50.46(a) and (c) shall be analyzed.

2. Where the fluid reaching the break is calculated to be subcooled or saturated liquid, a discharge model appropriate to these conditions shall be used to calculate break flow.

3. For the period of transition from saturated liquid to low-quality two-phase fluid at the break exit plane, a discharge model appropriate to these conditions shall be used to calculate break flow.

4. Where the fluid reaching the break is calculated to be a two-phase fluid, or saturated vapor, the Moody discharge model (Moody, F.J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Journal of Heat Transfer, Trans. American Society of Mechanical Engineers, 87, No. 1, February 1965) shall be used to calculate break flow.

5. Over the entire spectrum, the postulated break shall be assumed to occur instantaneously and shall be modeled as discharge from a single node, as though through an open pipe having the postulated break area, and with a Moody multiplier (MM) of unity were the Moody model is used.

6. For postulated breaks in pressurized water reactor inlet lines and boiling water reactor recirculation lines, analyses shall be made assuming that the pipe fails as a complete instantaneous severance (guillotine). Flow shall be assumed to occur unimpeded from both ends of the open pipe without interaction between the discharging fluid streams. This model shall be used with at least three constant values of MM ranging from 0.6 to 1.0. If the trend is for peak cladding temperatures thus calculated to increase as MM decreases, the range of MM shall be extended to smaller values until the maximum peak clad temperature has been reached; that is, until further reduction in MM results in a lower peak clad temperature.

7. Noding detail in the vicinity of the break shall be sufficient to assure that the flow discharge calculation is performed with appropriate local fluid conditions.

Discussion of Break Characteristics and Flow

The earlier recommendations by the Regulatory staff (Exhibit 1113, Section 5) regarding the calculation of flow from the broken pipe during blowdown have been simplified in the Proposed Rule. In Exhibit 1113 the staff recommended that certain blowdown analyses be performed with the Henry-Fauske critical flow model in addition to the break spectrum blowdown analyses with the Moody model. The Proposed Rule does not require the Henry-Fauske calculations because the staff has concluded that the Moody model in conjunction with analyses of a spectrum of break sizes is sufficient for the purposes of assessing ECCS performance. This conclusion is based on the evidence that the Moody model overpredicts the rate of blowdown, and, therefore, realistic predictions of blowdown, such as obtained with the Henry-Fauske model (Exhibit 1113, page 5-7), are necessarily contained within the required analyses of a spectrum of breaks (Exhibits 1069, 1148, 1006A, 1059, and Transcript pages 7484; 14,382; 20,789). Section E of the GE Concluding Statement gives a thorough delineation of the differences between PWR's and BWR's with respect to the calculation of break flow. As stated above, the Regulatory staff agrees with GE's conclusion concerning the lack of need to repeat blowdown calculations with the Henry-Fauske model (see pages E-12 to E-14 of GE Concluding Statement), but we have reached this conclusion with respect to both PWR's and BWR's.

The staff has further concluded that the discharge flow model of Fauske (Exhibit 1105) as proposed by CNI (Exhibit 1069, Section 8) is inappropriate for use in calculating blowdown from a postulated pipe break for either PWR's or BWR's for reasons set forth in Exhibit 1113, Section 5, page 5-3 to 5-4; the conditions described there, for which the Fauske model is applicable, cannot occur for pipe breaks in either PWR's or BWR's. B&W and GE also address the inappropriateness of the Fauske model for PWR's and BWR's on pages 108 to 111 and in Section E, respectively, of their Concluding Statements.

In its Concluding Statement GE concludes that BWR LOCA analyses are insensitive to the assumed break characteristics. The noding and modeling sensitivity studies required by Section III.A, Appendix K of the Proposed Rule provide a method for systematic evaluation of analytical models of such phenomena as break character (split or guillotine). The staff concludes that the recommendations of the GE Concluding Statement, pages E-14 to E-25 and specifically number (2) on page E-22, should be evaluated, just as other noding configurations should be evaluated, in accordance with the proposed Section III.A.

The Westinghouse Concluding Statement on page 79 recommends that a criticality flow check be incorporated in SATAN-V for internal choking calculations. The staff agrees in principle with this recommendation for application to all PWR evaluation models. However, the staff concludes that additional information is needed to describe the form of the flow check to be used. Technical review of various methods of performing such a flow check would be required (Transcript pages 14,498 to 14,507) and the methods should be evaluated in accordance with Section III.A Appendix K of the Proposed Rule.

The staff has concluded that the critical flow model of Moody is appropriate for use in break spectrum analyses of blowdown transients in BWR's and PWR's on the basis that it overpredicts blowdown flow (Exhibits 1148, Section I; 132; 1113, Section 5; and Transcript pages 21,530 to 21,537; 21,394 to 21,398; and 21,421 to 21,433) whenever the break exit plane quality is greater than about two percent (Transcript pages 21,422; 15,555; 10,790). However, for the blowdown period during which subcooled liquid, saturated liquid, or low quality two-phase fluid exists at the break exit plane, the Moody model underpredicts experimental discharge data (Exhibits 1151; 1113, Section 5; 1137, Section 3; and Transcript pages 21,422-23). Therefore, the Proposed Rule requires the use of a model which is more appropriate to these fluid conditions. One such model contained in the evidence of this proceeding is the modified Zaloudek model of Westinghouse (Exhibit 1151, Section 3). The Moody model may also be applicable for early times during blowdown before the exit plane quality reaches two percent if it is used with a Moody multiplier of greater than unity (Exhibits 1113, Section 5; and Transcript pages F1 to F6, and 12,949). The staff concludes on the basis of this evidence that models appropriate to these flow regimes do exist. However, additional information describing how those models are incorporated in the computer programs should be evaluated in accordance with the proposed Section III.A of Appendix K.

DECAY HEAT

The Proposed Rule, Appendix K, Section II.C.1-4

C. Chemical Reactions and Heat Sources. The following chemical reactions and sources of heat shall be accounted for as a function of time and other variables as follows:

1. The reactor shall be assumed to have been operating continuously at a power level no lower than 1.02 times maximum licensed power level (to allow for instrumentation error), with the maximum peaking factor allowed by the technical specifications. A range of power distribution shapes and peaking factors representing power distributions over the core lifetime shall be studied and the combination selected that results in the most severe calculated consequences for the spectrum of postulated breaks and single failures analyzed.

2. Fission heat shall be calculated using reactivity and reactor kinetics. Shutdown reactivities due to temperatures and voids shall be given their minimum values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors studied in paragraph 1 of this Section C. Rod trip and insertion may be assumed if they are calculated to occur.

3. Radioactive decay of actinides, including neptunium and plutonium generated during operation, as well as isotopes of uranium, shall be calculated in accordance with fuel cycle calculations and known radioactivity properties. The most unfavorable time in the fuel cycle shall be assumed, independent of whatever such assumption was made in connection with paragraph 1 of this Section C.

4. Radioactive decay of fission products shall be estimated using 1.2 times the values for infinite operating time in the ANS Standard (Proposed American Nuclear Society Standard - "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," Approved by Subcommittee ANS-5, ANS Standards Committee, October 1971). The fraction of the gamma decay energy generated that is deposited in the fuel (including the cladding) may be equal to or less than 1.0; if the value used is less than 1.0, it shall be justified by a suitable calculation.

Discussion of Decay Heat

Section II.C of Appendix K of the Proposed Rule requires, for calculations involving the fission product decay heat, the use of the proposed ANS standard for decay heat with an added factor of 20 percent for uncertainties. Additionally, the assumption of infinite operating time at maximum peaking factors allowed by the technical specifications is required. As was discussed in the Regulatory Staff Testimony (Exhibit 1001, 3-25) and Supplemental Testimony (Exhibit 1113, Section 22) the decision to use 1.2 times the proposed ANS standard was based on a study of available literature sources on theoretical and experimental work on the subject, including (but not limited to) the report by K. Shure (Exhibit 1001, page 3-25) on the correlation which was used as a base in the proposed ANS standard for the most representative values of decay heat, and experimental work subsequent to the Shure studies. The 1971 staff review (Exhibit 1001) concluded that the Shure curve of decay heat could serve as a reasonable best estimate of decay heat and that the calculational and experimental information suggested an uncertainty about this base of the order of 15 percent. Thus, the use of the proposed ANS standard with its stated uncertainty of 20 percent could serve as a convenient reference for a conservative standard for decay heat. The use of this model is intended to provide a reasonable degree of assurance that the actual decay heat will not exceed that used in calculations. It is conservative in the following sense: the available information indicates a high probability that the correct values are close to the Shure curve; therefore, there is a high probability that such a standard provides a considerable margin in calculations.

No experimental evidence concerning decay heat has been presented in the course of the hearings or elsewhere that is in disagreement with the staff conclusions in this regard. With one exception, the various participants agree that the ANS + 20 percent prescription constitutes an adequately conservative formulation (see Concluding Statements of: the ECCS Utility Group, 69; Combustion Engineering, 3-57; General Electric, H-3; Babcock and Wilcox, 87). The exception is that presented by CNI (CNI Concluding Statement, 5.15-17 and CNI Rebuttal Testimony, Exhibit 1153, page 2.1-6). This disagreement is stated to be based primarily on the doctoral thesis work of T. R. England. England presented predictions of decay heat which resulted from his calculations using an improved computer program of the type which combines contributions from individual fission products to synthesize the decay heat.

The staff carefully considered the England work in its Supplemental Testimony (Exhibit 1113, Section 22). The primary contribution to ECCS-related decay heat concerns resulting from England's calculation was an indication that neutron capture effects in the fission product chains for large neutron-flux-time histories cause important increases in decay heat release rates. However, as discussed in Section 22 of Exhibit 1113, it is important in using the England calculations to consider carefully the flux-time histories appropriate to power reactor operation. It is incorrect simply to peruse generalized introductory statements in England's thesis and pick out a 30 percent increase in decay heat, as was done in the CNI Supplemental Testimony (Exhibit 1153, page 2.3). The staff carried out an analysis of the England work, using the England thesis results, in conjunction with an attempt to get the England computer code operational for other studies. This analysis was presented in the Staff Supplemental Testimony (Exhibit 1113, Section 22).

The staff analysis showed that using the England results directly from the thesis, even though these results involve excessive fuel burnup (Exhibit 1113, pages 22-10 and 11), and applying them to appropriately conservative power reactor flux-time histories gave decay heat rates well within the ANS standard plus 20 percent.

The staff also received, near the end of its review (Exhibit 1113, page 22-14), the results of a recent analysis by K. Shure (Exhibit 1178), updating and evaluating the England work. Shure's work corrected errors in the computer code England had used, improved its input data, and carried out calculations using more appropriate fuel operating histories (without excessive burnup). This work is, of course, of special interest: it serves as an additional review by a person well known in the field and working at the laboratory (Bettis Atomic Power Laboratory) where England's work on fission products had begun. Shure found in the review that (Exhibit 1178, page 11),

> "The dramatic effects of neutron absorption . . . due to an increased neutron flux level are almost completely attributable to the very significant fission rate history difference as a consequence of increased fuel depletion."

The results of this study were summarized in the last sentence of the abstract of K. Shure's paper (Exhibit 1178, page 1),

> "For practical $\underline{U}^{235}$ fueled reactors, it is shown that neutron absorption effects on fission product energy release are unimportant and the $\underline{U}^{235}$ fission product energy release values in the proposed ANS standard on the subject are within a few percent of the values obtained from two recent programs and their updated libraries."

Thus, it can reasonably be concluded that a thorough review of the problem initiated by the England thesis confirmed that the ANS standard plus 20 percent is conservative for use in LOCA calculations.

The ORNL work reviewing the experimental data relevant to decay heat which was mentioned on pages 22-15 and 16 of Exhibit 1113 has been completed. Since this completed work is not on the record of this proceeding, it was not relied upon by the staff in reaching the conclusions stated above. However, it is comforting to know that in its final form it introduces no unexpected information sources and contains nothing inconsistent with our own reviews of the experimental information; i.e., expected values are close to the Shure curve (for

4483

shutdown times of interest) and include adequate allowance for uncertainties; decay heat is within Shure plus 20 percent.

It has been pointed out (B&W Concluding Statement, pages 84-87 and the B&W supplemental Testimony, Exhibit 1137, page 11-3) that the IPS prescription for determining decay heat contains an inherent conservatism above and beyond any consideration of data uncertainties. This conservatism results from the required use of an infinite irradiation time at design peaking conditions. Since no reactor, in general, and, in particular, no peak power density fuel pellet operates at design peaks for an infinite time, this requirement does indeed provide an additional conservatism, but one difficult to quantify. Discussion and examples of probabilities or fraction of time near design peaking factors have been given in the above cited B&W reports and in the Staff Supplemental Testimony, Exhibit 1113, pages 2-3 to 6.

Using the formulation given in Exhibit 1113, page 22-6 and Figure 22.2 for finite irradiation time, the decay heating rate is reduced by 2 and 4 percent 100 and 1000 seconds after shutdown, respectively, for continuous operation at design peaking for 10,000 hours rather than for infinite time. Operation for 10,000 hours at 0.8 of design peaking followed by 1 day operation at design peaking just before shutdown reduces the heating rate by 5 and 8 percent at 100 and 1000 seconds after shutdown. In an example cited by B&W (Exhibit 1137, pages 11-3 to 5) xenon-produced power peaking, resulting from a severe power change maneuver prior to the LOCA, reduces the decay heating compared to the "infinite" assumption by more than 15 percent in the same shutdown time range.

The staff recognizes the existence of this conservatism, and believes that it is prudent to preserve it as an additional, if not precisely defined, margin in the calculation.

ZIRCONIUM-STEAM REACTION

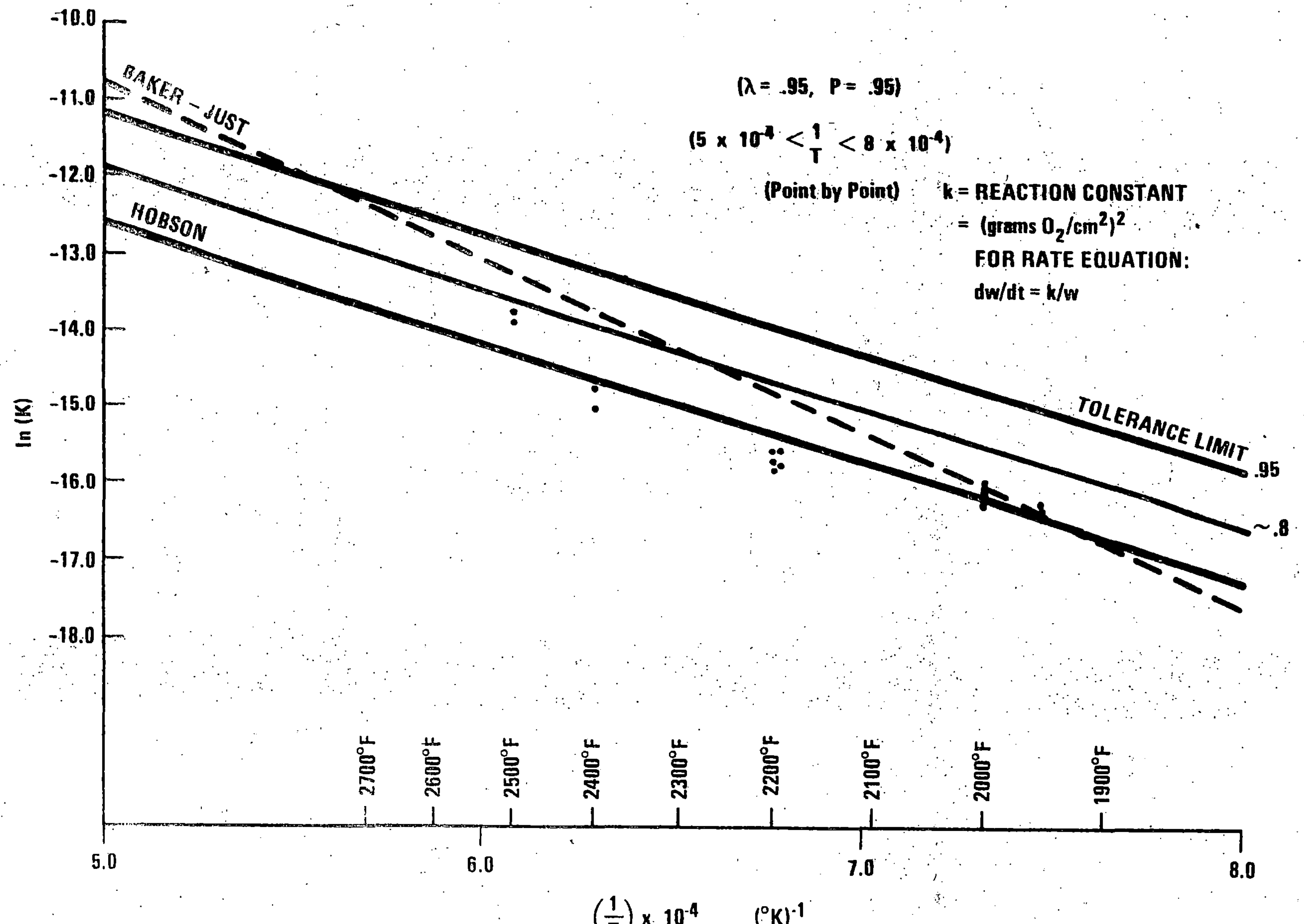
The Proposed Rule, Appendix K, Section II.C.5

5. The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation (Baker, L., Just, L. C., "Studies of Metal Water Reactions at High Temperatures, III. Experimental and Theoretical Studies of the Zirconium-Water Reaction," ANL-6548, May 1962), with a coefficient of unity. The reaction shall be assumed not to be steam limited. For rods whose cladding is calculated to rupture during the LOCA (see Section II.H), the inside of the cladding shall also be assumed to react. The calculation of the reaction rate on the inside of the cladding shall also follow the Baker-Just equation with a coefficient of unity, starting at the time when the cladding is calculated to rupture, and extending axially no less than 1.5 inches each way from the axial location of the rupture, with the reaction assumed not to be steam limited.

Discussion of Zirconium-Steam Reaction

In their Concluding Statements three reactor vendors (Westinghouse, B&W, and CE) and the Utility Group have suggested the use of an equation for calculating the zirconium-steam reaction that is more realistic than the Baker-Just equation which was specified by the Interim Policy Statement. Several data sets have been considered by these participants. The most comprehensive sets are from CE (data mentioned in CE Concluding Statement are not presently available to the staff or other participants), ORNL (Exhibit 509), and BMI (this data by Lemmon was considered by Westinghouse in its Direct Testimony). The data sets are consistent with one another. The Baker-Just equation is a good predictor of these data around 2000°F but becomes more conservative (i.e., higher reaction rate) at higher temperatures. At 2300°F the Baker-Just equation is higher than the mean of the data by about a factor of two. That is, for isothermal exposure at 2300°F, Baker-Just would predict about $\sqrt{2}$ times more total reaction than would be expected from the mean of the data. The data itself has a substantial spread. Statistical analyses were performed for the ORNL data by Westinghouse (Exhibit 1078, Section 3.5) and by ORNL. Best estimate equations were calculated by both organizations with similar results. One-sided tolerance limits were established for the data by ORNL (see Figure 1). The limit presented in Figure 1 is such that there is 95 percent confidence that 95 percent of the data in a true population would fall below the tolerance line. The Baker-Just equation over the range of interest falls about midway between the tolerance line and the Hobson (ORNL) best-estimate line. This indicates that the Baker-Just equation is approximately an 80 percent tolerance line for the data. The staff believes that this amount of tolerance to account for uncertainty in the application of this data is not excessive, and the staff concludes that use of Baker-Just should continue to be required. Confidence limits of 99 percent were also calculated for the Hobson best estimate line (see Figure 2). The Baker-Just equation falls above the 99 percent limit for the Hobson best-estimate line. That is, it can be stated with 99 percent confidence that Baker-Just would not correlate a true population in the range of the data. This is the meaning of the t-test performed by Westinghouse (Exhibit 1078, Section 3.5), which showed that lines falling above the confidence limit for a best

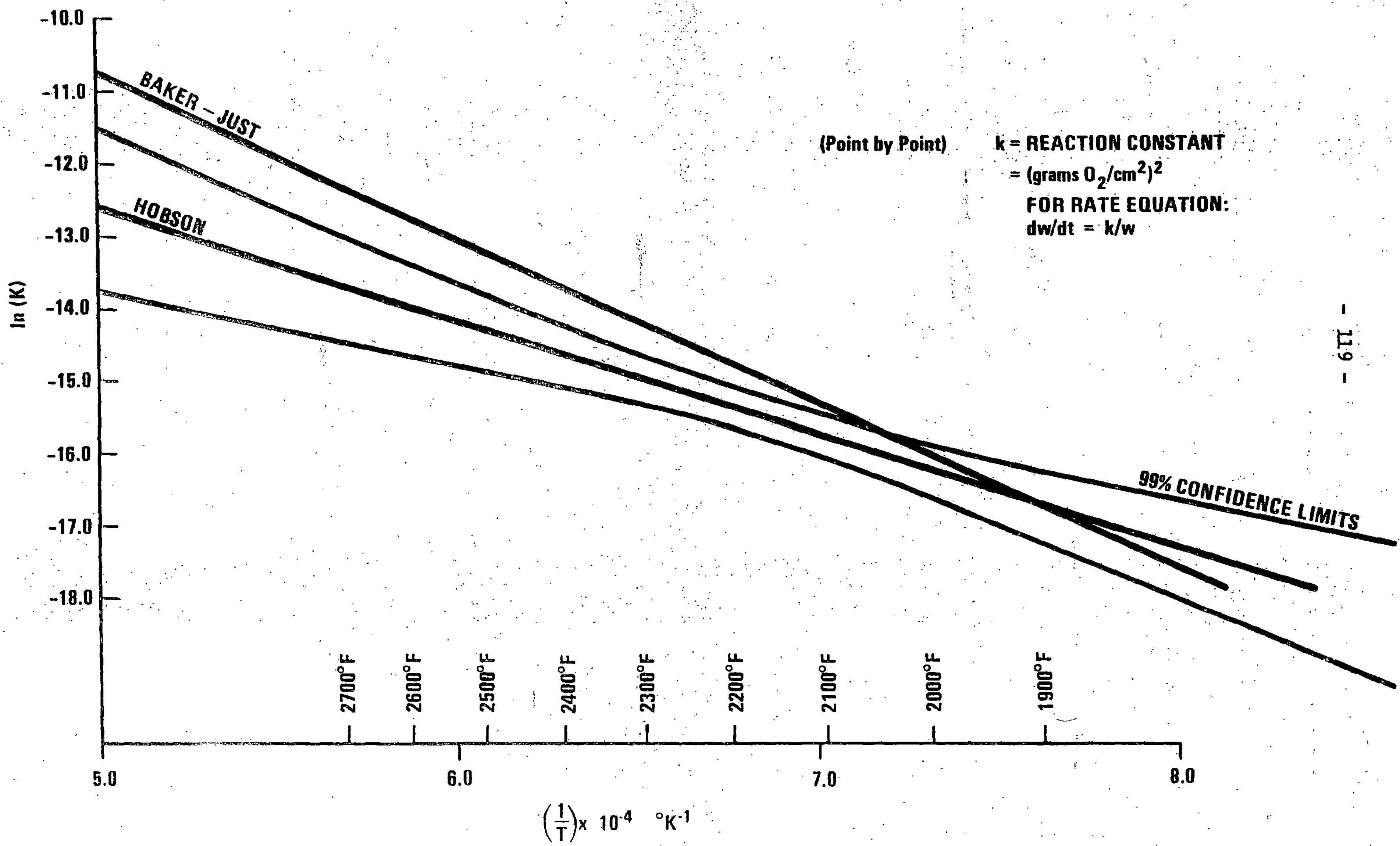
FIGURE . HOBSON DATA
One Sided Tolerance Limits



- 118 -

FIGURE 2 HOBSON 16 DATA POINTS

99% Confidence Interval About Regression Line



estimate correlation could not be considered true correlations of the population. That is, the t-test is used to determine "fit" of the correlation to the data. Tolerance limits on the spread of the data itself must be evaluated to determine conservatisms.

Some evaluations of the conservatism of the Baker-Just equation have been reconsidered in light of the Westinghouse and the ORNL analyses. In evaluating total oxygen consumed by the cladding, Westinghouse has considered the oxygen absorbed by the α and β phases and the thickness of the oxide (Exhibit 1078, section 3.5, Appendix A). B&W, in its comparison of measured oxide thickness to that calculated by Baker-Just, has apparently neglected the fact that all of the oxygen does not go to forming zirconium oxide, but some is absorbed by the α and β phases (Exhibit 1137, section 6). Therefore, the measured thickness should be increased by an amount equivalent to the amount of oxygen absorbed by the α and β phases in order to make a correct comparison with Baker-Just. By using methods similar to those outlined by Westinghouse (Exhibit 1078, section 3.5, Appendix A), the staff estimates that the measured values of the points in Figure 6.1 of B&W's comparison should be moved to the right (increased) by about 10 to 50 percent of their present values in order to account for dissolved oxygen. This would indicate far less conservatism than is implied by Figure 2.

GE in its Concluding Statement, page N-6, has sought to demonstrate the conservatism of Baker-Just by referring to the analysis in the PWR

FLECHT report (Exhibit 150, Appendix B). The staff agrees that, for one temperature transient of the 21 which appear in Figure B-12 of Exhibit 150, the Baker-Just prediction was high by nearly a factor of two. However, it appears that the same neglect of dissolved oxygen described above for the B&W comparison is applicable to the comparison in Exhibit 150. Therefore, proper assessment of conservatism cannot be made from these figures.

The statistical analysis of the ORNL oxidation data (Exhibit 509) indicated that additional oxidation rate data would surely lower the tolerance limits and allow better assessment of a proper rate equation. Since the B&W Concluding Statement, page 239, now suggests a possible non-conservatism because of steam limiting in the ORNL experiments, it is the staff's conclusion that additional experiments must be performed before Baker-Just is abandoned as the method for calculating energy release and hydrogen generation. The use of Baker-Just for assessing embrittlement was treated under the discussion of criteria (b)(1) and (b)(2), above.

As stated in the Staff Supplemental Testimony (Exhibit 1113, Section 18.0), the ratio of inside oxidation to outside oxidation is related to exposure temperature, azimuthal and axial location in the fuel element, and rupture opening size. For the FRF-1 transient with an 1800°F peak temperature the ratios of inside to outside oxidation were about 1.0 (Exhibit 1113, Section 18.0). For FRF-2 (Exhibit 1123) the ratios were substantially less but the scatter in the data is significant. More data is needed in order to realistically describe the functional dependence of inside reaction on temperature and other variables. Since inside-reaction data is missing over a wide temperature range and since many inferences must be made from the existing transient data in order to apply it to LOCA, the staff cannot now justify calculations performed in a less conservative manner than the full reaction rate recommend in the Proposed Rule. The available data does suggest, however, that reaction need only be considered for an axial region extending 1-1/2 inches in both directions from the center of the rupture (Exhibit 1123). Over this region, then, it is concluded that full reaction should be applied after rupture is postulated to occur.

REACTOR INTERNALS HEAT TRANSFER

The Proposed Rule, Appendix K, Section II.C.6

6. Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account.

Discussion of Reactor Internals Heat Transfer

Heat transfer from piping, vessel walls, and non-fuel internal hardware should be accounted for in the calculation because a large amount of stored energy is contained in the metal of the reactor system and can be significant in the evaluation of the smaller postulated pipe breaks (Transcript 10,087 and Exhibit 1031). This energy source is already included in most blowdown, refill, and reflood models so as to provide a realistic representation of the energy content of the coolant throughout the accident.

PWR PRIMARY-TO-SECONDARY HEAT TRANSFER

The Proposed Rule, Appendix K, Section II.C.7

7. Heat transferred between primary and secondary systems through heat exchangers shall be taken into account.

Discussion of PWR Primary-to-Secondary Heat Transfer

Heat transferred between primary and secondary systems through heat exchangers should be realistically accounted for in the blowdown, refill, and reflood calculations because energy loss or gain from the primary system coolant can effect the calculated course of the accident (Transcript 10,087 and Exhibit 1031). All PWR blowdown codes presently include provisions for modeling this energy source.

TWO-PHASE PRESSURE DROP

The Proposed Rule, Appendix K, Sections II.D and II.E

D. Frictional Pressure Drops. The frictional losses in pipes and other components including the reactor core shall be calculated using models that include realistic variation of friction factor with Reynolds number, and realistic two-phase friction multipliers that have been adequately verified by comparison with experimental data. The modified Baroczy correlation (Baroczy, C.J., "A Systematic Correlation for Two-Phase Pressure Drop," Chem. Engng. Prog. Symp. Series, No. 64, Vol. 62, 1965) or a combination of the Thom correlation (Thom, J.R.S., "Prediction of Pressure Drop During Forced Circulation Boiling of Water," Int. J. of Heat & Mass Transfer, 7, 709-724, 1964) for pressures equal to or greater than 250 psia and the Martinelli-Nelson correlation (Martinelli, R.C., Nelson, D.B., "Prediction of Pressure Drop During Forced Circulation Boiling of Water," Transactions of ASME, 695-702, 1948) for pressures lower than 250 psia is acceptable for calculating two-phase friction multipliers.

E. Momentum Equation. The following effects shall be taken into account in the conservation of momentum equation: 1. temporal change of momentum, 2. momentum convection, 3. area change momentum flux, 4. momentum change due to compressibility, 5. pressure loss due to wall friction, 6. pressure loss due to area change, and 7. gravitational acceleration. Any omission of one or more of these terms under stated circumstances shall be justified by comparative analyses or by experimental data.

Discussion of Two-Phase Pressure Drop

The conclusions of the Regulatory staff regarding the calculation of two-phase frictional pressure drop and the calculational representation of momentum conservation are exactly the same as the earlier suggestions contained in Section 4.0 of Exhibit 1113. These conclusions require realistic analytical treatment of two-phase friction and of momentum conservation, as reflected by Sections II.D and II.E.

Section II.E allows for simplification of the momentum equation when such simplification is warranted and has been demonstrated to be inconsequential to the LOCA calculation.

The reasons for requiring realism in these aspects of LOCA analyses were delineated in Section 4.0 of Exhibit 1113. The recommendations of that section were supported by the consultants to the Regulatory staff (Exhibit 1118 and page 4-9 of Exhibit 1113). However, the Concluding Statements of other participants have not agreed with the necessity for realistic treatment of these phenomena. The areas of disagreement are addressed below.

CNI discusses the momentum equation on page 7.5 of their Concluding Statement where they reach the conclusion that the Regulatory staff is lacking in "...diligence in gathering information..." because of its request for the derivations of the conservation of momentum equations used by the reactor manufacturers (pages 4-2 and 4-3 of Exhibit 1113). Technical judgment by qualified experts and comparison to appropriate experimental data are suitable bases for deciding which of the many possible mathematical representations of the momentum equation are acceptable for use in analysis of LOCA's. To aid in that technical judgment the Regulatory staff asked the reactor manufacturers (see page 4-10 of Exhibit 1113) to show in considerable detail their developments of the momentum equations contained in their ECCS evaluation models. These developments were reviewed by the staff and ANC, and the manufacturers were questioned at the hearing in this regard (e.g., Transcript pages 21,644-21,648, and 21,538-21,548). The staff has concluded as suggested in Exhibit 1113 that it is now possible to incorporate more computational detail in the momentum conservation equations. Whether the inclusion of that detail is important for particular LOCA's in particular reactors is a subject for investigation using the sensitivity studies required by Section III.A, Appendix K of the Proposed Rule. CNI has brought to this proceeding no new technical information concerning the momentum equation.

Several reactor manufacturers have disagreed with the position of the Regulatory staff regarding the need for more computational detail in the momentum conservation equations (i.e., the need for inclusion of momentum flux terms). The arguments are contained in their Concluding Statements as follows: B&W, pages 101-106; and GE, pages C-1 to C-8. The B&W and GE positions can be characterized in a few words - inclusion of momentum flux terms is inconsequential to most areas of LOCA analysis. The Regulatory staff position can also be briefly characterized - with the present state of technology momentum flux terms can be modeled in LOCA analyses, and the effects of adding these terms should now be systematically investigated. The Proposed Rule, Section II.E of Appendix K, and appropriate comparative analyses or sensitivity studies, Section III.A of Appendix K, will satisfy the concerns of the Regulatory staff in this regard. If the sensitivity studies confirm that the increased computational detail is unnecessary, then the vendors' concerns can be satisfied by the exclusion of these terms as delineated in Section II.E.

On a related point, the evidence described below is in disagreement with the assertion by GE at page C-5 to C-6 of their Concluding Statement that momentum flux effects are insignificant for the thermal-hydraulic characteristics of BWR's. Westinghouse also appears to disagree with GE in this regard (see pages 34 to 35 of Westinghouse Comment on Concluding Statements of Other Participants). According to testimony by GE witnesses, momentum flux and form loss are a larger fraction of total pressure drop in BWR's than in the case of PWR's which were described by Westinghouse and CE witnesses (compare Transcript pages 14,443 and 14,444 with Transcript pages D-22 to 23, and with Transcript page 15,591). The contradicting opinions among the vendors and the Regulatory staff consultants (Answers of L. J. Ybarrondo to CNI interrogatories, August 3, 1972) regarding the significance of momentum flux are the basis upon which the staff has proposed Section II.E of Appendix K of the Proposed Rule.

Section D of the GE Concluding Statement disputes the position of the Regulatory staff with regard to the need for realistic correlations of two-phase friction multipliers (pages 4-4 and 4-6 of Exhibit

4046

1113). GE claims (page D-1 and D-2) that Exhibit 1113 lacks analyses or data to support the position that the Thom correlation is preferable to the Martinelli-Nelson correlation (see Section II.D, Appendix K of Proposed Rule for sources of these correlations) for pressures greater than 250 psia. To the contrary, Exhibit 1113, page 4-4, clearly states that "The Thom correlation is superior to the Martinelli-Nelson correlation in accounting for the dependence of two-phase multipliers on fluid quality[4.9]." The analyses and data which GE claims as lacking were in fact presented by Thom in his publication of the correlation and by ANC on pages III 4.3-13 to III 4.3-15 of Exhibit 1033. The Thom correlation (published in 1964) is a refinement of the Martinelli-Nelson correlation (published in 1948). The statement by GE (page D-7 of Concluding Statement) that they have used the Martinelli-Nelson correlation since before the Interim Policy Statement was issued is certainly not a technical justification for its continued use. In summary, the Martinelli-Nelson correlation was made obsolete by the more recently published Thom and modified Baroczy correlations which are recommended by the staff. Section II.D will assure that all LOCA analyses are performed with the best information now available with respect to two-phase frictional pressure drop.

BLOWDOWN HEAT TRANSFER

The Proposed Rule, Appendix K, Sections II.F and II.G

F. Critical Heat Flux.

1. Correlations developed from appropriate steady-state and transient-state experimental data are acceptable for use in predicting the critical heat flux (CHF) during LOCA transients. The computer programs in which these correlations are used shall contain suitable checks to assure that the physical parameters are within the range of parameters specified for use of the correlations by their respective authors.

2. Steady-state CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following:

(a) W 3. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-uniform Heat Flux Distribution," Journal of Nuclear Energy, vol. 21, 241-248, 1967.

(b) B&W-2. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," Two-Phase Flow and Heat Transfer in Rod Bundles, ASME, New York, 1969.

(c) Hench-Levy. J. M. Healzer, J. E. Hench, E. Janssen, S. Levy "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors," APED-5186, GE Company Private report, July 1966.

(d) Macbeth. R. V. Macbeth, "An Appraisal of Forced Convection Burnout Data," Proceedings of the Institute of Mechanical Engineers, 1965-1966.

(e) Barnett. P. G. Barnett, "A Correlation of Burnout Data for Uniformly Heated Annuli and Its Uses for Predicting Burnout in Uniformly Heated Rod Bundles," AEEW-R 463, 1966.

(f) Hughes. E. D. Hughes, "A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia," IN-1412, Idaho Nuclear Corporation, July 1970.

3. Correlations of appropriate transient CHF data may be accepted for use in LOCA transient analyses if comparisons between the data and the correlations are provided to demonstrate that the correlations predict values of CHF which allow for uncertainty in the experimental data throughout the range of parameters for which the correlations are to be used. Where appropriate, the comparisons shall use statistical uncertainty analysis of the data to demonstrate the conservatism of the transient correlation.

4. Transient CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following:

(a) GE transient CHF. B. C. Slifer, J. E. Hench, "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," NEDO-10329, General Electric Company, Equation C-32, April 1971.

5. After CHF is first predicted at an axial fuel rod location during blowdown, the calculation shall not use nucleate boiling heat transfer correlations at that location subsequently during the blowdown even if the calculated local fluid and surface conditions would apparently justify the reestablishment of nucleate boiling. Heat transfer assumptions characteristic of return to nucleate boiling (rewetting) shall be permitted when justified by the calculated local fluid and surface conditions during the reflood portion of a LOCA.

C. Post-CHF Heat Transfer Correlations.

1. Correlations of heat transfer from the fuel cladding to the surrounding fluid in the post-CHF regimes of transition and film boiling shall be compared to applicable steady-state and transient-state data using statistical correlation and uncertainty analyses. Such comparison shall demonstrate that the correlations predict values of heat transfer coefficient equal to or less than the mean value of the applicable experimental heat transfer data throughout the range of parameters for which the correlations are to be used. The comparisons shall quantify the relation of the correlations to the statistical uncertainty of the applicable data.

2. The Groeneveld flow film boiling correlation (equation 5.9 of D. C. Groeneveld, "An Investigation of Heat Transfer in the Liquid Deficient Regime," AECL-3281, revised December 1969), the modified Dougall-Rohsenow flow film boiling correlation (D. H. Roy, "Direct Testimony on Behalf of Babcock and Wilcox, AEC Docket No. RM-50-1," March 23, 1972, page 7-8; and R. S. Dougall and W. M. Rohsenow, "Film Boiling on the Inside of Vertical Tubes with Upward Flow of the 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," U.S.A.E.C. 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) is suitable for use between nucleate and film boiling. Use of all these correlations shall be restricted as follows:

(a) the Groeneveld correlation shall not be used in the region near its low-pressure singularity,

(b) the first term (nucleate) of the Westinghouse correlation and the entire McDonough, Milich, and King correlation shall not be used during the blowdown after the temperature difference between the clad and the saturated fluid first exceeds 300°F,

(c) transition boiling heat transfer shall not be reapplied for the remainder of the LOCA blowdown, even if the clad superheat returns below 300°F, except for the reflood portion of the LOCA when justified by the calculated local fluid and surface conditions.

Discussion of Blowdown Heat Transfer

The subject of heat transfer from the fuel cladding to the reactor coolant during blowdown was treated in Sections 11, 12, and 13 of the Supplemental Testimony of the Regulatory staff (Exhibit 1113). Suggestions contained in the above sections of Exhibit 1113 are slightly different than the final conclusions of the Regulatory staff which are reflected in the Proposed Rule.

In Section II.F. the staff now recommends approval of six steady-state critical heat flux (CHF) correlations for use during blowdown. In recommending these six correlations the staff has concluded that earlier suggestions (Section 12.0 of Exhibit 1113) regarding statistical analyses of CHF data are not necessary for assurance of conservative treatment of blowdown heat transfer. The present recommendation of specifically approving six steady-state CHF correlations yields a realistic treatment of steady-state CHF data. The evidence presented in the course of this proceeding demonstrates that realistic correlations of steady-state CHF data, i.e., correlations such as those now recommended in the Proposed Rule, yield conservative predictions of the occurrence of CHF (i.e., at an earlier time than in reality) when used to analyze transient (blowdown) fluid phenomena which are within the ranges of the steady-state experimental data base (Exhibits 206; 613; 1059, Sections 4 and 7; 1113, Section 12; 1137, Section 2 and 8; 1144, Section 5; and Transcript pages 10,153-4; 20,274-6; and 21,118-135). Thus, experimentally verified steady-state correlations assure conservatism in the prediction of transient CHF.

In Section II.F the staff has recommended the approval of one transient CHF correlation on the basis of data and analyses contained in Appendix C of Exhibit 132. Section II.F also outlines procedures to be followed in the evaluation and potential use of other correlations of transient CHF data. Those procedures can be characterized as requiring conservatism in the treatment of transient CHF data. The reason for requiring this conservatism is related to the sources of uncertainty in the realistic prediction of transient CHF (Exhibits 1113, Section 12; 1137, Section 2; and Transcript pages 13,311; 15,525).

Westinghouse (Exhibit 1078, page 72) and B&W (Exhibit 1059, Section 4) have proposed that a "switching criterion" be used with DNB calculations to permit return to nucleate boiling (rewetting) if a DNB ratio of less than 1.0 is calculated to persist for less than 3 milliseconds (B&W) or 50 milliseconds (Westinghouse). Section II.F. of the Proposed Rule excludes such a criterion on the basis that time-hysterisis effects in rewetting-heat-transfer phenomena are not sufficiently well understood at this time (Exhibits 1001, page 4-32; 1113, page 11-2; and Transcript pages 14,412-417).

At pages 4.11 to 4.13 and 5.48 of its Concluding Statement CNI notes that the record of this proceeding contains a controversy about the realistic prediction of CHF and the realistic prediction of rewetting heat transfer during blowdown. However, CNI further suggests that there is controversy concerning the conservatism in the treatment of these phenomena in LOCA analyses. That suggestion is not supported by the evidence, as shown by the following. The Regulatory staff, the reactor manufacturers, and the laboratories consulting to the staff, have for some years agreed, and the Interim Policy Statement required, that in lieu of completely definitive information regarding the realistic treatment of these phenomena, they should be analyzed with conservative assumptions (Exhibits 1001, 1006, 1059, 1066, 1069, 1078). That is why, as discussed in the foregoing paragraphs, the transient CHF is required to be conservatively predicted either by use of realistic steady-state correlations or by use of conservative transient correlations, and the clad rewetting phenomenon during blowdown is conservatively required to be neglected. CNI also states at page 5.48 of their Concluding Statement that:

> "Moreover, if, as expected, there is flow oscillation during PWR blowdown, it is postulated that the critical heat flux may be exceeded locally and regions of dryout propagate within the core and especially accelerate dryout and aggravate temperature rises in the hotter regions."

Since this statement has no accompanying reference to the record, the staff interprets it to be a proposal by CNI to somehow calculate a delay CHF in the core hot region to allow for CHF propagation from the cooler core regions (i.e., to calculate "dryout propagation" to the "hotter regions"). Allowance for such propagation would decrease the conservatism of the methods required by the Interim Policy Statement and the Proposed Rule by increasing the calculated time of CHF in the hot (high power) regions. Such delay and corresponding decrease in conservatism is unwarranted by the evidence and should not be adopted by the Commission.

At pages 4.11 to 4.13 and 5.46 of its Concluding Statement, CNI claims that blowdown heat transfer data are not available for geometric and thermal/hydraulic conditions typical of large power reactors. Such is not the case as evidenced by the wealth of heat transfer data available to this proceeding for tube, annular and rod array geometries with steady and transient conditions of flow, pressure, and temperature (see for example, Exhibits 132; 206; 613; 1059, pages 4-54 to 4-58; 1061; 1113, Sections 11 and 12; 1137, Sections 2 and 8; 1151; 1152).

At pages 7.13 and 7.14 of their Concluding Statement, CNI presents:

> "...the capstone to the Regulatory staff's unprofessional approach to blowdown heat transfer is the section of Section 12 of their supplemental testimony in which they 'review briefly the new test data available from B&W' (Ex. 1113, pg. 12-1)."

The B&W blowdown heat transfer tests provide significant information which should not be diminished in stature by these CNI statements. In contrast to the CNI claims, the Regulatory staff and its consultants at ANC and ORNL reviewed the same information regarding the B&W blowdown heat transfer data and test conditions that was available to all participants in the hearing; namely, the Direct and Rebuttal Testimony of B&W and B&W's answers to staff questions (see Exhibits 1059; 1137; and Section 12 of Exhibit 1113). The Regulatory staff was the only participant in the ECCS hearing to question B&W concerning the test conditions, and to thereby demonstrate that the tests are applicable to LOCA analyses (Exhibit 1137 and Transcript pages 21,128-21,137).

In Section II.G the staff recommends approval of two flow film boiling heat transfer correlations for use during blowdown; the modified Dougall-Rohsenow equation and Groeneveld's equation for tubes and annuli (Equation 5.9 of AECL-3281). The recommendations are based on comparisons of these two correlations with statistical analyses of applicable heat transfer data (Exhibit 1177, 1144; 1127; 1113, Section 13; and Transcript pages 20,551-20,568 and 20,695-20,714). These references show that the modified Dougall-Rohsenow correlation and the Groeneveld correlation provide realistic estimates of stable flow film boiling heat transfer coefficients in rod bundle geometries typical of reactor cores. The modified Dougall-Rohsenow correlation also has the advantageous features of mathematically continuous behavior at the interface with single phase forced convection to steam and of continuous behavior at low pressure (Exhibit 1137). The staff concludes that the assumption of stable film boiling heat transfer coefficients during blowdown is the most conservative assumption possible because stable film boiling is the worst possible flow boiling heat transfer regime (Exhibit 1113, Section 13).

In Section II.G the Groeneveld correlation is restricted to fluid conditions which do not approach the correlation's point of singularity at low pressure. This condition can be satisfied in practice by graphs of heat transfer coefficient showing that no abnormally large coefficients result from using the Groeneveld correlation at low pressures (page 12-1 of Exhibit 1113).

The restricted use of transition boiling heat transfer coefficients specified in Section II.G is in accord with the recommendations of Section 11.0 of Exhibit 1113. The McDonough, Milich and King correlation provides a convenient method of assuring continuity between nucleate and film boiling heat transfer coefficients and thus promoting the computational stability of the computer solutions. The correlation is based on high pressure data in water, and the effect of the correlation on core heat transfer during a LOCA is small (Transcript page 13,306 and Exhibit 1001, page 3-34).

The Thom and Dittus-Boelter heat transfer correlations are consistently specified for use in all the evaluation models of Section III.C, Appendix K of the Proposed Rule. They were presented and discussed in Exhibit 1001, but their use was not a subject of contention in this proceeding.

CLADDING SWELLING AND RUPTURE

The Proposed Rule, Appendix K, Section II.H

> H. Cladding Swelling and Rupture. Calculations of gap conductance, cladding temperature, cladding embrittlement, and hydrogen generation from cladding-water chemical reactions shall take swelling and rupture of the cladding into account wherever the course of the postulated loss-of-coolant accident, calculated in accordance with an accepted evaluation model, leads to predictions of cladding swelling or rupture. Each evaluation model, therefore, shall where required include a model for predicting cladding swelling or rupture from consideration of the cladding axial temperature distribution and the pressure differential, both as functions of time. To be acceptable, a swelling and rupture model shall be based on applicable data in a conservative way.

Discussion of Cladding Swelling and Rupture

The staff recognizes that the methods required in the Proposed Rule for accounting for swelling, rupture, and zirconium-steam reaction inside the cladding (discussed above in Section II.C.5) in the fuel pin heatup calculation are likely to be restrictive (Exhibit 1113, Sections 10 and 18). In contrast to this restrictiveness, Westinghouse has suggested that because clad bulges have survived mechanical tests (Exhibit 1078, Appendix D; Exhibit 1151, Section 13), the bulges can be effectively ignored. It is the staff's understanding that in these tests the undeformed, unruptured regions also survived under expected duty. Therefore, it is not clear from these tests which region, bulged or unbulged, has less resistance to fragmentation. The staff is not aware of any experimental information to support the conclusion that the bulges are more resistant to fragmentation. However, we suggest that quench experiments of deformed and undeformed cladding could be performed to answer this question.

It has been pointed out that the bulged regions constitute a small fraction of the core and, therefore, need not be considered in LOCA analysis (Westinghouse Concluding Statement, pages 63, 64; GE Concluding Statement, Section M; B&W Concluding Statement, page 244). It is true that clad fragmentation of one local ruptured region of one pin would not impair the coolability of the core. However, the staff is not aware of any experimental information which would aid in quantitatively assessing the effect on core coolability of numerous ruptured regions with associated fragmentation. If the hot spot with its associated bulge were to be ignored along with other ruptured fuel elements, it is not clear that these other bulged regions would not fragment if the hot spot fragmented; nor is it known to what extent fragmentation of a widespread rupture distribution would impair ECCS effectiveness. Therefore, until realistic quantitative assessments can be made, the staff believes that swelling and rupture should be treated in LOCA analysis as specified in Section II.H so as to preclude fragmentation of the core hot spot.

If rupture occurs at elevated temperatures (greater than about 2000°F), enhanced energy release rates from zirconium-water reaction may result from the exposure to steam of a fresh, unreacted, inside cladding surface. The consequences of rupture at temperatures below about 2000°F were discussed in Exhibit 1113, Section 10, pages 10-16 to 10-22. For a given transient it is not obvious whether high- or low-temperature rupture results in the most severe cladding temperature transient. For a given pressure difference across the cladding, the rupture temperature data shows a spread of a few hundred psid (Exhibit 1007b, page 18). In order to assure conservatism, as required by the Proposed Rule, it is necessary to explore a range of rupture temperatures suitable to the applicable pressure differentials in order to determine the worst case for a given accident analysis.

INITIAL STORED ENERGY IN FUEL

Proposed Rule, Appendix K, Section II.I

I. Initial Stored Energy in Fuel. The steady-state temperature distribution and stored energy in the fuel before the accident shall be evaluated as a function of power density, fuel density, cold gap dimension, fuel thermal conductivity, fuel heat capacity, cold-fill gas composition and pressure, and burnup (cracking of fuel, sorbed gas and fission gas release, changes in fuel density, cladding creep). The values used and the burnup chosen (time in core lifetime) shall be such as to maximize the calculated initial stored energy in the fuel. For this calculation, the reactor operating power shall be assumed to be no less than 1.02 times maximum licensed power.

Discussion of Initial Stored Energy in Fuel

Under normal operating conditions a rather steep temperature distribution exists in fuel pellets due to the low thermal conductivity of uranium oxide and the high heat generation rate. Typically, a hot fuel pellet has centerline temperatures in excess of 4000°F, while the pellet surface temperature is in the neighborhood of 1000°F (Exhibit 110A, pages C-8 to C-10). Heat transfer between the oxide pellet and the metal cladding is controlled by the size of the gap between these two materials and the composition and density of the gas that fills the gap. The heat transfer coefficient from the cladding to the coolant is high resulting in cladding temperatures less than 100°F above coolant temperature (Exhibit 110A, page C-10). During the course of a postulated LOCA the heat generation is rapidly reduced to a lower rate (Exhibit 110A, page 7-23; Exhibit 132, page D-11) and a sharp decrease occurs in heat flux from the cladding to the coolant (Exhibit 110A, page 7-17; Exhibit 225, Figure VI-5; Exhibit 232, Figure 5-2). The temperature starts to equalize within the fuel pellet and between the pellet and the cladding, resulting in cladding heatup. Thus, stored energy from the fuel pellet is one of the primary heat sources that determine cladding temperature rise following a LOCA.

The stored energy varies with the characteristics of the fuel pellet (density, thermal conductivity, heat capacity) and the thermal conductance of the gap. The physical condition of the fuel pellet changes during operation due to thermal cycling and irradiation effects. During heatup the pellet expands, changing the fuel density and thus the size of the gap. At the same time the thermal conductivity and the heat capacity of the pellet also change because they are a function of temperature (Exhibit 132, pages D-7 and D-8). Repeated thermal cycling of the fuel results in cracking up of the pellets. This again will influence the thermal conductivity and the gap size.

Fuel element thermal characteristics are a function of many variables (Section 10, Exhibit 1113), as described below. Fuel irradiation, has two effects: irradiation-induced densification on a relatively fast time scale, and irradiation-induced swelling on a slower time scale. Both of these effects will change the pellet density and conductivity and will effect the gap size. The gap size depends on variations in the dimension of the cladding. The gap size also depends on thermal expansion and on the large external cladding pressure which will produce a slow creepdown of the cladding onto the fuel. The density and composition of the gas in the gap will depend on the sorbed gas content of the fuel, the fission gas release rate, and the initial composition and pressure of the fill gas. Since most of the parameters mentioned above are dependent on power density, further complications are introduced by changes in the power density of a given fuel pin during a fuel cycle.

The steady-state temperature distribution and stored energy of the fuel should be evaluated as a function of power density, fuel density, gap dimension, oxide thermal conductivity, oxide heat capacity and fill gas composition and pressure. The Regulatory staff is of the opinion that all of the above-mentioned effects are sufficiently important in the evaluation of the stored energy of the fuel that they should not be ignored in LOCA calculations (Exhibit 1113, Section 10). Furthermore, the influence of temperature and burnup should be taken into account in evaluating these parameters. Because of the importance and complexity of stored energy calculations, the analytical models used should be verified by experiments, and the combination of these parameters should be selected in a manner such that the calculations either predict or overestimate the stored energy at a point during core life when the calculated cladding temperature for a LOCA is a maximum. Also, uncertainties inherent in the measurement of the operating power level of the core should be accounted for in the stored energy calculations; thus, the Proposed Rule requires that the power level assumed for LOCA calculations should not be less than 1.02 times the licensed power.

These conclusions with regard to stored energy result in more detailed requirements than those of the Interim Policy Statement. When the IPS was prepared, the importance of stored energy on LOCA calculations was recognized. However, due to the variability of analytical methods employed by various manufacturers, no general rule was issued on stored energy; rather, the method to be used was specified in the description of the evaluation models. A considerable amount of information regarding stored energy has been introduced into evidence, and it is summarized in the Supplemental Testimony (Exhibit 1113, Section 10), where the following conclusions were reached:

(1) "Variations existed in steady-state gap models which could influence stored energy by an amount equivalent to about 200°F over the anticipated range of influential parameters."

(2) "Variations existed in UO_2 thermal conductivity used to calculate initial stored energy and transient energy release, which could affect clad temperatures by about 100°F over the range of values expected."

(3) "The actual steady-state gap conductance ... is a complicated function of power density, initial fuel pin dimensions, fill gas composition and internal fuel pin pressure, fuel density, fuel conductivity, burnup, fuel microstructure, fuel cracking, and clad creepdown onto the fuel. Therefore, no single value of gap conductance can be used to represent initial stored energy throughout fuel lifetime in LOCA calculations."

Such substantial differences exist (Exhibit 1113, page 10-12) in gap coefficients, even among one manufacturer's designs, so as to warrant the evaluation of steady-state gap coefficients on a case-by-case basis. This does not mean that generic models could not be approved at a later time, but, rather, that stored energy models should be integral parts of the manufacturers' respective evaluation models.

Further indication of gap conductance variation with linear power density and burnup was presented in the Redirect Testimony of Westinghouse (Exhibit 1151, page 24-7). The B&W Concluding Statement (pages 69 to 76), through the description of the B&W model, indicated the importance and complexity of the stored energy calculations. CNI emphasized the importance of gap conductance and stored energy calculations in LOCA evaluation (CNI Concluding Statement, pages 5.17 and 5.18).

GE (Initial Closing Statement, Section K) maintains that, since the IPS accepts a constant gap coefficient of 1000 Btu/hr-ft^2-°F for certain GE designs, all BWR's should be licensed in the future using this value. However, requirements set forth in the Regulatory staff's Technical Report on Densification of Light-Water Reactor Fuels, dated November 14, 1972 (pages 69-72), have already superseded any previous approval of a gap coefficient of 1000 Btu/hr-ft^2-°F. Also, gap conductance is a function of many variables which are in turn dependent on a specific design or on the specific operating condition of a plant. These dependences must be taken into account. In connection with specific GE-designed plants, there is sufficient evidence (Exhibit 253) to recognize that LOCA analysis of at least certain BWR designs is very sensitive to the selection of the gap conductance. It should also be noted that GE has not presented for the record the GE equivalent of the parametric study on gap conductance discussed in Exhibit 253. Therefore, the Regulatory staff concludes that the so-called "approved" gap heat transfer coefficient of 1000 Btu/hr-ft^2-°F for BWR's is only of historic significance and recommends that all vendors calculate, for each case, the value of the gap coefficient, accounting for all effects specified in the Proposed Rule.

FUEL ROD THERMAL PARAMETERS DURING POSTULATED ACCIDENT

The Proposed Rule, Appendix K, Section II.J

J. Fuel Rod Thermal Parameters During Postulated Accident.

1. The calculations of the fuel and the cladding temperatures as functions of time shall use values for gap conductance and other thermal parameters as functions of temperature and other applicable time-dependent variables.

2. If cladding swelling or rupture is calculated to occur, the gap conductance shall be varied in accordance with the change in its dimensions and any other applicable variables.

Discussion of Fuel Rod Thermal Parameters During Postulated Accident

During the postulated LOCA transient, the cladding of the hot fuel rod is being heated by transfer of energy from the fuel and cooled by transfer of energy to the coolant. In addition, heat generation occurs within the cladding due to chemical reaction with oxygen at elevated temperatures. Heat transfer from the cladding to the coolant depends on the thermohydraulic parameters of the coolant, and it is described in the discussions of Sections II.F, II.G and II.Q of Appendix K of the Proposed Rule. Heat transferred from the fuel to the cladding is controlled by the temperature of these two surfaces and by the existing gap conductance. The thermal parameters (thermal conductivity and heat capacity) of both the fuel and the cladding influence the respective surface temperatures. Since these thermal parameters are temperature dependent (Exhibit 110A, page C-6; Exhibit 123, pages D-7 and D-8), they vary during the transient. Depending on the surface temperatures and on the size of the gap, the significant mode of heat transfer through the gap can also vary. In small gaps heat is transferred by conduction through contact points and by conduction and convection through the filler gas. At elevated temperatures radiative heat transfer between fuel and cladding could become important (Exhibit 1113, page 10-21). For conditions where the gap size is very large (e.g., swollen or perforated cladding), and the fuel and clad surface temperatures are substantially different from one another, radiation becomes the principal mode of heat transfer. Thus, variations in the gas composition, the size of the gap, and the transient values of the thermal parameters could influence cladding heatup.

The Regulatory staff has concluded on the basis of the above evidence that during postulated LOCA transients the fuel pellet and cladding temperatures vary within a sufficiently broad range to necessitate the time-dependent evaluation of the thermal parameters (e.g., thermal conductivity and heat capacity). Furthermore, in some cases, the cladding temperatures reached during the transient are high enough to produce cladding swelling and perforation. The staff further believes that following the onset of clad swelling, the fuel pin heatup calculation should account for the increase in gap size, the thinning of the cladding, thermal radiation across the gap, and the presence of fission gases and/or steam in the gap (see also the discussion of Section II.H, above).

This position is an extension of the previous Interim Policy Statement requirements. The evaluation models approved under the IPS provide for calculation of the thermal parameters as a function of temperature and therefore time. Increase in gap size due to cladding strain, however, is not calculated in these models. One of the models (Westinghouse) attempted to account for gap size variations by changing the value of the gap conductance during the transient (Exhibit 1001, page 3-23). However, none of the presently accepted evaluation models has the capability to actually change the geometry of the cladding during the course of the postulated accident. Neither do the various evaluation models have a thermal radiation term to account for increased heat transfer across the gap at elevated fuel surface temperatures.

There is evidence to show that variations in thermal parameters and in gap conductance can alter the results of the cladding heatup calculations. This evidence is summarized in the Regulatory staff's Supplemental Testimony (Exhibit 1113, pages 10-1 to 10-4, 10-13 and 10-14). Probably the most important of the transient effects is the change in gap conductance. B&W (Exhibit 1059, page 6-9) showed that, for a severe LOCA, transient swelling and rupture could occur in 1 to 2 seconds. This is in agreement with the CE Redirect Testimony (Exhibit 1144, page 5-3) which predicts swelling and rupture during blowdown without specifying the exact time of rupture. This is also consistent with staff calculations which indicate rupture times as low as 3 seconds (Exhibit 1113, page 10-16). In addition, the Regulatory staff performed a parametric study in which the gap size was assumed to undergo a step change, and the gap coefficient was calculated as a function of time, including thermal radiation effects (Exhibit 1113, pages 10-16 to 10-23). The findings of the staff's study were:

1. If clad swelling occurs, the reduced gap coefficients for the remaining portion of the blowdown period are the controlling resistance to the removal of stored energy from the fuel pellet.

2. Because of thermal radiation, higher gap coefficients could develop just prior to and during reflood thus enhancing the stored energy transfer during this period.

3. The reduced heat capacity per unit heat transfer area, due to thinned cladding, contributes to the increase in clad temperature rise.

4. The onset of swelling is an important parameter. If swelling occurs early in the transient, the delayed release of fuel pellet stored energy increases the peak cladding temperature. If swelling occurs late in the transient, a lower gap

conductance during reflood can be beneficial to the clad temperature transient.

Westinghouse reported cladding temperature calculations where the heat transfer area of the cladding was increased at the time of rupture (Exhibit 1151, pages 13-1 to 13-4). The results of these calculations augment the Regulatory staff's findings. The Westinghouse conclusions are as follows:

> "Because of the decreased pellet-clad gap heat transfer coefficients, less energy is transferred to the cladding during blowdown. Thus, the effect of blockage in this case is primarily on pellet temperatures rather than on clad temperatures during the blowdown period. Clearly, the increased pellet temperatures of the blocked core cause faster cladding heatup during the adiabatic period between end of blowdown and recovery of the bottom of the core (30 seconds), and during the initial phase of reflooding."

B&W performed a dynamic blockage analysis for a design basis LOCA in a typical B&W plant (Exhibit 1137, pages 5-3 and 5-4). The effects of clad swelling resulted in a 51°F increase in peak cladding temperature. Based on this analysis, B&W concludes that (Concluding Statement, page 183):

> "fuel element clad swelling does not have a significant effect on cladding temperature response during the blowdown phase of the LOCA, and indeed, over the course of the entire transient."... "the effects of fuel element cladding swelling need not be considered in LOCA evaluations."

The Regulatory staff finds that B&W's conclusion is not justified at the present time for the following reasons:

4 0870

1. The B&W calculations did not account for clad thinning or for radiation from the fuel to the cladding.

2. A 51°F increase in peak cladding temperature can be significant.

3. The single calculation for a given system and break size which produced a 51°F increase in peak cladding temperature does not demonstrate that all other cases are conservative.

4. New calculations with new assumptions (considering for example zirconium-water reaction on both sides of the cladding) could lead to different results.

With respect to the staff parametric study on gap conductance (Exhibit 1113, pages 10-16 to 10-23), B&W states (Concluding Statement, page 187):

> "The study provides no support for the proposition that the conditions which could cause an adverse effect of cladding temperature response would be reasonably expected to occur during the LOCA."

On the contrary the range of parameters covered in the staff study was selected on the basis of present day LOCA calculations (Section 10, Exhibit 1113). The conclusion drawn from the study is that transient variations in gap conductance may produce significant variations in the calculated cladding temperature for some conditions.

The GE position (Initial Closing Statement of GE, Section K) is as follows: During the course of the hearing, the participants did not provide evidence that variations in gap conductance during the LOCA transient would have a significant effect on BWR clad temperature response. Therefore, the GE evaluation model (approved under the Interim Policy Statement), which did not account specifically for this effect, must be acceptable in the future. The Regulatory staff does not accept this position for the following reasons:

1. GE has not presented for the record BWR cladding temperature calculations that adequately accounted for possible variation in the gap conductance during the postulated LOCA. The only GE calculations available (Exhibit 1113, page 10-14) did not account for clad thinning due to clad swelling and did not account for an increase in gap heat transfer at higher temperatures due to radiation.

2. The GE evaluation model was not approved for more than an interim period. Before a more permanent approval can be granted, GE should demonstrate the acceptability of the method in light of today's knowledge concerning transient gap conductance.

CNI's opinion on variations in the gap conductance during a LOCA is expressed on pages 5.18 and 5.19 of the CNI Concluding Statement as follows:

> "CNI further believes that the changes in the gap conductance that will occur during LOCA conditions are poorly, indeed inadequately, established and that the reactor vendors are using values of conductivity and evaluations of gap changes which, although within the excessive range of presently assessed uncertainty, are most favorable to establishing ECCS effectiveness. (See staff Supplemental Testimony, Exhibit 1113, Reference 10.33)."

The CNI contention, that reactor vendors are using values of oxide conductivity and gap conductances during the transient which are most favorable to establishing ECCS effectiveness, is not supported by reference 10.33 of the staff Supplemental Testimony. The author of this reference presented a discussion of the difficulties associated with measurements of oxide conductivity and gap conductance. He also summarized the approaches that various reactor vendors have taken in the past in their safety evaluation. But, he did not reach a conclusion similar to the CNI contention, neither did he present information from which such a conclusion would follow. Furthermore, reference 10.33 is not in evidence.

PUMP MODELING

The Proposed Rule, Appendix K, Section II.K

K. Modeling of Rotating Pumps. The characteristics of rotating primary system circulating pumps (axial flow, turbine, or centrifugal) shall be derived from a dynamic model that includes momentum transfer between the fluid and the rotating member, with variable pump speed as a function of time. The pump resistance used for analysis should be justified. The pump model for the two-phase region shall be verified by applicable two-phase pump performance data.

Discussion of Pump Modeling

PWR's

The primary coolant pump models contained in the PWR evaluation models are a controlling factor in calculating core flow for a postulated cold leg break (Transcript 6309), because the system pressure distribution tends to reverse the flow through the core during part of the blowdown (Exhibit 1113, page 6-1). Since the blowdown core flow reversal is opposed by the driving head of pumps in the unbroken reactor-coolant loops, realistic pump modeling is necessary in order to properly calculate core flows, which then establish the rate of energy removal from the fuel rods during blowdown (Exhibit 1113, page 6-1). On the other hand, to the extent that blowdown heat transfer is dependent on turbulence and rate of system depressurization (Exhibit 1137, page 2.2; and Exhibit 1113, Section 12), dynamic pump modeling has less significance. B&W's Concluding Statement, page 118, also references this effect.

44874

The major question raised about pump modeling was pump performance during the two-phase portion of blowdown (Transcript 6312 and 7479; Exhibit 1176, page 5). This question results from the limited data on pump performance under two-phase conditions. Because of the limited data available at the time of the Supplemental Testimony, the staff performed sensitivity studies assuming conditions wherein the pumps; (a) were assumed to cavitate when the pump suction pressure was reduced to the saturation pressure, and (b) the pump continued to operate (pump head not degraded) in the two-phase region (Exhibit 1113, page 6-3). These analyses were performed to bound the effects of pump modeling, and they resulted in a maximum calculated cladding temperature difference of 100°F for the ten cases examined (Exhibit 1113, page 6-4). On the basis of this sensitivity study the staff has proposed a requirement for experimental pump performance data in the two-phase flow region (Transcript 21,000). Section II.K of the Proposed Rule reflects this staff conclusion.

The use of dynamic pump models is also supported in the Concluding Statements of both Babcock & Wilcox and Westinghouse; Combustion Engineering did not comment on pump modeling during blowdown. The Babcock & Wilcox and Westinghouse Concluding Statements state the following:

B&W (page 122) - "It is B&W's position that the dynamic pump model is, indeed, a better representation of the fundamental physical behavior governing pump performance during the two-phase regime in blowdown."

Westinghouse (page 72) - "...Westinghouse recommends that in the Westinghouse evaluation model, pump behavior during blowdown be analyzed by using the proposed pump model with the pump homologous curves to describe the pump behavior under transient conditions."

CNI in its Concluding Statement presented a series of quotations concerning pump models (page 5.42). CNI suggested no alternative to dynamic pump modeling, and it cited no evidence contrary to the adoption of dynamic pump models (pages 5.11-5.43). Also, CNI presented no opinion with regard to the staff position as previously stated in the Supplemental Testimony (Exhibit 1113, Section 6).

BWR's

The core flow in a BWR during blowdown is determined by the performance of both the jet pumps and the recirculation pumps (Exhibit 1113, page 6.2). The BWR model for recirculation pumps assumes that the pump head is not degraded until the pump suction pressure reaches saturation pressure (Exhibit 132). At this time in the LOCA, the pump head is assumed to decrease linearly, going to zero at one percent suction quality. Since calculations (Exhibit 132) indicate that the jet pumps uncover before this condition is reached, it is conservative to assume core flow goes to zero when the jet pumps uncover; i.e., before the recirculation pump suction conditions reach one percent quality (Exhibit 1113, page 6-2). Thus, two-phase pump performance for BWR recirculation pumps is not an issue by virtue of the evaluation model assumption of zero core flow when the jet pumps uncover.

The Concluding Statement of GE states (page G-4);

> "Clearly, the detailed technical evidence in the record fully supports the adequacy of the pump modeling in the GE evaluation model."

The staff agrees in part: the BWR evaluation model (Exhibit 132) is satisfactory for LOCA's where operation of the pump in the two-phase region does not control the calculation of core flow. However, the staff reserves judgment on the BWR pump model for potential conditions where core flow can be shown to be controlled by recirculation pump performance in the two-phase region.

CONTAINMENT PRESSURE

The Proposed Rule, Appendix K, Section II.L

L. Containment Pressure. The containment pressure used for evaluating cooling effectiveness during reflood and spray cooling shall not exceed a pressure calculated conservatively for this purpose. The calculation shall include the effects of operation of all installed pressure-reducing systems and processes.

Discussion of Containment Pressure

As shown in Exhibit 1113, Section 15, the containment pressure during the reflood stage of PWR's is a consideration in determining both the core reflooding rate and the fuel clad-to-fluid heat transfer coefficients as derived from the FLECHT experiments. A lower assumed or calculated containment pressure would tend to decrease both the core reflooding rate and heat transfer coefficients, thereby increasing the calculated cladding temperature. The Interim Policy Statement required that the containment back pressure assumed for the reflood analysis should not be higher than 80% of the increase in pressure calculated for the accident. The conservatism of the IPS requirement was questioned during the proceedings (Transcript 6503). The staff presented an evaluation in Exhibit 1113, Section 15, which demonstrated that the IPS method was conservative for reflooding considerations in large dry containments since the containment pressure assumed for analysis was lower than the realistically anticipated containment pressure with full containment safeguards in operation.

The PWR evaluation models in the Interim Policy Statement applied only to large dry containments. Ice condenser and subatmospheric containments were not considered. The Proposed Rule applies to all containment systems and requires that the pressure shall be conservatively calculated assuming all pressure reducing systems and processes are operable. This would be done for each LOCA and for each containment design.

The calculation of the containment pressure during reflood was discussed in the Concluding Statements of B&W and CNI. B&W agreed with the conservatism of the calculation and stated (page 5.47):

> "The B&W methods of analysis for containment back pressure in accordance with the IAC have thus been demonstrated to produce a lower than expected value of containment back pressure during the LOCA analysis."

CNI disagreed with the staff position (Section 15 of Exhibit 1113) and said in their Concluding Statement (page 5.47):

> "...the containment pressures are not known with adequate precision."

CNI offers no evidence to support this statement, nor does it present any alternate to the staff proposal. The staff rejects this CNI conclusion which is contradicted by the evidence (Exhibit 1113, Section 15).

BWR calculations of core cooling effectiveness conservatively assume that the containment is at atmospheric pressure (Exhibit 1113, page 16-26). Thus, containment pressure is not an issue for BWR's since no credit is given for improved heat transfer at higher containment pressure.

SPRAY COOLING HEAT TRANSFER

The Proposed Rule, Appendix K, Section II.M

M. Spray Cooling Heat Transfer (Applies Only to Boiling Water Reactors). Following the blowdown period, convective heat transfer shall be calculated using coefficients based on appropriate experimental data. For reactors with jet pumps and having fuel rods in a 7 x 7 fuel assembly array, the following convective coefficients are acceptable:

1. During the period following lower plenum flashing but prior to the core spray reaching rated flow, a convective heat transfer coefficient of zero shall be applied to all fuel rods.

2. During the period after core spray reaches rated flow but prior to reflooding (see paragraph 3), convective heat transfer coefficients of 2.0, 3.2, 1.5, and 1.7 $Btu\text{-}hr^{-1}\text{-}ft^{-2}\text{-}F^{-1}$ shall be applied to the fuel rods in the outer corners, outer row, next to outer row, and to those remaining in the interior, respectively, of the assembly.

3. After the two-phase reflooding fluid reaches the level under consideration, a convective heat transfer coefficient of 25 $Btu\text{-}hr^{-1}\text{-}ft^{-2}\text{-}°F^{-1}$ shall be applied to all fuel rods.

Discussion of Spray Cooling Heat Transfer

The heat transfer coefficients specified in the Proposed Rule are unchanged from those previously specified in the Interim Policy Statement and are based on the results of tests performed in the BWR Full Length Emergency Cooling Heat Transfer (FLECHT) program. Such coefficients conservatively represent the conditions expected in a BWR following a postulated LOCA only if the tests on which the coefficients are based are applicable to the specific LOCA conditions under consideration, and the coefficients are applied in such a manner that the temperature response of the fuel clad is conservatively predicted.

BWR FLECHT Test Adequacy

The BWR FLECHT program and the adequacy of the tests were discussed in Section 16 of the staff's Supplemental Testimony (Exhibit 1113); in Section III-B of the ANC Direct Testimony (Exhibit 1006); in Section IV of the GE Direct Testimony (Exhibit 1069); in Sections L, M, N and O of the GE Redirect and Rebuttal Testimony (Exhibit 1148); and in Chapter 5 of the Direct Testimony of CNI (Exhibit 1041). The PWR manufacturers have generally taken a parochial view of system-related ECCS problems so that only GE, CNI and now the staff have discussed the BWR FLECHT program in their Concluding Statements.

The Regulatory staff concludes that the BWR FLECHT tests are an appropriate basis for developing a BWR core heatup model because the tests closely simulated the reactor fuel bundle geometry and materials, and the test conditions were representative of the conditions expected during the spray cooling phase of a postulated LOCA in a BWR. (Exhibit 1113, Section 16).

GE has concluded (GE Initial Closing Statement, page P-1):

> "The BWR FLECHT program has provided major experimental verification for a number of the inputs in the BWR evaluation model, and has yielded information of interest on such other matters as flow blockage, the lack of significant oxidation on the inside of fuel rods, and overall BWR ECCS capability."

CNI has concluded (Exhibit 1041, page 5.2, and CNI Concluding Statement, page 5.37):

> "The FLECHT experiments do not provide a sound basis for assessing the efficacy of the core spray or reflooding system...[and the

FLECHT] program was characterized by narrow scope, limited range of parameters (many inappropriate to the tasks at hand), the use of incorrect materials, crude and incompetent instrumentation and operating techniques (with consequent major equipment malfunctions), and, as a culminating weakness, expansive and overgenerous interpretations."

Although the staff agrees with GE's rather general conclusion, they present no clear rationale for their conclusion with record citations. The staff disagrees with CNI's characterization which is unsupported by the record of the hearing, as is demonstrated below by an analysis of each item.

The BWR-FLECHT program consisted of 143 tests using ten heater bundles (Exhibit 137, page 2), which is indicative of a broad, not a narrow, scope. The broad range of parameters in the tests exceeded the range of parameters expected for a postulated LOCA and was appropriate to the task of determining on a quantitative basis the heat transfer mechanism in a fuel assembly during the operation of the spray cooling system (Exhibit 1113, pages 16-16 through 16-27). The materials (i.e., stainless steel bundles and Zircaloy bundles) were properly chosen, provided the maximum amount of heat transfer data from the program, and were the logical choice for reasons stated in Exhibit 1113, page 16-2. Characterizing the instrumentation and operating techniques as "crude and incompetent" is not based on facts, but on an unrealistic expectation that complex tests can be performed flawlessly, and an incorrect judgment that failures are due to incompetence or intent. No serious equipment malfunctions occurred during the stainless steel tests. Most of the malfunctions during the Zircaloy tests occurred more than three minutes after the spray flow was initiated. Since reflooding of the core following a postulated LOCA occurs within three minutes after initiation of the sprays, the malfunctions did not seriously affect the data during the test period of interest. Although CNI considers the interpretations of the tests to be "expansive and over-generous," nowhere is it demonstrated that such interpretations were used in developing the Interim Policy Statement.

CNI specifically faulted the tests because of the use of stainless steel heaters, the lack of energy balance computations, the inadequate simulation of flow blockage, the method of deriving heat transfer coefficients, the anomalous and inconsistent results, and the experimental problems (Exhibit 1041, page 5.2). CNI further stated that the spray flow and bundle power were not representative of BWR's (Exhibit 1041, pages 5.5, 5.35 and 5.38). GE specifically refutes the CNI contentions relating to the temperature response of the Zr2K bundle, the usefulness of the Zr2K, Zr3M, and Zr4M tests, and the use of stainless steel (GE Initial Closing Statement, Section P). Except for flow blockage which is discussed elsewhere in this Concluding Statement (Acceptance Criterion (b)(4), Coolable Geometry), the specific arguments are discussed in the following paragraphs.

Stainless Steel Heaters

Although "the true purpose of using stainless steel cladding in 138 out of 143 FLECHT [tests] is not apparent" to CNI (Exhibit 1041, page 5.9), the necessity and desirability of performing the majority of the tests using stainless-steel-clad heaters has been demonstrated. The stainless steel heater rods were chosen because of their durability which permitted a series of tests to be run with each test bundle. The capability to perform multiple tests with a single bundle improved the accuracy of the derived heat transfer coefficients (Exhibit 1069, page 17) and provided heat transfer data applicable to a range of reactor types and conditions (Exhibit 1113, page 16-2). The Zircaloy-clad heater test bundles did not have this capability (Exhibit 1113, page 16-2). Thus, the stainless-steel-bundle tests were performed so as to obtain parametric heat transfer information, whereas the Zircaloy-clad bundles were used to determine whether any significant anomaly existed in the transient heat transfer behavior of Zircaloy.

Energy Balance

The contention by CNI that the simplified energy balance calculated for the Zr-4 test by ANC (Exhhibit 1041, page 5.97) demonstrated that the power, spray rate, or drain rate might be significantly in error and result in incorrect conclusions is not supported by the record of the hearing. The test engineer responsible for the Zr-4 test testified that a simplified energy balance would be expected to be in error and in any case an energy balance of this sort "is not necessarily germane to the heat transfer test" (Transcript 13,872). By contrast, a detailed transient energy balance for each heater rod in the SS2N test bundle was made in order to derive the convective heat transfer coefficients from the data (Exhibit 1113, page 16-28; Exhibit 1069, page 18). Using these heat transfer coefficients to predict the temperature response of the Zircaloy bundles also required detailed transient energy balances for each rod in the Zircaloy bundles (Exhibit 1113, page 16-33). Therefore the determination that there were no significant errors present in the tests was based on the predictions of temperatures in the Zircaloy tests using the model and heat transfer coefficients which were derived from detailed energy balances rather than from a simplified energy balance.

Derivation of Heat Transfer Coefficients

The method of deriving heat transfer coefficients does arbitrarily assume a sink temperature (Exhibit 137), but this does not distort the results as suggested by CNI (Exhibit 1041, page 5.2). Because the sink temperature was arbitrarily assumed to be a constant 212°F (the saturation temperature of steam at atmospheric pressure), some heat transfer coefficients derived from the data were negative. The reason for CNI concern in this regard appears to relate to heating of the clad by the steam (Exhibit 1041, page 5.14) which may not be "allowed for in the utilization of the experimental results" (Exhibit 1041, page 5.15).

The mathematics of the core heatup model predict negative heat transfer coefficients when the sink temperature is higher than the rod surface temperature. Since the steam temperature must always be below the highest rod surface temperature, heating of rods by the steam, as indicated by negative heat transfer coefficients, cannot affect the hot spot temperature and need not be included in the clad heatup model. Therefore, the method of deriving heat transfer coefficients does not distort the ~~results~~ of the experiments or the calculation of peak clad temperatures following a postulated LOCA.

Anomalies

CNI contends that, "analysis of FLECHT data has not been able to reconcile several grossly anomalous and inconsistent results." (Exhibit 1041, pages 5.2) Inconsistencies between temperatures measured in different runs (Exhibit 1041, page 5.15) and different tests (Exhibit 1041, pages 5.19, 5.101), and interpretation of thermocouple traces (Exhibit 1041, pages 5.46, 5.48, 5.60) are the specific illustrations presented by CNI.

CNI cites the apparent inconsistency between the temperatures measured in run 33 of the SS2N test and the temperatures measured in run 6 and run 13 of the same test (Exhibit 1041, page 5.16; Exhibit 127, Figure 6). As explained in the original report (Exhibit 127, page 7) temperatures measured early in bundle life were slightly lower (50°F) than temperatures measured late in bundle life. This occurred because a series of runs with both spray cooling and bottom flooding (runs 24 through 31), in which the bundle was cooled more quickly, was performed just prior to the purportedly anomalous test run. Therefore the difference in temperatures was attributed to the distortion and bowing of the heater rods. Post-test inspection of the bundle confirmed that the rods were bowed and distorted. Rod distortion is further verified by noting that all the runs (33, 34 and 35) performed at the same power, but after the combined spray-flooding runs, were consistent among themselves (Exhibit 127, Table 1). In addition, other runs (2, 4, 9 and 17) performed prior to spray-flooding runs were also consistent among themselves (Exhibit 127, Figure 7). This minor perturbation is approximately the amount predicted to occur due to bundle distortion, it is well understood, it is not significant, and it was considered in developing the evaluation models of the Interim Policy Statement (Exhibit 1113, page 16-7).

CNI cites as another apparent inconsistency the difference between the temperatures measured in two tests using the SS2N and the SS4N bundles which CNI considers to be "closely the same" (Exhibit 1041, page 5.19B). However, the tests bundles were not the same since the axial power peaking factors differed (Transcript 7013; Exhibit 1113, page 16-38). This difference in peaking caused approximately half of the difference in measured temperatures. The remaining difference could be the result of experimental errors. Even though the measured temperatures differed, the heat transfer coefficients derived from the two sets of test data are not significantly different (Transcript 7019; Exhibit 136, Figure G-35). When each set of heat transfer coefficients is applied to a postulated LOCA, the calculated peak clad temperatures differ by only 40°F (Transcript 14,234). Therefore the difference between the SS2N and the SS4N tests, although considered in developing the IPS, is not significant.

The CNI argument that the interpretation of the thermocouple traces from the Zr2K tests "indicates a false temperature turnaround" (Exhibit 1041, pages 5.46, 5.48, 5.58-5.79) is repeated in the CNI Concluding Statement (page 5.11). The summary of the questioning and testimony presented in the Initial Closing Statement on Behalf of the General Electric Company, Volume 2, page P-2, agrees with the view held by the staff with regard to the Zr2K tests. In any case, the discussion is irrelevant to the determination of the validity of the Proposed Rule since the Zr2K bundle temperatures under contention occurred long after the corresponding reflooding of a fuel assembly in a reactor would have occurred. This portion of the Zr2K test is not representative of conditions expected in a BWR with jet pumps and was not considered in developing the heat transfer coefficients specified in Section II.M.

Experimental Problems

CNI contends that "several crucial FLECHT tests are compromised beyond salvation by experimental problems and incompetent or careless conduct" (Exhibit 1041, pages 5.2, 5.100). These problems include heater and thermocouple failure (Exhibit 1041, pages 5.40, 5.42, 5.64), the use of molybdenum heaters (Exhibit 1041, page 5.10), the reaction between the alumina insulator and the Zircaloy clad, and accumulation of water in the test assemblies.

The failure of heaters and the power shifting caused by the molybdenum heaters impaired the usefulness of the tests but did not compromise their validity for use in constructing an evaluation model (Exhibit 1113, page 16-34). Reflooding of fuel assemblies in a BWR following a postulated accident is calculated to occur within 2.5 minutes after initiation of the core spray. Reflooding of the fuel assemblies will provide a significant increase in heat transfer, and as a result clad temperatures will be quickly reduced. All but three of the heater failures in the Zr2K test and all of the heater failures in the Zr3M and Zr4M tests occurred later than three minutes after spray initiation. Therefore most of the failures did not affect the portion of the experiments which were relevant to a LOCA heatup transient in a BWR with jet pumps. The changes in power which did affect the relevant portion of the test were due to the initial three heater failures in the Zr2K test and the power shifting in the Zr3M and Zr4M tests. The changes in power were recognized and included in the calculations used to predict the test results (Exhibit 1113, page 16-35). The data used to test the predictive capability of the core heatup model was either unaffected by the problems cited by CNI, or the input to the model was properly corrected for their effect (Exhibit 137, page 28). Therefore evaluation of the predictive capability of the core heatup model, which was based on the comparison of the predictions of the model to the data, is independent of the problems cited by CNI.

In some of the Zircaloy tests, the alumina insulation material reacted chemically with the Zircaloy, thereby heating the cladding and causing some local melting. However, this reaction occurred, as shown in further tests, nearly 100°F above the Interim Acceptance Criterion clad temperature limit of 2300°F. Furthermore, the reaction involved a material unique to the test and not found in reactor fuels; i.e., alumina vs uranium oxide (Transcript 10,425). Therefore this phenomenon was not considered in the development of the GE core heatup model. It is worth noting that this inadvertent chemical reaction provides additional evidence of the effectiveness of the core spray. Even though some melting occurred in these high temperature tests, the temperature rise was halted and reversed by the action of the water spray (Transcript 14,270).

CNI contends that the spray water in some tests did not drain properly from the test rig, and thus it accumulated and produced some flooding of the test bundles which resulted in higher heat transfer coefficients than would be expected from spray alone (Exhibit 1041, page 5.37). However, the test engineer testified that although the possibility of water accumulation could not be ruled out, none was observed (Transcript 13,896; 13,901).

Spray Flow

CNI views the supplemental tests, used to determine the appropriate value of the spray flow rate for the FLECHT tests, to be very simple and unrepresentative (Exhibit 1041, page 5.5), and the simulation of the spray in the FLECHT tests to be in error (Exhibit 1041, page 5.39). The supplemental experiments which were performed to determine the distribution of spray flow in a reactor and the proper value to use in the FLECHT tests have been discussed by both the staff (Exhibit 1113, Chapter 16) and GE (Exhibit 1148, Section 0). The spray flow rate used in the FLECHT experiments was based on experiments which measured the distribution of the spray among the fuel assemblies in a reactor. A full-scale mockup of the top of a reactor core was used in these experiments including duplicates of the fuel bundle channels and lifting handles, the core spray sparger, and the nozzles used in a reactor. The steam separators were modeled using stand pipes (Exhibit 1148, page 0-3). The tests reproduced the expected spray distribution, velocity, and droplet size because the type and arrangement of the spray nozzles duplicated those of a reactor

(Exhibit 1113, page 16-23). Both spray flow distribution measurements and photographic observations of the core spray droplet size distribution were made during these tests (Exhibit 1148, page 0-5). Additional tests were performed with steam updraft simulated by air. Analyses demonstrate that the air flow selected for these additional tests properly simulated the expected steam flow (Exhibit 1148, page 0-5; Exhibit 1113, page 16-22). Additional tests with steam flow through a single channel at atmospheric pressure investigated the effect steam might have on droplets (Exhibit 1113, page 16-23). The results of all these tests demonstrate that the spray flow used in the FLECHT tests is less than the minimum expected in a reactor, and that steam flow up through the fuel assemblies will not have a significant effect on spray flow distribution (Exhibit 1113, page 16-23).

CNI expressed concern over the lack of experiments to determine spray droplet size distribution or the velocity and the degree of superheat of the ejected steam (CNI Concluding Statement, page 5.43). However the need for these experiments has not been demonstrated. Measurement of spray particle size distribution is unnecessary because, as discussed in the preceding paragraph, the tests made to assess the effect of updraft on the spray distribution reproduced the spray particle size distribution in a reactor (Exhibit 1113, page 16-23; Transcript 14,226). Tests to determine the steam velocity and temperature are unnecessary since the velocity increase due to the expanded specific volume of superheated steam is shown both qualitatively (Transcript 13,951) and quantitatively (Exhibit 1113, page 16-22) to be offset by the decreased steam generation rate resulting from the diversion of energy to the superheating of the steam.

"A hint that the exit velocities may be very great" (CNI Concluding Statement, page 5.44), and presumably greater than predicted, is based on an incorrect paraphrasing of a comment by a staff consultant, which comment is not on the record (Exhibit 1113, Reference 16.20). The consultant noted that the steam plume from the high temperature Zr3M and Zr4M tests (not the lower temperature Zr2K test cited by CNI) was significantly greater than previously observed. This subjective observation is only indicative that the higher rod temperatures observed in these tests may have resulted in steam velocities higher than predicted for a reactor. In any case, these temperatures are higher than predicted for a reactor, and they occurred late in the test transient - a period after reflooding occurs in a reactor. Therefore this observation is irrelevant to the development of the Proposed Rule.

Bundle Power

Although GE makes no mention of it in their Concluding Statement, both the staff and CNI recognized that the Zircaloy bundles in the BWR FLECHT tests were not tested at power levels representative of current BWR power levels (Exhibit 1113, page 16-41; and Exhibit 1041, page 5.38). However, the CNI Concluding Statement (page 5.13) incorrectly assumes that the staff ignored this fact and gave "the benefit of the doubt to system effectiveness". Although the thermal response of the test bundles is not completely representative of the highest powered BWR fuel assemblies, the heat transfer mechanisms operative during spray cooling were simulated since they are strongly dependent on temperature and not direct functions of bundle power (Exhibit 1113, page 16.17). Since the temperatures in the test bundles exceeded the range of temperatures calculated for any fuel assembly following a postulated LOCA (Exhibit 1113, page 16-20), the heat transfer coefficients based on this data and specified in Section II.M of the Proposed Rule are appropriate for calculating temperatures in any fuel assembly, including the highest powered. Use of these heat transfer coefficients in calculating the temperatures of bundles with various power levels correctly predicts the thermal response. This confirms the staff's conclusion that "the peak clad temperature was a strong function of power" in the BWR FLECHT test (Exhibit 1113, page 16-15). Therefore, the information cited by CNI reaffirms rather than contradicts the staff's conclusions.

Conclusion

The Regulatory staff concludes that the BWR-FLECHT tests are adequate for the purpose intended, the results are applied in a conservative manner when in accordance with Section II.M of the Proposed

Rule, and the resulting calculations conservatively predict the temperature response of the fuel cladding in a BWR core.

CHANNEL BOX WETTING

The Proposed Rule Appendix K, Section II.N

N. Channel Box Wetting (Applies Only to Boiling Water Reactors). Following the blowdown period, heat transfer from, and wetting of, the channel box shall be based on appropriate experimental data. For reactors with jet pumps and fuel rods in a 7 x 7 fuel assembly array, the following heat transfer coefficients and wetting time correlation are acceptable.

1. During the period after lower plenum flashing, but prior to core spray reaching rated flow, a convective coefficient of zero shall be applied to the fuel assembly channel box.

2. During the period after core spray reaches rated flow, but prior to wetting of the channel, a convective heat transfer coefficient of 5 Btu-hr^{-1}-ft^{2}-F^{-1} shall be applied to both sides of the channel box.

3. Wetting of the channel box shall be assumed to occur 60 seconds after the time determined using the correlation based on the Yamanouchi analysis ("Loss-of-Coolant Accident Emergency Core Cooling Models for General Electric Boiling Water Reactors," General Electric Company Report NEDO-10329, April 1971).

Discussion of Channel Box Wetting

The wetting time specified in the proposed rule is unchanged from that previously specified in the Interim Policy Statement, but the heat transfer coefficients now proposed are slightly lower. Both the wetting time and the heat transfer coefficients are based on test results from the BWR Full Length Emergency Cooling Heat Transfer (FLECHT) program (Exhibit 137).

The specified wetting times and heat transfer coefficients can conservatively represent the conditions expected in a BWR following a postulated LOCA only if the tests on which they are based are applicable to the specific LOCA conditions under consideration, and the coefficients and wetting time are applied in such a manner that the temperature response of the fuel clad is conservatively predicted. A discussion of the adequacy of the BWR FLECHT program is contained in the previous section of this Concluding Statement (Section II.N, Spray Cooling Heat Transfer) and in Exhibit 1113, Section 16. The results of those tests relating to channel wetting and the manner in which the results are applied are discussed below.

The results of the BWR FLECHT tests have been used to develop the correlation specified in Section II.N of the Proposed Rule. The correlation is based on a theoretical development by Yamanouchi. The correlation relates channel wetting time with a parameter which is a function of the channel temperature and a time-dimensioned grouping of channel thermal parameters (Exhibit 131, page D-19). Using measured channel temperatures from the tests, the correlation predicts well the channel quench times observed in the stainless steel bundle tests. The quench times of the channels in the Zircaloy bundle tests are not as well correlated due to distortion of the channel and variations in temperature between different sides of the channel (Exhibit 1113, page 16-33). If the correlation was modified to fit the Zircaloy bundle channel data alone, this modification would be less conservative (Transcript 14,048). A conservative estimate of the wetting time of a fuel assembly channel with a known temperature can be made using the procedure specified in Section II.N of the Proposed Rule, since all but one of the quench times observed in BWR FLECHT tests are less than 60 seconds after the time calculated using the specified correlation (Exhibit 1113, page 16-33).

The Zircaloy-test-bundle channel boxes were identical to a reactor-fuel-assembly channel box, thus assuring identical thermal properties. However, the fuel-assembly-channel temperature must be calculated before the correlation can be used to predict quench times. The temperature of the channel is a function of the net energy radiated from the rods and the energy convected to the surrounding steam atmosphere (Exhibit 1113, pages 16-8 and 16-20). Although the radiant heating of the channel in the tests closely simulated that expected in a reactor (Exhibit 1113, page 16-6), the energy assumed to be convected from the channel box included thermal radiation to the test rig structure (Exhibit 1113, page 16-8). A convection coefficient based on the uncorrected test data might be high by a factor of two. Therefore, the coefficient specified in Section II.N has been corrected for thermal radiation to the test rig structure.

CORE FLOW DISTRIBUTION DURING BLOWDOWN

The Proposed Rule, Appendix K, Section II.0

O. Core Flow Distribution During Blowdown (Applies Only to Pressurized Water Reactors).

1. The flow rate through the hot region of the core during blowdown shall be calculated as a function of time. For the purpose of these calculations the hot region chosen shall not be greater than the size of one fuel assembly. Calculations of average flow and flow in the hot region shall take into account cross flow between regions and any flow blockage calculated to occur during blowdown as a result of cladding swelling or rupture. The calculated flow shall be smoothed to eliminate any calculated rapid oscillations (period less than 0.1 seconds).

2. If fuel cladding swelling or rupture is calculated to occur in the hot region during blowdown, the hot region flow shall be multiplied by a flow reduction factor of 0.8 to form the flow input data for the hot channel heatup calculation.

3. A method shall be specified for determining the enthalpy to be used as input data to the hot channel heatup analysis from quantities calculated in the blowdown analysis, consistent with the flow distribution calculations.

Discussion of Core Flow Distribution During Blowdown

An integral part of the LOCA analysis is the prediction of local coolant characteristics (flow and enthalpy) at the hot portion of the core (Exhibit 1113, Section 7). During blowdown the rapidly flashing coolant serves as a heat sink for the removal of energy stored in the fuel rods during normal operation. Thus, the local coolant conditions influence the peak cladding temperature. The open lattice characteristics of current PWR's and possible swelling of some of the fuel rods during blowdown complicate the physical phenomena.

Mostly due to fluid density differences, relatively hot portions of the core will have different flow transients than cooler portions of the core. Flow variations may also exist within these two regions if the geometry of the individual coolant channels is distorted due to swelling of the cladding.

The Regulatory staff recommends evaluation models that (1) account for flow variations among the various regions of the core; and (2) apply a penalty to the calculated flow rate to account for local flow variations within a region once blockage is calculated to occur during blowdown. The recommended penalty is 20 percent reduction in the calculated flow rate. This reduction should apply only after clad perforation and swelling are predicted in the hot region by appropriate calculations. For the purpose of performing core flow distribution calculations, the hot region should not be larger than the size of an individual fuel assembly. Should calculational techniques be improved to a point that the flow distribution within a region can be predicted, revision of Section II.0 would be appropriate.

The present recommendation represents a change relative to the Interim Policy Statement, and it develops further the Regulatory staff position expressed in the Supplemental Testimony (Exhibit 1113, page 7-12). Evaluation models approved under the IPS take 80 percent of the calculated core average flow for use in the cladding temperature calculations (Exhibit 1001). It has been acknowledged that (1) the flow calculations are sensitive to numerical oscillations; and (2) the coolant enthalpy and flow in the hot regions of the core differ from those of an average region (Exhibits 1001, 1113). This knowledge led to the IPS requirements that the core flow rate predicted by the system decompression code should be smoothed and multiplied by a factor of 0.8. Regulatory staff calculations indicated that use of a smoothed flow rate is conservative (Exhibit 1113). The 0.8 factor accounted for uncertainties in core flow distribution (Exhibit 1001).

During the course of the hearing additional information on flow redistribution became available. The extent of flow blockage due to cladding swelling and perforation was discussed in detail. Various participants also presented predictions of hot region flow transients which accounted for the effect of flow blockage on flow distribution. These developments are summarized in Exhibit 1113, Sections 7 and 20. At the time that the Staff Supplemental Testimony was prepared (September 1972) all information available on core flow distribution was reviewed and the following conclusions were reached (Exhibit 1113):

1) The hot subchannel flow cannot be represented realistically by a single multiplier, such as 0.8.
2) The core enthalpy distribution should be considered.
3) The effect of postulated or potential blockage should be considered.

The state-of-the-art of calculational techniques was also reviewed and a three-step program was recommended, namely: (1) extend the use of the interim requirements for approximately one more year; (2) develop and use improved analysis techniques; and (3) verify the improved techniques with experimental data.

The Redirect and Rebuttal Testimony of various participants presented additional information on flow distribution during blowdown. B&W (Exhibit 1137, pages 8-1 to 8-3) performed a study with multi-region core representation using the CRAFT code. This study was repeated with calculated flow area reductions representing flow blockages in various parts of the core. CE (Exhibit 1144, pages 5-4 to 5-6) discussed the results of CEFLASH-4 calculations using a two-region core model. This study also included the explicit representation of flow blockage. Both the B&W and the CE studies selected one fuel assembly as the size of the hot region. No attempt was made to predict flow variations within this region. The Westinghouse Redirect Testimony (Exhibit 1151, page 15-1) presented experimental verifications of the THINC III code which is used to calculate flow redistribution due to blockage during blowdown. Calculations were compared with results of steady-state subchannel temperature measurements and with results of the Westinghouse isothermal flow blockage experiments.

The conclusions of each of the above-mentioned studies supported the first two suggestions of the Regulatory staff's Supplemental Testimony, namely that (1) the hot subchannel flow cannot be represented by a single multiplier applied to the average channel flow; and (2) the core enthalpy distribution should also be considered in determining criteria. Regarding flow blockages the studies concluded that, when blockages were explicity represented in the calculations (within the limitations of these calculations), the resulting peak cladding temperatures were approximately the same as calculated by the present evaluation models. B&W reported a 22°F reduction in peak cladding temperature when using calculated blockages (Exhibit 1137). CE stated that an assumed 65 percent blockage in the hot region and no blockage in the rest of the core gave approximately the same results as the present CE evaluation model (Exhibit 1144). The CE assumption is severe since no blockage greater than approximately 70 percent has been observed in 16-channel test assemblies or in test assemblies having more than 16 channels (Transcript 9166-67); current PWR fuel assemblies contain approximately 200 channels. Based on these findings, B&W, CE and Westinghouse recommended the continued use of 80 percent of the computed average core flow in their future evaluation models. The same position was taken by each of these three participants in their Concluding Statements. The most complete review of the evidence supporting the IPS requirement on core flow redistribution can be found in the B&W Concluding Statement.

The CNI Concluding Statement (Section V.E) referenced statements of witnesses expressing concern with the selection or justification of the IPS requirement that 80 percent of average core flow be used in the heatup calculations. Based on this evidence CNI concluded that the IPS requirements on core flow redistribution are unacceptable. CNI suggested no alternatives to the assumption of 80 percent of average core flow.

The Regulatory staff concludes that the technology is now available for the explicit representation of the hot assembly in the blowdown calculations. Furthermore, this representation has three advantages over the IPS requirements:

1) it provides a more realistic approach for the prediction of the time history of the hot assembly flow;
2) it provides the needed local enthalpy input to the cladding temperature calculations;
3) it permits the direct representation of assembly-wise blockage in the calculations.

For these reasons, the Regulatory staff proposes the incorporation of hot assembly flow calculations into the evaluation models. At the same time the Regulatory staff recognizes that no satisfactory subchannel-flow calculational techniques are presently available to predict flow variations during blowdown within a hot assembly in the presence of fuel swelling. Until appropriate methods are developed, and because of the insensitivity of this multiplier on peak clad temperature (Exhibit 1113, Sections 7.0 and 10.0), the Regulatory staff recommends a 20 percent reduction in the calculated hot assembly flow to account for possible reductions in local flow in the immediate vicinity of the swollen fuel region. For reasons given above, this approximate treatment of the coolant flow is acceptable now. However, the Regulatory staff expects that development of improved analytical models and performance of experiments will continue in the direction indicated in Section 7 of Exhibit 1113. The development of experimentally verified, three-dimensional subchannel codes for analysis of power- and blockage-induced flow redistribution was the goal suggested by the staff in that earlier document.

COOLING WATER INJECTED DURING BLOWDOWN

Proposed Rule, Appendix K, Section II.P

P. Cooling Water Injected During Blowdown (Applies Only to Pressurized Water Reactors). For postulated cold leg breaks, all emergency cooling water injected into the inlet lines or the reactor vessel during the bypass period shall in the calculations be subtracted from the reactor vessel calculated inventory. This may be executed in the calculation during the bypass period, or as an alternative the amount of emergency core cooling water calculated to be injected during the bypass period may be subtracted later in the calculation from the water remaining in the inlet lines, downcomer, and reactor vessel lower plenum after the bypass period. This bypassing shall end in the calculation at a time designated as the "end of bypass," after which the expulsion or entrainment mechanisms responsible for the bypassing are calculated not to be effective. The end-of-bypass definition used in the calculation shall be justified by a suitable combination of analysis and experimental data. Acceptable methods for defining "end of bypass" include, but are not limited to, the following: 1. Prediction of the blowdown calculation of downward flow in the downcomer for the remainder of the blowdown period; 2. Prediction of a threshold for droplet entrainment in the upward velocity, using local fluid conditions and a conservative critical Weber number.

Discussion of Cooling Water Injected During Blowdown

The Interim Policy Statement described a phenomenon identified as "accumulator bypass." Accumulator bypass is a term used to describe an assumption regarding emergency core coolant injected into the primary coolant system during the blowdown period of a postulated PWR cold-leg break. In the Interim Policy Statement the several PWR evaluation models require that ECC injected during the blowdown period should be considered lost. In that context, blowdown was defined as the time at which zero break flow was first computed. The Proposed

Rule, above, requires an approach to the calculated end of bypass which is based on the physical phenomena responsible for bypass. This requirement replaces the previous method which was arbitrary in nature, as explained below.

Two principal reasons were cited in Exhibit 1001 as necessitating the arbitrary, conservative bypass requirement. The computer codes were regarded as inadequate in their ability to account for the interaction of the cold ECC with the hotter primary fluid, and several small scale tests, the Semiscale 845-851 test series conducted at ANC, exhibited experimental behavior at variance with code predictions.

Testimony by the participants offered differing viewpoints on the subject of ECC bypass. Westinghouse discussed 1/16 scale tests they performed in relation to ECC bypass, but the test results were never fully reported (Transcript 14,618). The results were generally insufficient to permit quantification of a change in the bypass requirement (Transcript 14,629). Some qualitative results from the tests were that the dimensions of the lower plenum, the hot-leg nozzles, and the downcomer affected bypass (Transcript 14,638). Westinghouse witnesses further testified that they were not now proposing a change in the bypass requirement (Transcript 14,707). Combustion Engineering testified that the 100 percent bypass assumption is conservative, but that there is a possibility that some ECC is lost (Transcript 13,181). This opinion, apparently based on CE's engineering judgment as opposed to a calculation, was also offered earlier by the staff in Exhibit 1001.

Further discussion of ECC bypass was presented at the redirect/rebuttal phase of the hearing. The Regulatory staff, in Section 8 of Exhibit 1113, stated its belief that the bypass analysis could be improved if a method was used which related in an analytic fashion to the phenomena that might cause ECC rejection. The staff recommended that a time in the blowdown sequence be established as the "end of bypass" which would, for this purpose, supplant "end of blowdown." The suggestion was that ECC bypass would still be assumed, but it would be terminated on the basis of a mechanistic rather than an arbitrary model. This approach would permit a variation in analyses dependent on the extent to which analyses and experimental resources were applied to the problem. During the direct phase of the hearing, and in the redirect/rebuttal phase, only one other participant, B&W, offered specific suggestions as to how to best quantify the end-of-bypass phenomenon. B&W presented in their Redirect/Rebuttal Testimony (Exhibit 1137, Section 4) three separate formulas for quantifying the process wherein the upflow of steam can inhibit the downflow of liquid. These mechanisms are based on two-component flow theory and ignore the condensation effect that would accompany the interaction process. The Regulatory staff concludes that the B&W suggestions are useful and that the most conservative of the three, concerning droplet ejection, should now be used. In practice this would permit end of bypass for a small upward velocity in the downcomer. In the Proposed

Rule the use of droplet entrainment is recommended as an acceptable method of defining "end of bypass." Greater detail on this mechanism is given in Exhibit 1137. Alternatively, and even more conservatively, the staff concludes that when the direction of flow in the reactor downcomer is downward, no additional ECC bypass need be assumed. The staff believes that other means of calculating ECC bypass also exist. As more data become available the staff believes other methods should be proposed and, if acceptable, could be adopted for use.

B&W in its Concluding Statement suggested a model for ECC bypass which appears to be consistent with the staff's Proposed Rule, although they believe droplet ejection to be conservative. Westinghouse in its Concluding Statement suggested use of the Wallis flooding correlation. The Wallis correlation is also one of the three methods discussed in the B&W Redirect/Rebuttal Testimony. Westinghouse had previously testified that they did not have any experiments that were specifically addressed to the Wallis correlation (Transcript 14,690). The Wallis correlation is less conservative than two other procedures for computing ECC bypass (Exhibit 1137, page 4-5). Further, Westinghouse has not presented anywhere on the hearing record any comparisons of the Wallis correlation with steam-water data or with decompression data. As a result, the staff can not now recommend adoption of the Wallis correlation for ECC bypass, but we do not rule out such acceptance in the future, based on appropriate supporting information.

CNI in its Concluding Statement (page 6.3) commented on "accumulator exhaustion." Therein CNI expresses a concern that the residual accumulator inventory at the end of blowdown may be inadequate to fill the lower plenum volume. CNI is in error in its allegation that residual accumulator inventory is overlooked. The Interim Policy Statement in the approved evaluation models for PWR's required a calculation of the sequence of events subsequent to the end of blowdown based on the accumulator inventory at end of blowdown. For some breaks in some plants the result of that requirement is that the accumulator inventory is not sufficient to refill the lower plenum. In such cases reflooding is provided by ECC pumps, and the cladding temperature transient is computed accordingly. CNI offered no suggestions regarding minimum accumulator inventory during direct or redirect/rebuttal testimony. However, it is self-evident that what might be an adequate residual accumulator inventory at end of blowdown for one break size could well be nonconservative for another. That is why a spectrum of breaks must be considered in LOCA analyses, and that is why such consideration was a part of the IPS and remains a part of the Proposed Rule.

Combustion Engineering did not offer in its Concluding Statement any useful suggestion as to how to calculate ECC bypass. They suggest for the first time (pages 3-22) a "waterfall" concept, but they offer no quantitative modeling procedure. The ECCS Utility

Group in its Concluding Statement did not comment on ECC bypass, nor did the General Electric Company.

In summary, the Regulatory staff recommends the replacement of the IPS requirement concerning accumulator bypass during blowdown by the more realistic approach of Section II.P, Appendix K of the Proposed Rule.

REFLOOD HEAT TRANSFER

The Proposed Rule, Appendix K, Section II.Q

Q. Reflood Heat Transfer (Applies Only to Pressurized Water Reactors). For the early portion of the reflood period, during which droplet entrainment or fluid oscillations do not transport a two-phase mixture to the core hot spot, heat transfer calculations shall be for steam-only cooling and shall take into account any flow blockage calculated to occur as a result of cladding swelling or rupture as such blockage might affect both local steam flow and heat transfer. A transition to reflood heat transfer coefficients based on applicable experimental data, including FLECHT results ("PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," Westinghouse Report WCAP 7665, April 1971) shall be made when calculated conditions are sufficient to transport a two-phase mixture to the hot spot. The criteria for such transition shall be justified by analysis and/or experimental results. The use of a correlation derived from FLECHT data shall be demonstrated to be conservative for the transient to which it is applied; presently available FLECHT heat transfer correlations ("PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report," Westinghouse Report WCAP-7435, January 1970; "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Group II Test Report," Westinghouse Report WCAP-7544, September 1970; "PWR FLECHT Final Report Supplement," Westinghouse Report WCAP-7931, October 1972) are not acceptable. New correlations or modifications to the FLECHT heat transfer correlations are acceptable only after they are demonstrated to be conservative, by comparison to FLECHT data, for a range of parameters consistent with the transient to which they are applied.

Discussion of Reflood Heat Transfer

The staff's Direct Testimony (Exhibit 1001) stated that experimental data obtained from the PWR-FLECHT program (Exhibit 150) was sufficient to permit a realistic evaluation of the heat transfer phenomena during reflood. In reaching this conclusion, the staff recognized uncertainties with respect to:

LI 1918

1) PWR-FLECHT bundle-housing effects.

2) Potential differences between heat transfer coefficients for Zircaloy and stainless steel.

3) Potential in PWR's for oscillatory reflood behavior not observed in PWR-FLECHT.

4) Experimental measurement and correlation errors.

Before writing its Supplemental Testimony the staff reviewed the hearing record to that date and assessed the information and criticisms offered by participants with regard to the above considerations and the PWR-FLECHT tests in general. The results of that review are presented in detail in Section 17 of the staff's Supplemental Testimony (Exhibit 1113).

In view of continued expressions of concern regarding the validity of the PWR-FLECHT experiments and the application of the data, the comments which follow are offered to delineate the bases for the staff conclusion expressed in Section II.Q of the Proposed Rule.

FLECHT bundle-housing effects received considerable critical attention during the ECCS hearings (e.g., Exhibits 1041, pages 6.5-6.9; 1044; and Transcript pages 6778-83; 11,245-59; 11,382-6; 11,399-402; 10,676-7; 10,664-9; 10,690-6). Bundle-housing effects had been initially assessed by ANC as having about a 10 percent effect on computed heat transfer coefficients (Exhibit 1006A, page III-1). Reassessment by

ANC estimated the effect to be less than 5 percent (Exhibit 1113, page 17-3). Westinghouse estimated the effect to be less than 1 percent (Exhibit 1078, pages 46-52). The staff concludes that PWR-FLECHT heat transfer coefficients are insensitive to the bundle-housing (pages 17-2 to 17-4, Exhibit 1113).

CNI has criticized (Exhibit 1041, pages 6.7b, 6.8, 6.8a; Transcript pages 19,046-83; 19,437-38) the use of PWR-FLECHT heat transfer coefficients derived from tests using stainless steel. This concern is the CNI basis for suggesting a condition of "thermal runaway" in the tests (page 6.8a of Exhibit 1041). The staff's considerations and position regarding the use of stainless steel data is contained in Exhibit 1113, pages 17-5 to 17-10, where it is shown that the CNI concern is in error because it is based on a comparison between FLECHT Test 9573 and a reactor LOCA calculation for conditions of significantly different rod power densities. This error, and others, have been noted and commented on in the Westinghouse Concluding Statement of Position, pages C-74 to C-76; in B&W's Concluding Statement, pages 198-201; and in B&W's Response to CNI's Concluding Statement, pages 26-28. Furthermore, CNI's concluding statements regarding PWR-FLECHT, pages 5.31-5.36, no longer contest the validity of heat transfer coefficients derived from stainless steel rods. In summary, the staff conclusion is that heat transfer coefficients derived from PWR-FLECHT stainless steel experiments can be used for PWR reflood analyses.

CNI at one time expressed concern with possible errors in the derivation of PWR-FLECHT heat transfer coefficients (Exhibit 1041, pages 6.19-6.23). The evidence in the hearing record (Exhibits 150; 156; and 1113, pages 17-13 to 17-15) shows that experimental measurement errors and correlation uncertainties were understood sufficiently to permit use of the heat transfer coefficients in PWR reflood heat transfer analyses. CNI's Concluding Statement, pages 5.31-5.37, no longer contests this subject.

The potential for oscillatory flow behavior during the early portion of the reflood period has been noted (e.g., Transcript 6842; 21,021). The staff's conclusion is that the use of the FLECHT heat transfer data based on constant flooding rates underestimates heat transfer during early reflood oscillations in PWR's, as discussed in Exhibit 1113, pages 17-11 to 17-13. However, the staff has suggested (page 17-11 of Exhibit 1113) continued thorough review of the PWR-FLECHT-SET results to substantiate this conclusion for a wide spectrum of oscillatory reflood conditions.

In summary, the staff reiterates that PWR-FLECHT was designed to provide heat transfer data for use in PWR reflood analyses (Transcript 10589-91), and it was not a PWR reflood demonstration experiment as was erroneously suggested (Transcript 6798). The staff concludes in accord with its Direct Testimony (Exhibit 1001, page 3-50) that the FLECHT data offer a realistic representation of the heat transfer phenomena during reflood and should be used in the reflood period of LOCA calculations. Westinghouse's Concluding Statement (pages 111-113) and B&W's Concluding Statement (pages 205-206) reflect their support for continued use of PWR-FLECHT heat transfer data.

The staff noted in Exhibit 1113, page 17-15, that existing PWR-FLECHT heat transfer correlations, as opposed to data, do not adequately correlate heat transfer data early in reflood. Therefore, Section II.Q of the Proposed Rule requires use of the data applicable to the particular transients under consideration.

Based on the considerations discussed above, the staff concludes that PWR-FLECHT heat transfer data can be used in PWR reflood calculations, on the condition that evaluation models (see Sections II.Q and II.S of Appendix K) conservatively treat calculations of reflooding rates and associated phenomena.

STEAM-LIQUID INTERACTION IN PIPES

The Proposed Rule Appendix K, Section II.R

> R. Steam-Liquid Interaction in Pipes (Applies Only to Pressurized Water Reactors). During the refill and reflood periods, steam flow through primary coolant pipes is subject to potential interference by injected emergency core cooling water. This effect shall be included as appropriate in the thermal and hydraulic aspects of reflooding rate calculations. During refill and reflood, the calculated steam flow in reactor coolant pipes shall be taken to be zero during the time that accumulators are discharging water into those pipes, and emergency cooling water shall be assumed to mix homogenously with steam, unless experimental evidence is available regarding the realistic thermal-hydraulic interaction between the steam and the liquid. The thermal-hydraulic interaction between steam and all emergency core cooling water shall be taken into account in calculating core reflooding rate.

Discussion of Steam-Liquid Interaction in Pipes

The steam flow in the cold legs of a PWR during reflood is subject to increased resistance due to high accumulator-injection-flow rates into these legs (except for the B&W design which has accumulator injection into the downcomer). This increased resistance is due to complex interaction (Exhibit 1113, Section 14) between the injected accumulator water and the steam flow. The increased resistance has a potential for causing a decreased flooding rate (Transcript 11,825-7). To include these effects in the flooding rate calculations, it is necessary to account for the physical phenomena which occur. In the absence of experimental data, the IPS contained the conservative requirement to assume a blocked-loop (no steam flow) condition to exist during accumulator injection in Westinghouse PWR's and an increased steam flow resistance in Combustion Engineering PWR's. The staff has stated that total steam flow blockage is a worst case, and in reality steam flow past the accumulator injection point is likely (Exhibit 1001). The evaluation models of the Proposed Rule will allow for calculations of the effects of steam-water interaction during refill and reflood when test results are available for the particular configuration and conditions expected during accumulator injection (such as those reported in Exhibit 227, CE Redirect-Rebuttal Testimony, pages 8-10 to 8-18). These same thermal-hydraulic effects are expected to occur during the injection of other ECC (i.e., injection by both high- and low-head pumps), even after accumulators have been exhausted (Exhibit 1113, Section 8.0). The effects on steam flow due to all ECC injection should be taken into account, as provided in the Proposed Rule.

During ECC injection there is a momentum and energy exchange between the steam and water. (CE Concluding Statement, Section 3.3; Westinghouse Concluding Statement, pages 80-81). Both Westinghouse (1/14 Scale) and CE (1/5 Scale) have performed tests to determine effect of injection into the cold leg during reflood. CE in their Concluding Statement, page 3-64, has proposed a new prediction model that includes the effects of their injection design and of momentum and energy exchange between ECC and steam. Westinghouse at page 80 of its Concluding Statement has also proposed to submit, at a later date, a new refill-reflood model. These proposed changes will be evaluated by the staff when they are submitted. The staff's conclusion is that new refill and reflood models proposed by the PWR vendors should include the thermal and hydraulic effects on steam flow of all emergency core coolant.

The CNI Concluding Statement did not reference steam-water interaction as a problem in core-reflooding-rate calculations.

REFLOODING RATE CALCULATIONS

The Proposed Rule, Appendix K, Section II.S

S. Reflooding Rate Calculations (Applies Only to Pressurized Water Reactors). The refilling and reflooding flow rate shall be calculated as a function of time using an acceptable thermal and hydraulic model. Core reflooding calculations which neglect dynamic effects leading to fluid oscillations in the system are acceptable, as are calculations which include dynamic effects. For both calculational options, core and system thermal-hydraulic phenomena shall be modeled and reactor primary coolant pumps shall be assumed to have locked impellers. The ratio of total fluid flow at the core exit plane to the total liquid flow at the core inlet plane (carryover-rate-fraction) shall be used to determine the core exit flow and shall be determined in accordance with applicable experimental data (for example, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," Westinghouse Report WCAP-7665, April 1971; "PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report," Westinghouse Report WCAP-7435, January 1970; "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Group II Test Report," Westinghouse Report WCAP-7544, September 1970; "PWR FLECHT Final Report Supplement," Westinghouse Report WCAP-7931, October 1972).

The effects on the reflooding rate of the compressed gas in the accumulator, which is discharged following accumulator water discharge, shall also be taken into account.

Discussion of Reflooding Rate Calculations

The PWR refill-reflooding computer programs described in the Interim Policy Statement provide a means to calculate the average core reflooding rates. The staff has re-examined the evidence and arrived at the view (Exhibit 1113, page 14-17) that more sophisticated refill-reflood computer programs can and should be developed to more accurately represent the physical phenomena that would occur. An example of the present absence of sophistication is the failure of all but one vendor's reflood computer programs to predict the oscillatory nature of reflood observed in the FLECHT-SET experiments. (Exhibit 1113, pages 14-15 to 14-16). In addition, both core and system thermal and hydraulic effects which occur during refill and reflood should be modeled. A partial list of such phenomena to be considered was presented in the staff's questions to vendors (June 1972).

The degree of conservatism in reactor refill and reflooding calculations for PWR's has been discussed extensively in the hearing (Transcript pages 12,631; 13,053-5; 12,598-609; 19,197-215; 10,767-9; 10,272-4; 10,268-74 and others). Core reflooding models include calculations and assumptions concerning parameters such as containment pressure, loop seals, steam-water interactions during accumulator injection, carryover-rate-fractions, pump resistance, heat sources, and reflood initial conditions (Exhibit 1113, Sections 6, 8, 10, 14 and 15). Several examples of conservatism in these models can be given here. Blowdown heat transfer calculations are performed so as to maximize the stored energy in the fuel at the beginning of the reflood calculation (see discussions above of proposed Sections II.H, I, and J of Appendix K). Conservatism is also provided by the treatment of accumulator bypass with regard to minimizing the water remaining in the vessel at the end of blowdown and thereby delaying and decreasing the calculated initial core reflooding rate (Exhibit 1113, Section 8). Calculational conservatisms are also discussed by the PWR vendors (B&W Concluding Statement, pages 206-223; CE Concluding

Statement, pages 3-58 to 73; Westinghouse Concluding Statement, pages 74-82).

CNI in their Concluding Statement, page 5.20, states that present predictions of reactor flooding rates "are now very close to the expected conditions for a double-ended PWR inlet line break." On the contrary, the evidence shows that because of the conservatisms listed above and because the reactor reflooding rates predicted by current reflood codes are for average, not oscillatory, system thermal-hydraulic response, the calculated reflooding rates are lower than would be expected to occur in reality (Exhibit 1113, Section 17).

Both B&W (Concluding Statement, page 223) and CE (Concluding Statement, Section 3) have concluded that the use of carryover-rate-fractions leads to overprediction of core exit flow and under-prediction of reactor reflooding rate. Information required for calculating the exact carryover rate is not available since only the entrained water was measured as a function of time in the FLECHT experiments. However, use of the carryover-rate-fraction as proposed by the staff in Section II.S, Appendix K of the Proposed Rule insures that the core flow (steam plus water) will be conservatively estimated. Both B&W and CE use a heat transfer model to predict steam generation rates. They also assume constant liquid entrainment by the steam (B&W Concluding Statement, pages 213-217; CE Concluding Statement, pages 3-72 and 3-73). In these models the percent of the calculated fluid flow that is assumed to be entrained liquid is constant throughout the refloodinq transient. The staff believes that this assumption is not supported by the data obtained in the PWR FLECHT tests (Exhibits 148, 149 and 150). Carryover-rate-fraction data based on quench front velocity obtained from several applicable FLECHT runs (Exhibit 1137, pages 9-1 and 9-4) was used to develop the B&W correlation. The Westinghouse and the B&W carryover-rate-fractions are described as a function of pressure, flooding rate, subcooling, height in the core, and peak heat generation (Exhibit 1137, page 9-22 and Exhibit 1079, Appendix F). The staff agrees that these parameters influence entrainment.

Westinghouse has suggested that their correlation for carryover-rate-fraction should be included in an improved version of their refill-reflood code (Westinghouse Concluding Statement, pages 74-76). The staff recognizes that use of a carryover-rate-fraction determined from FLECHT hot bundle data provides a conservative estimate for calculating the core exit flow rate (Exhibit 1137, page 9-1) because correlations developed from FLECHT data do not account for the variation of mass stored above the quench front (Westinghouse Concluding Statement, pages 74-76), and the data corresponds to a hot bundle and not to the average core. Westinghouse has also indicated that preliminary results from PWR-FLECHT-SET carryover data, when compared to their carryover-rate-fraction correlation, show the validity of the correlation under oscillatory and gravity feed conditions (Westinghouse Concluding Statement, page 76).

The staff concludes that reflooding rates predicted by the carryover-rate-fraction are more conservative than those predicted by a constant liquid entrainment assumption, and, therefore, the staff has proposed that carryover-rate-fraction models be used in the Proposed Rule.

CE has concluded that the pump model used during reflood is overly conservative and should be modified for more realism. This would increase calculated reflooding rates (CE Concluding Statement, page 3-60). In the Staff Supplemental Testimony (Exhibit 1113) it was stated that the broken loop pump could overspeed during the blowdown, and its condition is not well-defined (Exhibit 1113, page 14-10). The pumps in the unbroken loop are not predicted to overspeed (Exhibit 1113, page 14-10) and probably will be left in a coastdown condition at the end of blowdown. The staff is still of the opinion that the speed of each pump at the end of blowdown is not well defined. In the absence of any acceptable tests of pump operation during blowdown conditions, a continuation of the conservative IPS assumption of locked pump rotor (i.e., speed = 0) during refill and reflood is proposed by the staff. The staff has also stated (Exhibit 1113, Sections 6 and 14) that tests should be performed to determine the condition of the pump and the pump resistance to steam flow at conditions expected during reflood. CE agrees (Concluding Statement, page 3-60).

After injection of accumulator water during reflood the gas (nitrogen) used to pressurize the accumulator is injected into the reactor system. Accumulator gas affects the thermal and hydraulic response of the steam-filled system. The time period for considering the effects of accumulator gas injection should extend from the time at which the accumulator liquid volume is depleted until the accumulator gas flow rate is negligible. The staff believes that these effects should be considered by appropriate hydraulic models or by using a conservative assumption concerning steam flow during the period of gas injection. Accumulator gas injection models are contained in the evidence (Exhibits 232, 225, 221) but they were not discussed elsewhere in the hearing.

DOCUMENTATION OF EVALUATION MODELS

The Proposed Rule, Appendix K, Section III.A

A. Documentation

1.(a) A description of each proposed evaluation model shall be furnished. The description shall be sufficiently complete to permit technical review of the analytical approach including the equations used, the assumptions made, and the values of all parameters or the procedure for their selection, as for example, in accordance with a specified physical law or empirical correlation.

(b) The description shall be sufficiently detailed and specific to require significant changes in the evaluation model to be specified in amendments of the description. For this purpose, a significant change is a change that would result in calculated fuel cladding temperatures different by more than 20°F than the temperatures calculated (as a function of time) previously for a postulated LOCA.

(c) A complete listing of each computer program, in the same form as used in the evaluation model, shall be furnished to the Atomic Energy Commission.

2. For each computer program, solution convergence shall be demonstrated by studies of system modeling or noding and calculational time steps.

3. Appropriate sensitivity studies shall be performed for each evaluation model, to evaluate the effect on the calculated results of variation in noding, phenomena assumed in the calculation to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items shown to be sensitive, the choices made shall be justified.

4. To the extent practicable, predictions of the evaluation model, or portions thereof, shall be compared with applicable experimental information.

Discussion of Documentation of Evaluation Models

Thorough documentation of evaluation models is required in view of the complexity of LOCA analysis methods. The IPS identified specific reports which describe or list calculational procedures carried out with the approved evaluation models. However, previous experience has shown that additional documentation would be useful (Exhibit 1031, page 2; CNI Concluding Statement, page 4.16; and Transcript pages 5643; 6675; 6691; 10,879 to 10,883).

Considerable hearing time was devoted to consideration of the adequacy of code analysis methods (see Exhibit 1043 and Transcript pages 8294; 8386; 11,065-11,112; 11,156). Time would be saved in the hearing process, in generic reviews, and in case reviews, if for each evaluation model a detailed description were provided which defined the analytical approach and equations, the assumptions, the references, the selection and justification for the input parameters, and the mathematical symbolism used to establish the corresponding computer programs.

With respect to the programming and the mathematical treatment, the staff believes that a complete description and listing of the computer programs, in identical form to those approved and being used (at a specific time) for LOCA analyses, should be provided to serve as an information "source file." Such a source file provides a formal code index which can be used to guarantee that the codes used for safety analyses always correspond to the approved, published evaluation models.

Revisions in evaluation models, and thus in computer programs, are recognized as inevitable. Westinghouse has agreed in this regard; see pages 71 and 80 of their Concluding Statement. Such revisions should be described in detail including a revised computer program listing to update the previous "source file."

The need for noding and sensitivity studies for the computer programs is clearly reflected by the hearing record (e.g., Exhibits 1006, 1043, 1044, 1001, 1113, 1148). The Proposed Rule formalizes the scope and intent of such studies.

The need for comparisons of analytical models with experimental data is discussed in the written testimony of all participants. The staff (Exhibits 1001 and 1113) has recognized the value of such comparisons, and therefore the Proposed Rule requires such comparisons.

C. Discussion of Conservatism

1. General

a. Definition - For this discussion, a conservatism is defined as an item less favorable than reality. The use of conservatism provides margin if the reality should ever occur. Favorable must be defined in context according to use; what is favorable in one situation or for one purpose can be unfavorable under different circumstances.

Conservatisms can in principle be found in postulates of events or failures believed to be so unlikely as to be beyond the bounds of reality, or in criteria stricter than realistically necessary, or in analysis methods and values of parameters giving calculated results less favorable than a realistic evaluation of the circumstances forming the basis of the calculation. More than one such aspect of conservatism can be present simultaneously.

For ECCS criteria, reality is the course of whatever loss-of-coolant accidents might eventually occur in light-water power reactors. The conservatisms under discussion are those aspects of the proposed acceptance criteria and evaluation models that are less favorable than this reality; that is, lead to predictions of results less favorable than those that might eventually be experienced.

b. Relationship with Defense in Depth - It should be recognized that the defense-in-depth approach to reactor safety has certain inherent conservatisms (Staff Testimony, Exhibit 1001, pages 1-2 through 1-10). Indeed, there is a conservatism in requiring high standards for design, construction, and operation of systems important to safety, and installing protective systems to shut down the reactor in case something goes wrong, and then requiring provision of engineered safety features designed on the basis of serious failures in spite of the precautions, including a mandatory quality assurance program, used to prevent such failures.

The Regulatory staff believes that the margins of conservatism inherent in defense in depth are appropriate in providing assurance for the health and safety of the public. But in the following sections the staff points out the conservatisms inherent in the approach used, as well as in the detailed provisions of the Proposed Rule.

For these reasons, the very existence of the LOCA as a design-basis accident and of the ECCS criteria are evidence of a conservative approach.

c. Variations in Reality - There is a wide range of possible "realistic accidents." For each possibility, the conditions governing what would happen ("reality") are different. One of the most important sets of parameters governing reality would be the initial conditions existing when the transient began. The characteristics (size, location, orientation) of the actual break would also be determining. After the break had occurred, the actual occurrence would evolve with time in a way determined by the properties of materials and the geometry of the facility, and also by how the equipment actually functioned and the actual actions of the reactor operator.

If a LOCA should actually occur, all the parameters would have the values that go with the actual conditions then in force, which thus would become the "correct" values for that LOCA; the choices actually made by the operator (pumps actually energized; valves actually opened or closed, etc.) would determine the course of events.

d. Relationship of Calculations and Actual Events - The thrust of Section c above is that there is a spectrum of possibilities. Each occurrence, real or postulated, is one of a large population of possible occurrences, different occurrences being characterized by different initial conditions, different break characterisitics, different equipment operability, etc., etc. The number of combinations, as previously stated, is far too large to allow investigation of all.

With each possible occurrence, it is possible in principle to associate a probability. (The sum of all probabilities is the probability of any LOCA per unit time.)

Although some pioneering work has been done on engineering statistical studies of the probability distribution of the population of possible LOCA occurrences, (Staff Supplemental Testimony, Exhibit

1113, Section 2), much remains to be done before such a distribution is well known. We do, however, have some knowledge of which postulated sequences of events are judged to have lower probabilities than others.

Besides the uncertainties in the parameters of the "real occurrence," as discussed above, there also are uncertainties in the presently available calculational techniques (Staff Testimony, Exhibit 1001 and Staff Supplemental Testimony, passim). These are uncertainties in arriving at a realistic prediction. That is why the conservatisms are put there, of course. Therefore, even for a particular postulated occurrence, for which all parameters are assumed to be known or specified, the course actually followed is not predicted with complete accuracy with present technology.

The reasons for this are discussed at great length in the record of the ECCS proceeding and in this Concluding Statement. Knowledge of physical phenomena is incomplete, and calculational techniques are not fully developed, either. It is not surprising that this be true of a LOCA; it is true to a greater or lesser extent of all calculational modeling of physical phenomena. Even the best calculations involve simplifications and approximations of the almost infinitely complex real world.

It is fortunately also true that an accurate, realistic prediction of the course of a LOCA, postulated or actual, is not needed for safety evaluation; rather, it is only necessary to have a conservative evaluation or prediction. The actual course of an eventual accident, if it occurred, would not be expected to follow such a conservative prediction; rather, its course and consequences would be more favorable than the calculation. This is the essence of our use of the word conservative; see Section a. The conservative calculation would therefore correspond to a less probable sequence of events. The choice of "suitable conservatism" is the choice of which conservatisms to apply so that the probability of the calculation being more favorable than the event is acceptably low. The Commission is sponsoring a study of probabilities of postulated reactor accidents to quantify these concepts as well as may be possible with present technology.

e. Uncertainty Analyses and Sensitivity Studies - The role and value of statistical uncertainty analyses in ECCS evaluations were noted by the Regulatory staff in Section 2.0 of Exhibit 1113. Other participants have also commented on the potential value of developing methods for performing statistical uncertainty analyses (see Exhibit 1148; Transcript pages 15,432 and 14,423). In addition, many participants to the hearing performed more realistic calculations of the LOCA than are required by the Interim Policy Statement (Exhibits 1113, 1059, 1066, 1069, 1078). These were not statistical uncertainty analyses (Exhibit 1113, Section 2), but they can be thought of as sensitivity analyses. Each such analysis showed significant reductions in peak cladding temperatures.

The Concluding Statement of CNI indicates that they have misunderstood what a statistical uncertainty analysis of a LOCA is, and what its uses and limitations are. The misunderstanding centers about an exchange of questions by CNI and answers by the Regulatory staff witness panel at Transcript pages 20,311 to 20,313. CNI makes reference to this exchange at several points in their Concluding Statement: pages 6.4, 6.8 to 6.11, 6.14, 8.2 to 8.3. The first full paragraph of page 6.9 presents the CNI view of Transcript pages 20,311 to 20,313. That paragraph reads as follows:

> "The question that was put to the Regulatory staff panel was the fundamental question of where did we stand with regard to the parameters and assumptions in the approved evaluation models. For cases where the approved evaluation models predicted acceptable transients, were these changes in the parameters of the approved evaluation models that would result in a prediction of an unacceptable LOCA transient within the range of uncertainties of those parameters? The Regulatory staff witnesses answered that they did not know. (Transcript 20312-20313)."

To clarify what the staff said at that point in the Transcript, we note that the response to the question put by CNI at lines 6 to 10 of Transcript page 20,312 was 24 lines long. The last line of the answer reads, "Therefore we cannot answer your question." The panel indicated in its answer that the question made no sense and therefore could not be answered. The panel referred in its answer (line 16, page 20,312) to a lengthy discussion with CNI on this subject during the previous day of questioning (Transcript pages 20,151 to 20,193). The discussion at those pages from the previous day explains why statistical uncertainty analyses of a LOCA are dependent upon first doing a best-estimate (realistic) analysis of the LOCA (see also Exhibit 1113, Section 2). Then and only then can the total uncertainty of the calculation be assessed and the contributions due to uncertainty in individual parameters be systematically studied. CNI has implied in their Concluding Statement that the total uncertainty of a calculation can be judged by addition of pessimistic assumptions regarding one parameter to the results of an already pessimistic calculation. This cannot be done because the sum of individual conservatisms does not equal the total conservatism (Exhibit 1059, page 7-1). Rather, total uncertainty must be judged with respect to best-estimate (realistic) calculations (Exhibit 1113, Section 2). The staff witness panel testified to this fact (Transcript pages 21,026 to 21,027) when asked by Westinghouse for clarification of the staff's earlier answer to the CNI question at page 20,311 of the Transcript.

2. Conservatisms in the Proposed Rule - Discussed here are the significant conservatisms in the Proposed Rule of the Regulatory staff presented in Chapter II of this Concluding Statement. Only the most important items of conservatism are discussed. Omission of an item may mean that the Regulatory staff believes its treatment to be

realistic rather than conservative, or that the conservatism is believed to be unimportant or not well established at present.

a. General Criteria Applicable to All Reactors

(1) Definition of Loss-of-Coolant Accident

The hypothesis of a spectrum of loss-of-coolant accidents, including large and even double-ended breaks, is a conservatism inherent in defense in depth, as discussed in paragraph 1.b of this Section. A large break occurring suddenly is highly improbable, although the occurrence rate of small leaks and breaks is not especially low. (Information on the probability of a LOCA, including applicable codes and quality assurance provisions, is outside the scope of this rule making.) Large and double-ended breaks of the sorts postulated have never occurred in Nuclear Class I piping such as that used for water reactor primary coolant pressure boundaries. Large breaks probably will be preceded in time by leaks that can and will be detected, and the plant shut down and depressurized before a serious break occurs. The entire spectrum of break sizes is nevertheless required to be considered, from small leaks up to the (highly improbable) double-ended severance of the largest primary-system pipe.

The break is assumed to occur instantaneously. This assumption increases the calculated blowdown forces for which PWR cores must be designed but has little effect on calculated ECCS performance.

(2) Peak Cladding Temperature

The value of 2200°F is believed by the Regulatory staff to be conservative by several hundred degrees (Transcript pages 19,992 to 19,993). In addition Zircaloy FLECHT bundles have survived temperatures as high as 2900°F and remained coolable (Staff Supplemental Testimony, Section 16).

The requirement that <u>no</u> cladding exceed the temperature and oxidation requirements is also a conservatism. It is well known that the neutron flux and the power density are not uniform throughout the reactor core (Exhibit 1113, Section 2). Besides the large scale variations due to neutron leakage, the loading of fuel of different enrichments into different regions of the core, and the fuel burnup, variations on a smaller scale occur because of the grids, control-rod proximity, local burnup, and refueling.

The result is that many reactors will never contain a fuel rod having the peaking factor assumed for ECCS calculations; in others, such a rod will be at the limiting conditions for only a small part of the fuel loading cycle (Exhibit 1113, Section 2).

Moreover, even where the hottest rod is as calculated, more of the fuel rods will be operating at a considerably lower rating. A calculated example (Babcock and Wilcox, "Answers to National Intervenor's Interrogatories," March 1, 1972, AEC Docket RM-50-1) for a PWR shows that if the peak cladding temperature for the hottest rod is of the order of 2200°F, almost 80% of the fuel pins will never exceed 1900°F, and fewer than half will exceed 1500°F, at the axial hot spot of each of the fuel elements.

The staff believes that this conservatism is appropriate, and accordingly the Proposed Rule provides that no single fuel rod may exceed the temperature and oxidation limits. Thus the Proposed Rule keeps even the hottest rod from failing; the great majority of rods have a large margin to failure.

b. Evaluation Models

(1) Single Failure Criterion

The assumption that the worst single failure will occur at the time of the LOCA is a combining of improbable events.

Although reliable offsite (utility grid) and onsite (emergency diesels) sources of electrical energy are required to be provided, the preformance of the ECCS is evaluated assuming only the onsite source available. The onsite source is also subject to the "worst single failure" conservative assumption. The offsite electrical power system is designed, calculated and in some instances tested to withstand the electrical system transient following a postulated LOCA. The probable availability of onsite and offsite power would energize more ECCS equipment, more quickly, than the assumed onsite power acting alone.

(2) Decay Heat

Use of 1.2 times the ANS Standard decay heat curves is higher than the best estimate (Staff Supplemental Testimony, Exhibit 1113, Section 22). The LOCA calculations are performed with power peaking factors (ratio between the power generation rate in the hottest fuel pellet and the power generation rate in an average fuel pellet) at the limiting value allowed by the technical specifications. The reactor is also assumed to have been operating for infinite time at design overpower conditions.

(3) Metal/Water Reaction

The Baker-Just equation has been shown to overpredict the best estimate oxidation rate at temperatures above 2000°F (see discussion of Section II.C.5, Appendix K of Proposed Rule, above).

(4) Critical Heat Flux

The time required to reach critical heat flux (CHF) is required to be conservatively calculated. The heat transfer calculations thereafter assume a less advantageous cooling mode. In reality, DNB should in a severe blowdown occur later than predicted, and high efficiency boiling heat transfer may be reestablished intermittently.

(5) Cladding, Swelling and Rupture

The Proposed Rule requires conservative predictions of cladding swelling and rupture. The Proposed Rule then requires analysis of the disturbed cladding, and the temperature calculations take this into account. A more realistic calculation (Staff Supplemental Testimony, Exhibit 1113, Section 2) would show fewer disturbed rods because of lower calculated rod temperatures.

(6) Initial Stored Energy in Fuel

The most unfavorable values of stored energy cannot exist simultaneously with the most unfavorable factors in, for example, after-heat, yet this unreal combination is imposed in the calculations.

(7) Accumulator Bypass

The calculations are required to assume the discard of all water injected during the bypass period; whereas realistically it is probable that some cooling water would remain in the system. The end of bypassing is also modeled conservatively (see discussion of Section II.P, Appendix K of the Proposed Rule).