UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

July 11, 2002

MEMORANDUM TO: ACRS Members

FROM: August W. Cronenberg, Cognizant Staff Engineer
Paul Boehnert, Designated Federal Official

SUBJECT: CERTIFICATION OF THE MINUTES OF THE JOINT MEETING OF THE ACRS SUBCOMMITTEES ON MATERIALS AND METALLURGY, THERMAL-HYDRAULIC PHENOMENA, AND RELIABILITY AND PROBABILISTIC RISK ASSESSMENT MAY 31,2002 - ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued July 11, 2002, have been certified as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc: J. Larkins
S. Bahadur
H. Larson
S. Duraiswamy
ACRS Staff Engineers

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

MEMORANDUM TO: August W. Cronenberg, ACRS-Senior Staff Engineer

FROM: William J. Shack,
Joint Subcommittee Chairman

SUBJECT: CERTIFICATION OF THE SUMMARY/MINUTES OF THE MEETING OF THE JOINT MEETING OF THE ACRS SUBCOMMITTEES ON MATERIALS AND METALLURGY, THERMAL-HYDRAULIC PHENOMENA, AND RELIABILITY AND PROBABILISTIC RISK ASSESSMENT - MAY 31, 2002 - ROCKVILLE, MARYLAND

I do hereby certify that, to the best of my knowledge and belief, the minutes of the subject meeting on May 31, 2002, are an accurate record of the proceedings for that meeting.

7/10/02

William J. Shack, Date
Joint Subcommittee Chairman

June 14, 2002

MEMORANDUM TO: William J. Shack, Sub-committee Chairman

FROM: August W. Cronenberg, Cognizant Staff Engineer
Paul Boehnert, Designated Federal Official

SUBJECT: WORKING COPY OF THE MINUTES OF THE JOINT MEETING OF THE ACRS SUBCOMMITTEES ON MATERIALS AND METALLURGY, THERMAL-HYDRAULIC PHENOMENA, AND RELIABILITY & PROBABILISTIC RISK ASSESSMENT MAY 31, 2002, ROCKVILLE, MARYLAND

A working copy of the minutes for the subject meeting is attached for your review. Please review and comment on them at your soonest convenience. If you are satisfied with these minutes please sign, date, and return the attached certification letter in the pre-addressed/stamped envelope attached.

Attachment: Minutes (DRAFT)

cc: S. Bahadur
P. Boehnert
J. Larkins
H. Larson
S. Duraiswamy

CERTIFIED BY:
W. Shack - 07/10/02

Date:06/14/02

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
JOINT MEETING OF THE ACRS SUBCOMMITTEES ON MATERIALS AND METALLURGY, THERMAL-HYDRAULIC PHENOMENA, AND RELIABILITY AND PROBABILISTIC RISK ASSESSMENT
MEETING MINUTES - MAY 31, 2002
ROCKVILLE, MARYLAND

INTRODUCTION

The ACRS Subcommittees on Materials and Metallurgy, Thermal-Hydraulic Phenomena, and Reliability and Probabilistic Risk Assessment met on May 31, 2002, at 11545 Rockville Pike, Rockville, MD, in Room T-2B3. The purpose of this meeting was to discuss the status of NRC staff and industry initiatives to risk-inform the technical requirements of 10CFR 50.46 for emergency core cooling systems (ECCS) for light-water nuclear power reactors.

The Subcommittees received no written comments from members of the public regarding the meeting. The entire meeting was open to public attendance. Mr. August W. Cronenberg was the cognizant ACRS staff engineer for this meeting. Mr. Paul Boehnert was the designated federal official. The meeting was convened at 8:30 a.m. and adjourned at 5:10 p.m.

ATTENDEES

ACRS

W. Shack, Chairman
M. Bonaca, Member
T. Kress, Member
G. Wallis, Member
G. Apostolakis, Member
P. Ford, Member
S. Banerjee, Consultant
D. Powers, Member
G. Leitch, Member
V. Ransom, Member
S. Rosen, Member
A. Cronenberg, Staff
P. Boehnert, Staff
V. Schrock, Consultant

Principal NRC Speakers

A. Kuritzky, RES
M. Drouin, RES
S. Lee, NRR
M. Cunningham, RES
S. Bajorek, RES
R. Tregoning, RES
N. Lauben, RES

Principal Industry Speakers

None.

There were approximately seven members of the public in attendance at this meeting. A complete list of attendees is in the ACRS Office File, and will be made available upon request. The presentation slides and handouts used during the meeting are attached to the office copy of these minutes.

OPENING REMARKS BY THE SUBCOMMITTEE CHAIRMAN

Dr. William Shack, Chairman of the ACRS Subcommittee Materials and Metallurgy convened the meeting at 8:30 a.m. Dr. Shack stated that this was a joint meeting of the ACRS Subcommittees on Materials and Metallurgy, Thermal-Hydraulic Phenomena, and Reliability and Probabilistic Risk Assessment. He noted that all ACRS Members were in attendance except for Dr. Powers. He then stated that the purpose of the meeting was to discuss the status of NRC staff efforts and industry initiatives at risk-informing the technical requirements of 10CFR50.46 for emergency core cooling systems (ECCS) for power reactors.

Dr. Shack noted that the Subcommittee had received no written comments from members of the public regarding the meeting. He then called upon Mr. Mark Cunningham of RES to begin the presentations.

DISCUSSION OF AGENDA ITEMS

NRC Staff Presentation

Mr. Mark Cunningham of RES opened the discussion for the NRC staff and introduced Ms. Mary Drouin and Mr. Alan Kuritzky of RES who were likewise seated at the presenter's table. Mr. Cunningham first outline the goals and purpose of the meeting and the proposed changes to 10CFR50/46 and schedule for completion of the technical work to support rule-making. He stated that there would be no request for an ACRS letter at this time. He then turned over the presentation to Ms. Drouin

Ms. Drouin described the background for risk-informing 10CFR50.46, including proposed changes to Appendix K and General Design Criteria 35 (GDC-35). At this point Dr. Kress asked if 10CFR50.46 specifically required ECCS for all light water reactors (LWRs). Ms. Drouin responded yes, that ECCS is spelled out as a requirement for all reactor designs in 10CFR50.46 and pointed to her overhead, which she indicated was direct wording from the code of federal regulations (CFR). Dr. Kress then asked if ECCS is a requirement for non-LWR designs. Mr. Drouin responded that she didn't want to get into discussions of requirements for unique/advanced reactor concepts.

The presentation then turned to staff recommendations on changes to 10CFR50.46, Appendix K and GDC-35, which are documented in SECY-02-0057: These recommendations were summarized as follows:

> ECCS reliability: Provide two voluntary performance-based options (one generic, one plant-specific) that would demonstrate reliable ECCS safety function without assuming LOOP and single additional failure in GDC-35.
>
> ECCS acceptance criteria: Change current prescriptive ECCS acceptance criteria in 50.46 to add a performance-based option.

ECCS evaluation model: Add an option to App. K decay heat requirement to permit use of 1994 ANS standard Staff intends to address subject of uncertainty and conservatism in App. K models separate from rule-making activity.

ECCS spectrum of break sizes and locations: Continue the feasibility study of redefining the maximum pipe break size required to be considered as part of the ECCS performance evaluation.

At this point ACRS Consultant Prof. Schrock noted that there are several options for decay heat curves, notably the 1971, 1978, and 1994 ANS curves, which produce different peak clad temperatures. He asked if this would be sorted out. Ms. Drouin responded that this would be addressed later in the presentations. The presentation was then turned over to Mr. Kuritzky., who's presentation centered on the technical work being done by RES in each of the above 4 technical areas to support rule-making.

With regards to changes in ECCS reliability requirements he noted that the revised approach would permit ECCS to be designed, operated, and evaluated based on quantitative reliability considerations instead of prescriptive assumptions on loss of off-site power (LOOP) and additional single failure. Mr. Leitch asked if new rule would allow for no diesel backup if licensee could demonstrate highly reliable off-site power. Mr. Kuritzky responded that this issue has not been decided at this point. Mr. Rosen asked if there would be changes in performance monitoring and corrective actions. Mr. Kuritzky responded no, not really and went on to discuss proposed changes to ECC reliability requirements, plant-specific approach for risk-informed alternative to GDC-35, and proposed core damage frequency (CDF) and large early release frequency (LERF) acceptance guidelines. Dr. Kress questioned if there was an absolute delta-CDF that the staff was proposing and Mr. Rosen asked if a full-scope PRA was would be required, with internal and external events, as well as seismic. Mr. Kuritzky responded that a full-scope PRA would indeed be required if a licensee wanted to change its licensing basis to risk-informed ECCS performance. Dr. Kress asked if the staff planned to also look at total site LERF if there were multiple plants on the site. Mr. Kuritzky responded that right now the staff has not specifically called out such a requirement.

Mr. Kuritzky then went on to describe staff thoughts on proposed CDF and LERF acceptance guidelines for design basis changes. He pointed out that this was essentially an analytical change not a physical one. He stated that a licensee would need to demonstrate that ECCS functional reliability is commensurate with the frequency of accidents for which ECCS success would prevent core damage or large early release. He said this could be accomplished by demonstrating compliance with the acceptance guidelines that CDF and LERF meet the quantitative guidelines and the change in risk form proposed ECCS changes not result in a significant risk increase. Members Apostolakis, Rosen, and Kress, questioned Mr. Kuritzky as to more specific CDF and LERF acceptance guidelines , with Mr. Glen Kelly of NRR interjecting from the audience that the presentation by Mr. Kuritzky was a description of the technical basis for proposed changes to 10CFR50.46 and associated guidance, but that these proposals need to be reviewed by an agency working group to go forward with rule-making and that the Commission has accepted Reg. Guide 1.174 as the process for risk-informing such decisions.

ACRS member, Dr. Ransom, asked if there would be any added benefit to the overall public health and safety by risk-informing 10CFR50.46 ECCS requirements, to which Mr. Kuritzky responded that quantification of the risk may have benefit from public confidence. Mr. Kuritzky concluded his presentation at about 10:45 a.m. , at which time Dr. Shack, the subcommittee chairman, called for a 15 minute break, until 11:00 a.m.

The presentation then centered on ECCS spectrum of break size and location, as well as LOCA frequency re-definition, which was provided by Mr. Robert Tregoning of RES. Mr. Tregoning began his presentation with an overview of large-break (LB) loss-of-coolant accident (LOCA) conditions and break size assumptions embedded in the current 10CFR50.46 prescriptive requirements and proposed changes LOCA frequency and break size redefinition. He stated that the historical approach to estimating the LOCA frequency was based on service history; however, this history did not include recent events such as leakage and failure from control rod drives such as noted at the V. C. Summer, Oconee and Davis-Besse plants. Mr. Paul Boehnert of the ACRS staff asked if pipe ruptures due to hydrogen buildup from coolant radioysis and resultant explosions, such as noted in recent events in Germany and Japan, were included in LOCA estimates. Member Wallis noted that one event occurred close to vessel and a nearby valve and could have prevented valve closure. Mr. Tregoning responded that he was aware of these events and that as such events occur, they will need to be considered in LOCA frequency estimates. Dr. Apostolakis questioned if the frequencies indicated in Mr. Tregoning overhead were historical LOCA frequencies, with Mr. Tregoning responding that yes these were historical LOCA frequencies.

Mr. Tregoning then went on to discuss how initial LOCA frequency estimates were determined from a trial staff expert panel opinion process. Member Wallis questioned if there was enough time for the panel members to do calculations and research or were they just asked to give their best guess on frequency. Mr. Tregoning responded that it was the latter, but that in the next step the intent would be to allow experts time to do some calculations to support their opinion. ACRS member Dr. Peter Ford also asked if time-dependent piping wear, associated with plant license renewal would be included. Mr. Tregoning responded that such events as noted by such questions would have to be factored into any future risk-based LOCA frequency evaluations. Dr. Apostolakis questioned if an expert in say thermal-hydraulics (T-H) can really provide a reasonable estimate of pipe rupture frequency and wouldn't such an T-H expert need to be familiar in statistical methods and PRA. Prof. Wallis added that T-H phenomena such as water-hammer seem to be absent from the elicitation questionnaire, while Dr. Bonaca interjected that the License Renewal Rule does not provide for added piping inspections although we seem to be seeing more breaks. Mr. Tregoning responded that members should not read too much into the result of this staff elicitation effort and that any licensee who chose to use the risk-informed option would need to justify their LOCA frequencies from the plant specific experience on piping wear estimates, ruptures, etc. Mr. Tregoning follow-on overheads pointed to a significant data base with regards to pipe breaks, including the familiar Skl-Swedish data base. At this point, Mr. Robert Osterrieder, of Westinghouse Electric Co. interjected that the members of the Westinghouse Owners Group would be involved in the next LOCA frequency elicitation effort, which was referred to in Mr. Tregoning's presentation as the "Intermediate-Term Elicitation". Dr. Ford also interjected that the LOCA frequency for pipe ruptures should be "binned" according to factors such as reactor operating experience, plant experience with water chemistry, piping type such as stainless-steel 304 versus 316, etc. Mr. Leitch questioned if pipe leakages versus true breaks would be included in the LOCA frequency estimates, to which Mr. Tregoning responded that yes a whole range of pipe conditions would be included, piping flaws requiring change-out to guillotine ruptures. Mr. Tregoning finished his presentation with several overheads indicating the necessary technical requirements to support any re-definition of the LOCA pipe break size that might be used by a licensee to change his licensing basis to a risk-informed ECCS basis. The subcommittee chairman, Dr. Shack, called for a lunch recess until 2:00pm.

The afternoon discussions began with a presentation by Dr. Stephen Bajorek (RES) who provided a briefing on research findings and recommendations regarding 10CFR50.46 acceptance criteria and ECCS evaluation model requirements for Appendix K. The primary focus of his presentation was on replacement of Appendix.-K requirements to use 1.2 times the 1971 ANS decay-heat standard with the 1994 ANS standard and the impact of the proposed changes in decay heat and metal-water reaction rate models on non-conservative aspects of the existing Appendix-K models. He then outlined RES recommendations including replacement of the 1971 decay heat standard with the 1994 standard, replacement of Baker-Just with Cathcart-Pawel model for metal-water reaction rate correlation, delete requirement for steam cooling at reflood rates below 1 inch/sec, and require new models to demonstrate sufficient conservatism to account for several identified non-conservatisms.

Dr. Wallis asked what was the criterion used in Appendix-K for run-away oxidation for either the Baker-Just or Cathcart-Pawel models. Dr. Bajorek (RES) responded that there wasn't any specified in Appendix-K and stated that it's not just a temperature criterion but related to a geometry and cooling dependent heat balance, where more energy is generated in the cladding then released. Prof. Banergee, ACRS consultant, interjected that in a run-away reaction it's not the absolute value of temperature that matters but the rate of change of temperature, with Dr. Bajorek concurring. The discussion then lead to fuel cladding failure criteria and that any changes to Appendix-K requirements would still stipulate maintaining a coolable geometry post-reflood. Chairman Shack stated that this in essence would require the cladding to maintain some ductility and noted that this is not really a risk-informed change but rather just a change in criteria that makes more sense. Prof. Banergee interjected that the essence of the research recommendations that it really didn't matter if Baker-Just or Cathcart-Pawel were used the real change to Appendix-K requirements actually relates to requirements on maintenance of clad ductility. Dr. Bajorek summarized that Appendix-K at this time requires the use of Baker-Just to calculate cladding oxidation and the associated heat release rate, but that research is recommending a change to require Cathcart-Pawel because it's better science though one could not say it was risk-informed.

At this point , 2:55 p.m. Chairman Shack requested a 15 minute break. The meeting reconvened at 3:10 p.m. with a presentation by Mr. Norman Lauben (RES) on the effect/impact of proposed changes for the decay heat correlation and Appendix-K. He noted that Research has proposed that the decay heat requirements in Appendix K and the best estimate guidance in Regulatory Guide-1.157 be replaced with requirements and guidance based on the 1994 ANS decay heat standard. He also stated that the Appendix K option in 50.46 currently requires fission product decay heat be modeled using the draft 1971 ANS standard with a multiplier of 1.2 and the assumption of infinite irradiation. A separate paragraph in Appendix K requires consideration of actinide decay heat.

The alternative would permit the use of the 1994 ANS decay heat standard, which involves more sophisticated uncertainty methods and a greater number of options left to the user. Mr. Lauben also noted that a number of model options in the 1994 standard were assessed by the NRC staff identified, including whether the reactor operating history should be represented by a histogram of multiple irradiation intervals and multiple fissile Isotopes or can be modeled as a single interval and a single fissile isotope, U-235. Other items investigated were the impact of values of the recoverable energy per fission for U-235, Pu-231, U-231 and Pu-241, an assessment of the impact of a correction factor G(t) for neutron capture in fission products, the actinide contribution to decay heat power, and the effect of various uncertainty methods and parameters. He then provided plots of the impact of these variables on decay heat.
He then summarized a bottom-line conclusion that risk-informed changes to 10CFR50.46 and Appendix-K should allow the use of the 1994 ANS decay standard.

Prof. Schrock questioned Mr. Lauben on weather or not Appendix-K requires the use of the ORIGEN code to calculate decay heats, including the contribution from actinide decay. Mr. Lauben responded that Appendix-K only requires that actinide decay be accounted for but does not specify how that is to be done. Dr. Wallis also asked if Appendix-K recommends a procedure for calculation or does it just say that certain things must be accounted for. Mr. Lauben responded that in some cases it's quite explicit as how to calculate something in other cases it just says that things like actinide decay must be provided for and justified by the user, but does not specify how. At this point Chairman Shack interjected that because of time constraints the committee needed to move on to the last two presentations on the agenda, that is by Dr. Bajorek and Mr. Sam Lee of NRR.

Mr. Lauben turned the presentation back to Dr. Bajorek, who provided the subcommittee with the rational behind the RES recommendation to change the metal-water reaction rate and heat from the current Baker-Just correlation to Cathcart-Pawel. He presented the committee with plots of oxide-thickness versus time at temperature measurements with both the Cathcart - Pawel and Baker-Just correlations, indicating batter comparison of the data with Cathcart - Pawel, particularly at lower pressures. He also discussed the justification for deleting Appendix-K requirements for steam cooling at reflood rates below 1 in/sec, and retention of the requirement to prohibit return to nucleate boiling assumptions in LOCA analysis. He closed his presentation with a summary overhead of the impact of various modeling assumptions on predicted peak cladding temperatures (PCT), showing both increases and decreases in PCT depending on models. Mr. Rosen questioned why a licensee can't take credit for steam cooling on reflood at 1 in/sec. Dr. Bajorek responded that tests show that you get droplets entrained in steam at low velocities which complicates the modeling of steam cooling. Mr. Rosen questioned if a licensee had some problems with his ECCS analysis would he be allowed under the new Appendix-K to perform some tests to justify some cooling at low reflood velocities. Dr. Bajorek said this would have to be considered on an individual basis. Dr. Bonaca noted that on the summary overhead showing the impact of various modeling assumptions on predicted peak cladding temperatures (PCT), that the present models allowed in Appendix-K might not produce necessarily conservative results to which Mr. Rosenthal and Bajorek of RES responded that the effects are non-additive. At this point Mr. Caruso of NRR interjected that there is a certain school of thought that use of the revised Appendix-K and 10CFR50.46 might be quite limited. Chairman Shack then noted that it was close to 5 p..m. And that the subcommittee still needed to hear from Mr. Sam Lee of NRR on thoughts on proposed rule-making.

Mr. Lee began his presentation by noting that the table he projected on the screen showed the proposed changes recommended by RES as previously discussed and the second column showing industry interest in such changes. He stated that a working group would be formed in the near future to evaluate and assess the suggested changes. He noted that there would be a separate rule making associated with each of the proposed changes, at which point Mr. Rosen asked how they would be linked together. Mr. Lee responded that although there would be separate rule making activities, they indeed would be linked by way of an oversight group that would make sure of such linkage. After some general comments by several ACRS members thanking the presenters on an informative and interesting subcommittee, Mr. Sieber adjourned the meeting.

SUMMARY of SUBCOMMITTEE COMMENTS, CONCERNS, AND RECOMMENDATIONS

Subcommittee members raised the following significant points during its discussion with NRC staff and industry representatives:

- Mr. Leitch asked if the proposed changes to 10CFR50.46 and Appendix-K would allow for no diesel backup if licensee could demonstrate highly reliable off-site power. Mr. Kuritzky responded that this issue has not been decided at this point.

- Mr. Rosen asked if there would be changes in performance monitoring and corrective actions regarding the proposed changes to 10CFR50.46 and Appendix-K. Mr. Kuritzky responded no.

- Dr. Kress questioned if there was an absolute delta-CDF that the staff was proposing and Mr. Rosen asked if a full-scope PRA was would be required, with internal and external events, as well as seismic. Mr. Kuritzky responded that a full-scope PRA would indeed be required if a licensee wanted to change its licensing basis to risk-informed ECCS performance.

- Members Apostolakis, Rosen, and Kress, questioned Mr. Kuritzky as to more specific CDF and LERF acceptance guidelines. Mr. Glen Kelly of NRR interjected that the presentation by Mr. Kuritzky was a description of the technical basis for proposed changes to 10CFR50.46 and associated guidance, but that these proposals need to be reviewed by an agency working group to go forward with rule-making and that the Commission has accepted Reg. Guide 1.174 as the process for risk-informing such decisions.

- Dr. Ransom, asked if there would be any added benefit to the overall public health and safety by risk-informing 10CFR50.46 requirements regarding ECCS, to which Mr. Kuritzky responded that quantification of the risk may have benefit from public confidence.

- Prof. Schrock questioned Mr. Lauben on weather or not Appendix-K requires the use of the ORIGEN code to calculate decay heats, including the contribution from actinide decay. Mr. Lauben responded that Appendix-K only requires that actinide decay be accounted for but does not specify how that is to be done.

- Dr. Wallis asked if Appendix-K recommends a procedure for calculation or does it just say that certain things must be accounted for. Mr. Lauben responded that in some cases it's quite explicit as how to calculate something in other cases it just says that things like actinide decay must be provided for and justified by the user, but does not specify how.

- Dr. Bonaca noted that on the summary overhead showing the impact of various modeling assumptions on predicted peak cladding temperatures (PCT), that the present models allowed in Appendix-K might not produce necessarily conservative results. Mr. Rosenthal and Dr. Bajorek responded that the effects are non-additive.

- Mr. Rosen noted that if the agency was going to make a separate rule for each of the proposed changes to 10CFR50.46 and Appendix-K, then how would such changes be linked together. Mr. Lee responded that although there would be separate rule making activities, they indeed would be linked by way of an oversight group that would make sure of such linkage.

SUBCOMMITTEE DECISIONS

None.

FOLLOW-UP ACTIONS

A follow-up briefing is anticipated sometime during the fall of 2002. A report to the Commission is anticipated at the time of public comment on rule-making related to changes to 10CFD50.46, Appendix-K requirements, and General Design Criteria-35.

CONSULTANT REPORTS (Attached)

Prof. Virgil E. Schrock (University of California-Berkeley), report dated June 5, 2002.
Prof. Sanjoy Banerjee (University of California-Berkeley), report dated May 31, 2002.

BACKGROUND MATERIALS PROVIDED TO THE SUBCOMMITTEE PRIOR TO THIS MEETING

1. Subcommittee agenda.
2. Subcommittee status report.
3. SECY-02-0057: "Update to SECY-01-0133: Fourth Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10CFR50 (Option-3) and Recommendations on Risk-Informed Changes to 10CFR50.46 (ECCS Acceptance Criteria)"
4. INTERNAL MEMO from S. Newberry (RES) to D. Matthews (NRR): "Transmittal of Technical Work to Support Rule-making for Risk-Informed Alternative to 10CFR50.46/GDC 35" (PRE-DECISIONAL).
5. DRAFT REPORT by G. M. Wilkowski et al, "Technical Evaluation of Probabilistic LBB Codes and Approaches", (Nov. 30, 2001).
6. VIEW-GRAPHS: "Re-evaluation of LOCA Frequency Distributions/Overview" , (PRE-DECISIONAL)
7. VIEW-GRAPHS & 1-PAGE QUESTIONNAIRE: "Elicitation of Results and Updated LOCA Frequency Distributions/Questionnaire" (PRE-DECISIONAL)
8. INTERNAL MEMO from F. Eltawila (RES) to J. T. Larkins (ACRS), May 23, 2002, "Supporting Documents for the Joint Meeting of the ACRS Subcommittee on Materials and Metallurgy, Thermal-Hydraulic Phenomena, and Reliability and PRA, May 31, 2002"

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Note: Additional details of this meeting can be obtained from a transcript of this meeting available for downloading or viewing on the Internet at "http://www.nrc.gov/ACRSACNW" or can be purchased from Neal R. Gross and Co., Inc., (Court Reporters and Transcribers) 1323 Rhode Island Avenue, N.W., Washington, DC 20005 (202) 234-4433.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
JOINT MEETING OF THE ACRS SUBCOMMITTEES ON
MATERIALS AND METALLURGY, THERMAL-HYDRAULIC
PHENOMENA, AND RELIABILITY & PROBABILISTIC RISK ASSESSMENT
MAY 31, 2002
ROCKVILLE, MARYLAND

- PROPOSED AGENDA -

		TOPIC	SPEAKER	TIME
I	Introduction		ACRS	8:30 a.m.
8.	NRC Staff Presentations			
	1.	Introductory Remarks	M. Cunningham, RES	8:40 a.m.
	2.	ECCS Reliability Requirements	M. Drouin/A. Kuritzky, RES	8:50 a.m.
	3.	ECCS LOCA Size Definition	R. Tregoning, RES	10:50 a.m.
			LUNCH	12:30 p.m. - 1:30 p.m.
	4.	ECCS Acceptance Criteria and Evaluation Model Requirements	R. Meyer/S. Bajorek, RES	1:30 p.m.
			BREAK	3:00 p.m. - 3:15 p.m.
	4.	ECCS Acceptance Criteria and Evaluation Model Requirements (Cont'd)		3:15 p.m.
	5)	Rulemaking Activities	S. Lee, NRR	4:15 p.m.
III	Meeting Wrap-up		ACRS	4:50 p.m.
IV	Adjournment			5:00 p.m.

Note: Presentation time should not exceed 50% of the total time allocated for a specific item.
Number of copies of presentation materials to be provided to the ACRS/ACNW - 35.

corrective actions that included re-instructing and re-training its employees; and the Licensee has had no prior violations of NRC regulations.

The Licensee also argues that none of the rationales set forth in the enforcement policy for issuing a penalty are applicable in this case. Specifically, the Licensee indicates that the penalty will not encourage prompt identification and prompt corrective action because the Licensee had already identified and corrected the violations. The Licensee also states that the penalty will not deter future violations because the theft of the radioactive device was the result of a criminal act by a third party. Finally, the Licensee maintains that the penalty will not focus the Licensee's attention on significant violations because the Licensee believes that the violation was insignificant.

3. NRC Evaluation of Licensee's Request for Withdrawal of the Civil Penalty

Notwithstanding the Licensee's contentions regarding the significance of the violation, the NRC maintains that the violation was appropriately classified at Severity Level III, consistent with the NRC enforcement policy. Since the gauge contained less than 1000 times the quantity of cesium-137 set forth in 10 CFR Part 20, Appendix C (the gauge contained approximately 800 times that quantity), the failure to secure the gauge and maintain surveillance over it might have been classified at Severity Level IV, in accordance with Section C.11 of Supplement IV of the enforcement policy, had the gauge not been stolen. However, since the failure to secure or maintain constant surveillance over the gauge, resulted in the gauge being stolen and radioactive material entering the public domain and being handled by members of the public, the violation is more appropriately classified at Severity Level III. Such violations are considered significant since, although the source is normally shielded within the gauge, significant radiation exposures could occur if the source becomes unshielded while in the public domain.

The NRC agrees that the gauge was properly labeled, the Licensee took appropriate actions once it discovered that the gauge was missing, the violation was not willful, and the Licensee's prior enforcement history has been good. As a result, consistent with the NRC enforcement policy, a civil penalty would not normally be warranted for a Severity Level III violation, as the NRC indicated in its February 27, 2002 letter transmitting the civil penalty. However, although the outcome of the normal civil penalty process in this case would not result in a civil penalty, a civil penalty is warranted, in accordance with Section VII.A.1.g of the enforcement policy since the case involved a loss/improper disposal of a sealed source. The Commission included Section VII.A.1.g. in the policy since it believes that normally issuance of a civil penalty is appropriate for cases involving of loss of a sealed source or device. This is necessary to properly reflect the significance of such violations.

Although the loss of the gauge was due to the criminal act of a third party, the Licensee is responsible for that occurrence since the gauge user left the gauge unattended and unsecured, which directly contributed to the theft. Accordingly, issuance of the violation, categorization of the violation at Severity Level III, and imposition of the related civil penalty, is appropriate in this case, and consistent with the NRC enforcement policy.

4. NRC Conclusion

The NRC has concluded that the Licensee did not provide an adequate basis for withdrawal of the civil penalty. Accordingly, the proposed civil penalty in the amount of $3,000 should be imposed.

[FR Doc. 02–11872 Filed 5–10–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Joint Meeting of the ACRS Subcommittees on Materials and Metallurgy, on Thermal-Hydraulic Phenomena, and on Reliability and Probabilistic Risk Assessment; Notice of Meeting

The ACRS Subcommittees on Materials and Metallurgy, on Thermal-Hydraulic Phenomena, and on Reliability and Probabilistic Risk Assessment will hold a joint meeting on May 31, 2002, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows: *Friday, May 31, 2002—8:30 a.m. until the conclusion of business.*

The Subcommittees will continue their review of the proposed risk-informed revisions to the technical requirements of the Emergency Core Cooling Systems Rule (10 CFR 50.46 and Appendix K). The purpose of this meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittees, their consultants, and staff. Persons desiring to make oral statements should notify the Designated Federal Official named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittees, along with any of their consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittees will then hear presentations by and hold discussions with representatives of the NRC staff, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor, can be obtained by contacting the Designated Federal Official, Mr. Paul A. Boehnert (telephone 301–415–8065) between 7:30 a.m. and 5 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda that may have occurred.

Dated: May 7, 2002.

Sher Bahadur,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 02–11870 Filed 5–10–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Draft Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission has issued for public comment a proposed revision of a guide in its Regulatory Guide Series. Regulatory Guides are developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits and licenses.

The draft guide is temporarily identified by its task number, DG–1118, which should be mentioned in all correspondence concerning this draft guide. Draft Regulatory Guide DG–1118, the Proposed Revision 1 of Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Safety Systems," is being developed to describe a method acceptable to the NRC staff for complying with the NRC's regulations with respect to satisfying the single-failure criterion for safety systems.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS SUBCOMMITTEE MEETINGS ON MATERIALS AND METALLURGY, THERMAL-HYDRAULIC PHENOMENA, AND RELIABILITY AND PROBABILISTIC RISK ASSESSMENT

MAY 31 2002
Date

PLEASE PRINT

ATTENDEES PLEASE SIGN-IN FOR THE MEETING

NAME	NRC ORGANIZATION
Alan Kuritzky	RES/DRAA/PRAB
BOB TREGONING	RES/DET/MEB
Amarjit Singh	RES/DRAA/PRAB
WILLIAM BECKNER	NRR/RORP
Syed A. Ali	RES/DET/ERAB
Mary Drouin	RES/DRAA/PRAB
DAVID DIEC	NRR/RPRP
Glenn Kelly	NRR/DSSA/SPSB
J. S. Hyslop	~~NRR~~ RES/PRAB
JIM LAZEVNICK	NRR/DE/EEIB
Jack Rosenthal	RES/SMSAB
Stephen Bajorek	RES/SMSAB
Anne Passarelli	NRR/DSSA
RALPH CARUSO	NRR/SRXB
Michael Johnson	NRR/SPSB
Samuel S. Lee	NRR/DRIP/RPRP
Nilesh Chokshi	RES/DET/MEB
UNDINE SHOOP	NRR/DSSA/SRXB
Debbie Jackson	RES/DET/MEB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS SUBCOMMITTEE MEETINGS ON MATERIALS AND METALLURGY, THERMAL-HYDRAULIC PHENOMENA, AND RELIABILITY AND PROBABILISTIC RISK ASSESSMENT

MAY 31 2002
Date

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ATTENDEES PLEASE SIGN-IN FOR THE MEETING

NAME	NRC ORGANIZATION
FAROUK ELTAWILA	RES/DSARE
Steve West	NRR
RALPH LANDRY	NRR/DSSA/SRXB
Mark Rubin	NRR
CHRIS GRIMES	NRR
JOHN CLARK	RES
Mark Kowal	NRR
FRANK ORR	NRR/SRXB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS SUBCOMMITTEE MEETINGS ON MATERIALS AND METALLURGY, THERMAL-HYDRAULIC PHENOMENA, AND RELIABILITY AND PROBABILISTIC RISK ASSESSMENT

MAY 31 2002
Date

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ATTENDEES PLEASE SIGN-IN FOR THE MEETING

NAME	AFFILIATION
William Slagle	Westinghouse Electric
Robert Osterrieder	Westinghouse Electric
Ellen Anderson	NEI
Jerry Holm	Framatome
Nancy Chapman	SERCH/Bechtel
Jeff LaChance	Sandia National Labs
Charles Brinkman	Westinghouse

RISK-INFORMING 10 CFR 50.46

Presented to:
ACRS Subcommittees on Materials and Metallurgy, Thermal-Hydraulic Phenomena, and Reliability and Probabilistic Risk Assessement

Presented by:
Mark Cunningham, Mary Drouin, Alan Kuritzky, RobTregoning, Lee Abramson, Steve Bajorek, Norm Lauben and Sam Lee
U.S. Nuclear Regulatory Commission

May 31, 2002

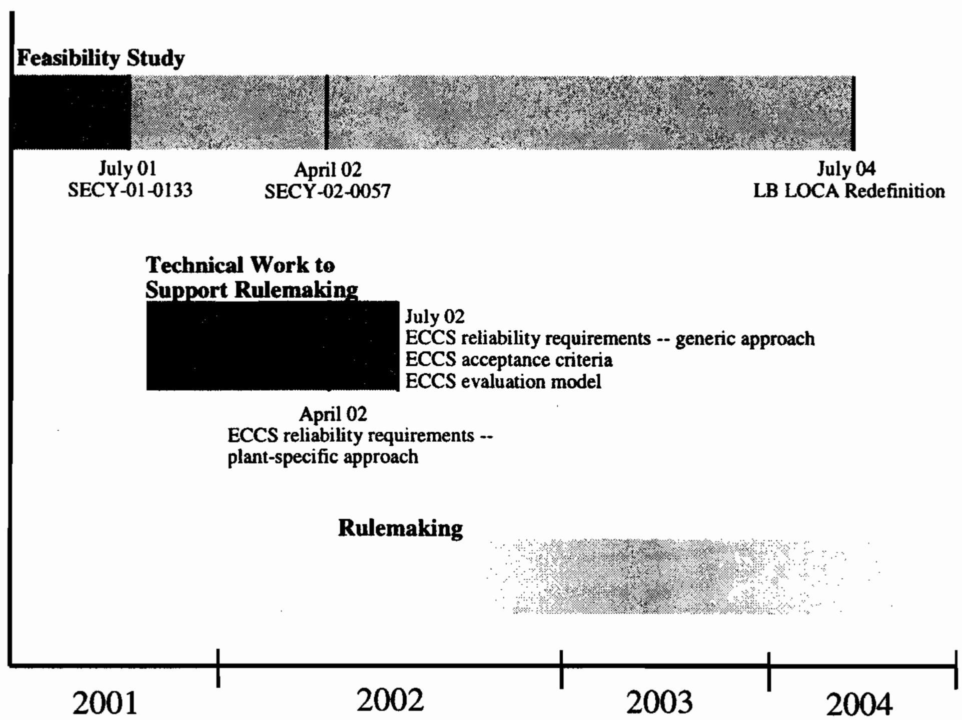
OUTLINE

- Purpose/goal of meeting
- Status and schedule
- Proposed changes to 10 CFR 50.46 (including Appendix K and GDC 35)
- Technical work to support rulemaking for changes to 10 CFR 50.46

PURPOSE/GOAL OF MEETING

- Provide status on staff's efforts to risk-inform 10 CFR 50.46
- Solicit feedback and comments from ACRS
- No letter requested (at this time)

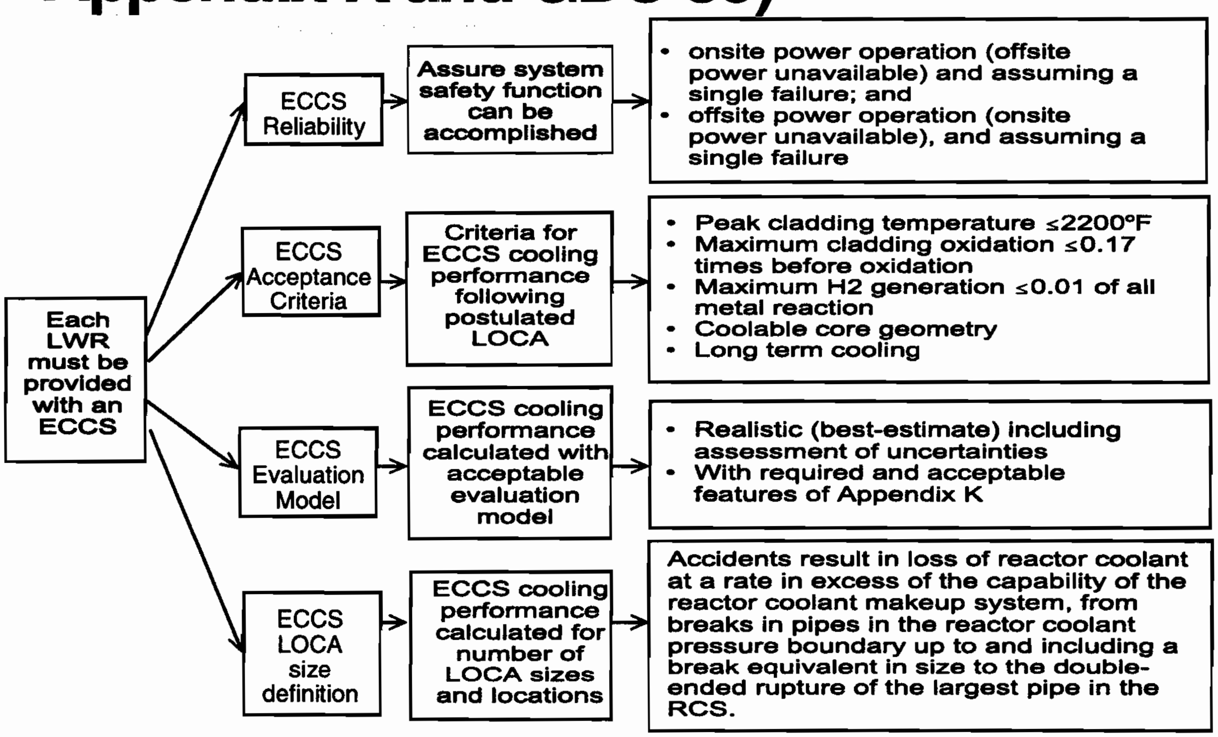
STATUS



JULY 2002 DELIVERABLE

- Memo from A. Thadani (RES) to S. Collins (NRR) will document technical work to support rulemaking for changes to:
  - ECCS reliability requirements (GDC 35)
  - ECCS acceptance criteria
  - ECCS evaluation model requirements

OVERVIEW OF 50.46 (including Appendix K and GDC 35)



PROPOSED CHANGES TO 50.46, APP. K AND GDC 35 (SECY-02-0057)

Staff recommendations on:

- ECCS reliability: Provide two voluntary performance-based options (one generic, one plant-specific) that would demonstrate reliable ECCS safety function without assuming LOOP and single additional failure in GDC 35
- ECCS acceptance criteria: Change current prescriptive ECCS acceptance criteria in 50.46 to add a performance-based option

PROPOSED CHANGES TO 50.46, APP. K AND GDC 35 (SECY-02-0057) (Cont'd)

Staff recommendations on:

- ECCS evaluation model: Add an option to App. K decay heat requirement to permit use of 1994 ANS standard
  - Staff intends to address subject of uncertainty and conservatism in App. K models separate from rulemaking activity
- ECCS spectrum of break sizes and locations: Continue the feasibility study of redefining the maximum pipe break size required to be considered as part of the ECCS performance evaluation

TECHNICAL WORK TO SUPPORT RULEMAKING FOR CHANGES TO 10 CFR 50.46

- ECCS reliability requirements
- ECCS spectrum of break sizes and locations
- ECCS acceptance criteria
- ECCS evaluation model requirements

TECHNICAL WORK TO SUPPORT RULEMAKING FOR CHANGES TO 10 CFR 50.46

- ***ECCS reliability requirements***
- ECCS spectrum of break sizes and locations
- ECCS acceptance criteria
- ECCS evaluation model requirements

PROPOSED CHANGES TO ECCS RELIABILITY REQUIREMENTS

Risk-Informed Alternative to GDC 35

- As part of proposed rulemaking, current approach of GDC 35 would be changed
- Revised approach would permit ECCS to be designed, operated or evaluated based on quantitative reliability considerations instead of prescriptive assumptions on loss of offsite power (LOOP) and additional single failure

PROPOSED CHANGES TO ECCS RELIABILITY REQUIREMENTS

Scope and Limitations

- Proposed changes apply only to ECCS requirements (GDC 35)
  - No changes proposed to requirements for containment design or equipment qualification
  - Changes to single failure criterion not generically extended to other systems
  - E.g., no changes to GDCs 17, 34, 38, 41 and 44
- Performance monitoring and corrective action strategies may need to be developed for specific applications

PROPOSED CHANGES TO ECCS RELIABILITY REQUIREMENTS

Approaches for Risk-Informed Alternative to GDC 35

- In place of loss of offsite power and additional single failure assumptions in current GDC 35, two options would be offered in a Regulatory Guide to ensure ECCS safety function reliability:

  1. Plant-specific approach where licensees, with appropriate consideration of uncertainties, demonstrate compliance with NRC-established acceptance guidelines, ***OR***

  2. Generic approach where a minimal set of ECCS equipment required to meet NRC-established acceptance guidelines would be specified by the NRC, by generic plant group.

- Approaches based on Option 3 framework

PLANT-SPECIFIC APPROACH FOR RISK-INFORMED ALTERNATIVE TO GDC 35

- Technical work included:
  - Determining proposed CDF and LERF acceptance guidelines
  - Determining acceptable LOCA frequencies (ongoing)
  - Developing possible method for plant-specific calculation of conditional probability of LOOP given LOCA

PROPOSED CDF AND LERF ACCEPTANCE GUIDELINES

- Two types of licensee-proposed ECCS-related changes envisioned:
  - Changes in ECCS design or operation (e.g., removal of a piece of equipment or relaxation of technical specifications)
  - Changes in the ECCS design basis (e.g., removal of an accident from the ECCS design basis analyses)

PROPOSED CDF AND LERF ACCEPTANCE GUIDELINES

Design/Operational Changes

- Need to demonstrate that ECCS functional reliability is commensurate with frequency of accidents for which ECCS success would prevent core damage or large early release
- Can be accomplished by demonstrating that the following acceptance guidelines are met:

(1) Baseline total plant CDF and LERF meet quantitative guidelines specified in Option 3 framework, AND
(2) Resulting change in risk from proposed ECCS-related change does not represent a significant risk increase

PROPOSED CDF AND LERF ACCEPTANCE GUIDELINES

Design/Operational Changes (Cont'd)

- Option 3 framework specifies CDF and LERF thresholds of 1E-4/yr and 1E-5/yr, respectively
  - Since values apply to full-scope PRA, total plant CDF and LERF need to be determined or addressed
  - Thresholds are flexible, consistent with RG 1.174
- RG 1.174 acceptance criteria used to limit change in risk, since Option 3 framework only specifies absolute risk guidelines
- Consistent with Option 3 framework, quantitative guidelines are only one part of risk-informed defense-in-depth approach
  - Defense-in-depth principles cannot be violated

PROPOSED CDF AND LERF ACCEPTANCE GUIDELINES

Design Basis Changes

- Proposed change must meet Option 3 framework and RG 1.174 criteria, same as for other types of changes
- Change in CDF and LERF are determined by assuming plant can no longer respond to the subject accident (i.e., subject accident assumed to lead directly to core damage)

PROPOSED CDF AND LERF ACCEPTANCE GUIDELINES

PRA Scope and Uncertainty Analysis

- Acceptance guidelines are intended for comparison with results of full-scope PRA
- Significance of out-of-scope items needs to be addressed
- Consistent with RG 1.174, mean values should be used to compare with the acceptance guidelines
- Formal propagation of uncertainties should be performed, where possible
  - Supplement with sensitivity studies or qualitative arguments, where necessary

LOCA FREQUENCY ESTIMATION

- Need to consider LOCA initiating events and transient-induced (or consequential) LOCAs
- LOCA initiating events include pipe-break LOCAs and non-pipe-break LOCAs (e.g., SG manway failure)
- Causes and frequencies of transient-induced LOCAs and very small LOCA initiating events are relatively well understood
- Causes and frequencies of medium and large LOCA initiating events (>~2 in.) are not as well understood

LOCA FREQUENCY ESTIMATION (Cont'd)

- Sources of medium/large LOCA frequencies in PRAs
  - WASH-1400/NUREG-1150
    - ❒ Based on old data, most not applicable to nuclear power plants
  - NUREG/CR-5750
    - ❒ Based on recent operating experience, some technical issues raised
- Several concurrent studies to evaluate LOCA distributions
  - Short-term: quick, in-house elicitation
  - Intermediate-term: formal expert elicitation
  - Longer term: redefine spectrum of pipe break sizes

CONDITIONAL PROBABILITY OF LOOP GIVEN LOCA

- In typical PRAs, occurrence of LOOP following a LOCA is assumed to be random, independent event
- More recent analysis (NUREG/CR-6538) concludes that a dependency exists
- Extremely limited data for consequential LOOP following a LOCA or major ECCS actuation (surrogate for LOCA)

CONDITIONAL PROBABILITY OF LOOP GIVEN LOCA (Cont'd)

- Plant-specific method for assessing conditional probability of LOOP given a LOCA provided in RES report (App. D)
- Continuing to work with industry on alternative approaches for quantifying conditional probability of LOOP given a LOCA
  - Industry expert elicitation
  - Staff review

GENERIC APPROACH FOR RISK-INFORMED ALTERNATIVE TO GDC 35

- Technical work includes:
  - Formulating plant groups
  - Performing reliability/risk calculations
    - PRA scope and quality issues
- List of minimum required ECCS equipment and need to consider LOCA-LOOP would likely appear in regulatory guide
- Plant equipment in excess of the minimum determined above, would be candidates for design or operational changes

TECHNICAL WORK TO SUPPORT RULEMAKING FOR CHANGES TO 10 CFR 50.46

- ECCS reliability requirements
- ***ECCS spectrum of break sizes and locations***
- ECCS acceptance criteria
- ECCS evaluation model requirements

LOCA FREQUENCY AND LB LOCA BREAK SIZE REDEFINITION

Presented to the ACRS Subcommittees on Materials & Metallurgy, Thermal-Hydraulic Phenomena, and Reliability & Probabilistic Risk Assessment

Presented by
Rob Tregoning, Lee Abramson
U.S. Nuclear Regulatory Commission

May 31, 2002

LOCA REEVALUATION: PROGRESS SINCE PREVIOUS ACRS BRIEFINGS

- Previous ACRS briefings
  - March, 2001: Last substantive briefing on LOCA technical issues which initiated reevaluation.
  - June, July, November, 2001: Only overviews of LOCA effort provided to outline its importance within 10 CFR 50.46 revision framework.
- Progress Since March 2001
  - Developed technical position paper documenting issues to address for LOCA reevaluation.
  - Formulated approach for realizing near-term and long-term goals outlined within SECY-01-0133 (later SECY-02-0057).
  - Completed near-term elicitation to develop interim LOCA frequencies.
  - Public interaction with stakeholders: August 2001, October 2001, and March 2002.

LOCA REEVALUATION: EXECUTIVE SUMMARY

- Historical LOCA estimates have been based on service history experience.
- There are several potential LOCA initiating failure events which were not part of the service history based estimates (e.g. VC Summer, Oconee, and Davis Besse).
- MEB has initiated several concurrent studies to evaluate LOCA frequencies.
  - Near-Term (Complete) : developed interim LOCA frequency distributions by staff expert panel. Results were 2 to 4 times higher than NUREG/CR-5750 estimates.
  - Intermediate-Term (within one year): develop final LOCA frequency distributions through formal elicitation process using a panel of academic, industry, and government experts.
  - Longer-Term (2 years): redefine the spectrum of pipe break sizes to consider ECCS capability changes within existing RI-ISI framework.

LOCA REEVALUATION: MOTIVATION

- NRC is investigating risk informed changes to the following ECCS areas within 10 CFR 50.46:
  - ECCS Reliability.
  - ECCS acceptance criteria.
  - ECCS evaluation model.
  - ECCS spectrum of break sizes and locations.
- **LOCA frequency distribution impacts ECCS reliability (near-term effort) and the ECCS spectrum of break sizes and locations (longer-term effort).**

LOCA REEVALUATION: OVERVIEW

Several concurrent studies initiated to evaluate LOCA frequencies.

- **Near-term elicitation (by April 30, 2002): support ECCS reliability revision and initiation of rulemaking (SECY-02-0057).**
- Intermediate-term elicitation (within one year): support final rulemaking decisions.
- Longer-term (by June 2004): redefine the spectrum of break pipe sizes.

LOCA REEVALUATION: NEAR-TERM ELICITATION

- Eleven staff (5 NRR, 6 RES) chosen to obtain broad expertise in relevant technical areas: probabilistic risk assessment; the ASME code; structural mechanics; thermo-hydraulics; piping systems; seismic, thermal and vibrational loading; environmentally assisted cracking; thermal aging; and alternative LOCA mechanisms
- Objectives:
  - Adjust NUREG/CR-5750, Appendix J frequency distributions to account for other LOCA contributions not considered in original study.
  - Prioritize issues and questions which potentially provide the greatest additional contributions to LOCA frequency estimates. These issues will be considered during the formal elicitation process.

NEAR-TERM ELICITATION: APPROACH

- Kick-off meeting.
  - Provide background for NUREG/CR-5750, Appendix J LOCA estimates.
  - Present technical concerns and motivation for updating frequencies.
- Issue development meeting.
  - Develop definitions and baseline case.
  - Identify important initiating mechanisms, systems, and components.
  - Identify important factors affecting future LOCA frequencies.
- Elicitation questionnaire.
  - Decompose technical issues.
  - Evaluate expected changes up through license renewal.
  - Obtain rationale for quantitative responses.
- Wrap-up meeting.
  - Present results and summarize important findings.
  - Obtain feedback for intermediate-term elicitation.

NEAR-TERM ELICITATION: ISSUE DEVELOPMENT

BWR LOCA Contributing Systems

ID	LOCA Initiating System	Materials	Failure Mechanisms	(S, M, L) LOCA Contributions
B1	Jet Pump Risers	IN 182/600	IGSCC	S, M, L
		SS Weld	IGSCC	
		Wrought SS	IGSCC	
B2	Recirculation Loops	IN 182/600	IGSCC	S, M, L
		SS Weld	IGSCC	
		Wrought SS	IGSCC	
B3	Core Spray	IN 182/600	IGSCC	S, M, L
		SS Weld	IGSCC	
		Wrought SS	IGSCC	
B4	RHR/LPI	SS Weld	IGSCC, THFAT	S, M, L
		Wrought SS	IGSCC, THFAT	
B5	Feedwater	Carbon Steel	THFAT, FAC	S, M, L
B6	Drain Lines	SS Weld	IGSCC, THFAT, MEFAT	S, M
		Wrought SS	IGSCC, THFAT, MEFAT	
		Carbon Steel	THFAT, MEFAT	
B7	RWCU	IN 182/600	IGSCC, THFAT	S, M
		SS Weld	IGSCC, THFAT	
		Wrought SS	IGSCC, THFAT	
		Carbon Steel	THFAT, FAC	
B8	Instrument Lines	SS Weld	IGSCC, MEFAT, TGSCC	S
		Wrought SS	IGSCC, MEFAT, TGSCC	
B9	SRV		Stuck Open Relief Valves	S, M
B10	External Events		Failure caused by human error (bumping) instrument lines	S
B11	ISLOCA		Failure of Class 1 interfacing system	S, M, L

- Bin piping systems by functionality, material, potential degradation mechanisms, loading history, and transient similitude.
- Discus LOCA potential of other (non piping) components.
- Examine global issues which influence all systems equally (RI-ISI, leak detection threshold, future degradation mechanisms, and mitigation).

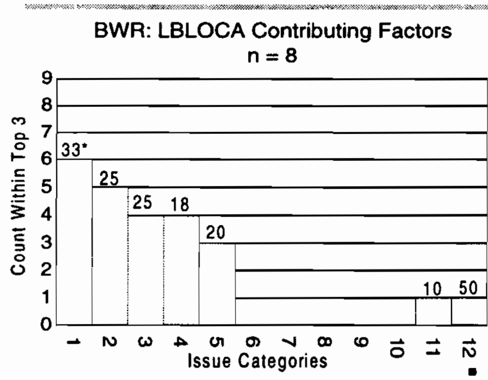
NEAR-TERM ELICITATION: QUESTIONNAIRE

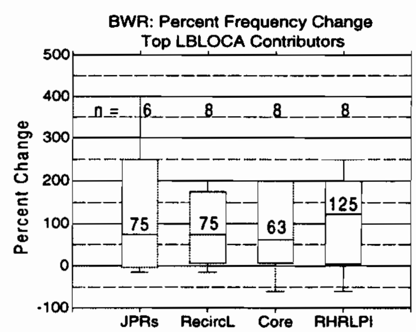
BWR LOCA "Relative Change" Table

ID	System or Components	SBLOCA (%)	MBLOCA (%)	LBLOCA (%)	Rationale and Comments
B1	Jet Pump Risers				
B2	Recirculation Loops				
B3	Core Spray				
B4	RHR/LPI				
B5	Feedwater				
B6	Drain Lines			NA	
B7	RWCU			NA	
B8	Instrument Lines		NA	NA	
B9	SRV			NA	
B10	External Events		NA	NA	
B11	ISLOCA				

- Each panel member completed an individual questionnaire.
- Evaluated relative changes in frequencies over next 35 years.
- Separately considered SB, MB, and LB LOCA changes.
- Utilized quantitative responses and rationale to determine most important LOCA contributors and LOCA frequencies.
- Combined responses in several ways (absolute changes, ratios, global changes) to conduct sensitivity analysis.

NEAR-TERM ELICITATION RESULTS: BWR LB LOCA CONTRIBUTORS



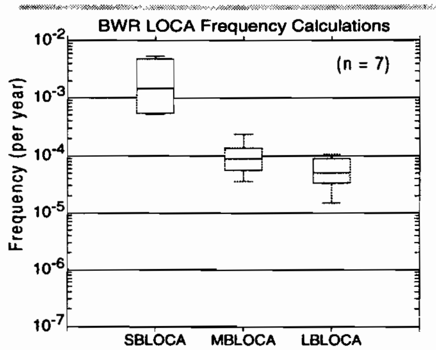


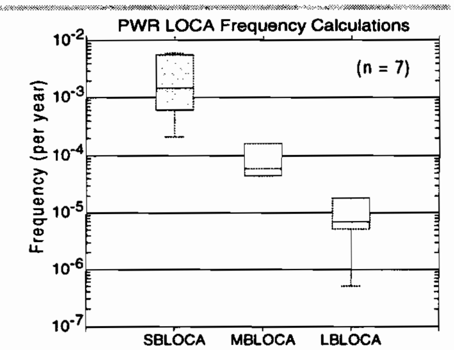
B1: Jet Pump Risers
B2: Recirculation Loops
B3: Core Spray
B4: RHR/LPI
B5: Feedwater
B6: Drain Lines (NA)
B7: RWCU (NA)
B8: Instrument Lines (NA)
B9: SRV (NA)
B10: External Events (NA)
B11: ISLOCA
B12: Stub Tubes

*median of non-zero respondents

- Large class-1 diameter piping failures expected to dominate LB LOCA freqs.
- Frequency increases expected to be relatively independent of system.
- More variability than MB LOCA changes; similar to SB LOCA.

NEAR-TERM ELICITATION RESULTS: INTERIM LOCA FREQUENCIES





- Larger variability in SBLOCA numbers driven by the non-piping initiating components.
- Variability among estimates is generally less than an order of magnitude.
- Initial 5750 differences between the BWR and PWR MB and LB LOCA frequencies are retained.

NEAR-TERM ELICITATION RESULTS: INTERIM LOCA FREQUENCIES

Comparison of Mean Results with Previous Studies (per Reactor Calendar Years)				
Reactor Type	**Analysis**	**SBLOCA**	**MBLOCA**	**LBLOCA**
BWR	**Current**	**15×10^{-4}**	**9×10^{-5}**	**5×10^{-5}**
	NUREG-5750	4×10^{-4}	3×10^{-5}	2×10^{-5}
	WASH-1400	30×10^{-4}	80×10^{-5}	30×10^{-5}
PWR	**Current**	**15×10^{-4}**	**6×10^{-5}**	**7×10^{-6}**
	NUREG-5750	4×10^{-4}	3×10^{-5}	4×10^{-6}
	WASH-1400	30×10^{-4}	80×10^{-5}	30×10^{-5}
Comparative Increase in 5750 Results				
BWR		3.7	3.0	2.6
PWR		3.7	2.0	1.7

- Interim results fall between NUREG/CR-5750 and WASH-1400 estimates.
- MB and LB LOCA frequencies are closer to the NUREG/CR-5750 estimates.

NEAR-TERM ELICITATION: CONCLUSIONS

- LB LOCA frequency < MB LOCA frequency < SB LOCA frequency.
- Dominant initiators are apparent for SB and LB LOCA frequencies for both BWR and PWR systems.
- The effect of other (non-piping) component failure is important for SB LOCAs, and to a lesser extent for MB LOCAs.
- The effects of the global issues explicitly considered was not significant in terms of the median update. However, there was substantial difference of opinion about the role of future mechanisms & mitigation, ISI, and hydrogen combustion.

NEAR-TERM ELICITATION: CONCLUSIONS, cont.

- The SB LOCA 5750 frequencies are expected to change to the greatest extent. This is a direct reflection of the addition of the failure of non-piping components.
- Failure of piping components is expected to increase in the future to a greater extent than non-piping components.
- Aging mechanisms are expected to substantially affect the LOCA frequencies in the future.
- Failure without a precursor event is a significant consideration.

LOCA REEVALUATION: OVERVIEW

Several concurrent studies initiated to evaluate LOCA frequencies.

- Near-term elicitation (by April 30, 2002): support ECCS reliability revision and initiation of rulemaking (SECY-02-0057).
- **Intermediate-term elicitation (within one year): support final rulemaking decisions.**
- Longer-term (by June 2004): redefine the spectrum of break pipe sizes.

LOCA REEVALUATION: INTERMEDIATE-TERM ELICITATION

Process

- Designed and implemented by NRC elicitation team with contractual support provided by Battelle and Emc^2.
- Panel to be solicited from non-NRC participants from industry, academia, contracting agencies, other government agencies, and international agencies.
- Panel members to represent the full range of relevant technical specialties.
- The elicitation process utilized in the flaw distribution determination for 50.61 (PTS) reevaluation will be used as a model.

LOCA REEVALUATION: INTERMEDIATE-TERM ELICITATION

Service History Baseline

- The SKI-pipe database will serve as pipe break baseline.
- This database will be updated through the CSNI-sponsored OPDE project.
- PRA estimates for other LOCA initiating failures and components (e.g. SRV/PORV, pump seal, ISLOCA, steam generator tube) will be combined.
- Relevant information from other industries (e.g. commercial fossil plants, petrochemical plants, oil and gas transmission) can be utilized to provide bounding estimates.
- Elicitation will be utilized to determine if any modifications to the service history baseline are required.

LOCA REEVALUATION: INTERMEDIATE-TERM ELICITATION

Updated LOCA Frequency Development

- Probabilistic fracture mechanics modeling will be utilized to base expectations on future changes in the LOCA frequencies resulting from aging mechanisms.
- ISI and mitigation strategies will be factored into the final result based on historical strategies and effectiveness.
- The effect of unique events and the emergence of additional mechanisms will also be considered.
- All decomposed contributors will be analytically recombined to determine the final LOCA frequencies from the elicitation process.

LOCA REEVALUATION: OVERVIEW

Several concurrent studies initiated to evaluate LOCA frequencies.

- Near-term elicitation (by April 30, 2002): support ECCS reliability revision and initiation of rulemaking (SECY-02-0057).
- Intermediate-term elicitation (within one year): support final rulemaking decisions.
- **Longer-term (by June 2004): redefine the spectrum of break pipe sizes.**

LOCA REEVALUATION: PIPE BREAK SIZE REDEFINITION

- **Objective:** determine the maximum pipe break size to use as design basis accident.
- **General Approach:**
  - Couple state-of-the-art fracture mechanics modeling with understanding of historical, recent, and potential degradation mechanisms to determine the likelihood of a double ended guillotine break in the largest primary system pipes.
  - Utilize philosophy consistent within current risk-based guidelines to determine the maximum allowable pipe break size.
- **Support:** Contract to be initiated with Battelle/Emc^2 (May or June 2002).
- **Goal:** completion by June 2004 as outlined in SECY-02-0057.

LOCA REEVALUATION: PIPE BREAK SIZE REDEFINITION

Necessary Technical Advancements

- Evaluate and update as needed probabilistic fracture mechanics (PFM) models and codes to include latest deterministic models for accurately modeling pipe-failure mechanisms.
- Utilize, where possible, realistic loading histories and frequencies. Also combine these loads with realistic/conservative residual stress distributions and pipe boundary conditions.
- Incorporate up-to-date material aging and environmental effect models to account for material degradation.

LOCA REEVALUATION: PIPE BREAK SIZE REDEFINITION

Necessary Technical Advancements

- Develop scheme to incorporate potential/surprise future failure mechanisms based on service history experience.
- Consider the effect of failure from transients (earthquake, thermal) and their event frequencies as well as from normal operating loads.
- Update fabrication flaw distributions developed for RPVs to reflect expected differences in piping manufacture. Also consider flaw initiation for relevant mechanisms.
- Assess likelihood of LB LOCA from other initiating failure modes and combine with LB LOCA frequencies from pipe failures.

TECHNICAL WORK TO SUPPORT RULEMAKING FOR CHANGES TO 10 CFR 50.46

- ECCS reliability requirements
- ECCS spectrum of break sizes and locations
- ***ECCS acceptance criteria***
- ECCS evaluation model requirements

TECHNICAL WORK TO SUPPORT RULEMAKING FOR CHANGES TO 10 CFR 50.46

- ECCS reliability requirements
- ECCS spectrum of break sizes and locations
- ECCS acceptance criteria
- ***ECCS evaluation model requirements***

Risk-Informed Revision of 10 CFR 50.46 Acceptance Criteria and ECCS Evaluation Model Requirements (Appendix K)

Presentation to the ACRS Subcommittees on Materials and Metallurgy, Thermal-Hydraulic Phenomena, and Reliability & Probabilistic Risk Assessment

May 31, 2002

Stephen M. Bajorek, G. Norman Lauben, Ralph O. Meyer
Safety Margins and Systems Analysis Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

OBJECTIVES

1. Update the Subcommittees on status of staff efforts related to risk-informing 10 CFR 50.46 acceptance criteria and Appendix K as envisioned by SECY-01-133 / SECY-02-0057.

2. Summarize and discuss near term actions.

BACKGROUND

SECY-01-133 states:

"The staff recommends that rulemaking should be undertaken to change the current 50.46. ....
....
..... In the near term, this revision would involve an update of Appendix K requirements based on more current and realistic information. ....

As part of this update, the staff will also consider the recognized non-conservatisms and model limitations to insure that proper safety focus is incorporated in any new rule.

.......; in summary, the staff will undertake work to:

support removal of unnecessary conservatisms from Appendix K."

The principal focus of this effort has been on:

1. Replacement of the Appendix K requirement to use 1.2 X 1971 ANS decay heat standard with a requirement based on the 1994 ANS decay heat standard.

2. Determining the impact of decay heat & metal-water reaction rate models and effect of accounting for non-conservatisms in existing Appendix K evaluation models.

- **Staff efforts have been in three areas:**

  - **Reviewing basis of exising 10 CFR 50.46 acceptance criteria for:**

    Peak Cladding Temperature (< 2200 ^{o}F),
    Maximum Cladding Oxidation (< 17% of total cladding thickness before oxidation)

  - **Reviewing 1994 Decay Heat Standard for incorporation into Appendix K, and feasibility of revising criteria related to Metal-Water Reaction, Steam Cooling, and Return to Nucleate Boiling During Blowdown**

  - **Evaluating known conservatisms and non-conservatisms in Appendix K EMs**

Outline: Recommendations to be Presented

1. Revise the 10 CFR 50.46 acceptance criteria for PCT and ECR to be "performance-based".
2. Replace 1971 ANS Decay Heat Standard with 1993 Standard
3. Replace the Baker-Just correlation with Cathcart-Pawel for metal-water reaction heat release.
4. Delete the requirement for steam cooling only at reflood rates below 1 inch/sec.
5. Retain the prohibition on assuming a return to nucleate boiling during blowdown.
6. Require that the new Evaluation Models to demonstrate sufficient overall conservatism and that they account for several identified non-conservatisms.

United States Nuclear Regulatory Commission

ACCEPTANCE CRITERIA
AND
METAL-WATER REACTION CORRELATIONS

Ralph Meyer
Office of Nuclear Regulatory Research

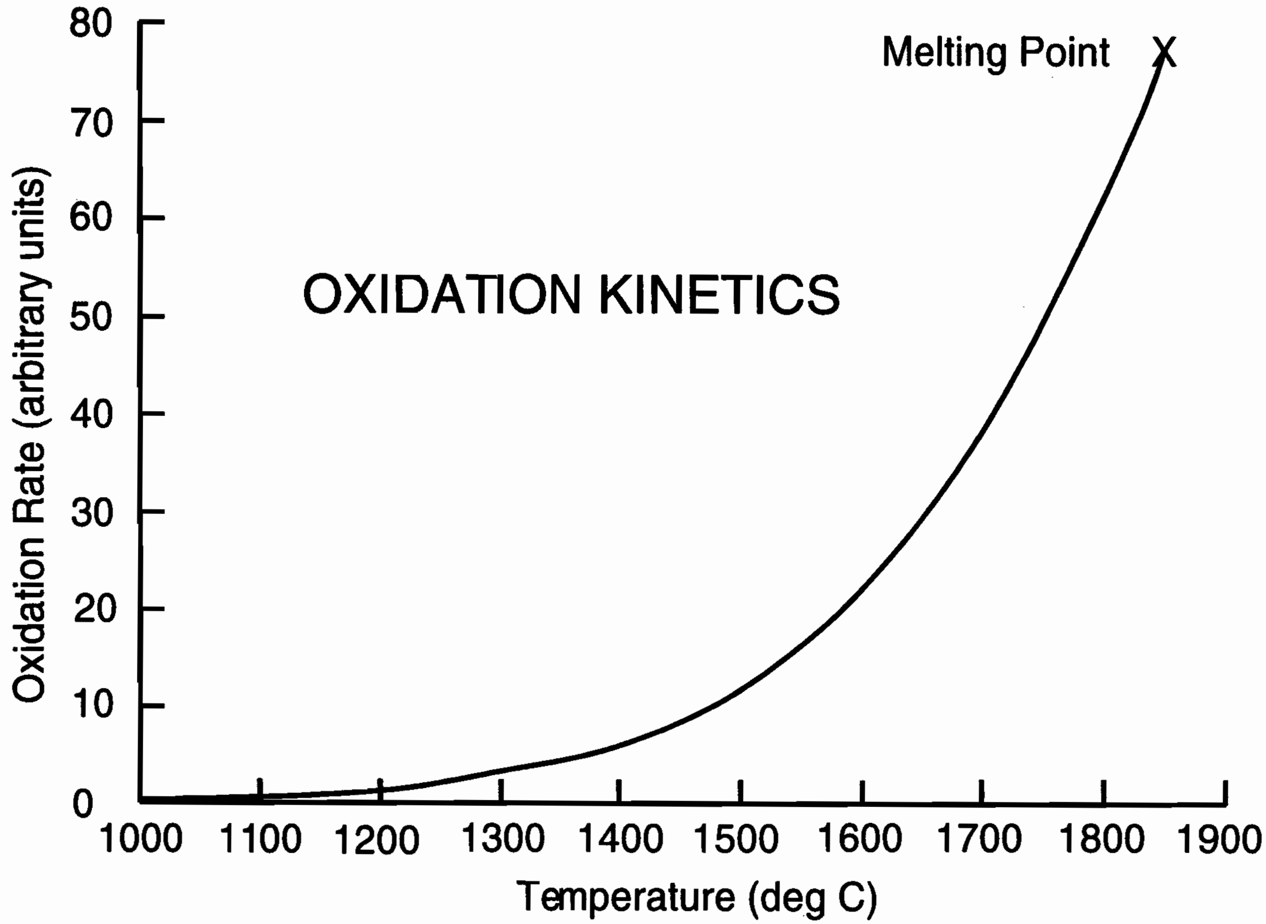
ACRS Meeting
May 31, 2002

ORIGIN OF PEAK CLADDING TEMPERATURE LIMIT

- Comes from temperature at which 17% ECR limit breaks down
- There was a second consideration related to runaway temperature escalation

STATEMENT ON TEMPERATURE LIMITS FROM 1973 HEARING

Westinghouse proposed a maximum calculated temperature limit of at least 2700°F; Combustion Engineering and the Utility Group agreed on 2500°F as the peak allowable calculated temperature on the basis that much of the data on oxidation and its effects stops at 2500°F. Babcock and Wilcox suggested a more conservative 2400°F as the peak calculated temperature to be allowed, presumably because “significant eutectic reaction and an excessive metal-to-water reaction rate would be precluded below 2400°F.” General Electric argued strongly that the limit should not be reduced to 2200°F; that 2700°F is really all right as far as embrittlement is concerned, but that the Interim Acceptance Criterion value of 2300°F should be retained. In addition to being consistent with their expressed desire not to change any of the criteria, the GE recommendation of retaining the 2300°F limit is intended to ensure that the core never “gets into regions where the metal-water reaction becomes a serious concern.” (USAEC, Opinion of the Commission....., CLI-73-39, Dec. 28,1973, p. 1097)


OXIDATION KINETICS
Melting Point X
Oxidation Rate (arbitrary units)
0
10
20
30
40
50
60
70
80
1000 1100 1200 1300 1400 1500 1600 1700 1800 1900
Temperature (deg C)

HEAT GENERATION RATE

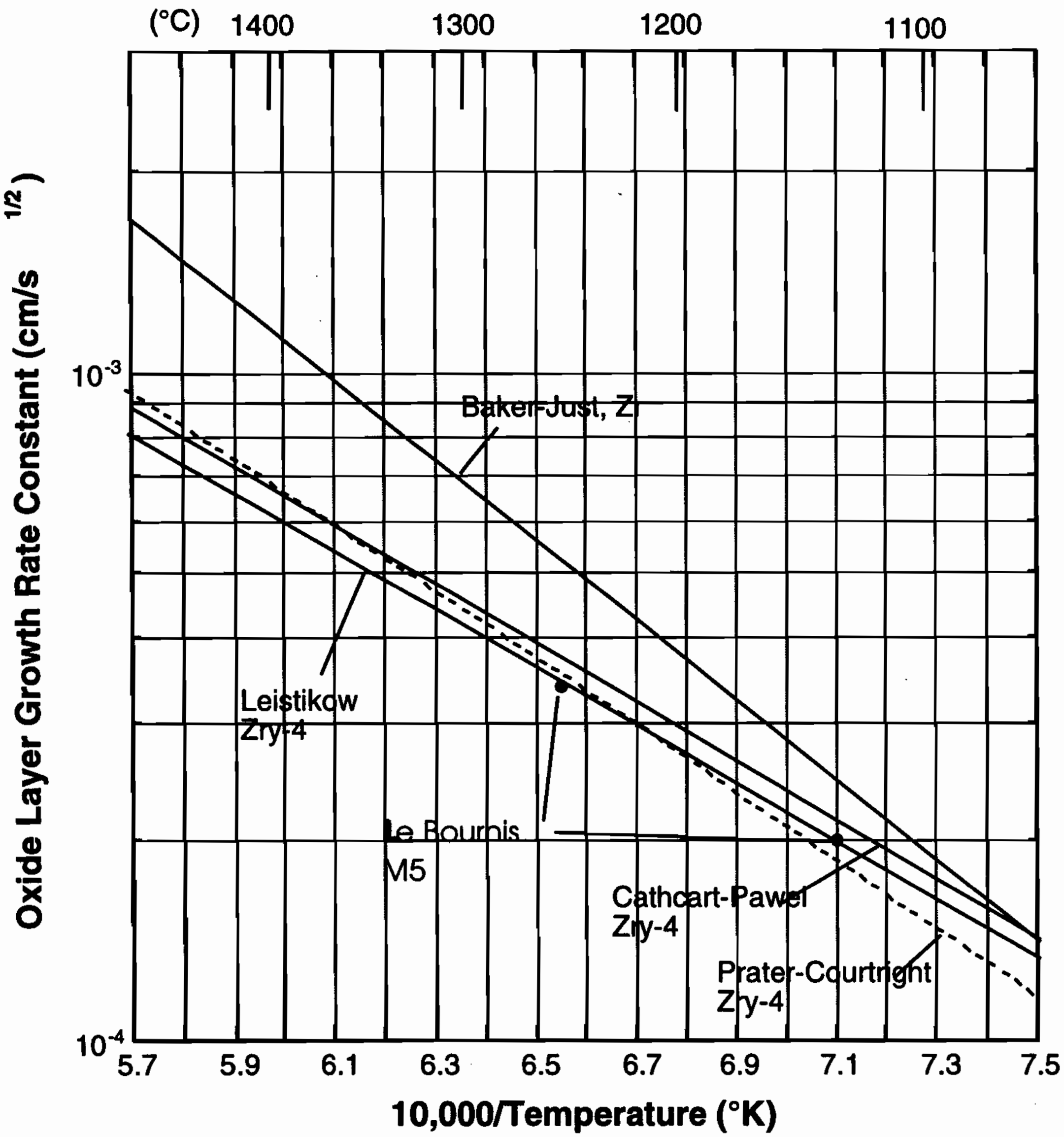
- When reaction heat becomes a significant part of total, positive feedback causes runaway

$$\text{Heat Rate}_{B\text{-}J}(2200^\circ F) = \text{Heat Rate}_{C\text{-}P}(2307^\circ F)$$

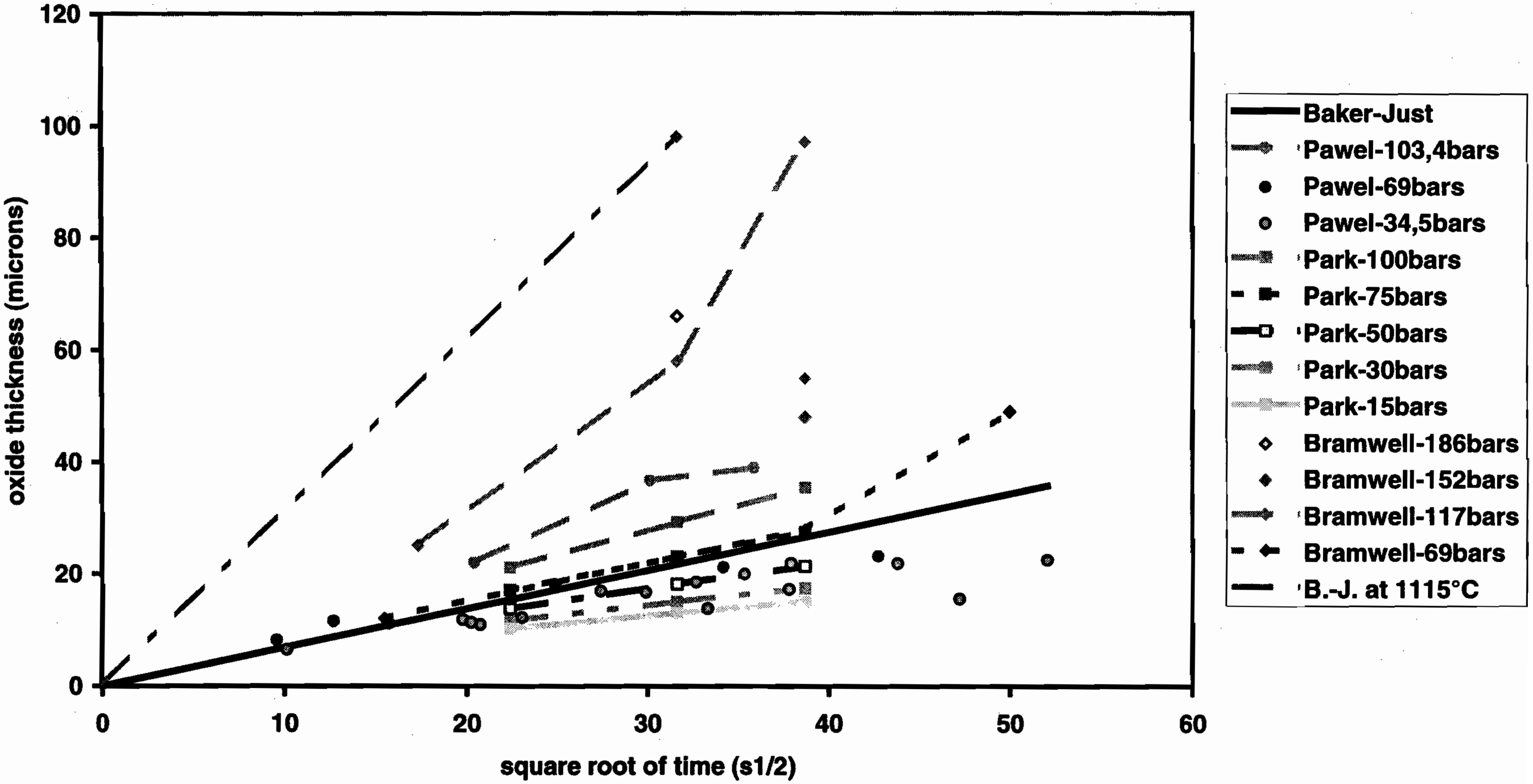
- Because Cathcart-Pawel is accurate, PCT could be increased to 2300°F with same margin to runaway as perceived in 1973

HIGH-TEMPERATURE OXIDATION MEASUREMENTS
(Approximately the same rate around 2200°F)

Investigators	Metal
Baker and Just	Zr
Lemmon	Zr
White	Valoy (Zr-1.3Cr-0.1Fe)
Urbanic	Zircaloy-2, Zircaloy-4, Zr-2.5Nb
Cathcart et al.	Zircaloy-4
Chung and Kassner	Zircaloy-4
Grandjean et al.	Zircaloy-4
Yan et al.	Zircaloy-2
Waeckel and Jacques	Zircaloy-2
Le Bourhis	M5
Leech	ZIRLO
Yegorova et al.	E110 (Zr-1Nb)


(°C) 1400 1300 1200 1100
Oxide Layer Growth Rate Constant (cm/s 1/2)
10-3
10-4
Baker-Just, Zr
Leistikow Zry-4
Le Bournis M5
Cathcart-Pawel Zry-4
Prater-Courtright Zry-4
5.7 5.9 6.1 6.3 6.5 6.7 6.9 7.1 7.3 7.5
10,000/Temperature (°K)

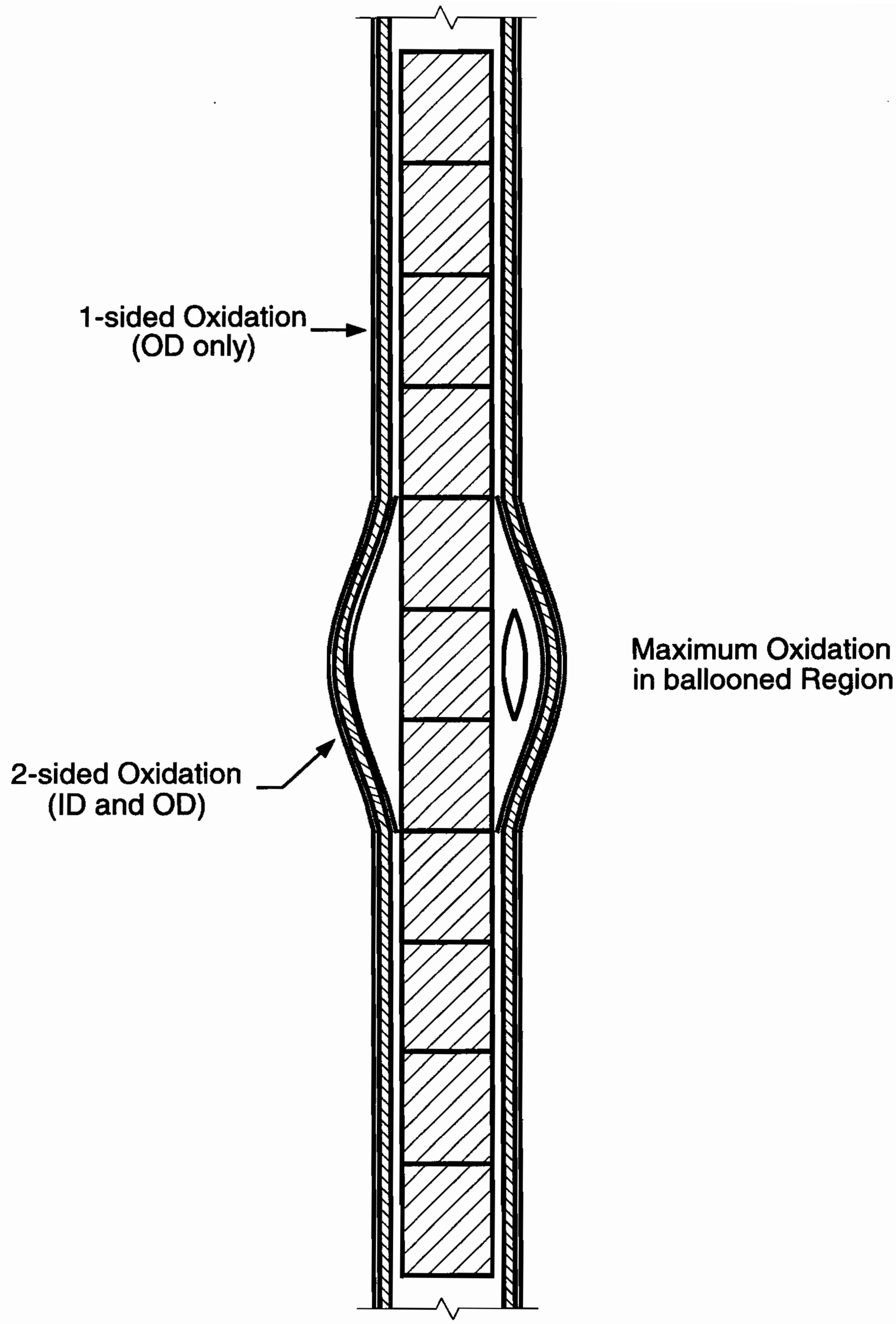
Zry - oxide thickness at 900°C as a function of square root of time and steam pressure



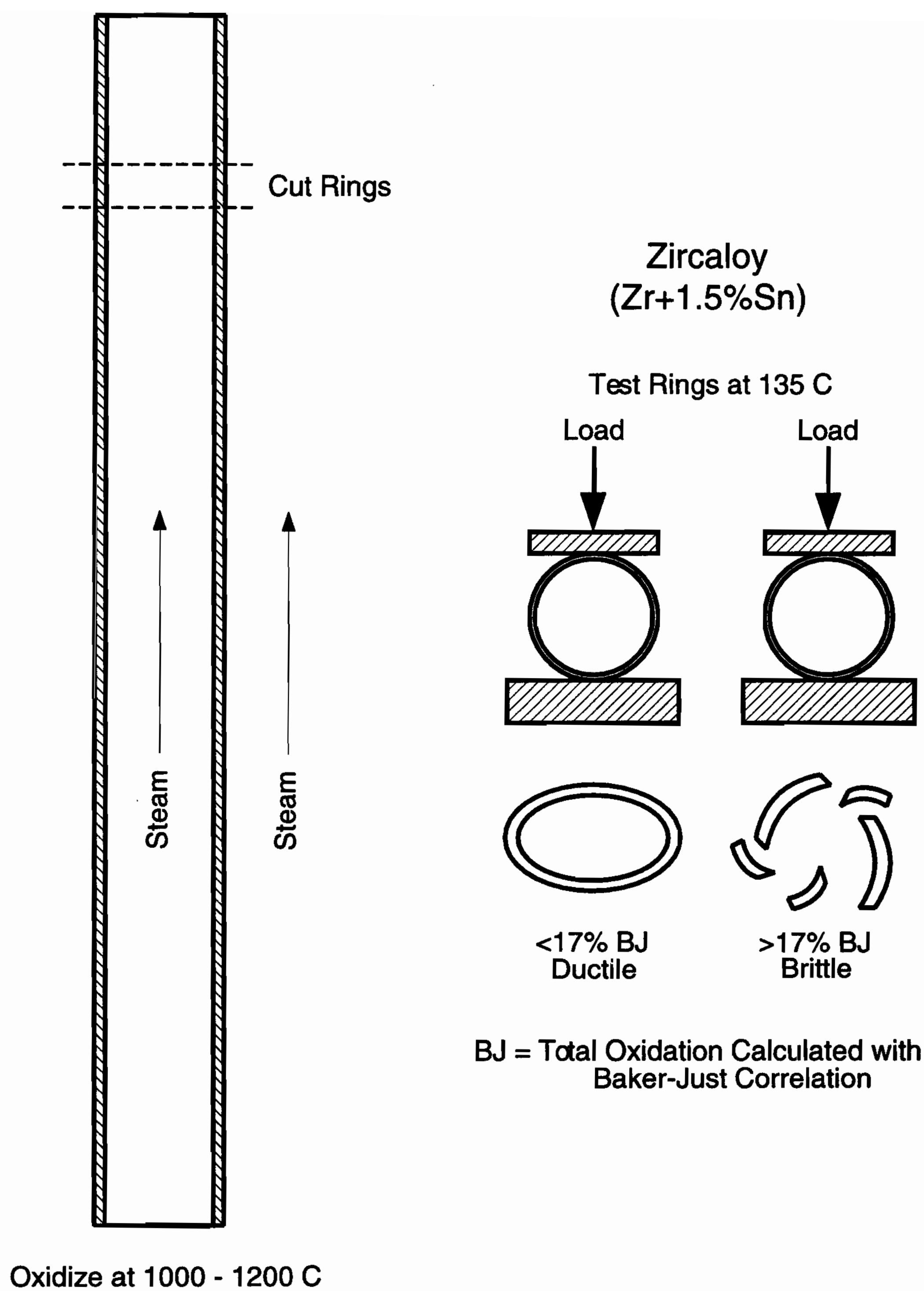
THERMAL SHOCK TESTS

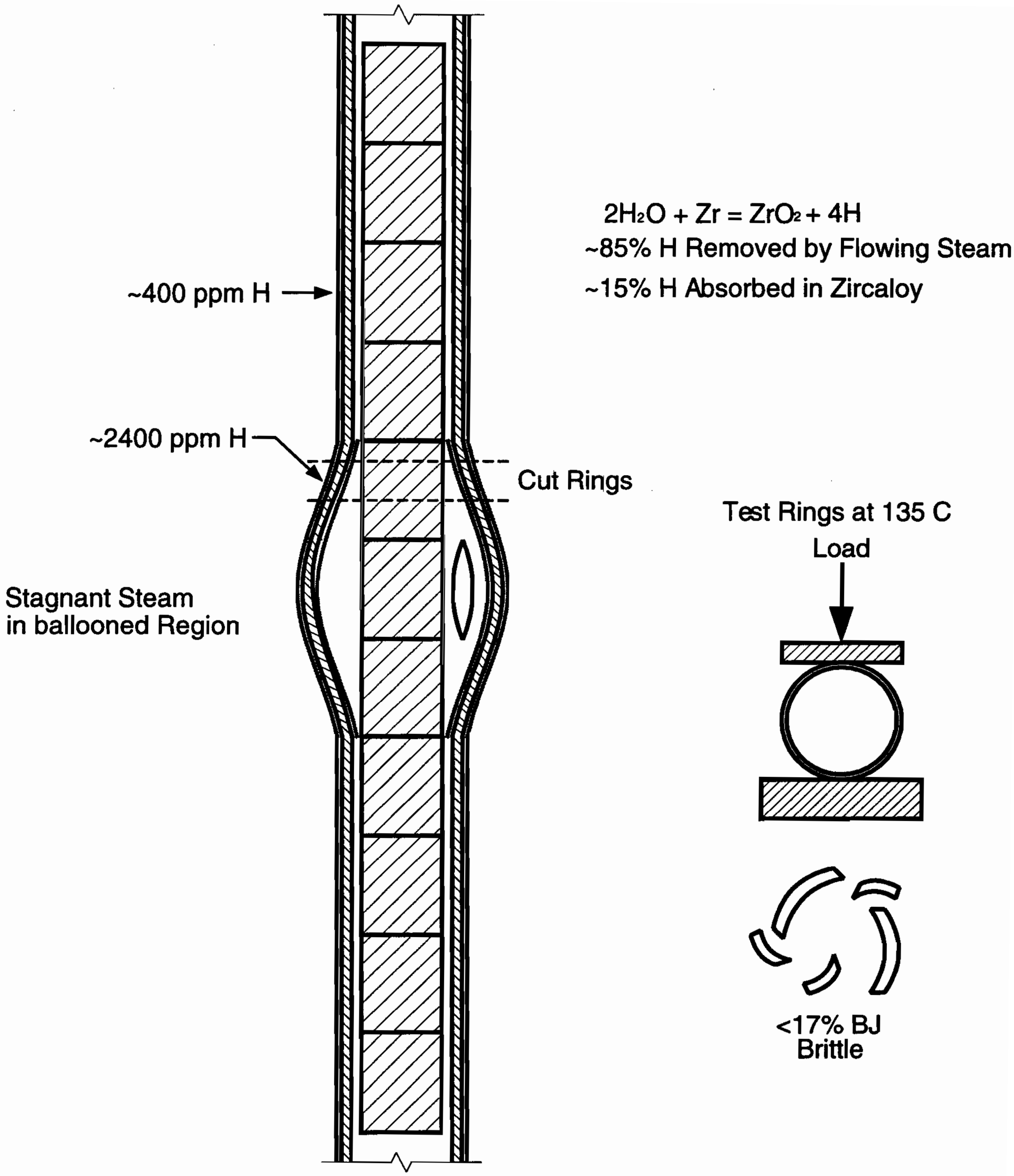
Not adequate according to U.S. AEC Commissioners in 1973

"Our selection of the 2200°F limit results primarily from our belief that retention of ductility in the zircaloy is the best guarantee of its remaining intact during the hypothetical LOCA. The stress calculations, the measurements of strength and flexibility of oxidized rods, and the thermal shock tests all are reassuring, but their use for licensing purposes would involve an assumption of knowledge of the detailed process taking place in the core during a LOCA that we do not believe is justified."

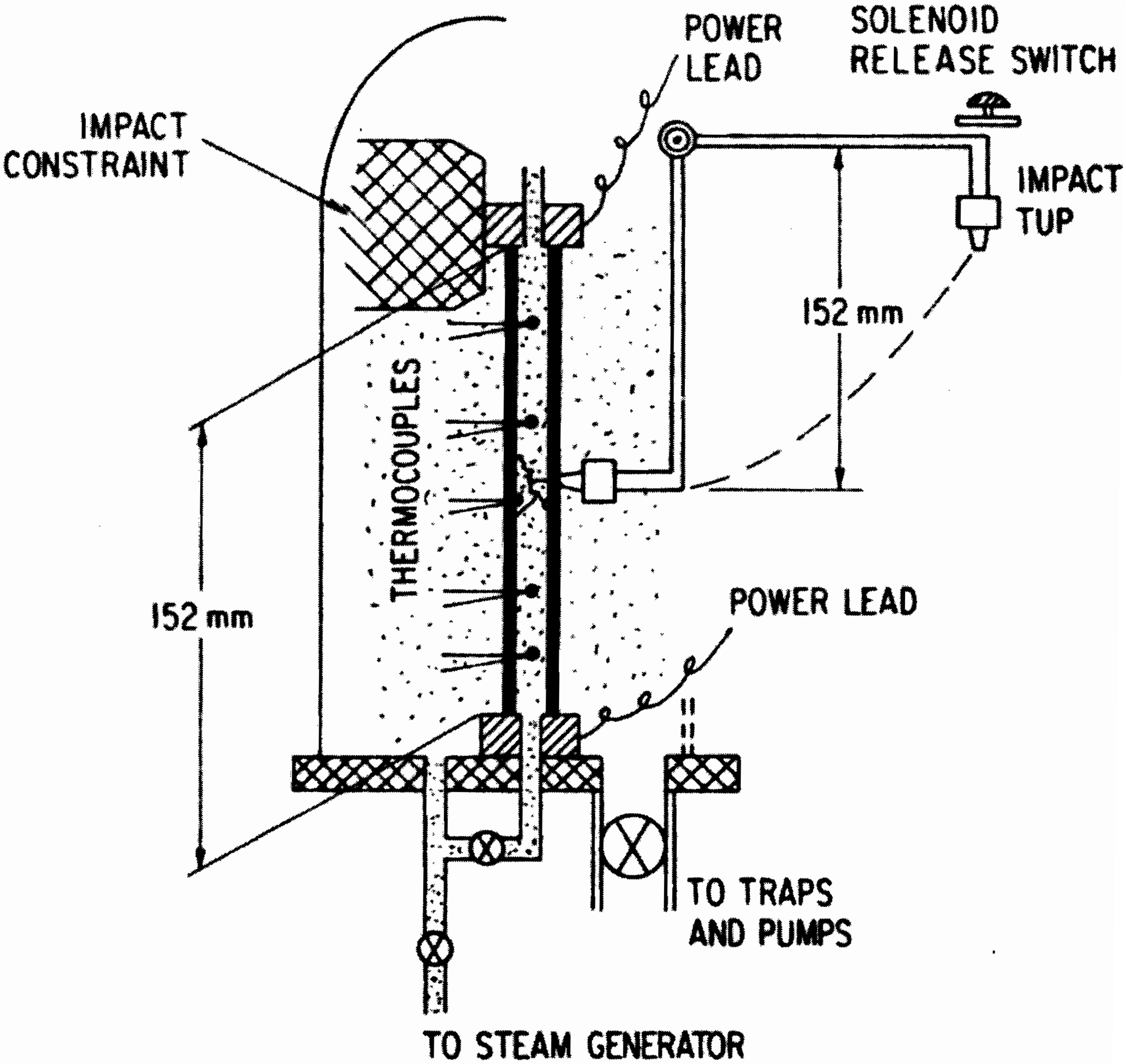


Significant Oxidation 1000 - 1200 C
(2200 F = 1204 C)

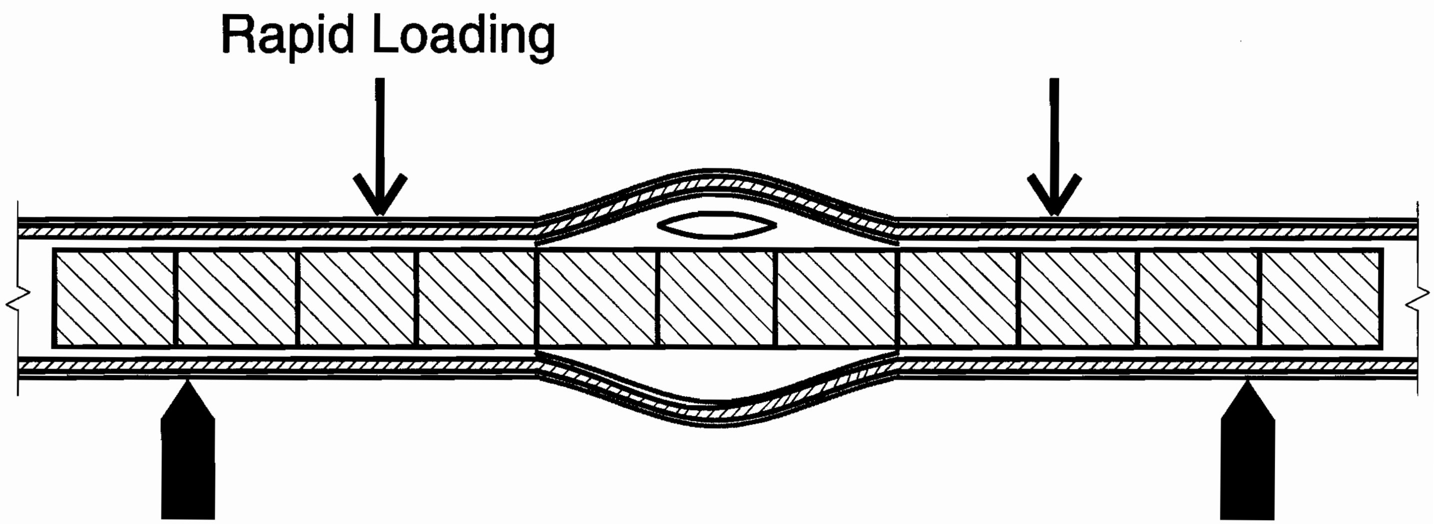

Cut Rings
Steam
Steam
Oxidize at 1000 - 1200 C
Zircaloy
(Zr+1.5%Sn)
Test Rings at 135 C
Load
Load
<17% BJ
Ductile
>17% BJ
Brittle
BJ = Total Oxidation Calculated with
Baker-Just Correlation



Hydrogen Effect Discovered ~1980



POWER LEAD
SOLENOID RELEASE SWITCH
IMPACT CONSTRAINT
IMPACT TUP
152 mm
THERMOCOUPLES
152 mm
POWER LEAD
TO TRAPS AND PUMPS
TO STEAM GENERATOR



4-Point Bend Test

CONCLUSIONS

- New PCT and ECR limits can be derived from mechanical property tests for all burnups and different alloys
- Simple ductility test (ring compression) may be adequate, as shown for unirradiated Zircaloy
- Confirmation of ductility test to be investigated with 4-point bend or pendulum impact test
- PCT should not exceed 2300°F to retain margin to avoid runaway temperatures
- Cathcart-Pawel may work adequately for all alloys and burnups (TBD) provided pressure enhancement is added for SBLOCA analysis

Decay Heat Changes to 50.46 and Appendix K

Joint Meeting of the ACRS Subcommittees on Materials & Metallurgy, Thermal-Hydraulic Phenomena, and Reliability & Probabilistic Risk Assessment

May 31, 2002

G. Norman Lauben
Safety Margins and Systems Analysis Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

“IF YOU DON’T HAVE A BEST ESTIMATE, HOW DO YOU KNOW WHAT IS LABELED AS CONSERVATIVE IS CONSERVATIVE?”

MILTON LEVINSON - ACNW
AUGUST 28, 2001

THE 1994 ANS DECAY HEAT STANDARD

- It is proposed that the decay heat requirements in Appendix K and the best estimate guidance in Regulatory Guide 1.157 be replaced with requirements and guidance based on the 1994 ANS decay heat standard.

- The Appendix K option in 50.46 currently requires fission product decay heat be modeled using the draft 1971 ANS standard with a multiplier of 1.2 and the assumption of infinite irradiation. A separate paragraph in Appendix K requires consideration of Actinide decay heat.

  - An alternative would permit the use of the 1994 ANS decay heat standard, which involves more sophisticated uncertainty methods and a greater number of options left to the user.

  - The 1994 ANS standard considers more recent available data and methods.

  - Model options in the 1994 standard have been identified and studied. They are:

    1. Whether the reactor operating history should be represented by a histogram of multiple irradiation intervals and multiple fissile isotopes or can be modeled as a single interval and a single fissile isotope, ^{235}U,
    2. Values of the recoverable energy per fission (Q_i) for ^{235}U, ^{239}Pu, and ^{238}U, and ^{241}Pu,
    3. Calculation of the correction factor G(t) for neutron capture in fission products,
    4. The actinide contribution to decay heat power
    5. The effect of various uncertainty methods and parameters.

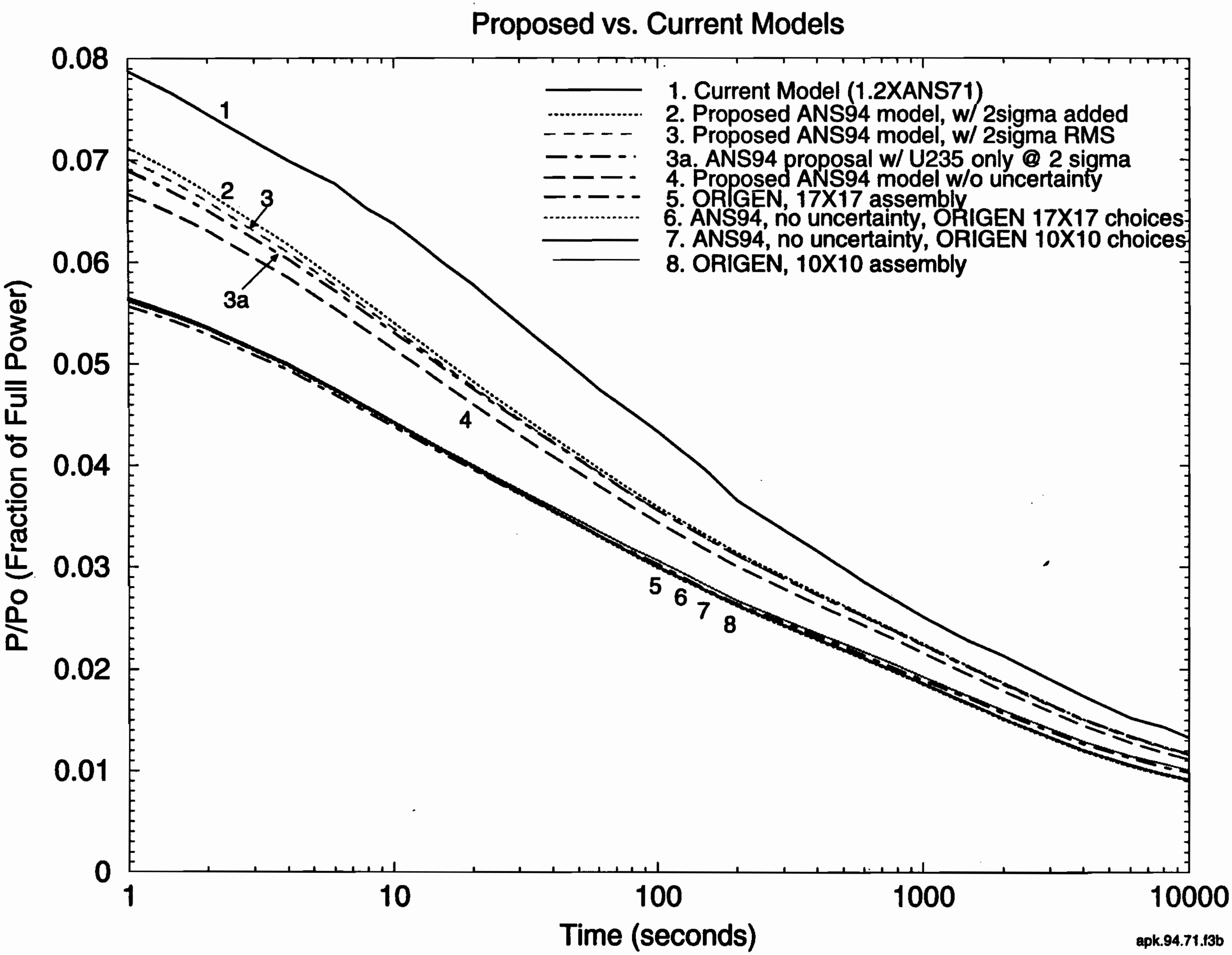
- The performance based realistic option in 50.46 would allow use of the 1994 standard today. Specification of the 1994 standard as an acceptable method in Reg. Guide 1.157 would facilitate its use.

ASSUMPTIONS FOR NINE DIFFERENT DECAY HEAT CALCULATIONS

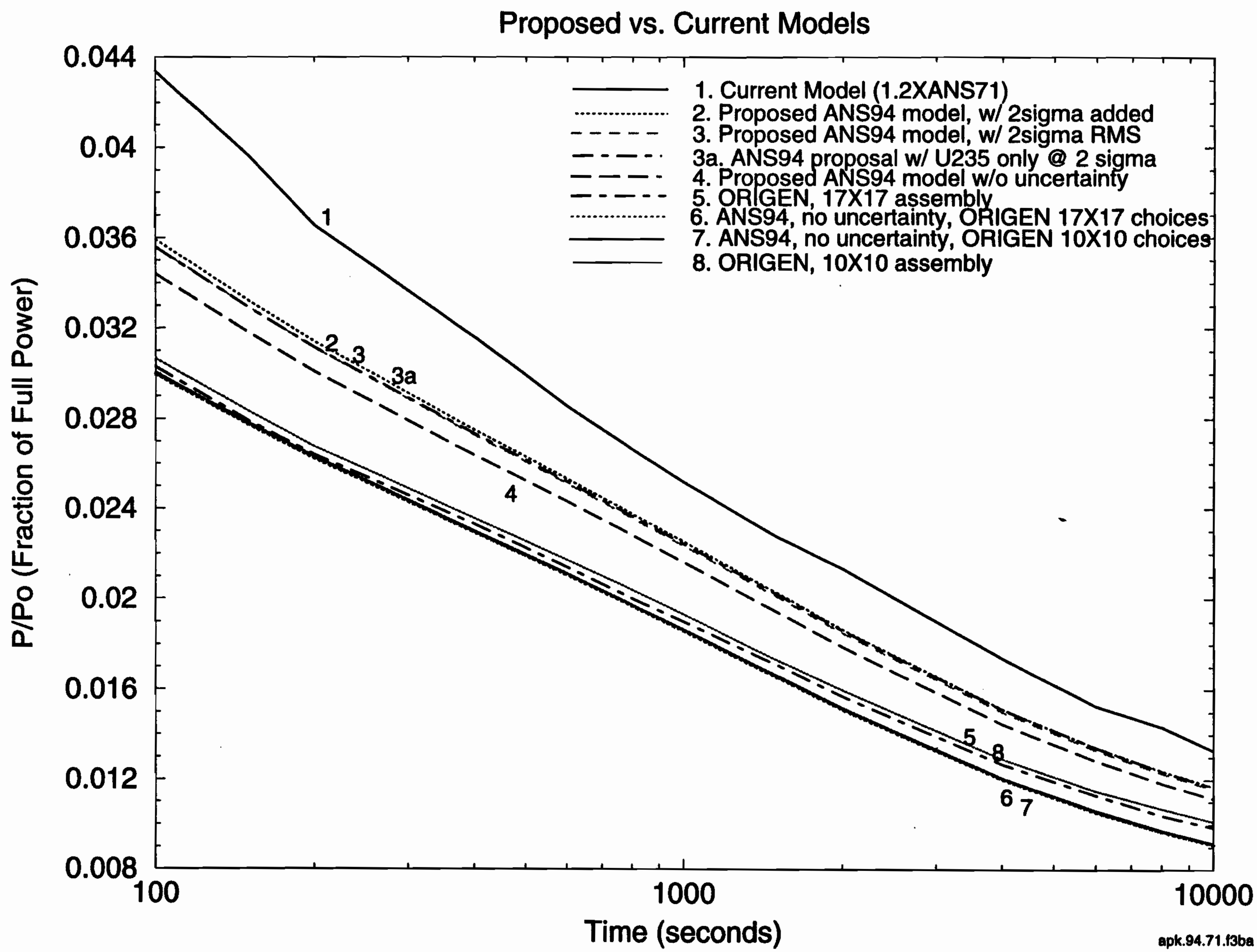
Case No.	Model	Multiplier	Operating Time	Fiss. Fractions	Capture Time (Sec.)	Ψ	Fission Energy MeV/f.	Actinide Yield	Isotope Tables	Isotopic Uncertainties
				Current Appendix K						
1	ANS73	1.2	∞	100%^{235}U	N/A	N/A	N/A	0.7	N/A	N/A
				Appendix K Proposals						
2	ANS94	2σ,add	∞	Note 3	2.e8	1.0	200	0.7	Note 7	Note 8
3	ANS94	2σ,RMS	∞	Note 3	2.e8	1.0	200	0.7	Note 7	Note 8
3a	ANS94	2σ	∞	100%^{235}U	2.e8	1.0	200	0.7	Note 7	Note 9
4	ANS94	mean	∞	Note 3	2.e8	1.0	200	0.7	Note 7	N/A
				Best Estimate						
5	ORIGEN[1]	mean	Calc.	Calc.	Calc.	Calc.	Calc.	Calc.	Calc.	N/A
6	ANS94	mean	ORIGEN[5]	Note 4	1.2e8[5]	1.0	ORIGEN[5]	.514[5]	Note 7	N/A
7	ANS94	mean	ORIGEN[6]	Note 4	1.2e8[6]	1.0	ORIGEN[6]	.508[6]	Note 7	N/A
8	ORIGEN[2]	mean	Calc.	Calc.	Calc.	Calc.	Calc.	Calc.	Calc.	N/A

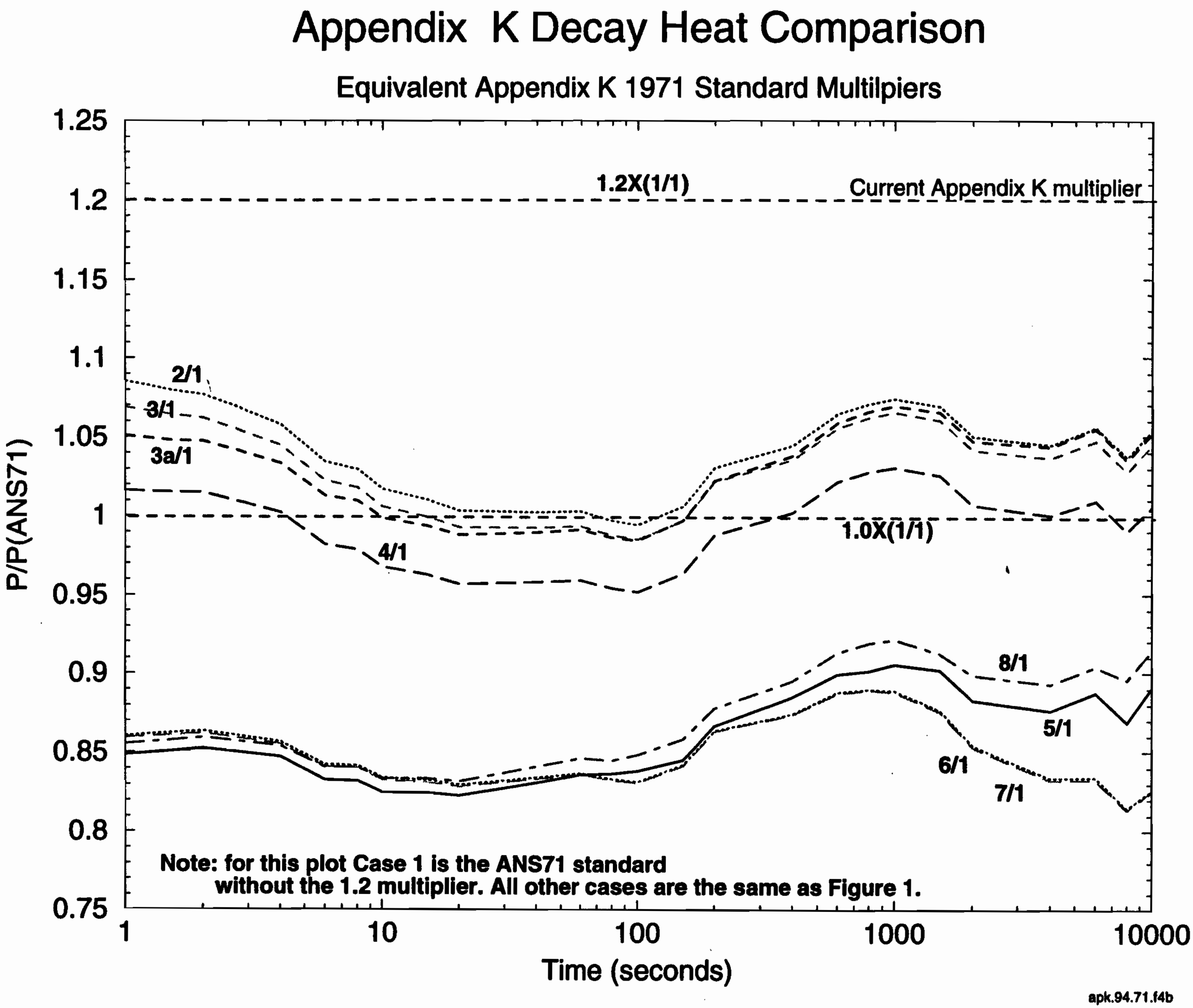
Note 1 17X17 PWR assembly
Note 2 10X10 BWR assembly
Note 3 Assumes fissioning fractions are 90% ^{235}U and 10% ^{238}U
Note 4 Cycle average values from ORIGEN calculations for four isotopes
Note 5 From 17X17 PWR ORIGEN calculation
Note 6 From 10X10 BWR ORIGEN calculation
Note 7 23 decay group exponential fits for $F(t,\infty)$ in ANS94 standard
Note 8 Used curve fits from Figures 1 and 2
Note 9 Used curve fit from Figure 1

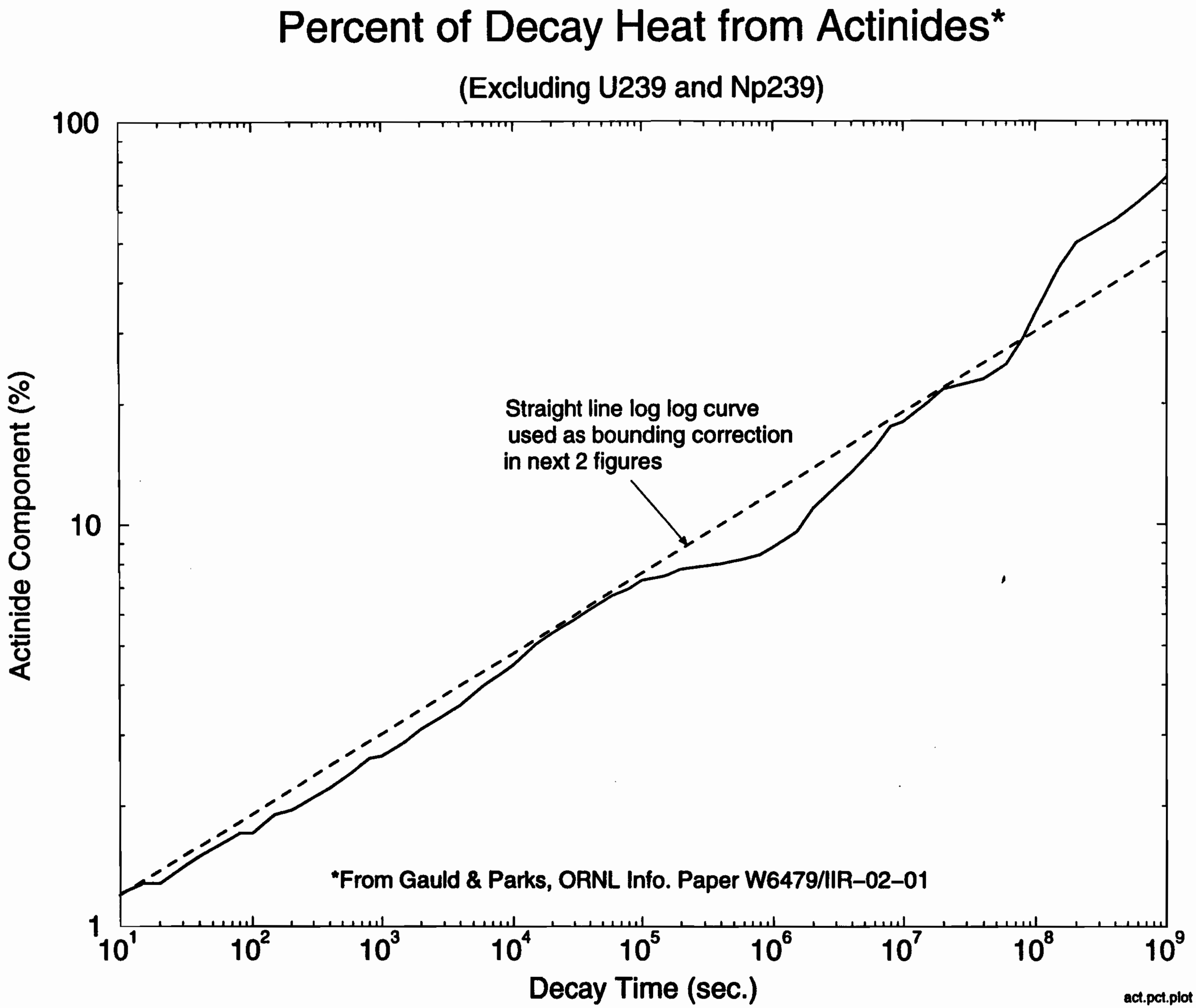
Appendix K Decay Heat Comparison



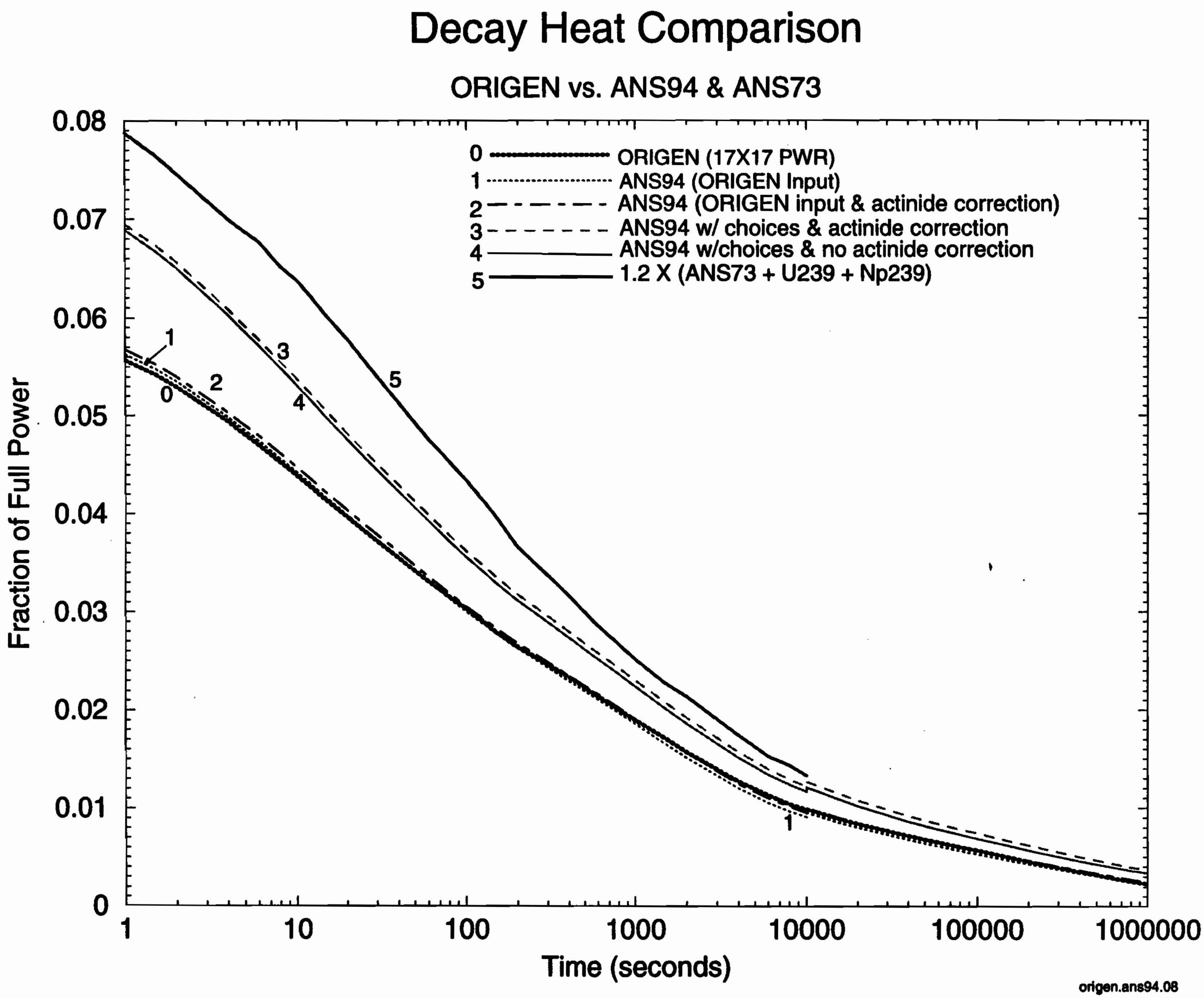
Appendix K Decay Heat Comparison



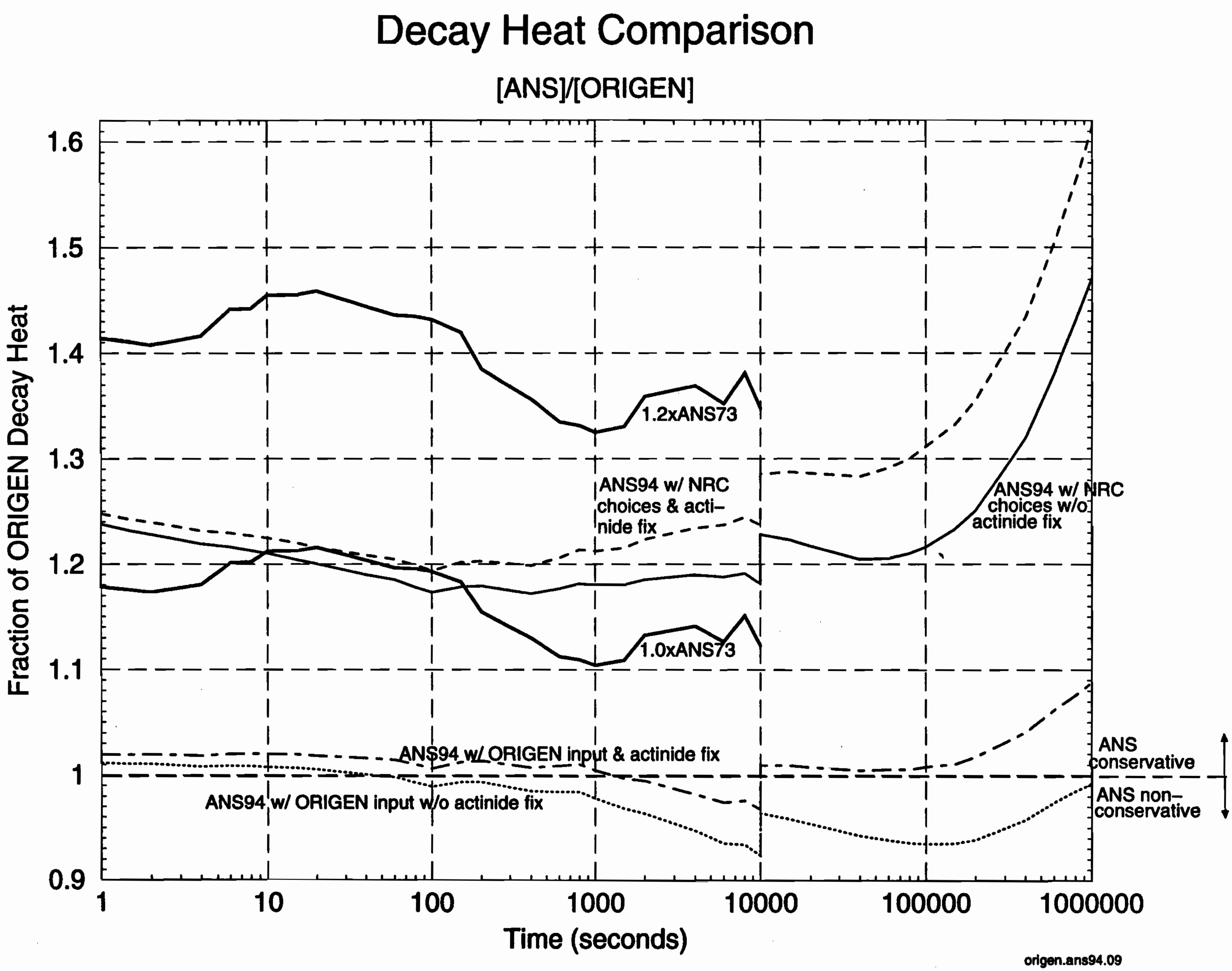

Appendix K Decay Heat Comparison
Equivalent Appendix K 1971 Standard Multilpiers
1.2X(1/1)
Current Appendix K multiplier
2/1
3/1
3a/1
4/1
1.0X(1/1)
8/1
5/1
6/1
7/1
Note: for this plot Case 1 is the ANS71 standard
without the 1.2 multiplier. All other cases are the same as Figure 1.
P/P(ANS71)
Time (seconds)
1.25
1.2
1.15
1.1
1.05
1
0.95
0.9
0.85
0.8
0.75
1
10
100
1000
10000
apk.94.71.f4b



Percent of Decay Heat from Actinides*
(Excluding U239 and Np239)
Actinide Component (%)
100
10
1
Straight line log log curve
used as bounding correction
in next 2 figures
*From Gauld & Parks, ORNL Info. Paper W6479/IIR-02-01
10^1
10^2
10^3
10^4
10^5
10^6
10^7
10^8
10^9
Decay Time (sec.)
act.pct.plot



Decay Heat Comparison
ORIGEN vs. ANS94 & ANS73
0 ORIGEN (17X17 PWR)
1 ANS94 (ORIGEN Input)
2 ANS94 (ORIGEN input & actinide correction)
3 ANS94 w/ choices & actinide correction
4 ANS94 w/choices & no actinide correction
5 1.2 X (ANS73 + U239 + Np239)
Fraction of Full Power
0.08
0.07
0.06
0.05
0.04
0.03
0.02
0.01
0
1
10
100
1000
10000
100000
1000000
Time (seconds)
origen.ans94.08



Decay Heat Comparison
[ANS]/[ORIGEN]
Fraction of ORIGEN Decay Heat
1.6
1.5
1.4
1.3
1.2
1.1
1
0.9
1
10
100
1000
10000
100000
1000000
Time (seconds)
1.2xANS73
ANS94 w/ NRC choices & acti- nide fix
ANS94 w/ NRC choices w/o actinide fix
1.0xANS73
ANS94 w/ ORIGEN input & actinide fix
ANS94 w/ ORIGEN input w/o actinide fix
ANS conservative
ANS non- conservative
origen.ans94.09

RECOMMENDATIONS

- "Grandfather" the current Appendix K decay heat requirements.

- Add an Appendix K option to use the 1994 ANS standard with the following pre-selected choices, which are equivalent to Case 3a in Figures 3 and 4:

  1. Assume ^{235}U is the only fissioning isotope.
  2. Assume infinite operating time.
  3. Assume 200 MEV/fission recoverable energy.
  4. Use Equation 11 in the standard for neutron capture effect for shutdown times less than 10^4 seconds. Use 2.e8 seconds operating time for this equation. Use 1.0 as the value for Ψ.
  5. Use Table 13 in the standard for neutron capture for shutdown times greater than or equal to 10^4 seconds.
  6. Apply Section 4 in the standard for the decay heat contribution for ^{239}U and ^{239}Pu. Use a value of 0.7 for R.
  7. Use a 2σ value of uncertainty for ^{235}U based on the bounding curve of Figure 1. Along with options 1 and 2, this obviates the need to consider methods to combine uncertainties.

- Add another Appendix K option to allow use of a subsequent consensus standard and/or selection of user choices other than those shown above.

- Use of the new Appendix K options would be subject to a model review as required in 50.46. A model review is prudent to assure retention of sufficient remaining conservatism in any revised Appendix K model in which a substantial amount of conservatism has been removed. This subject is discussed in more detail by Steve Bajorek.

- Allow use of the 1994 ANS standard in best estimate Reg. Guide 1.157

Risk-Informed Revision of ECCS Evaluation Model Requirements (Appendix K)

Presentation to the ACRS Subcommittees on Materials and Metallurgy, Thermal-Hydraulic Phenomena, and Reliability & Probabilistic Risk Assessment

May 31, 2002

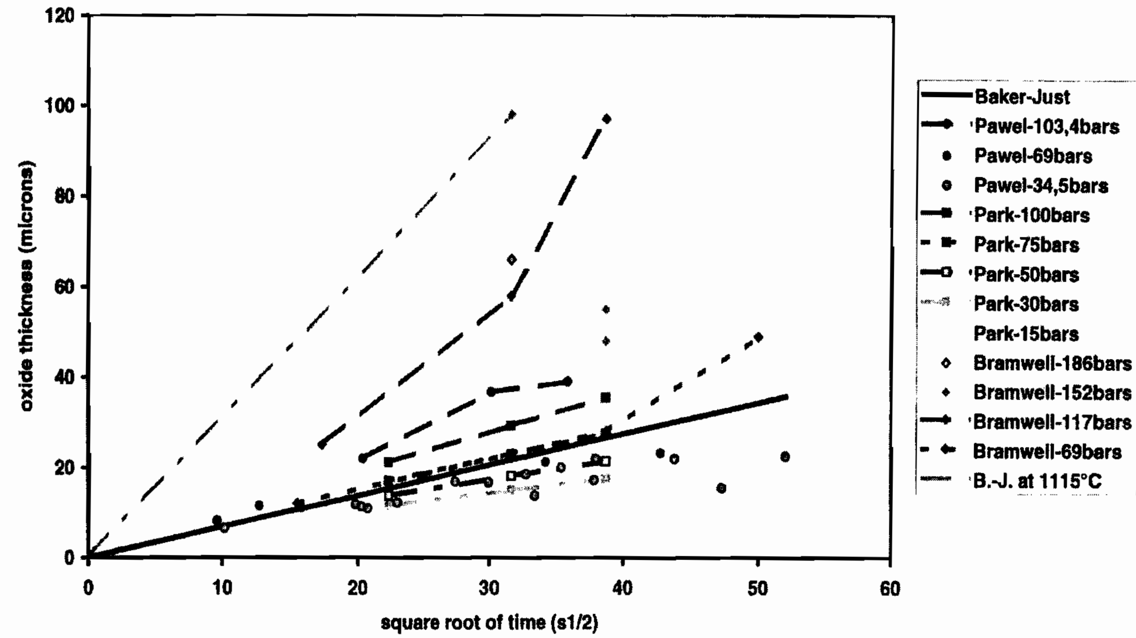
Stephen M. Bajorek
Safety Margins and Systems Analysis Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

Appendix K Modeling Requirements
Metal-Reaction Heat Release

- Original rulemaking assumed Baker-Just was conservative at 2000 ^{o}F, but was approximately correct at 2200 ^{o}F.

- Baker-Just equation based on pure Zr data - not alloys. Review of more recent data covering several different Zr based alloys shows low experimental data scatter and good agreement with Cathcart-Pawel.

- All Zr-based alloys exhibit about the same oxidation kinetics. Reason: Dominant rate-controlling step at high temperatures is diffusion of oxygen through ZrO_2 surface layer.



- Experimental data however, exhibits enhanced oxidation rates at high pressure. Cathcart-Pawel correlation is non-conservative for heat release at high pressure.

Recommendation:

The Baker-Just correlation for exothermic heat release can be replaced with the Cathcart-Pawel (at low pressures) or with a suitable realistic correlation shown applicable to a specific alloy. An adjustment to Cathcart-Pawel or other correlation if used at high pressure.

Appendix K Modeling Requirements
Steam Cooling Below 1 inch/sec

- Paragraph I.D.5.b. of Appendix K states that:

"During refill and during reflood when reflood rates are less than one inch per second, heat transfer calculations shall be based on the assumption that cooling is only by steam, ...

- Experimental data from FLECHT series of tests demonstrated high rates of entrainment & carryover, even for VIN < 1 ips.

Recommendation:

Delete the requirement for steam cooling only at reflood rates below 1 inch/sec.

Appendix K Modeling Requirements
Return to Nucleate Boiling During Blowdown

- Paragraph I.C.4.e. in Appendix K prohibits the return to nucleate boiling heat transfer even if the fluid and surface conditions apparently justify the return.

- Rewet during blowdown supported by LOFT experiments. However, overall database demonstrating blowdown rewet is sparse for Zr cladding and Tmin can be predicted only with very high uncertainty.

Recommendation:

Retain the prohibition on assuming a return to nucleate boiling dur-ing blowdown.

Appendix K "Non-Conservatisms"

Sources of potential non-conservatism:

1. **Thermal-hydraulic processes and fuel behavior that have been observed in experimental programs since 1973, but are not specifically addressed by Appendix K.**

2. **Large calculational uncertainties that are on the order of the overall conservatism of the EM. This was a main concern of SECY-86-318, ("Revision of the ECCS Rule Contained in Appendix K and Section 50.46 of 10 CFR Part 50) which recommended that the Appendix K decay heat guidelines <u>not</u> be revised unless model uncertainties were accounted for.**

Non-Conservative Processes Identified:

- **Downcomer Boiling**
- **Reflood ECC (Downcomer) Bypass**
- **Fuel Relocation**

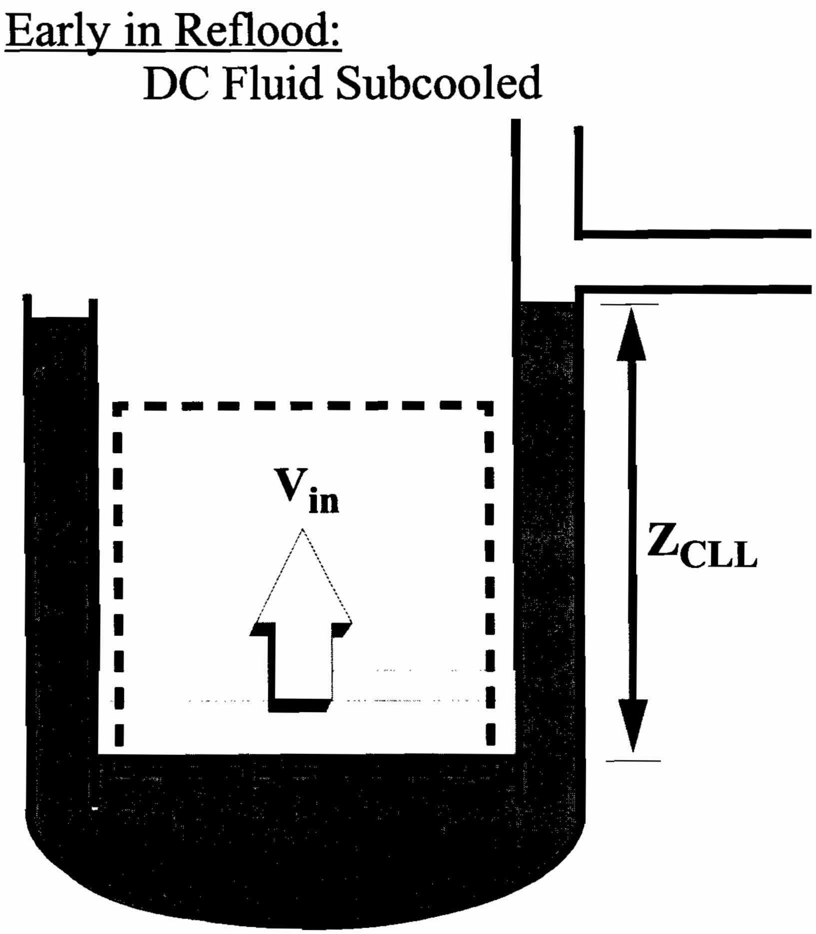
◆ Downcomer Boiling

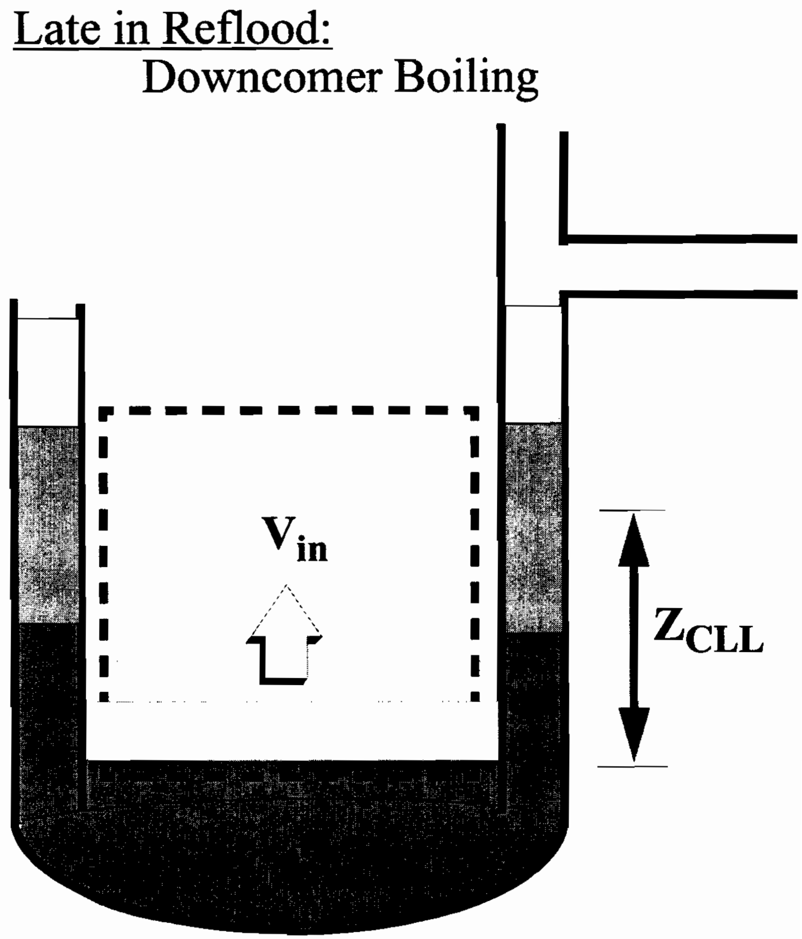
- **Experimental data from several facilities, and simulations using "Best Estimate" thermal-hydraulic codes show that stored heat in vessel walls, core barrel and lower plenum structures can cause coolant in the downcomer to boil during reflood.**

- **Voiding in the downcomer can result in a significant reduction in downcomer head. This reduces the flooding rate and increases the PCT.**

- **PWR Appendix K reflood models do not model downcomer boiling. Yet, for at least some plants in all three PWR vendor designs, the existence of downcomer boiling has at least been acknowledged.**

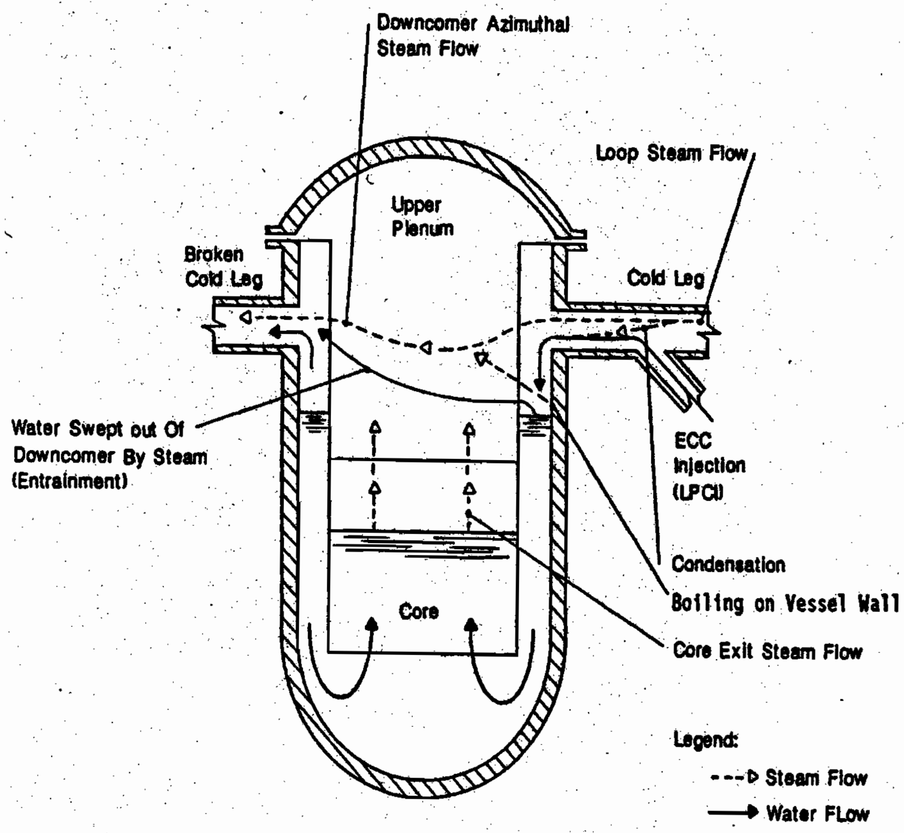
DOWNCOMER BOILING





Downcomer Boiling: Causes Net Loss of Driving Head & Reduces Reflood Rate

◆ Reflood ECC (Downcomer) Bypass



- **Experimental tests in the full scale UPTF facility showed that steam from intact loops could entrain significant amounts of water from the downcomer during reflood.**

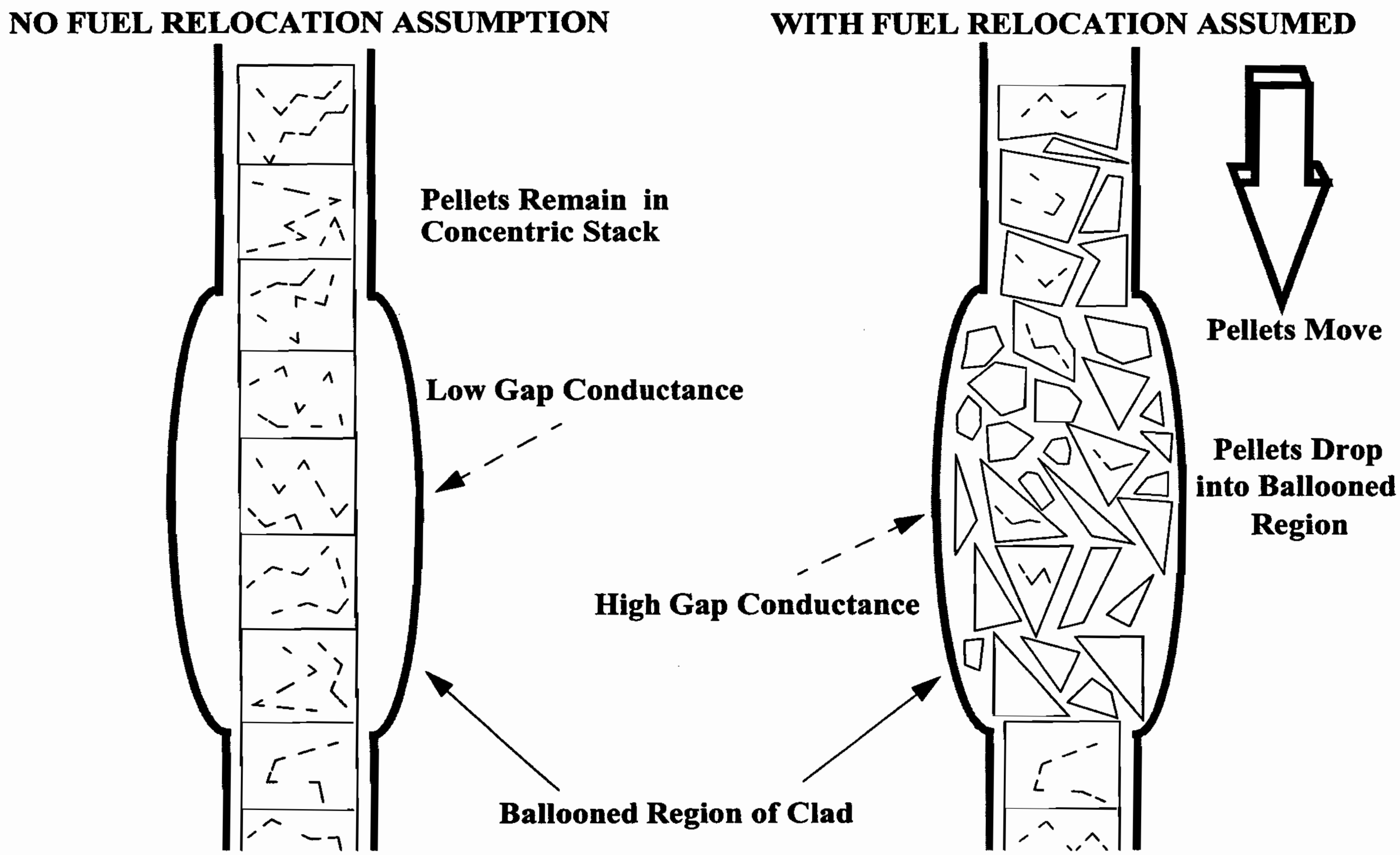
- **High entrainment and carryover to the break reduced the downcomer water level and can result in a reduction in downcomer head. This reduces the flooding rate and increases the PCT.**

- **Process is a strong function of the downcomer water level and oscillations.**

◆ Fuel Relocation

- Experiments in PBF-LOC, FR2 (Germany) and FLASH5 (France) showed significant fuel movement in regions where clad has ballooned.

- Relocation of additional fuel into ballooned region increases local power and increases conductance between pellets and clad.



Estimation of Evaluation Model Significance

- Proposed revisions to Appendix K requirements will have major impact on predicted peak cladding temperature (PCT) and equivalent cladding reacted (ECR).

- Since large break LOCA is generally the most limiting accident scenario, more information is available on effect of changing decay heat, metal-water heat release etc. for that accident. Very little information is available for SBLOCA. Results are plant dependent.

- The following tables list documented sensitivities for various models & assumptions. Reference numbers are identified in the "Research Information Letter."

Table 1: Large Break LOCA ΔPCT Estimates

Process	ΔPCT	Basis/Comments
Decay Heat	-260 to -450 °F	Recent Westinghouse estimate based on App. K EM calculations [2]. ANS 1971 + 1.20 replaced with ANS 1979 + 2σ. Calculations performed using BASH-EM.
Decay Heat	-372 °F	NRC contractor RELAP calculations for CE 2700 MWt (Millstone 2) plant [3]. ANS 1971 + 1.20 replaced with ANS 1979 + 2σ.
Decay Heat	-460 °F	1984 Westinghouse study on Appendix K relaxation [4] .
Metal Water Reaction	-45 to -55 °F	Recent Westinghouse estimate assuming the Baker-Just correlation is replaced with Cathcart-Pawel for metal-water reaction heat [2]. Calculations performed using BASH-EM.
Metal Water Reaction	-75 °F	NRC contractor RELAP calculations with Baker-Just replaced by Cathcart-Pawel [3].
Metal Water Reaction	-65 °F	1984 Westinghouse study on Appendix K relaxation [4] .
Downcomer Boiling	+400 °F	Westinghouse estimate from Best Estimate EM calculations for W 4-loop PWR [5].
Downcomer Boiling	+810 °F	NRC contractor calculations using RELAP5 for a CE System 80+ (3800 MWt) unit [6].
Downcomer Boiling + Reflood Bypass	+63 °F	Estimate based on WCOBRA/TRAC calculations for an uprated CE System 80+ unit [7]. Both downcomer boiling and ECC bypass during reflood were found to be important and contribute to increases in PCT.

Table 1: Large Break LOCA ΔPCT Estimates

Process	ΔPCT	Basis/Comments
Fuel Relocation	+46 °F	EG&G estimate based on experimental tests in PBF (Power Burst Facility) to address Generic Safety Issue (GSI) 92 [8].
Fuel Relocation	+313 °F	Results reported in technical paper by IPSN [9] using CATHARE for a Framatome PWR (similar to a Westinghouse 3-loop PWR). A burst zone 70% filling fraction assumed.
Code Uncertainty	+340 °F	W ΔPCT between 95th and 50th percentile uncertainty in a W 4-loop PWR for WCOBRA/TRAC calculation [10].
Code Uncertainty	+300 °F	Difference between the 95th and 50th percentile PCTs for a Westinghouse RESAR-3S plant using TRAC-PF1/MOD1 [11].
Code Uncertainty	> +275 °F	Framatome ANP large break code uncertainty using realistic version of RELAP [12]
Code Uncertainty	> +400 °F	GE code uncertainty using SAFER/GESTER [13]

Table 2: Small Break LOCA ΔPCT Estimates

Process	ΔPCT	Basis
Decay Heat	- 1000 °F	NRC contractor citation of CE sensitivity to decay heat using CE EM for CE 2700 MWt (Millstone 2) plant [3].
Decay Heat	- 859 °F	NRC contractor citation of W sensitivity EM to decay heat standard for CE 2700 MWt (Millstone 2) plant [3].
Decay Heat	-500 to -1000 °F	NRC contractor estimate based on RELAP5 calculations for typical plants [3].
Decay Heat + Metal Water Reaction	-500 °F	Calculations performed using a SBLOCA version of WCOBRA/ TRAC for Indian Point Unit 2 [14]. The ΔPCT is the difference between the limiting SBLOCA case in the paper and current plant (Appendix K based) analysis of record.
Metal Water Reaction	-11 to -76 °F	NRC calculations using RELAP with Baker-Just replaced by Cathcart-Pawel.
Fuel Relocation	Not known	Clad swell and rupture and fuel relocation may occur in SBLOCA. However, no calculations have been found documenting the effect.
Nodalization	+600 °F	NRC RELAP calculations w and w/o crossflow for CE 2700 MWt plant.
Operator Action	+ several 100 °F	Pump trip with off site power available depends on operator recognition and adherence to EOPs. This is a known post-TMI pump trip issue. Trip at inopportune time can cause deep uncovery.
Level Swell Uncertainty	+ several 100 °F	NRC contractor (verbal) estimate. Mixture level swell (code interfacial drag) is highly ranked PIRT process.
Loop Seal Clearance	+/- several 100 °F	Affects pressure drop through loop(s) and core level depression.

Recommendations:

A. Evaluation Models making use of a new, optional Appendix K should account for the non-conservatisms of downcomer boiling, downcomer ECC bypass, and fuel relocation.

B. These new Evaluation Models must demonstrate sufficient overall conservatism in their results.

Conclusions & Recommendations

1. Revise the 10 CFR 50.46 acceptance criteria for PCT and ECR to be "performance-based".
2. Replace 1971 ANS Decay Heat Standard with 1993 Standard
3. Replace the Baker-Just correlation with Cathcart-Pawel for metal-water reaction heat release.
4. Delete the requirement for steam cooling only at reflood rates below 1 inch/sec.
5. Retain the prohibition on assuming a return to nucleate boiling during blowdown.
6. Require that the new Evaluation Models to demonstrate sufficient overall conservatism and that they account for several identified non-conservatisms.

Planned Actions

1. Information presented will be documented in a "Research Information Letter" to NRR to provide a basis for pursuing new rulemaking consistent with SECY-01-0133 and SECY-02-0057.

2. RES is continuing to work with NRR to identify appropriate paths for removing unnecessary conservatisms in a new, optional Appendix K while insuring Appendix K retains sufficient conservative.

3. Public meeting is planned to discuss findings.

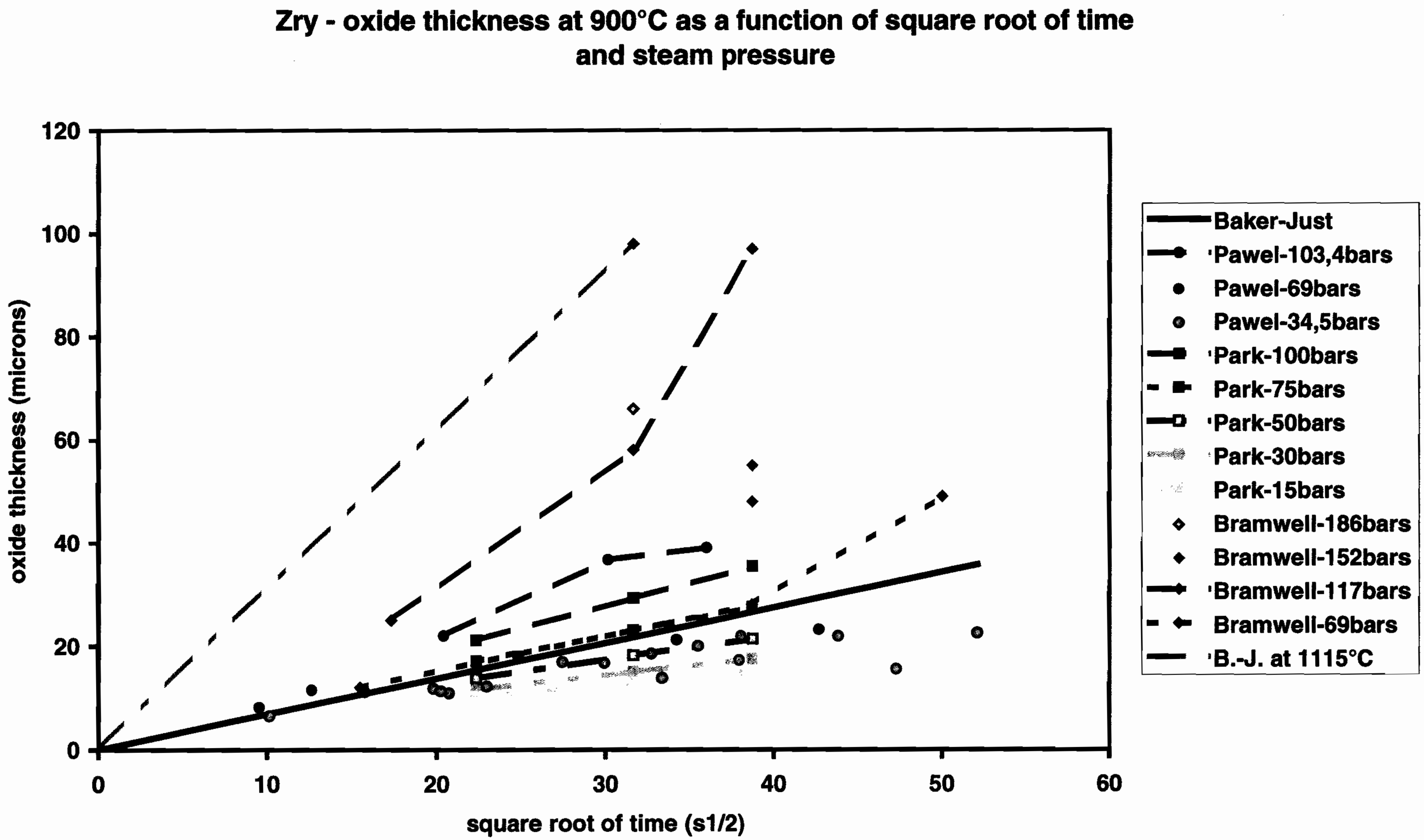
4. Continue the high burnup fuel research program, which is expected to provide supporting information on fuel relocation, cladding oxidation and cladding ductility/toughness following quench.

5. Resolve technical issues associated with 1994 ANS Decay Heat Standard uncertainty and user selected parameters.

Rulemaking Activities for
Risk-Informing 50.46 and Related Rules
(Samuel Lee, Policy & Rulemaking Program, NRR)

Changes Proposed in SECY-01-133 / SECY-02-0057		Related Industry Petitions and Interest	Staff Actions
1	Voluntary alternative to the technical requirements related to ECCS evaluation model	PRM 50-74 (Sept. 6, 2001): NEI filed petition for rulemaking that requested NRC to amend Appendix K to Part 50 to allow licensees to adopt 1994 revision of the ANS consensus standard for decay heat power, and to give licensees standing option to adopt any future NRC-endorsed revisions to the standard, without NRC having to conduct a separate rulemaking to incorporate by reference each subsequent revision.	Technical studies ongoing. Technical report to support rulemaking expected by July 2002. Upon completion of the technical report, staff plans to initiate rulemaking phase along with drafting relevant regulatory guides.
2	Voluntary alternative to ECCS acceptance criteria	PRM 50-71 (March 14, 2000): NEI filed petition for rulemaking that requested NRC to amend 50.46 to allow licensees to use zironium-based cladding materials other than zircaloy or ZIRLO, provided the cladding materials meet the requirements for fuel cladding performance and have received approval by the NRC staff. (Benefit: eliminate need for licensees to obtain exemptions to use advanced cladding materials that have already been approved by NRC)	Technical studies ongoing. Technical report to support rulemaking expected by July 2002. With respect to making the acceptance criteria performance-based, staff plans to initiate rulemaking phase along with drafting relevant regulatory guides once the related technical studies are completed. In the interim, the staff is considering a direct rulemaking to add fuel "M5" to the list of zirconium-based cladding materials in the current rule.

3	Voluntary risk-informed alternative to ECCS reliability requirements in GDC 35: Two options include (1) generic plant binning according ECCS accident mitigation reliability, (2) plant-specific assessment of ECCS accident mitigation reliability.	PRM 50-77 (May 2, 2002): Performance Technology filed petition for rulemaking that requested NRC to amend 50.46, Appendix A, GDC 17, "Electric Power Systems", to delete the requirement of assuming that offsite electrical power is not available for postulated accidents. (changes also proposed for GDC 35, 38, 41, and 44 to conform to GDC 17). (Stated benefit: Allows required EDG start time to be increased)	Draft technical report for plant-specific approach to assess ECCS safety function reliability was delivered on May 1, 2002. Technical work for generic approach is expected by July 2002. Complete technical work to support development of regulatory guide expected by July 2002. (Note: Completion of technical work is dependent on resolution/determination of LOCA frequency and conditional LOOP probability to be used for the analysis). Staff formed a working group to review the report on plant-specific approach and is developing an implementation plan for rulemaking.
4	LBLOCA redefinition	PRM50-75 (February 6, 2002): NEI filed petition for rulemaking that would allow alternate break size to currently required double ended rupture of largest pipe in RCS. (Stated benefit: "enable" technical discussions on redefining LBLOCA to proceed without being in conflict with current rules. Also may expedite schedule by up to two years).	Technical studies ongoing. Technical studies expected to be completed by July 2004. Staff is currently reviewing the petition. Rulemaking effort will follow accordingly.



Zry - oxide thickness at 900°C as a function of square root of time and steam pressure
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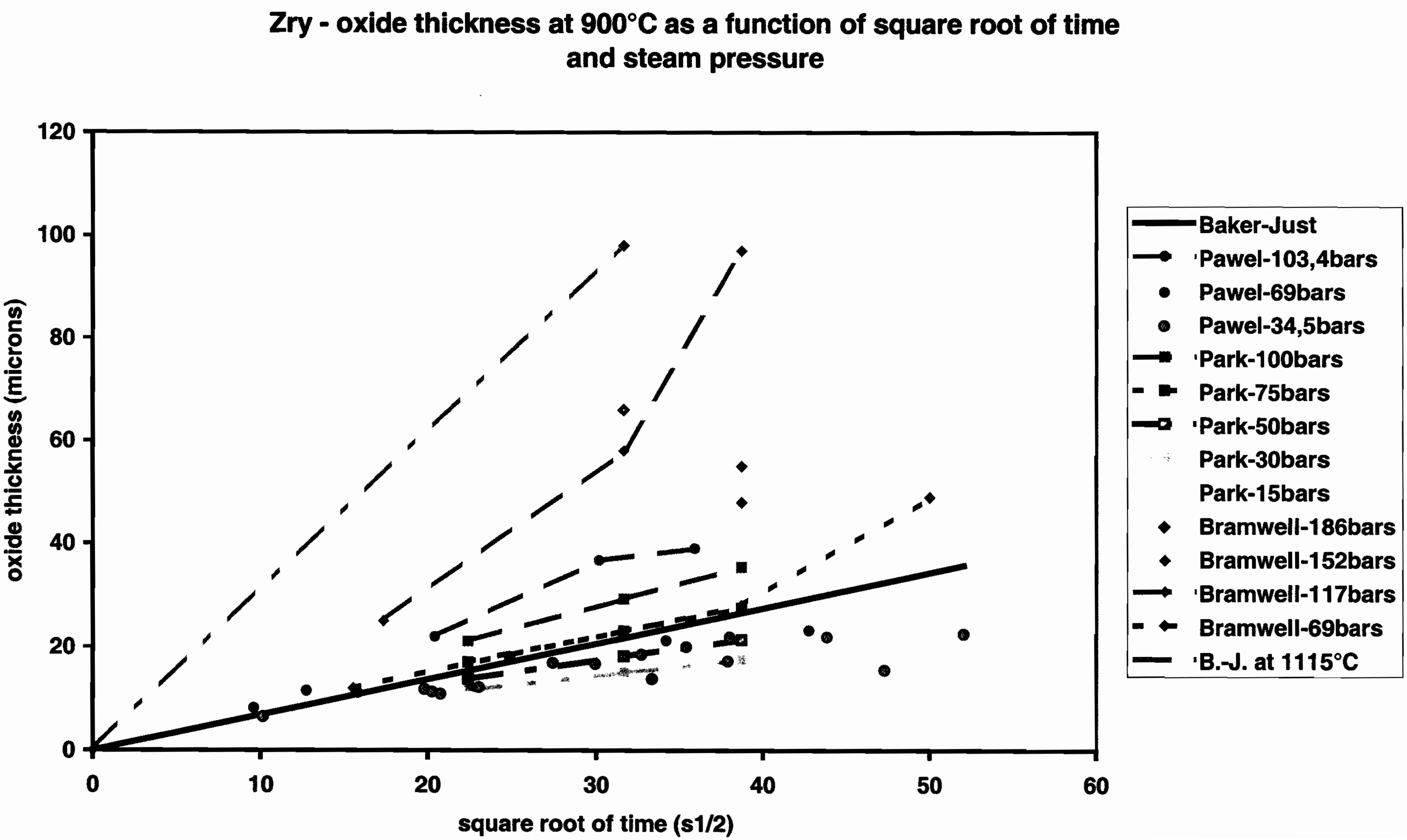

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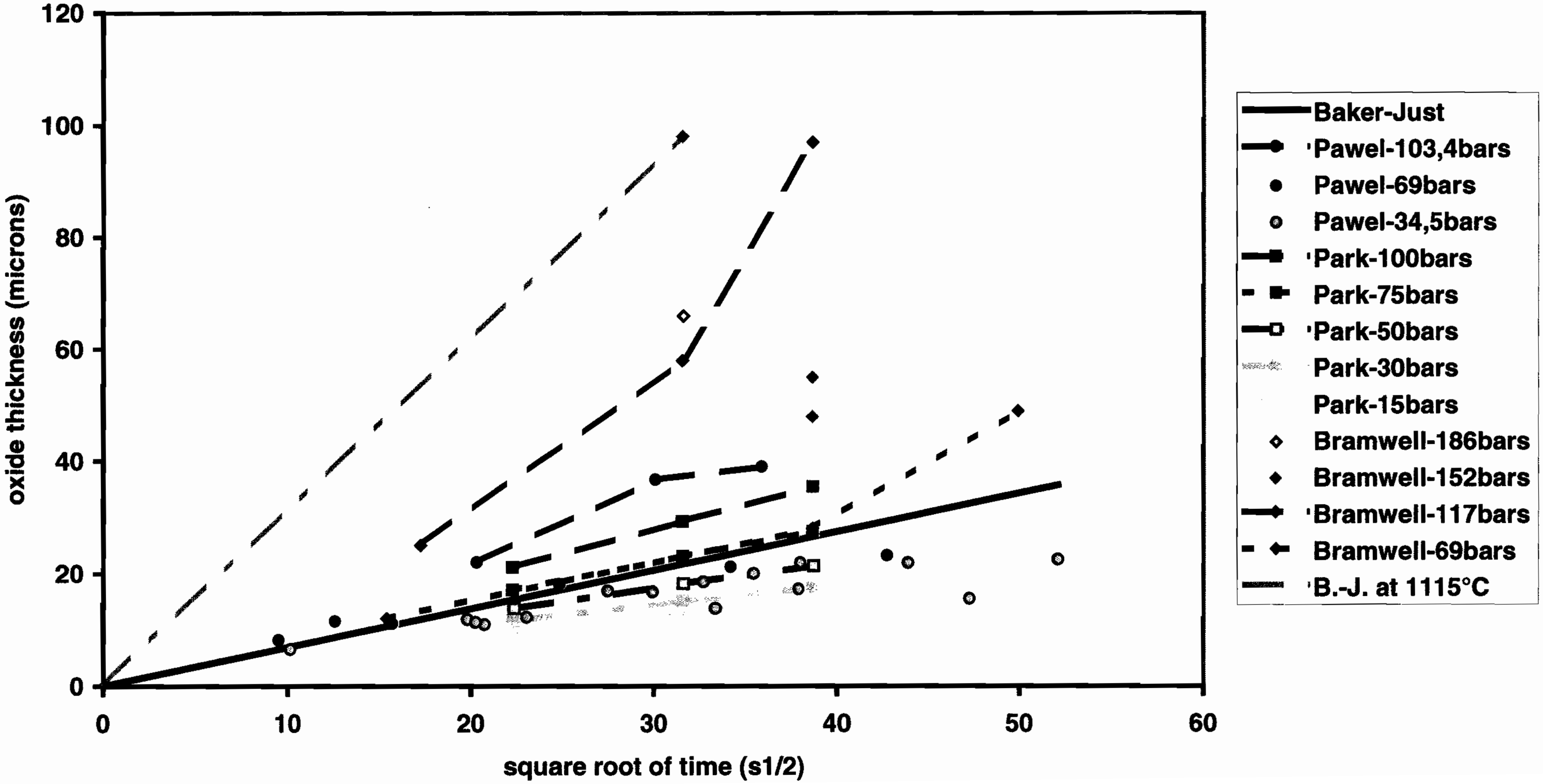

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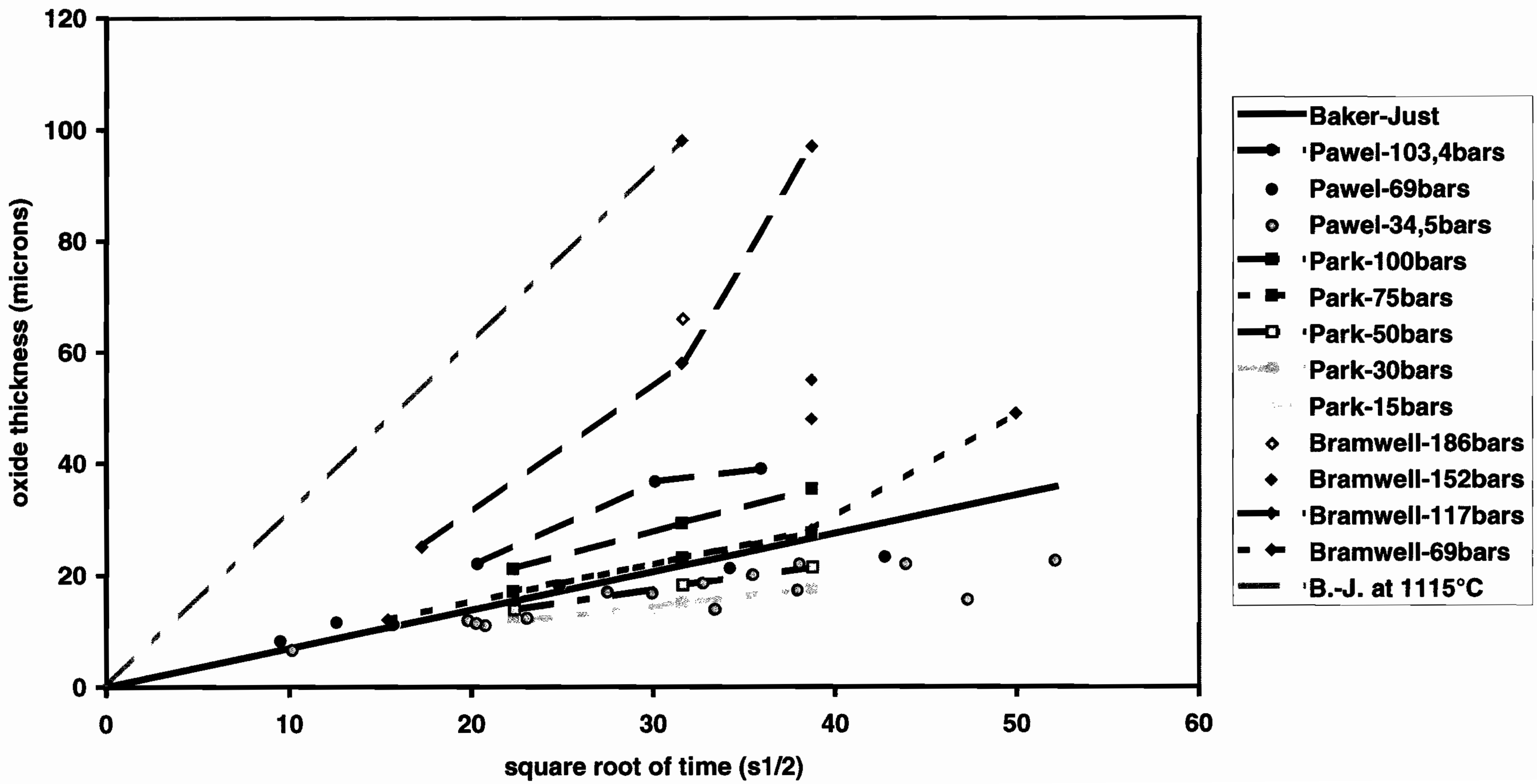

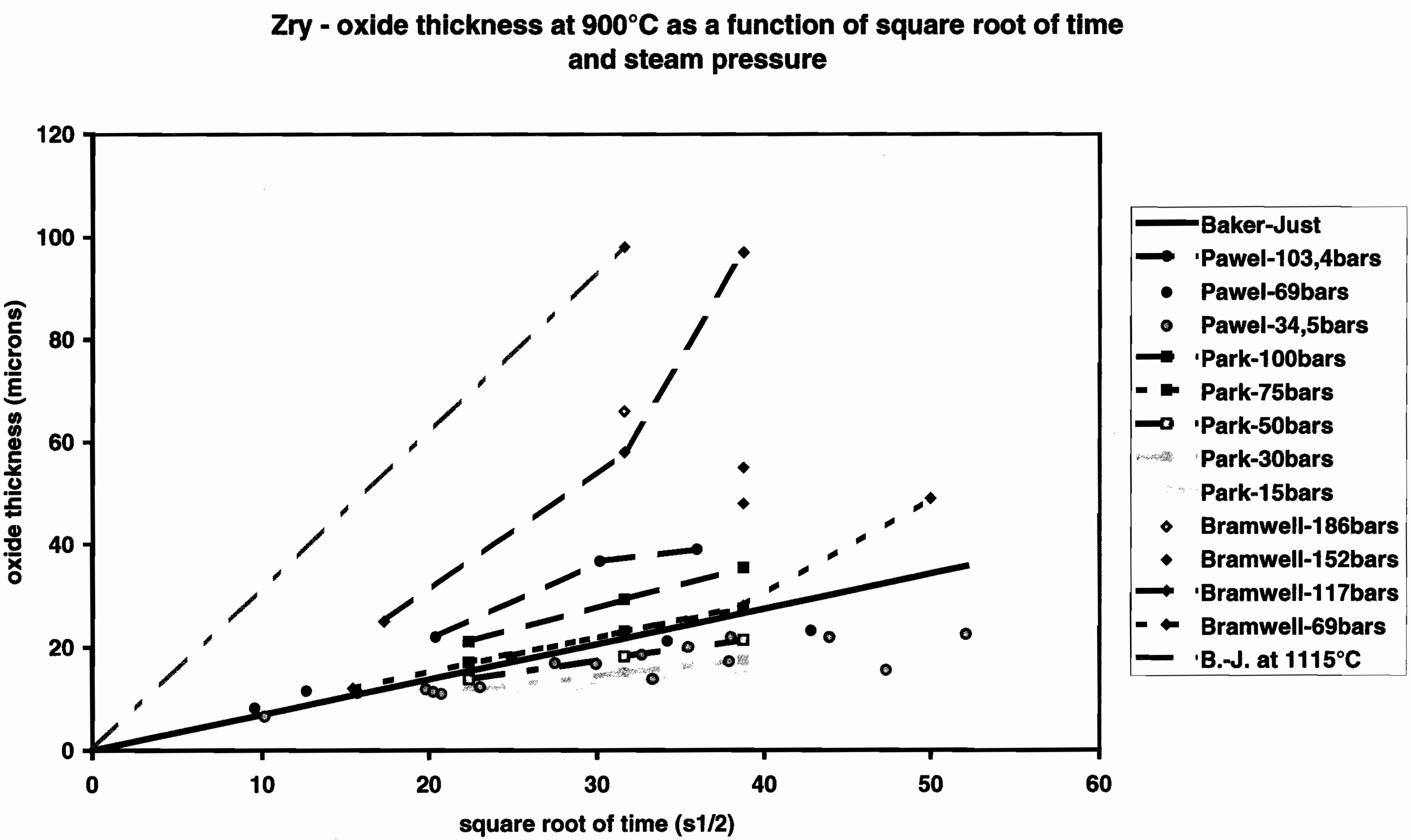
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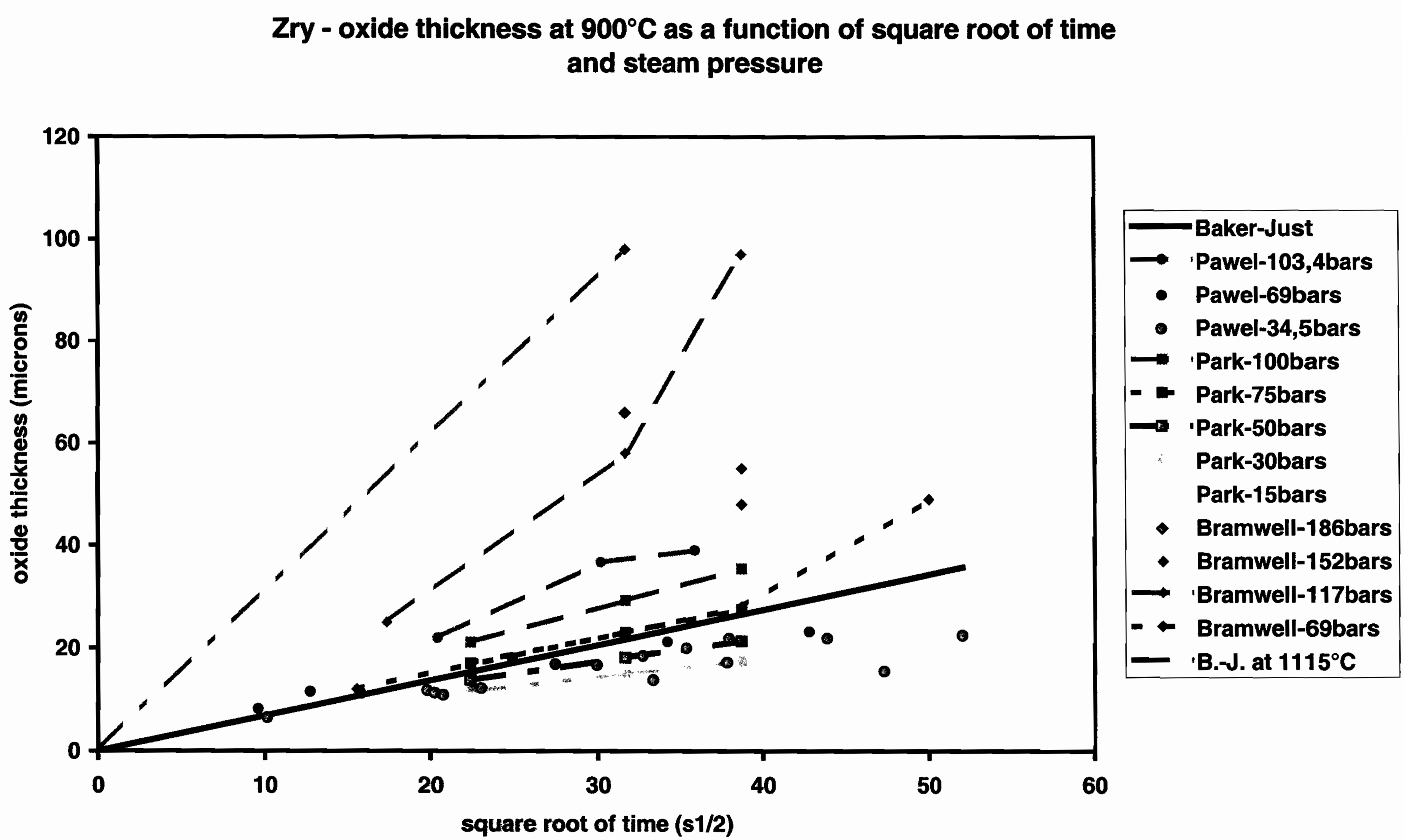
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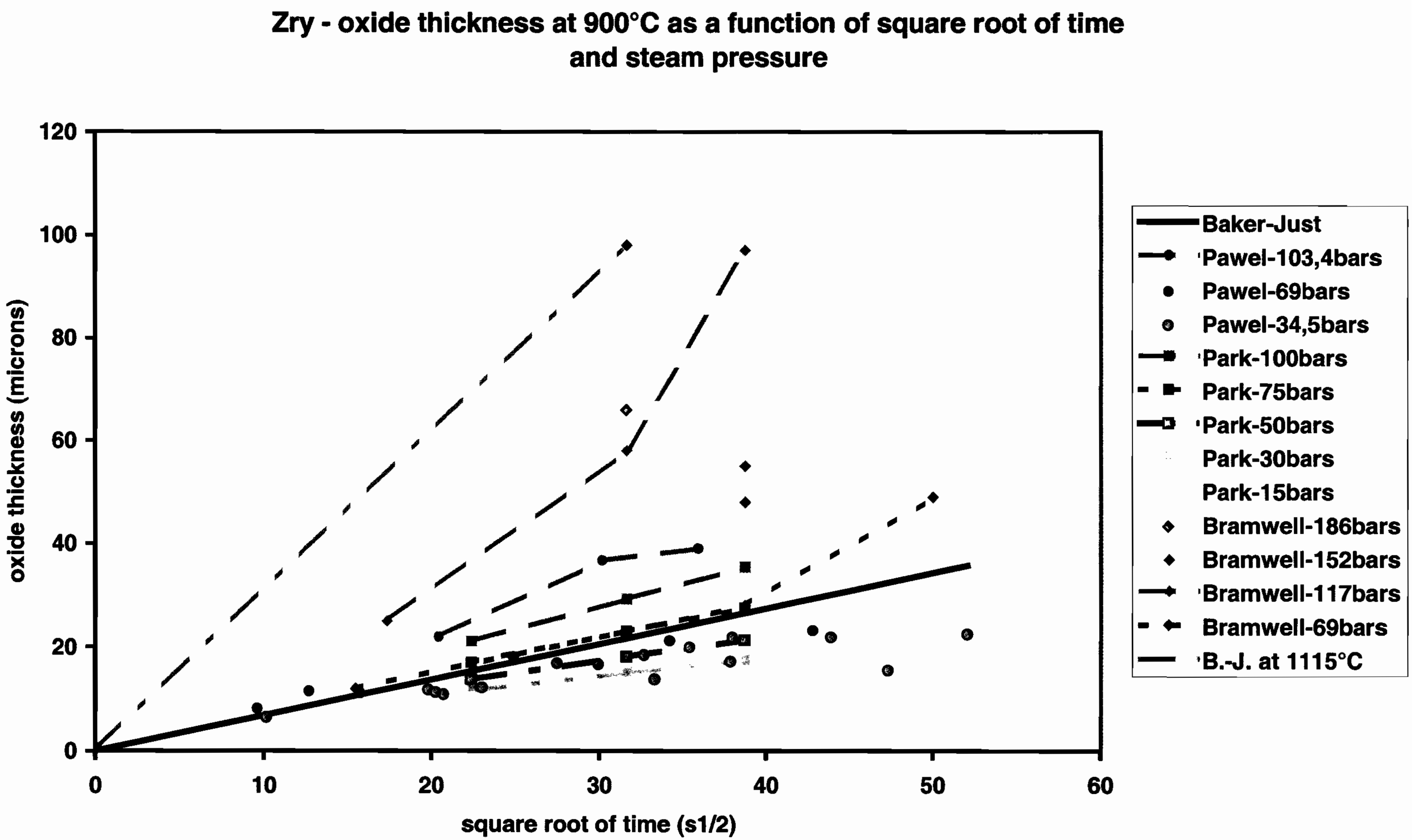
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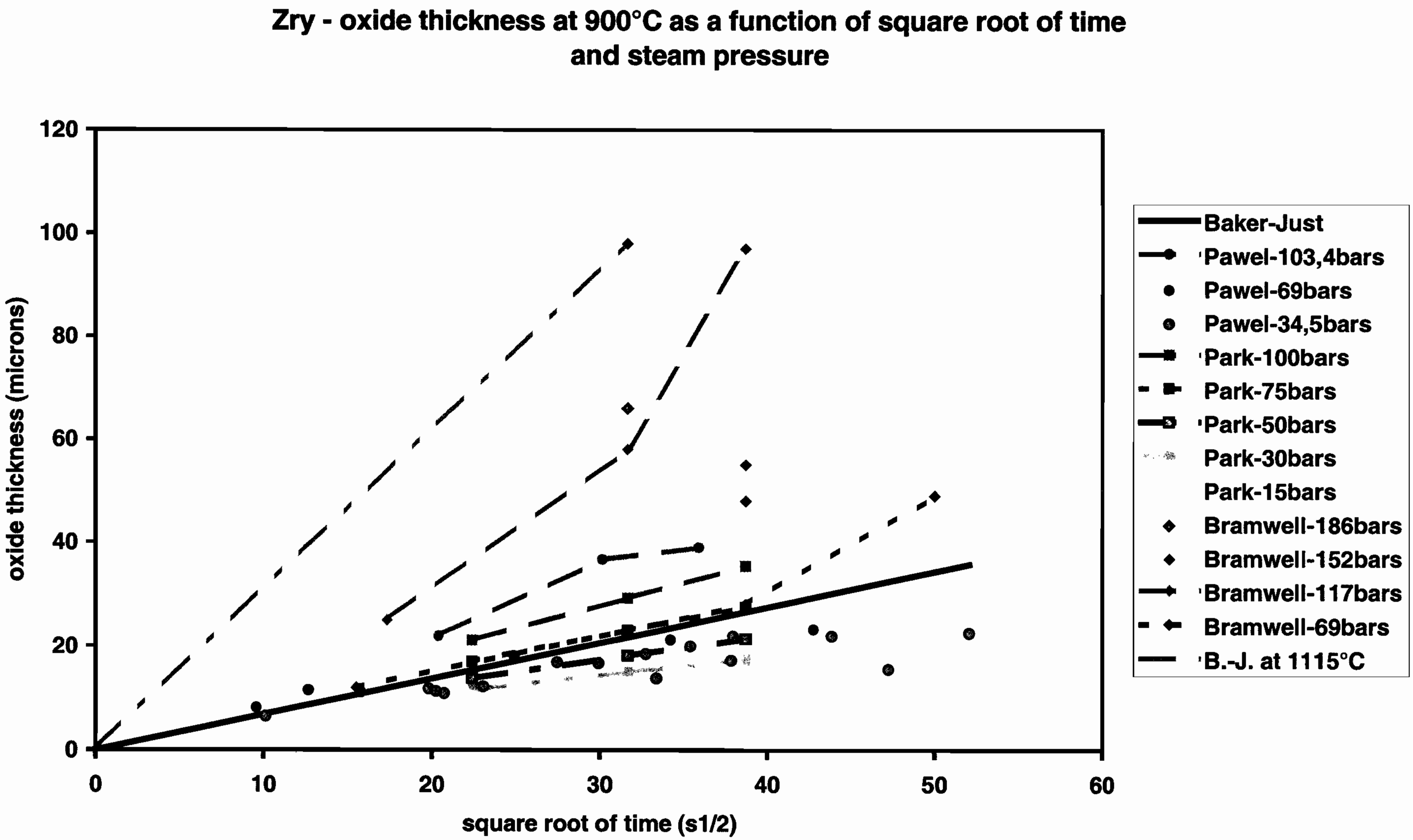

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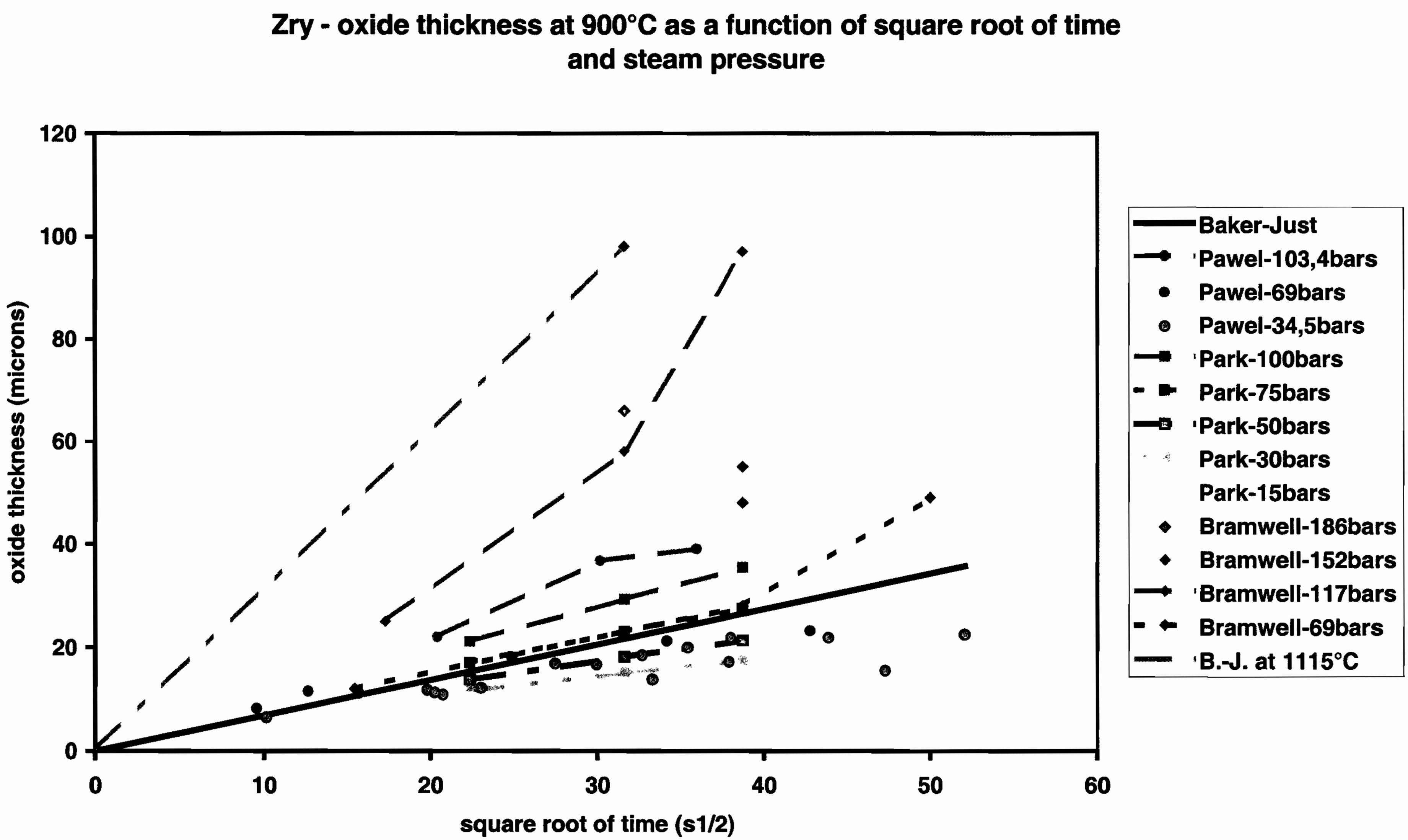

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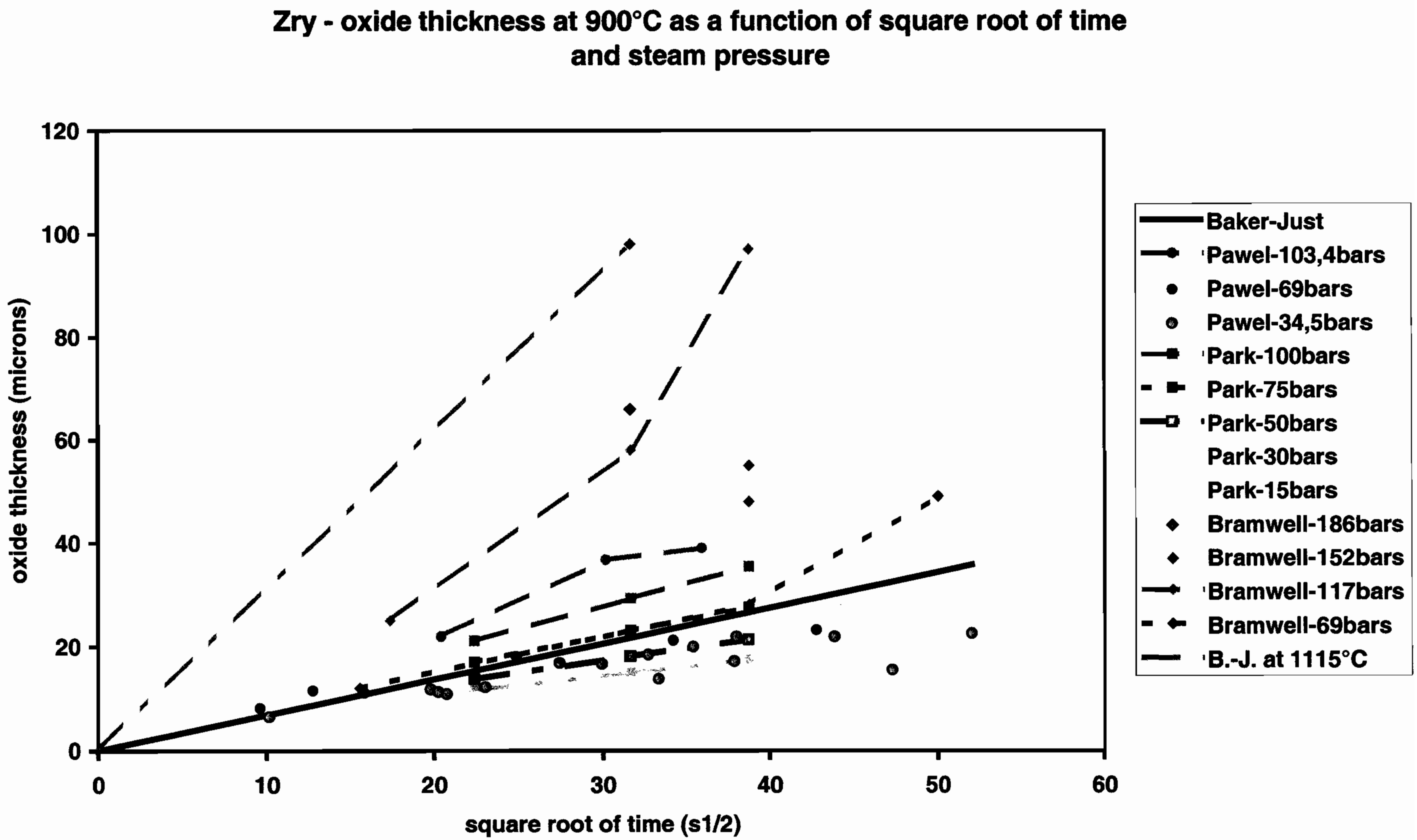

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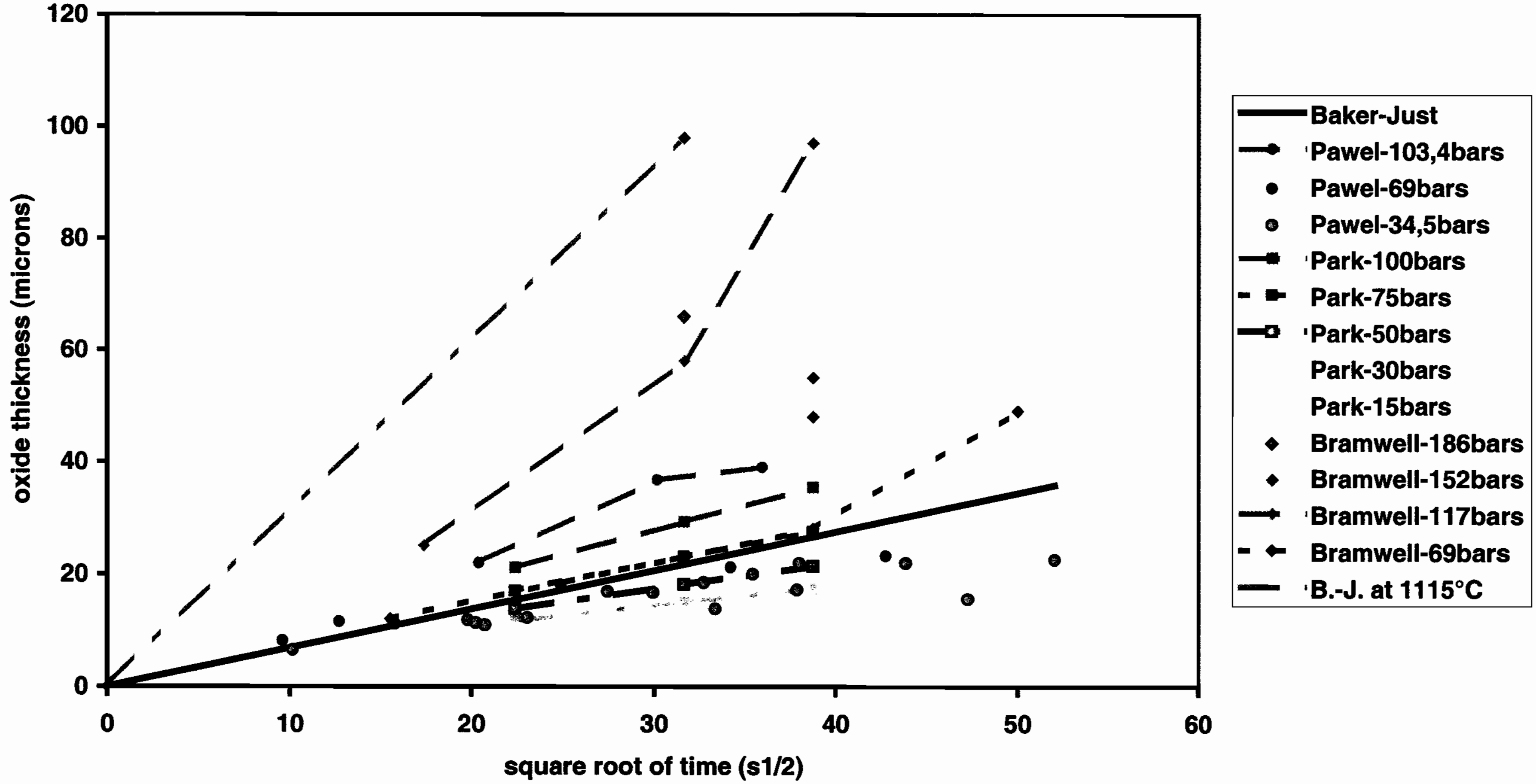

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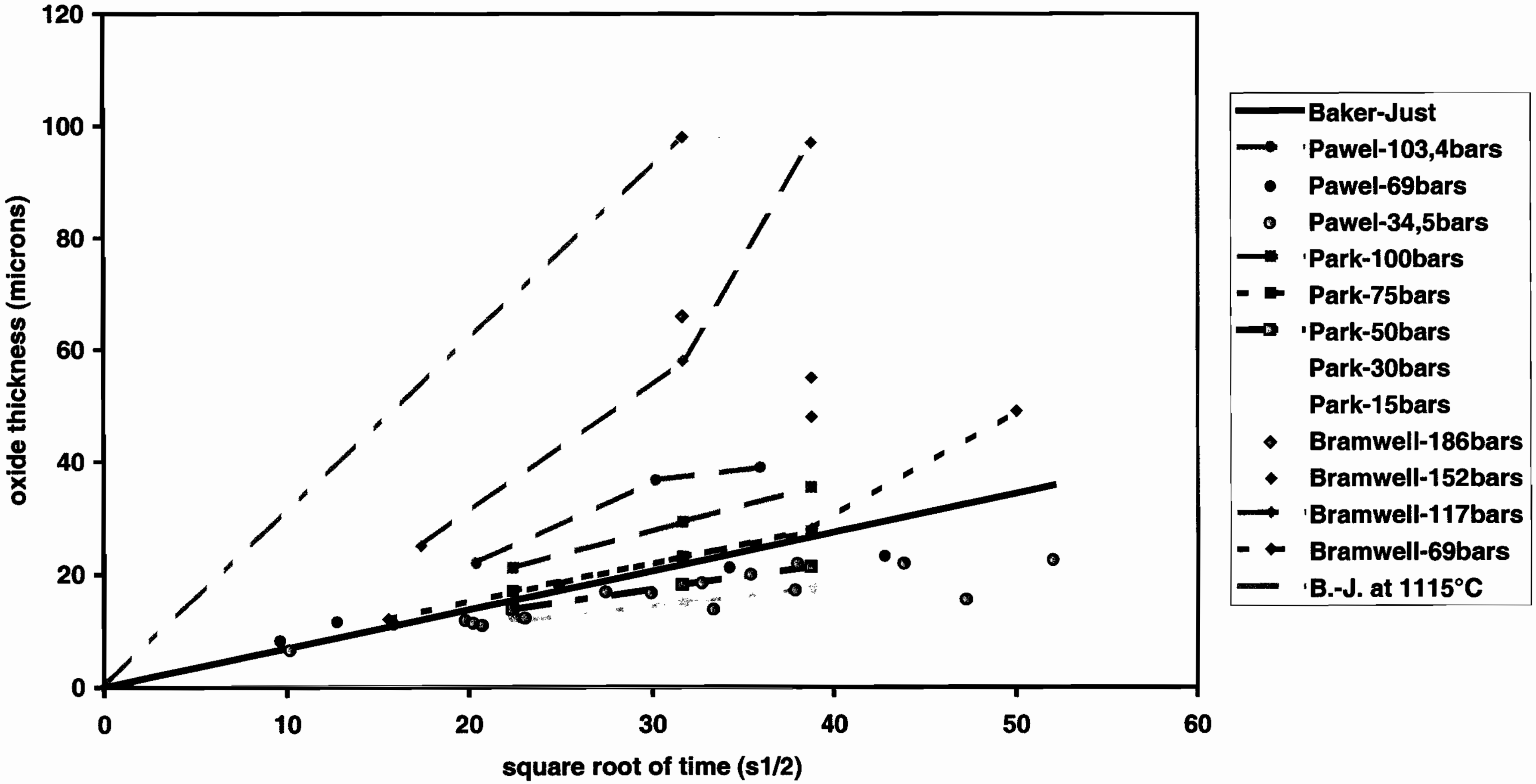

ATTACHMENTS

CONSULTANT REPORTS

Prof. Virgil E. Schrock (University of California-Berkeley), report dated June 5, 2002.

Prof. Sanjoy Banerjee (University of California-Berkeley), report dated May 31, 2002.

From: Virgil E. Schrock
258 Orchard Road
Orinda, California 94563
(925) 254-3252
schrock@nuc.berkeley.edu
virgilschrock@attbi.com

June 5, 2002

To: Dr. Graham B. Wallis
Chairman, Subcommittee on Thermal Hydraulic Phenomena
Advisory Committee on Reactor Safeguards

Via: Paul Boehnert
From: Virgil E. Schrock, Consultant

Subject: Joint Subcommittee Meeting on Risk Informing 10CFR50.46, May 31, 2002

In response to an industry petition NRC issued two SECY papers (SECY-01-0133 and SECY-02-0057) setting out the broad conditions /aims for rule-making to implement changes to Technical Requirements of 10CFR50 and Risk-Informed changes to 10CFR5.46 Acceptance Criteria. A staff team from NRR and RES has been working on a plan for the past year or more to develop the technical basis for such action.

EVALUATION MODEL CHANGES to 50.46 and APPENDIX K

In addition to the material distributed to the members and consultants for the meeting, I had previously received from Norm Lauben of RES, a draft copy of a document being prepared under the above title (evidently it will be an enclosure to a larger document). The Decay Power provision is probably the single most important issue addressed. Currently 10CFR50.46 (1988 Revision) has the options of using the original App. K requirements or using best estimate methodology with evaluated uncertainty. The original Appendix K requirement of 1.2 times the ANS draft Standard (1971/73) for "infinite reactor time" was a major source of conservatism in the rules for analysis of ECCS performance during LOCA. The best estimate option allowed (via RegGuide 1.157) the use of the 1979 ANS Standard for Decay Power. The 1979 Standard provided a sound technical basis for evaluation of decay power and its uncertainty. The Standard provides mathematical fits in the form of sums of exponentials (23 terms) to results evaluated from the combination of experimental measurements (calorimetric and radiometric) and summation calculations (codes like ORIGEN, CINDER, AND RIBD) using the Evaluated Nuclear Data File version ENDF BIV. It recognized and accounted for the many dependencies including:

1. Differences in fission yields among fuel nuclides which give rise to unique decay powers for each fissionable nuclide. The Standard provided data and uncertainties for 235-U, 238-U and 239-Pu and said that other contributors (notably 241-Pu) are to be treated as 235-U. Input for a

Standard calculation includes information on the partitioning of total fission rate among the contributors. Such information requires detailed knowledge of the evolution of the core composition in space and time (including initial data) such as developed for lattice physics and fuel cycle/management purposes. The Standard simply states that the relative fission rates for fissionable nuclides must be supplied and justified by the user.

2. The power history. The Standard suggested use of a histogram representation of the power history and gave guidance on acceptability of its fineness.

3. Total energy release associated with one fission event varies significantly for individual fuel nuclides and depends upon reactor specific material (fuel and structure) composition. The Standard requires the user to provide these parameters and to justify them.

4. Neutron absorption in fission products causes transmutations which increase the complexity of the radio nuclide mix and produces a small increase in decay power. The Standard treats this phenomenon by a small time dependent correction factor which depends upon the number of fissions per initial fissile atom and the reactor operating time. The Standard provides a choice between three options for this: a. an empirical formula including dependence on reactor operating time , b. a table of conservative values (based on four year operating time) or c. a first principle (implied) calculation and its justification provided by the user of the Standard.

5. The Standard states that it does not cover the issue of spatial distribution of deposition of decay power in the shutdown reactor. The Standard relates local power density in the shutdown reactor to local power density history during operation. It points out that the radiation spectra in the shutdown reactor is time dependent (gamma spreading phenomenon is of greater importance in the shut down reactor than during operation). Not mentioned in this regard is the fact that fuel composition is dependent upon time and position, thus impacting the user supplied data required under items 1 and 2 above.

6. The Standard provides equations for the evaluation of decay power from actinides 239-U and 239Np, the conversion products not included in the fission product source. These are the principle actinides of importance in LOCA analysis. "Other actinides" may be significant in some applications. The Standard requires that their contribution be evaluated, justified and included in the total actinide contribution to be added to the decay power from fission products. Reg Guide 1.157 gave no guidance on what the staff would find acceptable for dealing with the user supplied data/requirements. In Norm Lauben's presentation these requirements were referred to as "options".

So far the potential for more exacting decay power evaluations using the Standard method has not been fully utilized for LOCA analysis. There seems to be a mind set, perhaps carried over from the simplicity of the 1971/73 Standard single curve for infinite operating time. The complexities of the true physics acknowledged in the 1979 Standard have been largely viewed as too great a burden for licensing calculations and the best estimate methodologies so far put forth by industry all are short cut approaches that leave the uncertainty just that, uncertain.

I have recently become aware of Reg. Guide 3.54 (1984 and Rev. 1, 1999) which deals with Spent Fuel Heat Generation. Surprisingly that Guide is far more sophisticated (than 1.157) in terms of understanding the physics and in using the Standard in a rather complex evaluation of decay power for individual fuel bundles. "Other Actinides " play an increasingly important role in waste storage applications.

The ANS Standard was revised in 1994 with newly developed standard functions for 241-Pu included and with updated data for the existing nuclides based upon new experiments and summation calculations using improvements in the Evaluated Nuclear Data File. The structure of the 1994 Standard is the same as that of 1979 version. Shortly after its release I recommended that NRC should use it in the regulatory process. In fact, I think there is need for an ongoing program to keep the agency current with developing standards and other technological developments. The staff is now recommending this approach in connection with new rule making.

Regarding decay heat, the staff is recommending that 10CFR50.46, Appendix K be changed to include a prescriptive (implied conservative) method based on the 1994 ANS Standard. The original Appendix K requirement would again be grandfathered. The new "model" conditions are listed on Norm Lauben's slide 31 as:

1. 235-U is the only fissioning isotope
2. Infinite operating time
3. 200 MeV per fission
4. Restrictions on the use of the Standard equations for the neutron capture effect for times less than 104 sec..
5. Use the Standard Table 13 for t > 104 sec.
6. Use the Standard formulae for 239-U and 239-Np decay with R = 0.70
7. Use the 2 sigma value of uncertainty for 235-U.

This is intended to produce a bounding or conservative result with the pedigree of being an offspring of the 1994 ANS Standard. Norm Lauben presented comparisons of results for a variety of assumptions in support of the "model" choice. I think there remains a question as to whether it is bounding because 238-U products have significantly higher decay power than 235-U for the first 200 seconds and 238-U has a much higher uncertainty than 235-U. In addition the uncertainties are much larger for both Pu isotopes tending to offset their lower nominal values. In his slide 25 it appears that curve 3a corresponds to the proposed new Appendix K model (100% 235-U fissioning). Case 2 assumes 10% of fissions are in 238-U and the curve is a little above curve 3a. This seems to confirm that the model is slightly non conservative in the early time.

Another facet explored in the staff work is the issue they raised earlier concerning the 1979/1994 ANS Standard method for combining uncertainties of the various contributing fuel nuclides. In the Standard, the dimensional uncertainties are added whereas the NRC staff statisticians reviewing the Standard suggested that an rms combination would be more appropriate. The RES contractor at PNL, Dr. Brady-Rapp who is also the current Chair of the ANS5.1 Working Group on Decay Power, was inclined to agree with the rms approach. However, after consulting members of the WG who developed the 1979 Standard and 1994 revision, I believe she is convinced that the straight summation is more appropriate. Dr. Kirk Dickens (now retired from ORNL), Dr. Robert Schenter of PNL, Dr. Frank Schmittroth, PNL and Dr. Tal England (now retired from LANL) were principle architects of the uncertainty evaluations. As Dickens points out, the underlying fission-product data have substantial correlations , and so the uncertainties need to reflect the known correlations and estimates for the unknown correlations. The only correct method would be to report complete variance-covariance matrices, but these would completely overwhelm the Standard and very likely not be used or useful. So the uncertainties are not uncorrelated nor are they fully correlated. I think this means that there is some conservatism built into the Standard method. On the other hand the proposed rms combination would definitely underestimate the uncertainty (be non conservative). I understand that the matter will be clarified in the next Standard revision which is to be finalized at the ANS meeting June 9 in Florida. In any case, the impact on the result is quite small, as shown in Lauben's comparisons (case 2 vs. case 3). The comparisons also include some best estimate uses of the 1994 Standard and calculations done using ORIGEN, a summation code, with the current ENDF-B data. As expected, ORIGEN and the 1994 Standard are in close agreement since ORIGEN was one of the bases for the 1979/1994 Standard. The Best Estimate results are lower than the proposed Appendix K model by more than 10%. I would like to see more detail on the best estimate calculations in order to better understand how they were done. Specifically there is no comment on evolution of spatial variations in fuel composition. The calculations are described as for PWR and BWR fuel bundles with 3 typical cycles. ORIGEN uses a point reactor model and it is evidently coupled to SAS2D to in some way describe bundle average conditions in relation to core wide average conditions. This type of calculation is important to the best estimate application of the 1994 ANS Standard and needs to be more fully explained.

What does SAS2D supply to ORIGEN?

The Staff also recommends adding another Appendix K option to allow use of a subsequent standard (as new versions are developed) and changing the reference in Reg Guide 1.157 from the 1979 Standard to the 1994 Standard for best estimate applications. The decay power conservatism given up by the proposed change to Appendix K is of concern to the staff in the context of the overall conservatism.

Summary Comments on Decay Power

I think that RES and Norm Lauben deserve praise for their work in reviewing the decay power issue and development of information for proposed changes to Appendix K. The proposed change has technical merit although it may need some polishing to be sure it is bounding. It is a reasonable response to SECY 01-0133. The staff is obviously not enthusiastic about the proposal and is concerned to tie it to the question of overall conservatism. They think, perhaps hope, that nobody will choose to use the new option when it is so tied. I think this could backfire. It would be far better to recognize that Appendix K was not an optimum basis for licensing reactors. Make clear that it has been retained as an option only to sustain existing licensing commitments and devote developmental resources to the best estimate approach. I would strongly recommend setting a time limit on the continued grandfathering of Appendix K. I have said this many times before.

The staff work on decay power is very encouraging for the potential to at last find the right way to use the technical basis of the ANS Standard in best estimate analysis. I understand that

Norm Lauben is now a member of the ANS 5.1 Working Group on decay power. I think that should help. Spatial dependence is a key issue. Also how the knowledge of decay power is applied in the accident analysis codes needs to be reexamined.

Other Appendix K Models and Overall Conservatism

Baker-Just Metal-Water Reaction: The proposal is to replace Baker-Just with Cathcart-Pawel correlation. There seems to be sufficient technical evidence to support this change. It is estimated that the result would be a reduction in PCT predictions of between 50 and 100 deg. F.

Known Non-Conservatisms - Downcomer Boiling, Reflood Bypass, Fuel Relocation, etc. : These (at least some) appear to be real phenomena that are not treated by the current Appendix K requirements. Studies show that they have the potential to increase LBLOCA PCT by several hundred degrees F. The staff feels that the old decay heat conservatisms cover known and perhaps unknown conservatisms. They want new Evaluation Models to demonstrate sufficient conservatism and require that they account for identified non-conservatisms. This seems like a real can of worms to me. One might ask "why not simply add new requirements to eliminate known non-conservatisms". It is not well justified technically. "Sufficient conservatism" is ill defined. It is likely to be a playground for interveners should they get involved in new rulemaking. As I have already said, it will be far better to relegate the old Evaluation Model concept to history and move ahead with realistic or best estimate analysis in the licensing process. The information the staff has developed should be very useful in showing industry why it is not reasonable to chip away at individual EM conservatisms by revising the rules.

Concerning "Planned Actions", documentation is important. Some of the contractor supplied information is now in an undocumented form. For example, the PNL work on decay power ought to be developed into a NUREG/CR report meeting appropriate standards of completeness and clarity. I have already discussed issues associated with the 1994 Decay Heat Standard uncertainty and required user suplied information and justification. I think the staff is wrong in viewing the latter as "user selected parameters" or user "options". These matters are of greatest concern for implementation of the ANS Standard in best estimate analysis.

REPORT ON ACRS SUBCOMMITTEE MEETING

May 31, 2002

Sanjoy Banerjee

I. Introduction

The ACRS Subcommittees on Thermalhydraulic Phenomena, Reliability and Probabilistic Risk Assessment, and Materials and Metallurgy, met on May 31, 2002 to review NRC staff technical work related to risk-informing 10CFR 50.46 (including Appendix K and GDC35—all of which address ECCS for LWRs).

The technical work was meant to support rulemaking related to risk informing

— ECCS reliability requirements, both for generic and plant-specific approaches

— ECCS acceptance criteria

— ECCS evaluation model

The quality of the presentations and the supporting material were good, and the meeting was very useful.

II. General Comments

1) I had the impression that the work had been conducted in response to industry petitions to reduce conservatisms with regard to the various aspects of ECCS, viz. reliability requirements, acceptance criteria and the evaluation model. The NRC staff presented the pros and cons in terms of what had been found out since the original rulemaking. The picture, however, is mixed, since some conservatisms, such as in decay heat level assumptions, can be reduced, but others must be introduced, e.g., boiling in downcomers and ECC bypass. The net benefit for industry may, therefore, be quite small.

The staff clearly recognized this situation, and it is well explained in an enclosure (Thadani to Collins) to Eltawila's May 23, 2002 letter to Larkins conveying supporting documents for the ACRS Subcommittee meeting.

2) Why all this work was necessary (other than in response to industry) was not clear. The best estimate route (with allowance for uncertainties) already exists, and should assumptions that are allowed need modification, then it would be best to address them via this methodology. The current prescriptive framework incorporates many safety factors to compensate for things unknown at the time of the original rulemaking. Reducing these safety factors piecemeal could lead to a situation where many new safety factors would have to be introduced and the whole ad hoc, patched-together structure then defended in front of the public. Some of the conservatism reductions, particularly with regard to LOOP, LOCA break sizes and reflood heat transfer, may be very difficult to defend.

III. Specific Comments

1) I would like to have more quantitative information with regard to the effect of deregulation and weather (hot summers, etc.) on probability of LOOP with LOCA. The BNL report (BNL-NUREG-5228), on which much of the information provided on this subject is based, dates from July 1997—well before the California brownouts.

I would also like to know whether offsite power reliability would be monitored on a continuing basis for each plant and, should conditions deteriorate, whether changes in required equipment, e.g., bringing EDGs back into service, would be enforced. The presentations did not make clear how changes in LOOP/LOCA probability during the life of a plant would be dealt with.

2) With regard to the ECCS spectrum of break sizes and locations, it is clear that the current requirement to analyze pipe breaks with sizes up to double-ended guillotine breaks for the largest pipes, serves as surrogates for all sorts of possible accidents. With the current situation, where we receive a surprise every few months (David Besse and Hamaoka 1, to name a couple), it is hard to see how any reliable

predictions regarding break size and location can be feasible. In this context, where almost all the recent events are "unexpected", I have very little faith in the elicitation process.

I have also heard that Hamaoka 1 developed a pressure-vessel-bottom (CRD) leak that was not found during inspection. We were not told about this at the meeting—indeed, almost nothing was said even about the Hamaoka hydrogen explosion (which was almost certainly a detonation). I would like to have more information on both these aspects as they may impact the requirements for thermalhydraulic analyses.

3) With regard to the ECCS evaluation model, changing from Baker-Just to Cathcart-Pawel appears justified, but I do not agree that the PCT should be increased to 2300 ⊠F. This may or may not be justified. In general, the criterion for a runaway chemical reaction is

$$T_R - T_e = R\,T_R^2 / E_A$$

where T_R is the reaction temperature, T_C is the coolant temperature, R is the ideal gas constant, and E_A is the activation energy for the reaction. (The derivation is straightforward for an exponential reaction rate). In evaluating whether PCT should be revised upwards, it is the activation energy that is important, not the heat production rate. I would prefer to see an analytical approach to the runaway problem, rather than parametric studies with RELAP, which is what appears to have been conducted. The work we did with the AP600 indicated questionable reliability of RELAP heat transfer rates and core velocities at low pressures and flows.

4) There also appears to be an inconsistency in accepting 17% oxidation when Cathcart-Pawel is used. The original tests used Baker-Just to calculate the 17% limit. If Cathcart-Pawel is substituted, then the amount allowed should be correspondingly reduced. If this criterion was entirely removed and substituted with performance-based criteria, then it would be acceptable.

5) I am uncertain with regard to the removal of the restriction on "steam

cooling only" for reflood rates of up to one inch per second. This is the sort of thing that can be done for a best-estimate calculation, but is more problematic in the prescriptive framework. I would like to review more detailed evidence than was presented at the meeting before making a recommendation.

IV. Summary

As mentioned earlier, I am strongly in favor of "risk-informing" the best estimate route, rather than trying this with the prescriptive framework. Assuming, however, that something has to be done about the prescriptive framework, I suggest:

1) Further analysis of the PCT and oxidation limit criteria before they are changed (see Specific Comments 3 and 4).

2) Further review of the steam cooling-only requirement for reflood rates of less than one inch per second (see Specific Comment 5).

3) A clear strategy be put forward for dealing with changes in offsite power reliability during the life of a plant before revising the rule related to LOOP (see Specific Comment 1).

4) Reducing the priority of work on LOCA break size and location, as predictions of probability are likely to have low accuracy and credibility in view of recent surprises (see Specific comment 2).

5) The proposed changes in decay heat level and the Baker-Just correlation for metal-water reaction are justified, provided correlation of ECC bypass and downcomer boiling are included.

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The equation is:

$$T_R - T_C = R T_R^2 / E_A$$

T_R – reaction temperature (temp at which runaway initiates) °K

T_C – coolant temperature °K

R – ideal gas constant

E_A – activation energy for reaction