NUREG - 1230 title: Compendium....Realistic.....
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CSNI SOAR Dec 86 http://www.nea.fr/html/nsd/docs/1986/csni86-129.pdf
Based on ND-R-1351 Sept 86

NUREG - 1230 chapter 6.13 Metal-water reaction
recommends Cathcart-Pawel

NUREG - 1230 chapter 6.14 Fuel rod performance
recommends >0.3 mm thickness w < 0.7 % O

NUREG - 1230 chapter 8.1 Conservatisms in Existing ECCS Rule

Chung & Kassner report NUREG/CR-1344 date
data points near 17% ECR and near 0.3 mm thickness

Williford report NUREG/CR-4412 April 86

Hache & Chung paper NSRC Oct 2000 (+ Aix meeting Mar 01 & LOCA PIRT Dec 01)
plot of Hobson data

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** SEE INSTRUCTIONS ON THE REVERSE	1 REPORT NUMBER (Assigned by TIDC, add Vol. No., if any) NUREG-1230
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13. ABSTRACT (200 words or less)

Emergency Core Cooling Systems (ECCS) are required on all light water reactors (LWRs) in the United States to provide cooling of the reactor core in the event of a break in the reactor piping. These accidents are called loss-of-coolant accidents (LOCA), and they range from small leaks to a postulated full break of the largest pipe in the reactor cooling system. Federal government regulations require that calculations of the LOCA be performed to show that the ECCS will maintain fuel rod cladding temperatures, cladding oxidation, and hydrogen production within certain limits. The Nuclear Regulatory Commission (NRC) and others have completed an extensive investigation of fuel rod behavior and ECCS performance. The technology has been advanced to the point that it is now possible to make a realistic estimate of ECCS performance during a LOCA and to quantify the uncertainty of this calculation. This report serves as a general reference for ECCS research. The report (1) summarizes the understanding of LOCA phenomena in 1974, (2) reviews experimental and analytical programs developed to address the phenomena, (3) describes best-estimate computer codes developed by the NRC, (4) discusses the salient technical aspects of LOCA phenomena and our current understanding of them, (5) discusses probabilistic risk studies and (6) examines the impact of research on the ECCS regulations.

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FOREWORD

Emergency Core Cooling System (ECCS) design, analysis, and research has progressed steadily over the past twenty years. We have seen relatively simple computer codes developed at Bettis Atomic Power Laboratory mature into much more sophisticated predictive tools, such as the Transient Reactor Analysis Code (TRAC). Simple experiments involving a one foot heated core have now been supplemented with data from a full-size pressurized water reactor simulator.

As the data and analysis methods advance, so must the regulator. In the early 1970s, regulatory staff were constrained to making highly conservative assumptions in their analyses of loss-of-coolant-accidents (LOCA) because of the limited experimental and analytical data base. This resulted in very stringent licensing requirements. In 1988, after a strenuous (and expensive) era of data-gathering and model-building, it now appears that regulation should move along the lines of best-estimate assessment. Thus in step with the technical developments, the Nuclear Regulatory Commission (NRC) has revised the ECCS rule contained in 10 CFR 50.46 and Appendix K (published in the Federal Register 53 FR 35996) to permit more realistic analyses of ECCS performance. However in keeping with the admonition by the Commission in its 1973 issuance of Appendix K, we propose to retain a margin of safety by a requirement that calculational uncertainties be explicitly taken into account. This together with the demonstrated safety margin that will be retained in the 10 CFR 50.46 limits will provide a suitably conservative regulatory posture.

In addition to the rule, a regulatory guide, "Best-Estimate Calculations of Emergency Core Cooling System Performance," has also been published. The guide indicates features of ECCS evaluation codes that are acceptable to produce realistic calculations of ECCS performance, it lists models, data and model evaluation procedures that are acceptable to NRC staff, and it describes methods acceptable to NRC staff for performing uncertainty evaluations.

An important regulatory change such as this requires a technical basis. This document contains a distillation of the research that, taken as a whole, is the technical basis.

We gratefully acknowledge our contractors, the Electric Power Research Institute, our international associates, and the many dedicated researchers who have advanced our understanding of ECCS performance.

Denwood F Ross

Dr. Denwood F. Ross
Deputy Director for Research

The majority of this LOCA research is complete and has greatly improved the understanding of ECCS performance during a LOCA. The methods specified in Appendix K are now known to be highly conservative; that is, the actual temperatures during a LOCA would be much less than the temperatures calculated using Appendix K methods. This fact is illustrated by comparisons showing temperatures during LOCA simulations in LOFT (Reference 2-2) that are more than 600°F lower than the temperature calculated using Appendix K procedures as shown in Figure 2-1. The ECCS research has gone beyond showing that Appendix K is conservative; it has allowed quantification of that conservatism. The results of experiments, computer code development, and code assessment now allow more accurate calculations of ECCS performance during a LOCA, along with reasonable estimates of uncertainty.

2.2 Purpose of the ECCS Compendium

The data from twelve years of research since the ECCS hearing record was established is widely scattered in technical reports, meeting transactions, and professional journals. In view of the large amount of data involved and its dispersion, justification for changes to Appendix K of 10 CFR Part 50 is simplified if the most important and pertinent data justifying those changes are consolidated in a single report. Therefore, the purpose of the compendium is to summarize this data and to serve as a general reference for this extensive research effort. This is the first time such an extensive summary compilation has been attempted for reactor thermal-hydraulics.

2.3 Historical Perspective

Safety research for nuclear reactor plants has seen a succession of phases, each of which reflected the level of engineering and understanding at the time and led to additional research. At first, about twenty-five years ago, programs such as the SPERT experiments in Idaho were undertaken because of concern about the ability to control the neutron chain reaction in extreme circumstances. This focus on the possibility of what were called reactivity-induced accidents was a holdover from the technical perceptions in early days of nuclear weapons design reinforced by two accidents in critical experiments that killed the experimenters, by an accident at the experimental breeder reactor EBR-I, and by the fatal accident at the SL-1 reactor. Research on dynamics of the neutron chain reaction led to design principles that can be followed to prevent damaging reactivity-induced accidents.

A second phase of safety research was started after the issuance of WASH-740 in 1956, which reported the results of the first attempt to estimate what would be the effects of a severe accident in a nuclear power reactor. This analysis was done at a time before any commercial nuclear plants had been built. It was necessary to assume a highly idealized kind of accident to an undefined plant with the results calculated using assumptions as to basic data, many of which were still not known. This resulted in the effects of assumed accidents being overestimated by large

ACKNOWLEDGMENTS

The research summarized herein is the result of the efforts of thousands of managers, scientists, engineers, faculty, and technicians working in government, national laboratories, universities, and private companies and institutions. Many of these researchers are identified by the extensive reference to their research reports. This report is a compilation of summaries contributed by a large number of authors. A list of the principal contributors and reviewers is included below.

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Significant parts of several sections covering fuel behavior have been taken verbatim from an Organization for Economic Cooperation and Development - Committee on the Safety of Nuclear Installations (OECD-CSNI) report authored by C. A. Mann, E. D. Hindle and P. D. Parsons of the United Kingdom Atomic Energy Authority. The approval of the authors and the OECD-CSNI to take these sections verbatim, obtained with the assistance of R. Landry of the OECD, is gratefully acknowledged.

A report summarizing the ECCS research sponsored by the Electric Power Research Institute has been included as an appendix to this report. The assistance of B. Chexal of EPRI is acknowledged for obtaining permission to include this report.

E. Hill of the NRC provided many valuable hours editing and coordinating the preparation of this report.

U.S. NUCLEAR REGULATORY COMMISSION May 1989

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.157
(Task RS 701-4)

BEST-ESTIMATE CALCULATIONS OF EMERGENCY CORE COOLING SYSTEM PERFORMANCE

A. INTRODUCTION

Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that light-water nuclear reactors fueled with uranium oxide pellets within cylindrical zircaloy cladding be provided with emergency core cooling systems (ECCS) that are designed in such a way that their calculated core cooling performance after a postulated loss-of-coolant accident (LOCA) conforms to certain criteria specified in paragraph 50.46(b). Paragraph 50.46(b)(1) requires that the calculated maximum temperature of fuel element cladding not be greater than 2200°F. In addition, paragraphs 50.46(b)(2) through (b)(5), which contain required limits for calculated maximum cladding oxidation and maximum hydrogen generation, require that calculated changes in core geometry remain amenable to cooling and that long-term decay heat removal be provided.

On September 16, 1988, the NRC staff amended the requirements of § 50.46 and Appendix K, "ECCS Evaluation Models" (53 FR 35996), so that these regulations reflect the improved understanding of ECCS performance during reactor transients that was obtained through the extensive research performed since the promulgation of the original requirements in January 1974. Paragraph 50.46(a)(1) now permits licensees or applicants to use either Appendix K features or a realistic[1] evaluation model. These realistic evaluation models[2] must include sufficient supporting justification to demonstrate that the analytic techniques employed realistically describe the behavior of the reactor system during a postulated loss-of-coolant accident. Paragraph 50.46(a)(1) also requires that the uncertainty in the realistic evaluation model be quantified and considered when comparing the results of the calculations with the applicable limits in paragraph 50.46(b) so that there is a high probability that the criteria will not be exceeded.

This regulatory guide describes models,[3] correlations,[4] data, model evaluation procedures, and methods that are acceptable to the NRC staff for meeting the requirements for a realistic or best-estimate calculation of ECCS performance during a loss-of-coolant accident and for estimating the uncertainty in that

[1]For the purpose of this guide, the terms "best-estimate" and "realistic" have the same meaning. Both terms are used to indicate that the techniques attempt to predict realistic reactor system thermal-hydraulic response. Best-estimate is not used in a statistical sense in this guide.

[2]The term "evaluation model" refers to a nuclear plant system computer code or any other analysis tool designed to predict the aggregate behavior of a reactor during a loss-of-coolant accident. It can be either best-estimate or conservative and may contain many correlations or models.

[3]The term "model" refers to a set of equations derived from fundamental physical laws that is designed to predict the details of a specific phenomenon.

[4]The term "correlation" refers to an equation having empirically determined constants such that it can predict some details of a specific phenomenon for a limited range of conditions.

operation as well as isotopes of uranium, should be calculated in accordance with fuel cycle history and known radioactive properties. The actinide decay heat chosen should be appropriate for the facility's operating history. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.2.4 Fission Product Decay Heat

The heat generation rates from radioactive decay of fission products, including the effects of neutron capture, should be included in the calculation and should be calculated in a best-estimate manner. The energy release per fission (Q value) should also be calculated in a best-estimate manner. Best-estimate methods will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses. The model in Reference 10 is considered acceptable for calculating fission product decay heat.

3.2.4.1 Model Evaluation Procedure for Fission Product Decay Heat. The values of mean energy per fission (Q) and the models for actinide decay heat should be checked against a set of relevant data.

3.2.5 Metal-Water Reaction Rate

The rate of energy release, hydrogen generation, and cladding oxidation from the reaction of the zircaloy cladding with steam should be calculated in a best-estimate manner. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses. For rods calculated to rupture their cladding during the loss-of-coolant accident, the oxidation of the inside of the cladding should be calculated in a best-estimate manner.

3.2.5.1 Model Evaluation Procedure for Metal-Water Reaction Rate. Correlations to be used to calculate metal-water reaction rates at less than or equal to 1900°F should:

a. Be checked against a set of relevant data, and

b. Recognize the effects of steam pressure, preoxidation of the cladding, deformation during oxidation, and internal oxidation from both steam and UO_2 fuel.

The data of Reference 11 are considered acceptable for calculating the rates of energy release, hydrogen generation, and cladding oxidation for cladding temperatures greater than 1900°F.

3.2.6 Heat Transfer from Reactor Internals

Heat transfer from piping, vessel walls, and internal hardware should be included in the calculation and should be calculated in a best-estimate manner. Heat transfer to channel boxes, control rods, guide tubes, and other in-core hardware should also be considered. Models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.2.7 Primary to Secondary Heat Transfer (Not Applicable to Boiling Water Reactors)

Heat transferred between the primary and secondary systems through the steam generators should be considered in the calculation and should be calculated in a best-estimate manner. Models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.3 Reactor Core Thermal/Physical Parameters

3.3.1 Thermal Parameters for Swelling and Rupture of the Cladding and Fuel Rods

A calculation of the swelling and rupture of the cladding resulting from the temperature distribution in the cladding and from the pressure difference between the inside and outside of the cladding, both as a function of time, should be included in the analysis and should be performed in a best-estimate manner. The degree of swelling and rupture should be taken into account in the calculation of gap conductance, cladding oxidation and embrittlement, hydrogen generation, and heat transfer and fluid flow outside of the cladding. The calculation of fuel and cladding temperatures as a function of time should use values of gap conductance and other thermal parameters as functions of temperature and time. Best-estimate methods to calculate the swelling of the cladding should take into account spatially varying cladding temperatures, heating rates, anisotropic material properties, asymmetric deformation of cladding, and fuel rod thermal and mechanical parameters. Best-estimate methods will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.3.2 Other Core Thermal Parameters

As necessary and appropriate, physical and chemical changes in in-core materials (e.g., eutectic formation, phase change, or other phenomena caused by material interaction) should be accounted for in the reactor core thermal analysis. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

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The U.S. Nuclear Regulatory Commission (NRC) has issued a revised rule for loss–of–coolant accident/emergency core cooling system (ECCS) analysis of light water reactors to allow the use of best–estimate computer codes in safety analysis as an option. To support the revised ECCS rule and illustrate its application, the NRC and its contractors and consultants have developed and demonstrated an uncertainty evaluation methodology called code scaling, applicability, and uncertainty (CSAU).

The CSAU methodology and an example application described in this report demonstrate that uncertainties in complex phenomena can be quantified. The methodology is systematic and comprehensive as it addresses and integrates the scenario, experiments, code, and plant to resolve questions concerned with: (a) code capability to scale–up processes from test facility to full–scale nuclear power plants; (b) code applicability to safety studies of a postulated accident scenario in a specified nuclear power plant; and (c) quantifying uncertainties of calculated results. The methodology is able to address both uncertainties for which bias and distribution are quantifiable and uncertainties for which only a bounding value is quantifiable.

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PWR FUEL BEHAVIOUR IN DESIGN BASIS ACCIDENT CONDITIONS

THE DEFORMATION, OXIDATION AND EMBRITTLEMENT OF PWR FUEL CLADDING IN A LOSS-OF-COOLANT ACCIDENT

A State-of-the-Art Report by the TASK GROUP ON FUEL BEHAVIOUR OF CSNI PRINCIPAL WORKING GROUP No 2

DECEMBER 1986

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS
OECD NUCLEAR ENERGY AGENCY
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4. PBF rods that operated with breached cladding during film boiling testing were embrittled to a greater extent than intact rods oxidized under similar conditions. The handling fractures of those rods were not predicted by any of the embrittlement criteria.

5. The failure boundary (based on time at oxidation temperature for thermal shock) for out-of-pile data is more conservative than that drawn for the limited number of in-pile quenching failures.

6.14.4 Conclusions

After Appendix K was established, experimental and analytical programs have been completed that provide an increased understanding of fuel rod performance during both the heatup and reflood periods of a LOCA. The results of these programs have provided a significant amount of information in the following areas: (a) heat transfer across the fuel-cladding gap, (b) rod ballooning and rupture, and (c) oxidation induced rod embrittlement.

Experimental results show that fuel pellets crack, relocate, and are eccentrically positioned within the cladding. As a result, the heat transfer across the fuel-cladding gap is significantly greater than what is calculated when the fuel pellets are modeled as solid concentric cylinders.

Experimental results show that the extent of cladding ballooning is increased as the circumferential temperature variation of the cladding is decreased. Factors that influence the circumferential temperature variation are: (a) circumferential variation in the size of the fuel-cladding gap, (b) rate of heatup, (c) distance from cold walls, and (d) grid spacers.

Experimental results show that cladding embrittlement is a function of the oxygen content in the unoxidized portion of the cladding. The embrittlement criterion for thermal shock (thickness of the cladding with <0.9 wt% oxygen should be greater than 0.1 mm) fits the in-pile data. For cladding to have the capability to withstand fuel handling transport and storage, the calculated thickness of the cladding with less than 0.7 wt% oxygen should be greater than 0.3 mm. In addition to oxidation, embrittlement may be caused by hydrogen absorption due to stagnant steam within the rods. This source for embrittlement was observed in fuel rods with breached cladding operated at high power with film boiling.

Based on the experimental results, Chung and Kassner embrittlement criterion is recommended for predicting fuel rod cladding failure due to thermal shock or handling of embrittled cladding. The fraction of the 0.6-mm-thick cladding that has oxidized and the oxygen content of the oxidized cladding can be calculated based on time at temperature. Operation with breached cladding is so infrequent that it is unimportant.

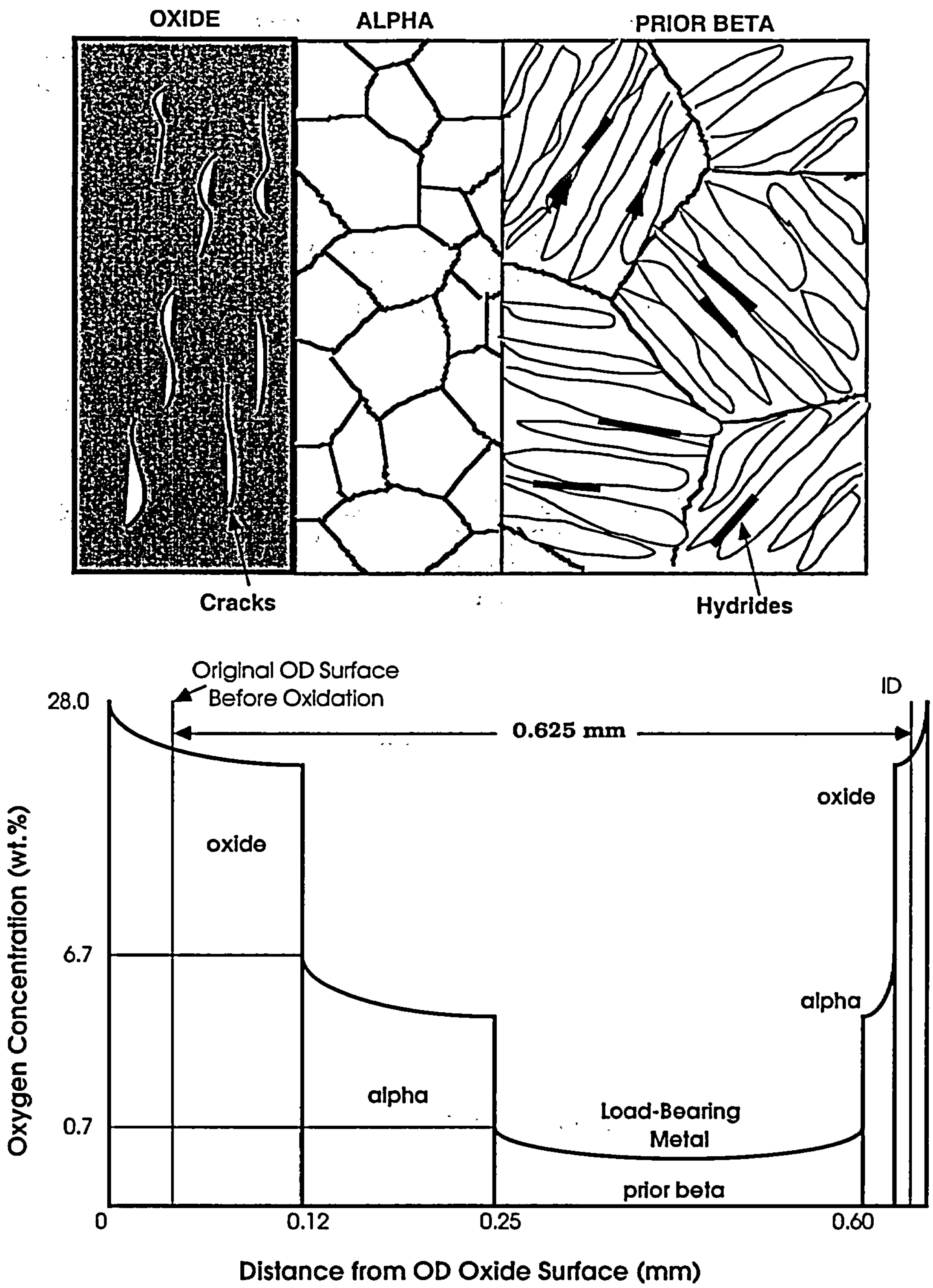

Fig. 1. Schematic illustration of microstructure (top) and oxygen distribution (bottom) in oxide, stabilized alpha, and prior-beta (transformed-beta) layers in Zircaloy cladding after oxidation near 1200°C.

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EMBRITTLEMENT CRITERIA FOR ZIRCALOY FUEL CLADDING APPLICABLE TO ACCIDENT SITUATIONS IN LIGHT-WATER REACTORS: SUMMARY REPORT

by

H. M. Chung and T. F. Kassner

Materials Science Division

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EMBRITTLEMENT CRITERIA FOR ZIRCALOY FUEL CLADDING APPLICABLE TO ACCIDENT SITUATIONS IN LIGHT-WATER REACTORS: SUMMARY REPORT

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ABSTRACT

The capability of Zircaloy cladding to withstand thermal-shock loads during the reflood stage of a loss-of-coolant-accident (LOCA) transient as well as anticipated loads during handling and transport of heavily oxidized fuel assemblies has been evaluated. Although the type and magnitude of the forces on the cladding under the latter situations have not been quantified, the critical fracture loads under conditions of impact, tension, and diametral compression have been determined as functions of the degree of oxidation of the material and microstructure produced by cooling through the temperature range of the $\beta \rightarrow \alpha'$ phase transformation at different rates. The effects of ballooning and rupture (i.e., wall thinning) and hydrogen uptake by the cladding during oxidation in steam on the deformation characteristics at room temperature have also been evaluated. The best correlation of the thermal-shock failure characteristics, the failure-impact energy, and the diametral-compression properties with an oxidation-related parameter was obtained relative to the thickness of the transformed β-phase layer, with a maximum oxygen content, for cladding that was oxidized at temperatures between 1200 and 1700 K for various times. Embrittlement criteria, which encompass the mechanical response of the cladding under different loading modes, have been formulated relative to the thermal-shock and 0.3-J impact resistance of the material.

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A2017	Mechanical Properties of Zircaloy

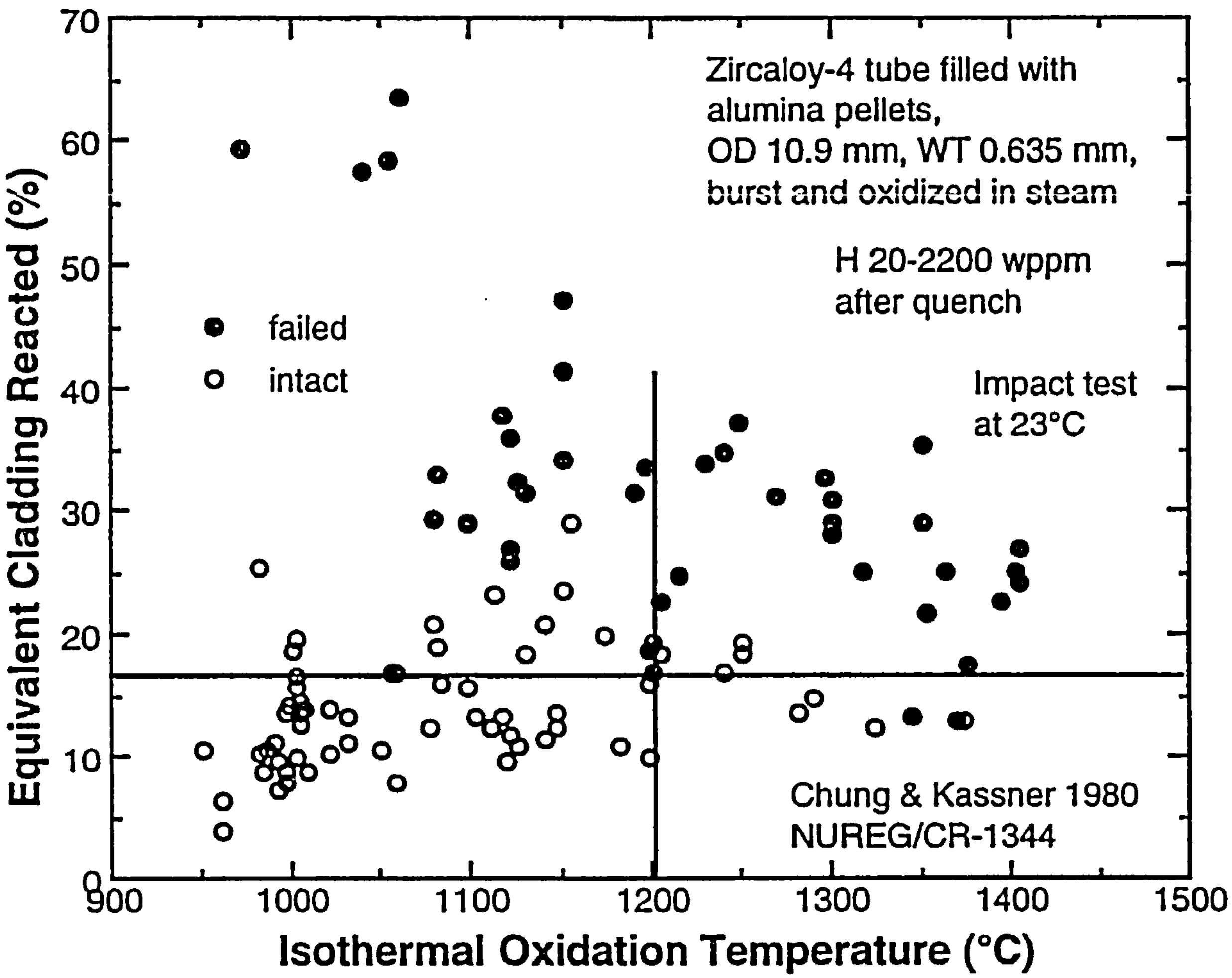

Figure 12.

Impact failure threshold as function of equivalent cladding reacted and oxidation temperature of burst, oxidized, slow-cooled, and quenched Zircaloy-4 tube containing 20-2200 wppm hydrogen (from Ref. 17).

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13. ABSTRACT (200 words or less)

Current Emergency Core Cooling System (ECCS) Acceptance Criteria for light-water reactors include certain requirements pertaining to calculations of core performance during a Loss of Coolant Accident (LOCA). The Baker-Just correlation must be used to calculate Zircaloy-steam oxidation, calculated peak cladding temperatures (PCT) must not exceed 1204°C, and calculated oxidation must not exceed 17% equivalent cladding reacted (17% ECR). The minimum margin of safety was estimated for each of these criteria, based on research performed in the last decade. Margins were defined as the amounts of conservatism over and above the expected extreme values computed from the data base at specified confidence levels. The currently required Baker-Just oxidation correlation provides margins only over the 1100°C to 1500°C temperature range at the 95% confidence level. The PCT margins for thermal shock and handling failures are adequate at oxidation temperatures above 1204°C for 210 and 160 seconds, respectively, at the 95% confidence level. ECR thermal shock and handling margins at the 50% and 95% confidence levels, respectively, range between 2% and 7% ECR for the Baker-Just correlation, but vanish at temperatures between 1100°C and 1160°C for the best-estimate Cathcart-Pawel correlation. Use of the Cathcart-Pawel correlation for LOCA calculations can be justified at the 85% to 88% confidence level if cooling rate effects can be neglected.

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Safety margins

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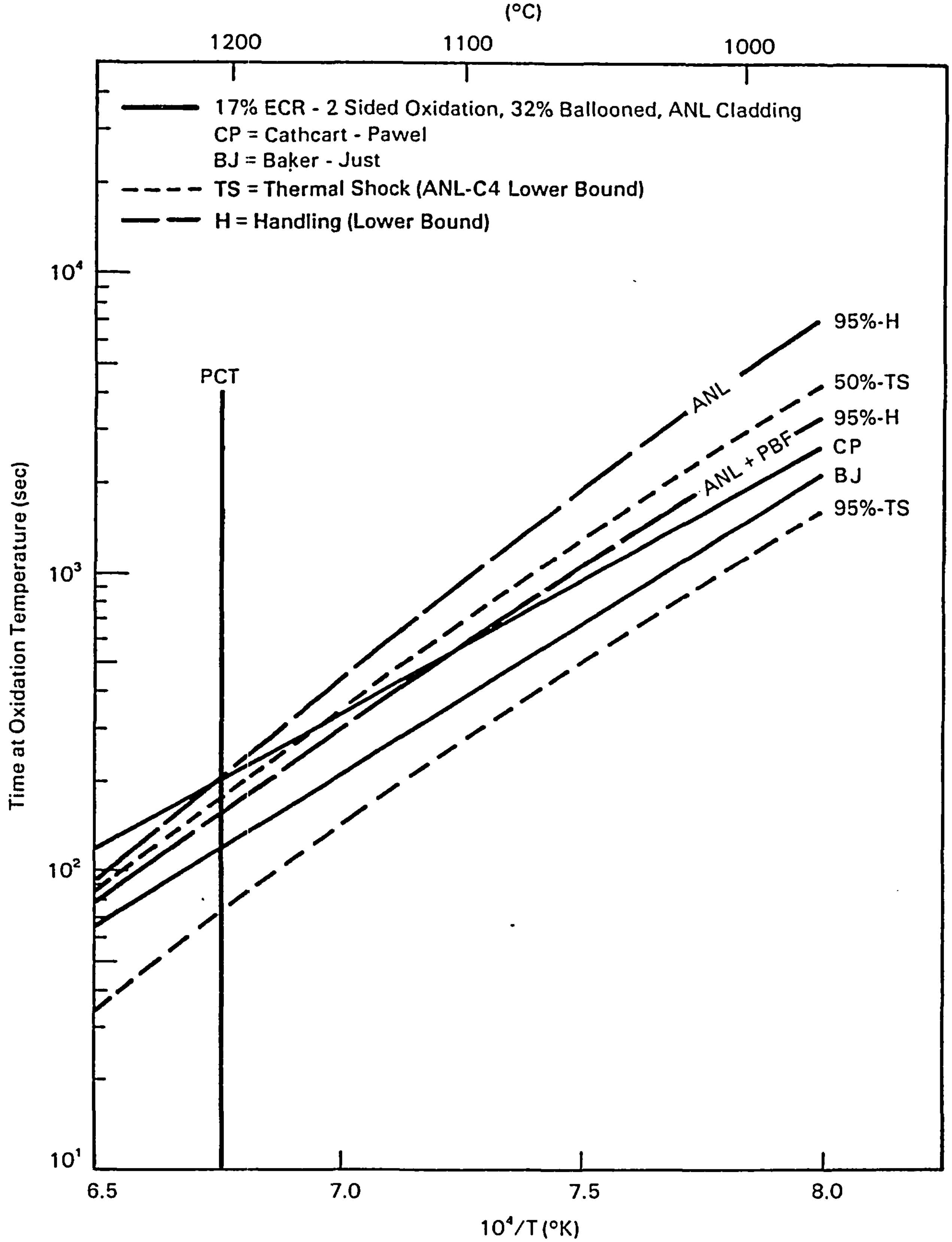

FIGURE 20. Thermal Shock and Handling Failure Boundaries at 50% and 95% Confidence Levels

THE HISTORY OF LOCA EMBRITTLEMENT CRITERIA

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Abstract

Performance of high-burnup fuel and fuel cladding fabricated from new types of alloys (such as Zirlo, M5, MDA, and duplex alloys) under loss-of-coolant-accident (LOCA) situations is not well understood at this time. To correctly interpret the results of investigations on the performance of the old and new types of fuel cladding, especially at high burnup, it is necessary to accurately understand the history and relevant databases of current LOCA embrittlement criteria. In this paper, documented records of the 1973 Emergency Core Cooling System (ECCS) Rule-Making Hearing were carefully examined to clarify the rationale and data bases used to establish the 1204°C peak cladding temperature and 17% maximum oxidation limits. A large amount of data, obtained for zero- or low-burnup Zircaloy cladding and reported in literature only after the 1973 Rule-Making Hearing, were also evaluated and compared with the current criteria to better quantify the margin of safety under LOCA conditions.

1. Introduction

Because of major advantages in fuel-cycle costs, reactor operation, and waste management, the current trend in the nuclear industry is to increase fuel discharge burnup. At high burnup, fuel rods fabricated from conventional Zircaloys often exhibit significant degradation in microstructure. This is especially pronounced in pressurized-water reactor (PWR) rods fabricated from standard Zircaloy-4 in which significant oxidation, hydriding, and oxide spallation can occur. Thus, many fuel vendors have developed and proposed the use of new cladding alloys, such as low-tin Zircaloy-4, Zirlo, M5, MDA, duplex cladding, and Zr-lined Zircaloy-2. Performance of these alloys under loss-of-coolant-accident (LOCA) situations, especially at high burnup, is not well understood at this time. Therefore, it is important to verify the safety margins for high-burnup fuel and fuels clad with new alloys. In recognition of this, LOCA-related behavior of various types of high-burnup fuel cladding is being actively investigated in several countries [1-6]. However, to correctly interpret the results of such investigations, and if necessary, to establish new embrittlement thresholds that maintain an adequate safety margin for high-

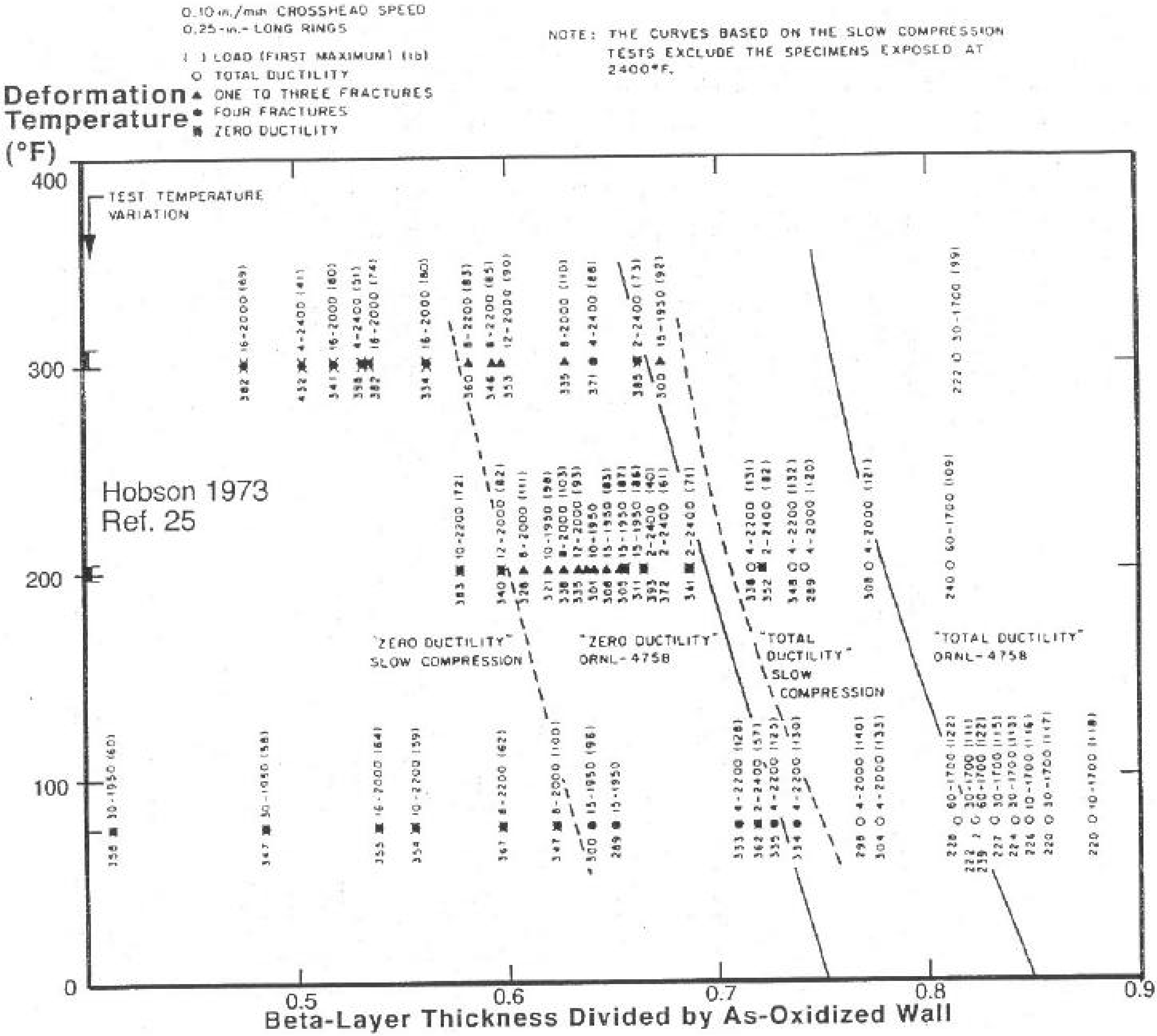

Figure 7.

Ductility of two-side-oxidized Zircaloy rings as function of slow- or fast-compression temperature and fraction of transformed-beta-layer (from Hobson, Ref. 25 and 26).

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