Attachment 1

Appendix A

TECHNICAL EVALUATION OF DRAFT NUREG 0630*

Prepared by Yankee Atomic Electric Company
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*D. A. Powers and R. O. Meyer; "Cladding Swelling and Rupture Models for LOCA Analysis"; November 8, 1979.

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1. Introduction

On November 8, 1979, a draft report prepared by the staff of the Office of Nuclear Reactor Regulation, "Cladding Swelling and Rupture Models for LOCA Analysis" (DRAFT NUREG 0630), was issued by the USNRC. It is presently being circulated for review by the technical community for the purpose of obtaining a technical critique. This report presents such a technical evaluation by Yankee Atomic Electric Company.

The importance of increasing the industry's existent understanding with respect to fuel rod behavior during LOCA and of improving the applicable data bases and phenomena correlations and models is recognized. These goals define the basis of this critique.

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2. Comments on the Data Base

2.1 Applicability of the Data

The selected data base, which has been restricted to experiments in aqueous environments that utilize either internally heated fuel-pellet simulators or fuel pellets in-reactor, is generally appropriate. The extent of Zircaloy cladding swelling at rupture has been demonstrated to be influenced strongly by the degree of cladding oxidation which exists prior to and occurs during cladding heatup.[1,2] According to Kassner et al., oxidation and temperature nonuniformities in cladding as would occur in-reactor will decrease the ballooning of the Zircaloy cladding, in contrast to data obtained in inert-gas atmospheres.[3] Data obtained in steam with direct heating of cladding may still be appropriate for determining a cladding rupture criterion (see Section 3.1); however, such methods are generally designed to produce uniform cladding temperatures and hence result in unrealistic cladding strains which should not be used in devising or qualifying a LOCA predictive model.[4]

2.2 Data Base Limitations

This restriction of the data base results in a sparse data field, especially in the regime of high temperature/low pressure burst. Whereas the report attempts to quantify the effects of heatup ramp rate on rupture, swelling, and blockage, the data base is incomplete. No data are included which are within the high temperature regime (>1000°C) for slow cladding temperature ramps.

2.3 Data Qualification

The final report should provide some detail with respect to experimental measurement techniques and data interpretation

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methods. For instance, the ORNL Multi Rod Burst Test Program reports burst temperature measured by external and internal thermocouples as the highest observed thermocouple reading regardless of burst location.(5) The FR2 nuclear tests report burst temperature calculated from thermocouples above and below the burst and corrected for the extent of burst deformation.(6) Estimates on the underprediction or accuracy of the reported data with respect to actual temperatures at external and internal cladding surfaces should be provided. Similar evaluations of other measured parameters (e.g. pressure and strain) should be included. Since the research programs and data evaluations examining these phenomena are not complete, this information should be considered both necessary and valuable.

2.4 Data Evaluation

Data have been reported which show large discrepancies in important measured parameters. Such data values should be explicitly identified in the analysis, correlation development, and associated figures. For example, Rods #B1.5 and #B1.2 of the FR-2 tests (6) showed almost identical ramp rate, burst pressure (engineering burst stress), and burst temperature measurements; however, burst strains are reported as 60 and 25 percent, respectively. Rods #A1.1 and #B3.2 exhibited similar engineering rupture strains at equal burst pressure; the burst temperature measurements which differ by 105°C are not explained by the slight difference in temperature ramp rate. For all data sets, the analysis should attempt to resolve these types of discrepancies and/or clearly identify that the discrepancies exist.

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3. Comments on the New Correlations

The correlations in the report are based upon a relatively restricted data set. Reliance on three multirod burst test experiments at two cladding heat rates is apparent. The objectives of the ORNL MRBT program have been to "delineate the deformation behavior of unirradiated Zircaloy cladding under conditions postulated for a loss-of-coolant accident (LOCA) and to provide a data base to facilitate assessment of the magnitude and distribution of geometrical changes in the fuel rod cladding in a multirod array and the extent of flow channel blockage that might result."(7)

To fulfill these objectives the program developed a test matrix designed to examine effects of fuel pressure and temperature at rupture in a multirod array. Several features of the experimental setup have been chosen to characterize performance in relation to independent variables. These include steam flow rate, shroud heating and heatup rate, and uniform axial heat generation. To this extent specific, anticipated in-reactor LOCA conditions were not intended and have not been produced. Program goals of providing a test data matrix for predictive model validation are being met.

The correlations to "predict" cladding strain at rupture and assembly blockage tacitly assume that the data base includes results representative of LOCA conditions. The analysis then envelopes this information with respect to the cladding temperature parameter. This process treats data within the envelope as dependent variable scatter with respect to the modeling parameter and ignores analysis of that data. The report should express this philosophy of correlation development, attempt to identify its limitations, and suggest potential model improvements which would predict, rather than interpret, the data.

3.1 Rupture Temperature

3.1.1 Statistical Evaluation of the Correlation

The rupture temperature correlation was developed within the ORNL MRBT program[8] as a function of rupture pressure and translated to engineering stress at rupture to apply to the extended data base of the DRAFT NUREG report.[9]

$$T = 3960 - \frac{20.4\ S}{1 + H} - \frac{8.51 \times 10^6\ S}{100(1+H) + 2790\ S}$$

where: H = Heat rate (^{o}C/sec)/(28^{o}C/sec)

T = rupture temperature (^{o}C)

S = engineering stress at rupture (KPSI)

Since the data base for the correlation is mainly from the ORNL program, an evaluation has been performed to benchmark the correlation against the extended data base of the DRAFT NUREG report. In this study, data points were eliminated from consideration if they failed either of the following tests:

1) ramp rate accurately known and

2) all information for a data point is known.

Applying these two conditions, a total of 112 data points were applicable to the analysis. The analysis consisted of evaluating the NRC correlation at each data point and comparing the calculated value of burst temperature to the measured value. General observations include:

(1) the RMS value of the difference tends to decrease with increasing ramp rate and

(2) the RMS values and numbers of points which are underpredicted by the correlation are greater than those which are overpredicted.

This difference indicates a bias of the correlation to predict lower burst temperatures. The value for the RMS difference ($\sim 25^{\circ}C$) indicates that temperature is a major factor which must be further quantified. Table 3.1 lists the RMS values for different subsets of the data. A bias of underprediction of the rupture temperature is observed for all data partitions.

3.1.2 Systematic Uncertainty in the Data

As discussed in Section 2, systematic uncertainties are apparent in the ORNL MRBT data for burst temperature. Chapman has indicated that such uncertainties can be evaluted and reduced by careful analysis of the burst versus thermocouple locations. Burst temperature data values may be on the order of $15^{\circ}C$ to $35^{\circ}C$ higher than those presented in the report.[10] Since approximately 80 percent of the data analyzed in Section 3.1.1 were developed in ORNL tests, this potential bias should be considered to be additive to that which has already been demonstrated.

3.1.3 Limits and Extrapolation of the Data Base

As indicated in Section 2, data are severely limited for low temperature ramp rate conditions which have recently been identified as applicable to most anticipated reactor conditions during a LOCA event.[11] No data have been presented in the temperature regime above $1000^{\circ}C$ for slow heating rates. Therefore, the heating rate dependence of the correlation evaluated in Section 3.1.1 is derived from data taken primarily in the $700^{\circ}C$ - $900^{\circ}C$ temperature range. Such extrapolation

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ignores the additional physical processes (i.e., cladding oxidation) which are known to occur in this regime and to influence the rupture process.

At least some data in this temperature range which was employed in the correlation development [8] was not obtained in an aqueous environment.[12] Preliminary investigations indicate that heating rate dependence may not exist in the high temperature, cladding oxidation regime. One data point excluded from the data reference by Karb[6] (Rod #A2.3) ruptured at an engineering burst stress of 2.5 KPSI and a burst temperature of 1015°C. The ramp rate for the data set is reported to be between 6°C/sec and 19°C/sec. This data point falls above the 28°C/sec correlation in this high temperature regime.

3.1.4 Applicability of Isothermal Data

To derive the low temperature dependence of the correlation, isothermal (0°C/sec) data have been used. Examination of four of these experiments shows that hold-times of these tests ranged from 49 to 250 seconds.[13] Such tests present data which are not consistent in that hold-time is an additional variable. Very low temperature ramp rate data rather than isothermal tests should be used to develop the correlation.

3.1.5 Ramp Heating Rate Accuracy

Cladding heatup ramp rates reported in the data base are those experienced during heatup prior to the time-of-burst. As the burst time is approached, instantaneous heatup rates are consistently much less than that experienced during the preliminary heatup ramp.[5,8] This observation has not been addressed in the data analysis.

3.1.6 Burst Temperature in Direct Heating Experiments

Observations in Sections 3.1.1 through 3.1.3 and Section 3.1.5 suggest that the burst temperature/engineering stress correlation may be greatly improved by increasing the accuracy of the temperature data. Experiments designed to accomplish this by reducing temperature variation (e.g., direct heating experiments) tend to produce large rupture strains. However, rupture temperature/pressure data tend to fall above the correlation curve in the range of that indicated by the systematic biases. The DRAFT NUREG study should investigate these data to determine if such experiments are valid for establishing a rupture criterion.

3.1.7 Other Burst Criteria

Hagrman has recently evaluated several cladding stress at failure/burst criteria and has determined a true tangential stress versus temperature relationship to be most appropriate.[14] Heating rate and strain rate do not affect this criterion. Hagrman used some data in the development of this correlation which were from direct electrical heating experiments and which covered a wide range of heating rates. The DRAFT NUREG report should investigate this criterion and compare its applicability and accuracy with respect to the DRAFT correlation.

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3.2 Burst Strain

Reference to the Chung and Kassner data [3,15] to quantitatively establish the temperature ranges for location of superplastic peaks and valleys appears valid, although explanations related to the physical phenomena of phase transformation and Zircaloy oxidation would provide a more appropriate basis. This type of information (phase transformation temperature, Zircaloy oxidation rate effects, Zircaloy creep tests) should be incorporated to justify the peak and valley positions with respect to temperature. The present approach limits accuracy of these positions to that achieved in this particular experiment.

Scaling the peaks observed by Chung and Kassner to the present data set assumes a quasi-linear relationship exists between burst strain results from direct heating experiments versus those from internal rod simulator heating experiments. Setting peaks to enclose all the data assumes (1) that LOCA conditions have been accurately simulated in the experiments and (2) that peak values at given temperatures are accurate and represent real upper bounds of swelling at rupture. The data which fall below the derived bounding curve are then ignored; therefore, the interpretation process used in the DRAFT REPORT to address the "bewildering" scatter in the more realistic data base is unsuccessful. A more fundamental examination of the data which identifies experimental uncertainties and uncontrolled independent variables should be incorporated in correlation development for the final report.

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3.3 Assembly Flow Blockage

The axial distribution of cladding rupture in the ORNL MRBT cannot be considered unbiased in that the location of burst is influenced by the performance of the internal heater element.[5,8] The suppressing effects of spacer grids exist in reactor bundles and should be considered in the data extrapolation. It does not appear that the exact scaling factor, SF, has been used in the comparison to vendor models in that the derived blockage curves appear the same for each comparison.

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4. Summary and Recommendations

1. The experimental program should not be limited in scope of temperature and pressure conditions, but should continue to focus on providing a test matrix for qualifying LOCA predictive models.

2. Data in the high temperature/low pressure burst regime for slow-ramp conditions are required.

3. Data qualification with respect to measurement techniques, accuracy, and interpretation of each of the experimental data sets should be included in the final report.

4. Apparent discrepancies in data within experimental data sets used in the report should be identified, discussed, and resolved.

5. Statistical evaluation of the rupture correlation demonstrates bias toward underprediction of the burst temperature, especially at low cladding heatup rates. Such an analysis should be included in the final report.

6. The difference in temperature measurement techniques for the experiments comprising the data base should be identified. An attempt should be made to examine these differences and to quantify resultant temperature uncertainties.

7. The rupture temperature/pressure correlation extrapolation to the high temperature regime is not qualified by any data in the slow

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ramp range. The correlation in this regime, especially for low ramp rates, should be re-examined.

8. Isothermal data may not be appropriate and should be re-examined for applicability.

9. Accuracy of cladding heat-up rates should be re-examined for each data set and identified in the report.

10. Applicability of direct electrical heating experimental results for burst temperature/pressure prediction only should be re-examined.

11. The derived rupture correlation should be quantitatively compared to Hagrman's failure criterion.

12. Reference to Chung and Kassner experiments should be deleted in determination of cladding strain. Temperature regimes of strain peaks and valleys should be derived from fundamental Zircaloy property/performance and from the current, applicable data base.

13. Magnitudes of peaks and valleys in cladding strain versus temperature curves should be derived from the current, applicable data base with attention focused on inconsistencies, uncertainties, and accuracies of the data sets.

14. The derivation of the assembly blockage correlation should recognize the systematic locations of rupture in the ORNL MRBT data sets and the effects of assembly spacer grids.

Table 3.1

Evaluation of the Rupture Temperature Correlation Preditive Capability

Cladding Temperature Ramp Rate (°C/sec)		Total Number of Points	Total RMS Deviation (°C)	Underprediction		Overprediction	
Min.	Max.			Number of Points	RMS Deviation (°C)	Number of Points	RMS Deviation (°C)
0	10	40	24	27	23	13	24
10	20	29	21	23	24	6	7
20	30	63	18	45	21	18	12
0	15	48	24	34	24	14	24
15	30	64	19	46	21	18	12
0	30	112	21	80	22	32	17

Note: Underprediction implies that the draft NUREG correlation predicts burst to occur at a lower temperature than observed in the test.

References

(1) R. R. Biederman, et al.; "A Study of Zircaloy-4--Steam Oxidation Reaction Kinetics"; EPRI NP-734, Project 249-2 Final Report Part 2; April 1978.

(2) R. E. Westerman, et al.; "Zircaloy Cladding ID/OD Oxidation Studies"; EPRI NP-525, Project 251-1 Final Report; November 1979.

(3) T. F. Kassner, et al.; "Development of Embrittlement Criteria for Zircaloy Cladding"; paper presented at the 5th NRC Water Reactor Safety Research Information Meeting, Gaitherburg, Md; November 7-11, 1977.

(4) D. A. Powers and R. O. Meyer; "Evaluation of Simulated-LOCA Tests that Produced Large Fuel Cladding Ballooning"; NUREG-0536; March 1979.

(5) R. H. Chapman, et al.; "Bundle B-1 Test Data - Multirod Burst Test Program"; ORNL/NUREG/TM-322; June 1979.

(6) E. H. Karb; "Results of the FR2 Nuclear Tests on the Behavior of Zircaloy Clad Fuel Rods"; paper presented at the 6th NRC Water Reactor Safety Research Informtion Meeting, Gaithersburg, Md; November 7, 1978.

(7) R. H. Chapman; "Multirod Burst test Program Quarterly Progress Report for January-March 1976"; ORNL/NUREG/TM-36; September 1976.

(8) R. H. Chapman; "Multirod Burst Test Program Quarterly Progress Report for April-June 1979"; NUREG/CR-1023; November 1979.

(9) D. A. Powers; Personal Communication; November 15, 1979.

(10) R. H. Chapman; Personal Communication; November 30, 1979.

(11) D. G. Eisenhut; Letter to All Operating Light Water Reactors concerning NSSS vendor and fuel supplier evaluations of cladding heating-rate effects; November 27, 1979.

(12) P. Hoffman; "Influence of Iodine on the Strain and Rupture Behavior of Zircaloy-4 Tubing at High Temperature", in ASTM-STP-681; 1979.

(13) R. H. Chapman, et al.; "Effect of Creep Time and Heating Rate on Deformation of Zircaloy-4 Tubes Tested in Steam with Internal Heaters"; NUREG/CR-0343; October 1978.

(14) D. L. Hagrman; "Cladding Stress at Failure"; paper presented at the 7th NRC Water Reactor Safety Research Information Meeting, Gaithersburg, Md; November 8, 1979.

(15) H. M. Chung and T. F. Kassner; "Deformation Characteristics of Zircaloy Cladding in Vacuum and Steam Under Transient - Heating Conditions: Summary Report"; NUREG/CR-0344: AWL-77-31; July 1978.