March 25, 1982

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Docket No.: 50-537

Mr. John R. Longenecker
Licensing and Environmental Coordination
Clinch River Breeder Reactor Plant
U. S. Department of Energy, NE-561
Washington, D.C. 20545

Dear Mr. Longenecker:

SUBJECT: CLINCH RIVER BREEDER REACTOR PLANT, REQUEST FOR ADDITIONAL INFORMATION

As a result of our review of your application for a construction permit for the Clinch River Breeder Reactor Plant, we find that we need additional information in the area of Core Performance. Please provide your final responses by May 15, 1982.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

If you desire any discussion or clarification of the information requested, please contact R. M. Stark, Project Manager (301) 492-9732.

Sincerely,

Original Signed by
Paul S. Check

Paul S. Check, Director
CRBR Program Office
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: Service List

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MECHANICAL DESIGN QUESTIONS (4.2)

QUESTIONS TO THE APPLICANT ON PSAR SECTION 4.2

CS490.1 The plutonium concentration in the mixed oxide driver pins has been changed from 20 and 25% in the FFTF and the previous CRBR design to 33% in the current CRBR design. This gives rise to concern over whether any of the data base on integral fuel pin performance or on fueled-cladding behavior is relevant to the current design of CRBR. Specific concerns include:

1) How does the change of Pu concentration affect fuel cladding chemical interaction? What is the basis for this assessment?

2) The thermal conductivity and solidus and liquidus temperatures of the fuel (plus probably other phenomena that enter into thermal performance) are affected significantly by the change in Pu concentration. It seems inescapable, therefore, that the power-to-melt tests and the thermal performance models based thereon do not apply to the revised CRBR fuel design. If the applicant agrees with this assessment, how does he plan to replace these two key pieces of the fuel design evaluation methods? If he does not agree with this assessment, he is requested to justify his position.

3) How does the change of Pu concentration affect the applicability of properties and models that are dependent on stoichiometry? Is it anticipated that hitherto unimportant or unsuspected effects of Pu redistribution will become significant? How is it anticipated that these changes will affect fuel performance? What is the basis for this assessment?

4) How does the higher concentration of Pu affect the fission gas retention characteristics of the fuel? If significantly different, how does this affect the applicability of the fuel pin evaluation models? How does the changed Pu concentration affect both time dependent and time independent deformation behavior of the fuel? How will this affect fuel performance? What are the bases for both answers? If an assessment is not possible now, how does the applicant propose to resolve the issue?

5) How does the change of Pu concentration affect fuel swelling as a function of burnup? What is the basis for this assessment?

6) Why doesn't this change invalidate both the CDF and the Ductility Limited Strain models for evaluating fuel performance? If it does invalidate both models, how does the applicant plan to evaluate fuel pin performance?

7) It would seem to be a minimum requirement that some check tests of 33% Pu concentration fuel pins be performed in FFTF and some of those pins be given transient tests in TREAT to confirm the predicted effects of the higher Pu concentration. What plans does the applicant have for such tests? Please be specific in the

response. If there are no such plans, how does the applicant plan to justify his assessment of the effect of the change? Are there any data at all on the behavior of irradiated mixed oxide fuel with this high a concentration of Pu?

CS490.2 The current data base for fuel pin response and cladding failure threshold under transient overpower (TOP) conditions includes no data at all in the ramp range from the power-to-melt tests (about 0.005 cents/s) to the W-2 test (about 5 cents/s), and very little data for ramp rates between 5 cents/s and 50 cents/sec. Please delineate the testing planned to provide data in the cited ramp range. If no testing is anticipated in the slow ramp rate range, how is it planned to determine what the cladding failure threshold is and what the threshold is for molten fuel expulsion (not necessarily the same)?

CS490.3 Current state-of-the-art analysis methods use stress-rupture based correlations for predicting fueled and unfueled cladding breach under both steady-state and transient conditions. There appears to be a lack of fundamental understanding of cladding failure mechanisms, as evidenced by:

1) very large difference in load bearing capability and ductility of fueled cladding between steady-state and transient conditions, yet stress-rupture formulations being used for both classes.

2) Hints of fission product assisted stress cracking propagation.

3) The elimination of much of the damage with regard to transient capability when steady state irradiated above about 1050 ^{o}F.

What testing plans are there to better identify the mechanisms of cladding failure, to define how steady-state and transient behavior mesh together, and to develop more appropriate failure criteria, particularly under overpower conditions?

CS490.4 Continuing questions in predicting and understanding fuel pin response to overpower conditions are the ductility and load bearing capability of fueled irradiated cladding under fuel cladding mechanical interaction (FCMI) conditions, and whether cladding response to these conditions is significantly different than exhibited under gas pressure loading. What testing is planned to define response to FCMI transient loading? Please forward whatever data may exist in this area, along with the project evaluation of the data.

CS490.5 There appears to be evidence that cladding ductility and load bearing capability is much less affected by irradiation at temperatures above 1050 to 1100 ^{o}F than by irradiation below that temperature level. This implies that under transient overpower conditions the site of the cladding breach, if breach occurs, is virtually certain to be below the axial level of the transition temperature. This may, depending on inlet coolant temperature and coolant flow rate, force the site of fuel expulsion in an unterminated reactivity insertion accident low enough that the fuel movement causes substantial additional reactivity ins.rtion, significantly exacerbating the accident. Please provide:

1) The data which support the cited cladding behavior;

2) A comparison of irradiated cladding ductility and strength above and below the transition temperature;

3) The best estimate of the transition temperature and its uncertainty, and the range of temperature involved in the transition;

4) Three sigma (high side) estimates of:

a) Fraction of fuel and blanket pins wherein the transition temperature occurs on the cladding below X/L* = 0.75

b) Fraction of fuel and blanket pins wherein the transition temperature occurs on the cladding below X/L = 0.65

c) Fraction of fuel and blanket pins wherein the transition temperature occurs on the cladding below X/L = 0.55

What would be the impact of lowering the inlet coolant temperature by 50 to 100 ^{o}F if this were necessary to avoid the transition temperature being reached at too low an axial location?

CS490.6 Non-prototypic factors in TREAT transient overpower (TOP) tests of EBR-II irradiated fuel pins seriously compromise translation of the results of these tests to the CRBR. What plans are there to evaluate those factors experimentally, particularly radial power depression and short vs. long pins, and now the additional factor of 33% Pu concentration as versus the 25% Pu concentration of the tests? Other factors include non-prototypic fluence to burnup ratio and U235 to Pu fission ratio, preconditioning, and static capsule non-prototypic cladding temperature. The ratio of U235 fissioning to Pu fissioning is of concern because 30% more Zr fission product would be produced from

* X/L: Fraction of fuel column measured from bottom.

Pu fissioning and might adversely affect fuel cladding chemical interaction through its effect on oxygen potential.

If there are no tests planned to evaluate the effects of the non-prototypic factors in the data base, how does the applicant plan to account for these factors in applying the data base to CRBR design evaluation?

CS490.7 Several of the EBR-II pins that have been TREAT tested were irradiated in subassemblies from which other pins have exhibited metallurgical evidence of far higher temperatures than can be accounted for during irradiation by thermal hydraulic means. Please enumerate the TREAT tests that involved pins from such assemblies and the apparent temperature defect in each case. Please provide your evaluation of what effect this may have had on the results of the subject tests.

CS490.8 The FCTT data are generated at constant load and increasing temperature. Permanent straining occurs as the yield strength decreases with increasing temperature. Thus straining and annealing are inextricably intertwined in the data obtained. The data are probably relevant for loss-of-flow events; however, in overpower events, straining is more likely to be dependent on differential expansion of the fuel against the cladding and only mildly dependent of cladding temperature. What plans are there to perform FCTT tests in which the strain rate is independently controlled? Please supply what ever data may be available in this area.

CS490.9 Essentially no data exist for either steady-state or transient performance of blanket pins. What are the current testing plans to obtain blanket pin data? If no tests are planned for blanket pins under either or both steady-state or transient conditions, how is it planned to confirm predicted cladding failure thresholds and margins to cladding failure?

CS490.10 What plans are there to incorporate transient fuel mechanical interaction loads into the CDF fuel pin evaluation method for overpower events? Are there any plans for incorporating FCTT test results for fueled cladding into the CDF method? Has the method been used to analyze TOP tests (especially those in which cladding breach occurred), and if so, what were the results?

Has the criterion been calibrated to the high fluence data now available? If so, please tabulate the additional data that have been incorporated.

CS490.11 The basis for construction of 99% confidence bands for the CDF fuel evaluation model was criticized by the partial draft safety evaluation (SER) report prepared in 1977 on pages 4.2-44 through 4.2-46. The question involved has not been resolved. Please discuss the relative merits of the method used in reference 58 to Section 4.2 and the method suggested in the partial SER. Please also perform the evaluation of the two methods suggested on page 4.2-46 and provide the result. All page numbers refer to the partial draft SER.

CS490.12 What plans are there to evaluate predictions for duct dilation and fuel and blanket subassembly refueling loads against FFTF experience? If it is not planned to use FFTF experience in this area, or CRBR predictions methods do not predict FFTF experience, how does the applicant plan to demonstrate the adequacy of his design with regard to the effects of subassembly bowing and distortion?

CS490.13 After reading the description of the FRST code in appendix A, it is apparent that the ductility limited strain model for evaluating fuel performance is very different from the old "design procedure", or as it was also called, the "design recipe" or "FCF213". The information provided in the PSAR on this model is quite limited. Please provide a detailed description of the model, including material properties used and the data to back those properties. A listing of the FRST code would be very helpful. Please also indicate how the model has been calibrated and qualified.

CS490.14 Substantial power jumps are anticipated in some fuel and blanket pins on starting up after refueling. The impact of these power jumps is to cause a sharp increase in fuel cladding mechanical interaction (FCMI) which then decays off as fuel and cladding deform under stress. The phenomenon was evaluated for the PSAR using a version of the LIFE code which had not been calibrated as to its prediction of FCMI or the manner in which FCMI decays. How is it planned to calibrate the method for predicting FCMI and the effect thereof on fuel damage?

CS490.15 Experience with the FFTF fuel system will be essential to resolving several licensing issues for the CRBR. Please detail the plans for surveillance of FFTF driver fuel pins (including both non-destructive and destructive examinations) and for transient tests of FFTF irradiated fuel pins. To what extent will the project have influence over the base technology program?

CS490.16 Several transient tests have been conducted since the most recent CRBR licensing activity. Documentation (data report, final report) available now or which becomes available in the future is requested for the following tests:

HEDL	ANL
HOP PTO 1-2A	J1
HOP PTO 3-2E	P2
HUC PTO 2-2A	P3
HOP 1-6A	P3A
HUT 3-5B	P4
HUT 5-5B	H6
HUT 3-6A	
HUT 3-6B	
W-2	
W-1	

All HEDL tests above bear the new test designation.

CS490.17 The behavior of irradiated core 1 steel for the FFTF duct and fuel pin cladding was found to be significantly different in swelling and steady-state stress rupture strength from N-lot and other test steels upon which much of CRBR design was based. How does the applicant plan to demonstrate the conservatism of CRBR design in view of the unexpected additional uncertainty in material behavior that this implies? Does the applicant plan to initiate irradiation tests early-on to discern whether a similar deviation will occur for CRBR duct and fuel pin cladding material? If no such testing is planned, how does the applicant plan to discern such a deviation before it has become a problem in CRBR?

CS490.18 The discussion of operation with defected fuel pins in the PSAR was apparently current as of September, 1979. Please update this discussion as may be warranted by new information.

What additional defected pin test results have become available since 1979, and how do they apply to the CRBR? Have any transient tests been performed (or are any planned) on defected pins? If no transient tests on defected pins are planned, how does the project plan to demonstrate that continued operation with failed pins would be safe, since this entails exposure to anticipated and unlikely events?

How is the change to 33% Pu concentration expected to affect continued operation of failed fuel pins?

CS490.19 There are a number of thermal and mechanical fuel performance codes, both steady-state and transient, that more or less satisfactorily predict data available on thermal performance, cladding breach, and cladding inelastic strain. Yet these codes can vary wildly as to fuel-cladding interface pressure, gap conductance, fission gas release, etc., particularly in extrapolations outside the calibration data base. This situation arises because the only data available for calibration are integral-pin, post-test data including thermal data relatively remote from the region of interest. It is virtually impossible to qualify indivdual phenomena models; this coupled with the number and complexity of models and the uncertainty of material properties allows an unlimited number of solutions to the problems of predicting extrapolated performance.

It is of particular concern, therefore, that all temperatures and performance predictions be performed in a consistent fashion for the purpose of reviewing fuel performance. Correlations and models that use calculated input parameters (for instance, the Failure Potential Correlation, or any of the SIEX code correlations) should be used only in keeping with the manner in which they were developed. Otherwise their use may yield totally invalid results. Alternatively, when assessing compliance with a criterion based on independent data, use of different models may give very different assessments.

What steps has the applicant taken, or does he plan to take, to ensure that all evaluations will be done in a consistent fashion, and that inputs to all empirical correlations will be determined in a fashion consistent with their development?

CS490.20 Cladding breach for undercooling conditions is generally considered to occur when the current burst pressure declines below the plenum pressure due to increasing cladding temperature. There are few data, if any, available to conservatively confirm just when breach would be expected. Virtually all FCTT data were obtained for either very high gas pressure or for low fluence or nonfueled cladding, and the loss of coolant tests conducted by ANL were all for low burnup pins. Please supply any additional data that may now be available (either FCTT or integral pin data) that are more relevant than the data quoted above to end-of-life, undercooli failure threshold conditions - that is, at plenum pressures in the 1000 to 1500 psi range, and at cladding irradiation damage levels approaching end-of-life conditions.

If no more relevant data than that quoted above are available, are there plans to obtain such data? If so, please describe those plans. If not, how does the applicant plan to demonstrate the conservatism of the fuel and blanket design for undercooling conditions?

CS490.21 In PSAR section 4.2, the reader is referred frequently to PSAR section 15.1.2 for details of the CDF fuel evaluation model and its development and qualification. However, the portions of Section 15.1.2 dealing with those subjects appear to have been deleted. Does this presage a decision to abandon the CDF fuel evaluation method for preparation of the FSAR and the operating license application? If so, what does the applicant plan to use in its place to evaluate fuel performance?

CS490.22 The CDF method for fuel performance evaluation provides a model for determining the accumulated cladding damage due to steady-state operation and all anticipated transient events plus one unlikely event at the end of life, and includes auxiliary models to account for cladding wastage, corrosion, and irradiation damage. All of these models, however, appear to depend on input from other sources as to plenum pressure, fuel cladding mechanical interaction loads, time-temperature history, etc. The manner in which this input is generated is also important to the validity of the method. Was the generation of input data for determination of the transient limit curves (TLC's) accomplished in a manner consistent with the generation of data for individual events that were compared with the TLC's?

CS490.23 The criterion for preservation of coolable geometry in extremely unlikely events is no sodium boiling. The coolant saturation temperature at the top of the CRBR core is about 1800°F (1255K). Presumably, therefore, no phenomenon has been identified up to 1800°F that could affect coolable geometry. However, there may be a mechanism to compromise coolable geometry short of coolant boiling under loss-of-flow conditions. In the space between 1600 and 1800°F, significant numbers of end-of-life fuel pins could breach, releasing large amounts of fission gas. Studies have shown that at full flow, release of all of the gas in the fuel pins in one subassembly at the end of life could uncover the core portion of the subassembly, or the top part of it, for approximately 0.1 to 0.2 seconds. Presumably, the cladding that was uncovered would be without cooling during this time. The uncovered time would be much less than the approximate 0.8 seconds without cooling required at full power to reach the cladding solidus temperature. However, under loss-of-flow conditions (low flow), the time some portion of the core would be uncovered would undoubtedly be much longer, probably more than long enough to melt cladding at full power.

At the other extreme, if the power were instantaneously reduced to zero from full power, the fuel and cladding with no cooling would equilibrate to a temperature above the cladding solidus for all powers above about 20 kW/m. It therefore follows that indefinite loss of cooling is not necessarily tolerable even with an instantaneous scram. In short, the existence of some combination of flow coastdown, residual heat generation rate, and residual stored energy that would culminate in cladding melting, cannot be ruled out now for a loss-of-coolant event that penetrates the temperature space between 1600°F and coolant saturation.

Therefore, a no-boiling limit does not necessarily preclude loss of coolable geometry under loss-of-flow conditions. Rather, protection against loss of coolable geometry is ensured by scramming the reactor rapidly enough to avoid breach of any fuel pins. With these considerations, explain the adequacy of the no-boiling limit for undercooling events. The fact that there are no identified protected loss-of-flow events in which cladding (let alone coolant) exceeds 1600°F for end-of-life (high plenum pressure) fuel pins does not answer this question.

CS490.24 The W-2 test is a "slow" overpower test (about 5 cents/s ramp rate) conducted on full length FFTF geometry fuel pins in the Sodium Loop Safety Facility (SLSF) by the Hanford Engineering Development Laboratory (HEDL). It has not been fully examined or analyzed; nevertheless, the test has several important implications for the CRBR.

First, there is the puzzle of very early cladding breaches, possibly as early as 10 seconds into the transient, and with a breach definitely confirmed at about 15 seconds into the transient. These early failures were unexpected because of the low fluence that had been accumulated by the cladding.

Second, gross fuel expulsion occurred about as predicted by all of the prediction methods (as to time) at about 22 seconds into the transient. However, the site of the expulsion was apparently at axial midplane, which was unexpected.

Third, it is speculated that the site of expulsion may have been influenced by the early failure, which is presumed to have occurred at midplane.

The applicant is requested to comment on:

1) the implications of the early cladding breaches with respect to the adequacy of performance evaluation models and cladding failure criteria being used for the CRBR, and

2) the implications of the midplane site of the fuel expulsion, and of the influence the early failure may have had on the location of the site, for beyond-design energetics.

CS490.25 Plenum pressures for the FFTF were determined for design purposes assuming 100% release of fission gas. However, the CRBR takes credit for retained fission gas, predicting the fraction of release using a correlation to data (PSAR Page 4.4-40). In developing the correlation, were peak or average values of linear heat rating and burnup used to represent the overall pin? If peak values were used, will not the correlation underpredict the fractional release? This would not necessarily be detected by comparing predicted with observed values in Table 4.4-13 unless the pins in Table 4.4-13 had axial power distributions that were significantly different from those of the calibration pins listed in Table 4.4-12. Were the predictions shown in Table 4.4-13 made for nominal parameters or 2 sigma parameters?

CS490.26 What is the predicted plenum pressure for the absorber rods? How do predicted pressures compare with test data on absorber rods?

CS490.27 It appears that the CDF method for fuel pin performance evaluation is primarily oriented toward prediction of design life. Is the method used for evaluating the extent of damage short of design life, or is it used strictly as a fail, no-fail indicator? In applying the CDF method is the 1.0 for design life partitioned into separate allocations for steady-state and/or anticipated transients?

CS490.28 Please provide the available B_4C test data and documentation for the tests identified in Table 4.2-46A, PSAR page 4.2-413, as supporting the CRBR control assembly design. What relevant experience has been gained thus far in FFTF startup testing and operation with regard to CRBR control assembly design?

CS490.29 Although not necessary for review of the PSAR, all codes used in design and evaluation of the fuel and blanket rods will need to be reviewed. Based on our current understanding, the codes to be reviewed will include:

FURFAN
LIFE-III
any of the LIFE-IV series
anticipated to be used for the FSAR
FORE-2M
FRST

CS490.30 The coolant flow rate is automatically cut back upon the initiation of a reactor scram to minimize thermal shock. Are there any conceivable circumstances under which a scram could be called for, the rods fail to be inserted, and the flow cutback is still executed? What is the outcome of such an event?

CS490.31 On page 11 of reference 58 to PSAR Section 4.2 it states that "...all possible emergency events are divided into two broad categories according to the physical processes involved, viz., undercooling and rapid reactivity insertions." The definitions of the two categories appear to exclude any consideration of transient fuel cladding mechanical interaction on a slow time scale, that is, on a time scale much greater than one second. Yet, the possibility of such occurring clearly cannot be ruled out. In fact, all reactivity insertion (or overpower) events are fundamentally different from loss-of-flow events regardless of speed. How, then, does the CDF model evaluate "slow" reactivity insertion events?

ADDITIONAL QUESTIONS ON PSAR SECTION 4.2

CS490.32 If for whatever reason one or more absorber rods breached and B_4C were washed or eroded out of the breached rods, how would this be detected? What is the maximum reduction in shut down capability in either the primary or secondary control system that could occur through either burnout or washout at the detection threshold? Is any surveillance planned to ensure that the functional capability of neither the primary nor the secondary control system has degraded unacceptably?

CS490.33 Please show the quantitative delays in effecting a reactor scram starting with the time a real variable or quantity reaches its scram trip point and ending with the time that the power just starts to decline.

CS490.34 It is our understanding that rod bundle-duct interaction can cause substantial cladding stresses, at least for blanket rod bundles. Are these loads considered in evaluating fuel rod performance by either the CDF or the ductility limited strain model? If so, please provide a specific description of how this is done, including examples for both steady-state and transient conditions.

PRELIMINARY QUESTIONS TO APPLICANT ON PSAR SECTION 4.4

CS490.35 Many computer codes were used by the CRBRP designers to perform the thermal and hydraulic analyses presented in section 4.4 of the CRBR PSAR. Some of these codes are proprietary and some were developed by the CRBRP or its contractors and are not widely available. To evaluate the applicability of these codes to the thermal and hydraulic analyses presented in section 4.4, substantially more information is needed than is presented in section 4.4, in Appendix A, and in the references cited in Appendix A. Therefore, please provide code manuals and/or detailed descriptions along with code listings for the following codes.

a. CATFISH
b. CORINTH
c. COTEC
d. CRSSA
e. DEMO
f. FATHOM-360
g. FATHOM-360S
h. FLODISC
i. FORE-2M
j. NICER
k. OCTOPUS
l. TRITON

CS490.36 Part A: In the uncertainty analyses presented in section 4.4.3.2 of the CRBR PSAR, the rationale used to determine 2σ and 3σ uncertainty factors for thermal and hydraulic data is discussed. The discussion does not include a quantitative justification for non-statistical factors nor does it provide information about the methods used to determine statistical factors. Please indicate for the data presented in Tables 4.4-18A through 4.4-31 which of the uncertainty factors are determined statistically and which are not. Also, for the non-statistical factors please provide a quantitative basis and for the statistical factors please provide a detailed description of the methods used and of the data base.

Part B: In addition to uncertainties in material property data, design tolerances, and similar data there are uncertainties associated with the numerical methods (including models) used in the various computer codes. Are uncertainites in numerical methods (including models) included in the uncertainty factors presented in Tables 4.4-18A through 4.4-31? If uncertainties in numerical methods are included in the overall uncertainties, please provide a detailed mathematical description of the methods used to determine these uncertainties. If numerical method uncertainties are not accounted for, please explain why they are not.

CS490.37 According to the CRBR PSAR (section 4.4.2.5) the procedure used to determine assembly orificing for the heterogeneous core is based on a 3 loop natural circulation transient with an imposed maximum coolant temperature of 1550^{o}F. Using this method, minimum required flows are calculated and used to determine flows for 12 orificing zones. The above procedure resulted in a minimum core flow of 93.07% of total flow out of a maximum allowed core flow of 94% of total flow. What would the result have been if, instead of using PEOC, THDV at 3 had been used to define the temperature T_M?

CS490.38 In section 4.4.2.6 of the CRBR PSAR there is a discussion of reactor coolant flow distribution at low flow conditions. It is stated there that a system of three computer codes (DEMO, COBRA-WC, and FORE-2M) was used to assess the effect of all natural circulation cooling on the maximum coolant temperatures in CRBR. Please provide a detailed description of the geometry modeled by each of the codes and of the data coupling between them, i.e. output used as input, for the calculations discussed in the above section. The geometry model information should include the number and type of assemblies modeled, the number of fuel or blanket rods in each assembly that are modeled explicitly, the LIM model, and the UIS model. Also, please provide detailed results, i.e. temperature distributions and flow rates as a function of time, for the calculations used to arrive at the conclusions presented. No experimental evidence of natural circulation cooling for the CRBR heterogeneous core geometry is presented in this section. Are there any experimental data? If not, what type of experiments are planned to demonstrate the conclusions presented?

CS490.39 In section 4.4.2.8.5 there is a discussion of fuel-cladding gap effects on peak cladding temperatures reached during an undercooling transient. The discussion concludes that under LOF conditions with scram it is conservative to overestimate heat transfer to the cladding early in the transient, i.e. a higher peak cladding temperature will be calculated. Please provide quantitative justification, i.e. transient temperature results, for this conclusion.