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EXPERIMENT-SPECIFIC ANALYSES IN SUPPORT OF CODE DEVELOPMENT

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Boiling Water Reactor Core Melt
Progression Phenomena Program*
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ABSTRACT

Experiment-specific models have been developed since 1986 by Oak Ridge National Laboratory Boiling Water Reactor (BWR) severe accident analysis programs for the purpose of BWR experimental planning and optimum interpretation of experimental results. These experiment-specific models have been applied to large integral tests (ergo, experiments) which start from an initial undamaged core state. The tests performed to date in BWR geometry have had significantly different-from-prototypic boundary and experimental conditions because of either normal facility limitations or specific experimental constraints. These experiments (ACRR: DF-4, NRU: FLHT-6, and CORA) were designed to obtain specific phenomenological information such as the degradation and interaction of prototypic components and the effects on melt progression of control-blade materials and channel boxes.

The experiment-specific models supplement and support the systems-level accident analysis codes. They allow the analyst to accurately quantify the observed experimental phenomena and to reduce the effect of known uncertainties. They provide a basis for the efficient development of new models for phenomena that are currently not modeled (such as material interactions). They can provide validated phenomenological models (from the results of the experiments) that may be incorporated in the systems-level "whole-core" codes.

Applications of ORNL models specific to the ACRR DF-4 and KfK CORA-16 experiments are discussed and significant findings from the experimental analyses are presented.

1. INTRODUCTION

After the accident at Three Mile Island Unit 2, the United States Nuclear Regulatory Commission (NRC) initiated a severe accident research program,[1-3] for the general purpose of developing a basis for evaluating reactor core melt progression and ultimately, for assessing the release of fission products from the plant site and the ensuing threat to the public health. An important element of this program has been the development of state-of-the-art computer codes, such as MELPROG,[4] MELCOR,[5] and SCDAP/RELAP5,[6] which model the physical and chemical processes occurring in a reactor core that has lost cooling capability. A second important element of this program has been the generation of a database by which such codes may be assessed and validated. This database consists of experimental evidence obtained from several in-pile and out-of-pile phenomenologically-oriented programs[7-12] supported by the NRC and its foreign Severe Accident Research Program partners.

The roles of the experimental programs are first to provide data to understand the important in-vessel melt progression processes, and second to provide data for model (code) development and validation. The analytical tasks focus on employing the experimental information base to develop computational models and codes that provide the mechanism for translating the database that has been generated to the analysis of nuclear power plants (NPP). Computational models serve as the principal mechanism for performing NPP analyses.

In the current systems codes (MELCOR and SCDAP/RELAP5), only the key physical processes that occur during melt progression are treated in order to capture the global response of the reactor during a meltdown accident. Processes are not treated on a microscopic level; rather, the treatment is macroscopic, with the level of detail dependent on the particular code. Engineering judgment has guided the development of the phenomenological models employed in the systems codes. This judgment has been used to simplify processes as much as practical in order to develop computational tools (ergo, codes) that can be run at an affordable cost and at the same time capture the major response of the reactor system.

The ability to simplify the modeling in the NPP codes is a direct result of having prototypic experimental data <u>and</u> detailed models of phenomena. The phenomenological models clarify the uncertainties in key processes and provide a firm technical basis for the simplified models in the NPP codes. The detailed phenomenological models allow the performance of detailed analysis of the experiments to gain a good understanding of the important physical processes. This understanding is then used to form the foundation for the simplified NPP models. Without the detailed phenomenological models and the thorough analysis of the experiments, the models in the NPP codes cannot be adequately justified. It is important to note that inclusion of detailed phenomenological models in the NPP codes (as attempted in MELPROG) has not proven to be practical to date.

2. EXPERIMENT-SPECIFIC MODELS AND ANALYSIS

The combined effort of experimental and analytical work has not yet resolved all in-vessel melt progression issues. Two of the reasons for this are as follows: First, the phenomena and processes that occur during melt progression are complex, and it has proven to be virtually impossible to use first principle models to treat all processes. Second, it has been very difficult to obtain sufficient experimental data to resolve the issues and validate the modeling. Hence, uncertainty remains in the validity of the modeling that is currently used. In assessing models and experimental needs, the following question should be considered: How is the experimental data used to develop confidence (or validate) the system-level codes? Ideally the desired process would be:

1) identify information needs,
2) design best experiments to meet needs,
3) perform experiments,
4) analyze and interpret results, and
5) validate system-level modeling.

For this process to be effective, the experiments should be well-scaled, should use prototypic materials and heating methods, and should be designed so that the influence of experiment-specific features is minimized. Under these circumstances, the process would be straightforward to execute. Unfortunately, limitations in scale and the attainment of extremely high temperatures in meltdown tests have prevented the performance of experiments that do not inherently incorporate experiment-specific features. As a result, the transition from Step 4 to Step 5 is not straightforward.

For example, experiment-specific features usually lead to heat losses that cannot be accurately modeled unless a very detailed approach is used. This level of modeling is currently beyond that required in codes used for system analysis. Without predicting the correct thermal response, it is virtually impossible to predict any of the important phenomena that occur after melting begins. Hence, the real dilemma is that these experiment-specific features have a dramatic influence on the behavior of the experiments, and proper consideration of these features is a necessary prerequisite to separating their effects so as to permit the desired interpretation of the experimental data.

Therefore, for analyzing and interpreting the results of an experiment it is necessary to use models (or codes) that treat a broader range of phenomena than would be required in a system-level analysis (e.g., the exact experimental geometry, the behavior of the insulation material). The experiment analyses are needed to interpolate between data points (both temporally and spatially), and to extrapolate results when instrumentation (thermocouples) fails at high temperatures. Without analyses that are detailed and comprehensive, the usefulness of the "raw data" is greatly diminished.

It could be argued that the experiment-specific features should be added to the system-level code. However, simply adding the necessary modeling for the experiment-specific features does not resolve the whole problem. The reason that experiment-specific models are needed is that the data obtained from the experiments is not very useful without additional analysis. This is primarily due to the type of experiments that are performed and the tendency for experiments to be designed with the primary goal of addressing a particular phenomenological issue and less emphasis on obtaining data for code validation. If tests could be performed with better controlled boundary conditions, then the need for experiment-specific codes would be eliminated. Note, however, that there would still be the need to perform detailed experiment analysis to gain a thorough understanding of key phenomena of interest. This understanding is used to form the justification of simplified NPP models.

Experiment-specific models (codes) allow data analysis to reduce uncertainties and obtain a better understanding of the experimental data and the formulation of models. If analyses are performed with integral NPP codes, uncertainties in calculated results will inherently be larger than the range of phenomenology of interest and model formulation and validation will be more difficult.[13]

Results from the ORNL experiment-specific analyses for the DF-4 and CORA-16 BWR experiments will be presented in the two following sections of this paper.

3. DF-4 BWR EXPERIMENT

The DF-4 experiment was the fourth test in a series of in-pile experiments conducted in the Sandia (SNL) Annular Core

Research Reactor (ACRR) investigating severe fuel damage phenomena for core accidents. The first three ACRR experiments relate to pressurized water reactor accident sequences, while the DF-4 experiment was the first experiment to address the behavior of boiling water reactor (BWR) structural and control components in a high temperature environment under severe accident conditions.

The staff at Sandia recognized that the phenomenology of a severe accident in a BWR has a number of striking differences from the severe accident progression expected for a PWR. The proposed DF-4 test was aimed at providing a database for code validation of phenomenology unique to BWR severe accidents. Given the uniqueness of the DF-4 design (see Fig. 1) and the assumed high heat losses through the test section shroud, it was a consensus of a joint November 1985 meeting (NRC, SNL, and ORNL) that an experiment-specific model for the DF-4 be developed by the BWRSAT Program at ORNL. At the time, the possibility that the DF-4 test might be the only degraded core BWR experiment was real (FLHT-6 and the CORA tests were not even in the planning stages); therefore, extensive pretest planning and calculations were highly desirable. It was also felt that the generalities of the then-existing systems codes that could be applied to BWR calculations could not adequately model the specific structures of the DF-4. Thus, a code based on the core, chemistry and physical property modules developed for BWR applications at ORNL with specific models of the DF-4 structures was to be developed by the BWRSAT Program staff and then provided to Sandia for pretest planning calculations.

The BWR/DF-4 code[13,14] was used extensively by Sandia personnel[15] in the summer of 1986 in order to invoke the desired thermal performance of the test while keeping the operational parameters of the experiment within the limits of the experiment envelope. Specifically, the pretest analysis objectives included:

1) definition of achievable experimental parameters (that is, steam flow rates, neutronic and chemical power generation, hydrogen generation rate, assembly heating rate),
2) estimation of the overall response of the experiment,
3) definition of the timing criterion for sequencing events (i.e., power adjustments, steam flow rate adjustments, etc.),
4) assurance that the experiment would satisfy the overall objectives of the test, and
5) assurance that the experiment would satisfy the safety criterion for the facility.

The DF-4 experiment was successfully completed in November 1986. Subsequently, Sandia personnel initiated posttest calculations with the experiment-specific code.[15] These calculations were carried out to aid the characterization of the experimental data (i.e., the calculations facilitated understanding of the important phenomenology) and to interpolate and extrapolate for information at locations where instrumentation was either lacking or perturbed. The end result of their effort[14] is a well characterized data set by which developing codes could be evaluated/validated.

The April 1988 Severe Accident Research Program Partners Review Meeting provided the time and place for the assemblage of an inter-laboratory team for the analysis of the DF-4 experiment. The team was commissioned to perform posttest analyses of the DF-4 experiment utilizing state-of-the-art computer codes that treat severe reactor damage processes. The codes employed and the laboratories performing the calculations were: MELPROG (SNL), SCDAP/RELAP5 (INEL), MELCOR (ORNL), and BWR/DF4 (ORNL). MELPROG, SCDAP/RELAP5 and MELCOR are integrated, generalized, system-level severe accident progression codes, whereas BWR/DF4 is the experiment-specific code developed by ORNL and provided to the DFR staff at Sandia National Laboratories in 1986. A comparison of the posttest analytical results is available in Ref. 13.

Given the constraint (generally true of the system-level codes) that the experimental cross-section must be cast in a radial configuration, the fluid flow cross-sectional areas, structure volumes and masses can be closely approximated; but, the radiation view factors are modeled with less quantitative accuracy. The most crucial aspect of modeling the heat-up phase of the DF-4 test is the proper representation of the structure-to-structure radiation heat exchange. In system-level modeling the representation of the radiation exchange (via view factors) may or may not be representative of reality.

In the application of the BWR/DF4 code, care was taken to replicate the true DF-4 experimental geometry - the most important aspect of which is the ability to use accurate structure-to-structure radiative properties. This also forecloses the luxury of being able to blame lack of agreement on imprecise geometry; lack of agreement is an experiment finding that must be explained.

The BWR/DF4 code was designed to predict the heat transfer rates, chemical reaction rates, and the temperatures in the test section during the experiment. The heat transfer processes include conduction in solid structures, convection and radiation in the gas phase and radiation between the interacting surfaces. Metal/water reaction kinetics are modeled to determine the reaction rates of steam with Zircaloy in the rods and canister, with steel in the control blade, and with B_4C absorber in the control blade. From the metal/water reaction calculations are derived the hydrogen generation rates and the temporal and spatial distributions of oxide formations. The code is thus able to provide estimates of the temperature history of the test assembly as well as clad oxidation, hydrogen generation, and the extent of melting and relocation of assembly components. The primary limitation is that flow channel blockage due to the relocation of molten materials is not currently modeled, and therefore the code cannot be applied after significant relocation has occurred.

For the posttest analyses, measured information supplied the initial spatial temperature profile in the test section and boundary conditions were specified by the measured ACRR power and inlet steaming conditions. BWR/DF4 provides a solution to this initial/boundary value problem which may be assessed by comparing calculated and measured core structural thermal responses. Given true representation of the actual experimental geometry, one should expect close agreement of the code predictions with the experimental results, at least through the early-phase melt relocation. If such agreement is not attained, then it is highly probable that necessary phenomenology is missing in the modeling.

Comparisons of experimental results to BWR/DF4 predicted core structural thermal response are discussed for the 6000-7800 s time frame. Code calculations were stopped at 7800 s for the following reasons:

1) All of the control blade (except the top 11 cm) relocated to the -40 to 75 mm region of the test section,
2) Significant Zircaloy relocation (cladding and canister) had occurred, and
3) Blockage models have not yet been implemented in BWR/DF-4.

In other words, since massive structural relocation is predicted and observed by 7800 s, extending the calculations further in time would exceed the applicability of the current code models.

Comparisons of the observed and calculated fuel cladding and control blade sheath thermal responses are presented in Figs. 2 through 4 respectively for the nominal 96 mm, 254 mm, and 368 mm axially instrumented planes. At 7200 s, the differences between the observed and calculated control blade temperatures at the 96, 254, and 368 mm planes are +20, ±5, and ±10 K respectively; similarly, for the cladding the differences are ±3, +20, and +20K.

Overall, the models employed in BWR/DF4 for convective, conductive, and radiative heat transfer, early phase relocation (candling), and the fission and chemical energy components of the structural energy balances adequately predict the events and the timing of events for the DF-4 experiment.

It should be recognized that the results presented here represent the evolutionary end result of the DF-4 analyses and there were many lessons learned during this calculation process.

The more important benefits of this exercise (such as identification of the appropriate oxidation kinetics) gleaned from the analytical modeling of the DF-4 experiment will be discussed in the following paragraphs.

3.1 Control Blade Model

Given true representation of the actual experimental geometry, one should expect close agreement of the BWR/DF4 code predictions with the experimental results, at least through the early phase melt relocation. If such agreement were not attained, and given reasonable accuracy in the experimental measurements, then necessary phenomenology must be missing from the modeling. The initial control blade models employed in BWR/DF4 are a prime example of a case of missing phenomena.

Referring to Fig. 5, use of an accepted literature value for the melting temperature of stainless steel of 1672 K (2550° F) as the single failure temperature of the control blade results in a significant lag in the predicted time-to-melt for the control blade and in the predicted thermal response due to candled material at the 236 mm level. The video record taken during the experiment indicated significant melt relocation starting at ~7440 s and the control blade temperatures at the 236 mm level were the first instrumented positions in the core (at 7455 s) to respond to relocating material. however, using a melt temperature of 1672 K, incipient melting of the control blade was not predicted to start until 7545 s.

Consultation with Gauntt (Sandia) led to the supposition that the B_4C absorber chemically attacked and liquified the absorber tube sheath at temperatures substantially lower (100-200 K) than the normal melting temperature of stainless steel (this is consistent with Hagen's observations).[16,17] This liquified material then either breached the outer stainless steel sheath or simply flowed through existing perforations in the outer sheath. (The control blade was prototypic in design; BWR control blade perforations allow water cooling of the absorber tubes under normal conditions.) Thus, a liquified B_4C/steel eutectic mixture could candle down the control blade while the outer sheath was still intact. Also, the outer sheath would fail at a different (normally higher) temperature.

Models reflecting this hypothesis that allow failure of the control blade absorber tubes at a difference temperature than the control blade sheath were implemented in BWR/DF4. Simulations resulted in good agreement between the observed and calculated control blade response (Fig. 2-4). For these simulations, failure of the absorber tubes occurred at 1505 K (2250° F) and failure of the outer sheath occurred at 1672 K (2550° F).

Hofmann's research[18] (circa 1989) has shown that the chemical interactions of B_4C with stainless steel can be described by parabolic rate laws and that melting will occur above 1473 K (2191° F) with rapid liquefaction above 1523 K (2281° F). Liquefaction occurs below the melting points of the components due to eutectic interactions.

The BWR/DF4 simulations have indicated that the response of a BWR control blade structure (as in the DF-4 experiment) can be adequately predicted <u>only</u> if the control blade models allow for reduction of the absorber tube melt (liquefaction) temperature.

The phenomena associated with the B_4C/steel interaction must be included in severe accident modeling of BWRs. They directly impact accident management measures and criticality issues in BWRs since the control blades would fail and relocate within the core at temperatures lower than expected (by as much as 200 K).

3.2 <u>Zircaloy/Steam Oxidation Kinetics</u>

At the completion (~7500 s) of the pre-transient phase of the DF-4 experiment, fuel rod cladding temperatures over ~126 mm of the axial length are calculated to be between 1700 and 1750 K. The onset of rapid Zircaloy oxidation occurs in the 7535-7555 s time frame, shortly after the power increase at 7500 s that starts the oxidation transient phase of the experiment. Given the excellent agreement between the calculated and observed cladding temperatures at the 96, 254, and 368 mm planes (all within 25 K) at 7500 s, an attempt was made to evaluate different Zircaloy/steam oxidation kinetic models for the oxidation phase of the experiment.

The Zircaloy/steam oxidation rate model features employed in BWR/DF4 allow three choices for high-temperature solid-state oxidation kinetics. The high-temperature solid-state oxidation kinetics options that are available include:

1) Urbanic/Heidrick[19] correlations (two kinetic rate laws with a temperature switch at 1853 K (2876° F),
2) Cathcart/Pawel[20] for temperatures less than or equal to 1853 K and Baker/Just[21] for temperatures greater than 1853 K, and
3) Cathcart/Pawel[20] for temperatures less than or equal to 1783 K (2750° F) and Prater/Courtright[22] for temperatures greater than 1783 K.

Of these options, use of the Urbanic/Heidrick oxidation kinetics was found to yield a better experimental/calculational comparison than the Prater/Courtright kinetics as shown in Fig. 6. (The Baker/Just kinetics give results intermediate to Urbanic/Heidrick and Prater/Courtright, but are not good as the results obtained with Urbanic/Heidrick kinetics.)

4. CORA-16 BWR EXPERIMENT

In the Federal Republic of Germany at the Kernforschungszentrum Karlsruhe (KfK), the Severe Fuel Damage (SFD) Program[23-25] was coordinated by the Project Nuclear Safety (PNS), now the LWR Safety Project Group (PRS). In the CORA Program[26], which is an important part of this effort, out-of-pile experiments are performed to provide information on the behavior of Light Water Reactor (LWR) fuel elements under severe fuel damage (SFD) conditions, i.e., in the temperature range from 1200 °C to above 2000 °C.

For the experimental approach at KfK, the CORA facility was especially designed for the SFD experiments. The decay heat of fuel rods is simulated in CORA by electrical heating using central tungsten heaters within the fuel rod simulators.

The CORA operational staff requested from the USNRC, under the auspices of the Severe Fuel Damage Partners Program, that the ORNL BWR-specific models be applied in support of future BWR experiments to be performed in that facility. Accordingly, the current Statement of Work for the BWR Core Melt Progression Phenomena Program, a derivative of the BWRSAT Program at Oak Ridge, provides for application of the ORNL BWR models to the BWR experiments to be performed in CORA.

The development of a CORA-specific BWR experimental model to analyze the results of CORA BWR experiments and the planning of future experiments was completed in May 1990. The CORA/BWR code includes sophisticated electrical heater-rod models, capability for appropriate noding for experimental analyses and an accurate representation of CORA boundary conditions.

The CORA-16 BWR experimental cross-section is illustrated in Fig. 7. At each axial node, the CORA BWR experimental structures illustrated in Fig. 8 are modeled by the CORA/BWR code. The structures shown in Fig. 8 represent a full 1/4 symmetrical section of the CORA BWR cross-section (see Fig. 7).

In the test conduct of the CORA experiments, the test sequence can be distinguished by three periods. In the first period, preheated argon enters the bundle, thus heating the structures and allowing the test section components to equilibrate in temperature. In the second phase, electric power is increased to 20-30 kW to achieve an initial heatup rate of 1 K/s; also, a constant superheated steam flow is added to the argon flow. The tests are terminated by reduction of electric power and cessation of steam flow (for a "slow" cooldown) or by movement of a quench tank over the test bundle (for a "fast" cooldown).

For the posttest analyses performed subsequent to the experiment, measured information supplies the initial spatial temperature profile in the test section and the boundary conditions are specified by the measured electric power and inlet coolant conditions.

Posttest analyses[28] of the CORA-16 experiment is ongoing. Although there are some known uncertainties that are primarily associated with the structural physical properties (such as surface emissivities), the predicted early phase structural thermal response compares well with the experimental data as shown in Figs. 9-11. Important results learned in this preliminary analysis will be highlighted in the following discussion.

4.1 Model Treatment of the Gaseous Coolant

ORNL modeling treats the gaseous coolant as a gray/interacting (i.e., absorbing/transmitting) medium. The methods for handling such a medium were first developed by Hottel.[29] The validity of his methods, tables, and charts remain relatively unchanged to this day. ORNL does use an upgraded version of the water vapor emissivity tables (function of optical depth and temperature) developed by Ludwig.[30] The applicability of these methods (in core uncovery and degrading conditions) has been demonstrated previously in the ORNL BDHT experimental analyses[31,32] and in the DF-4 experimental analyses.[13]

In essence, the treatment of the gas as a gray/interacting medium does affect the structure-to-structure radiation heat transfer since absorption can be a dominant process and it basically augments the convective heat transfer; that is

$$h_{\text{conv effective}} = h_{\text{conv}} + h_{\text{rad}}.$$

However, when applied to the 2.2 bar CORA-16 experiment, the radiant structure-to-structure heat transfer is underpredicted and the convective transfer to the gas from the structures is overpredicted as shown in Fig. 12.

Reviewing Ludwig's emissivity data (Fig. 13) indicates that, for the CORA-16 test conditions and computed optical lengths (0.2

to 0.6 cm-atm), the predicted gas emissivities are low in the data range (although still within the experimental range).

For the base case simulations (Figs. 9-12) the gas has been treated as transparent and non-interacting. Also, the convective heat transfer correlation employed in these analyses is the Sozer[32] correlation which appears (as demonstrated in the base case calculations) to adequately predict the convective heat transfer.

In CORA-9 (a 10 bar PWR small bundle experiment), the experimenters encountered difficulties in heating up the bundle (as compared to the normal 1 bar overpressure tests) and greater electrical power was required. For the 10 bar tests, optical lengths of 2 to 6 cm-atm would be expected and, referring to Fig. 13, this is roughly the DF-4 operating range. Therefore, one could surmise that the gas is behaving as a gray/interacting medium; thus the convective heat transfer would be augmented and the rods would heatup slower (the 'gray gas' curves in Fig. 12).

4.2 Test Section Spacers

Spacer models are included in axial nodes 8, 21, and 28 in the base case simulations. Node 28 (from 875 to 925 mm) has no instrumentation within the fuel assembly other than two spacer grid thermocouples (T126 and T129). Node 21 (from 525 to 575 mm), however, encompasses the most instrumented level (550 mm) in the core.

At node 28, there is no way to judge how effective a CORA/BWR run simulates the experiment; at node 21 however, there are multiple measurements to aid in judgment. Consistently, when the spacer was not included in node 21, the heated rod temperature in node 21 was underpredicted; but at all other levels the comparisons (below rod temperatures of 900 °C) were very good. This position (550 mm) in the core is observed to heat up to 1000 °C faster than any other measured position in the core except for the spacer measurement at 903 mm.

Finally, it was decided to develop and implement explicit spacer models in CORA/BWR, the conjecture being that the spacer caused the accelerated rod heatup. Fig. 14 show the results from simulations with and without spacer grid models. The simulation which includes the spacers compares more favorably with the data than the simulation without spacers. Apparently, as shown in Fig. 15, the spacers act as radiation shields, causing higher local rod temperatures.

4.3 Materials Interaction

Thermocouple response in CORA-16 is sufficient to indicate material relocation/interaction. For example, Fig. 16 at the 750 mm level shows the reaction of these Zircaloy structures to molten materials (probably to molten control blade constituents at temperatures of 1400 - 1500 °C) other than liquid Zircaloy (otherwise the measured temperature would have flared).

Data does exist for the CORA-16 experiment that could be used to develop and validate material interaction and blockage models.

5. SUMMARY

In order to develop a firm understanding of in-vessel melt progression, NRC has conducted a coordinated effort of experimental and analytical tasks. The roles of the experimental tasks are to provide data to understand the important processes and to provide data for model (code) validation. The analytical efforts have resulted in systems-level codes (MELCOR and SCDAP/RELAP5) that can be employed to predict the response of a nuclear power plant in severe accident scenarios. In the systems codes only the key physical processes that occur during melt progression are treated in order to capture the global response of the reactor during a meltdown accident.

In general the systems-level codes lack the detailed modeling required to account for the experiment-specific features (such as high radial heat losses) that invariably are encountered in high temperature meltdown experiments. The tests performed to date in BWR geometry have had significantly different-from-prototypic boundary and experimental conditions because of either facility limitations and/or experimental difficulties. Thus, specialized tools (experiment-specific codes) have evolved which allow the experimental analyst more detailed modeling of the experimental structures, boundary conditions, and the core melt phenomena.

The experiment-specific codes have been applied successfully in the posttest analyses of the DF-4 and CORA-16 BWR experiments and these analyses have provided specific insights in the following areas:

1) material properties and their interactions, (such as the B_4C/stainless steel interaction which results in a lowered control blade failure temperature)
2) Zircaloy/steam oxidation models,
3) candling heat transfer coefficients,
4) relative importance of various heat transfer modes,

5) modeling treatment of the gaseous coolant, and
6) role of spacers during core heatup.

However, the major role of experiment-specific codes (such as CORA/BWR) is to provide physics guidance to integrated code developers by providing insights that are not biased by geometric modeling considerations, such as:

1) clarification of experimental phenomena and timing of events,
2) indication of what needs to be modeled and what does not, or which models work and which do not, and
3) provision of a platform for developing and validating models that can be incorporated in integrated codes.

The experiment-specific models supplement and support the systems-level accident analysis codes. They allow the analyst to accurately quantify the observed experimental phenomena and to reduce the effect of known uncertainties. They provide a basis for the efficient development of new models for phenomena that are currently not modeled (such as material interactions). They can provide validated phenomenological models (from the results of the experiments) that may be incorporated in the systems-level "whole-core" codes.

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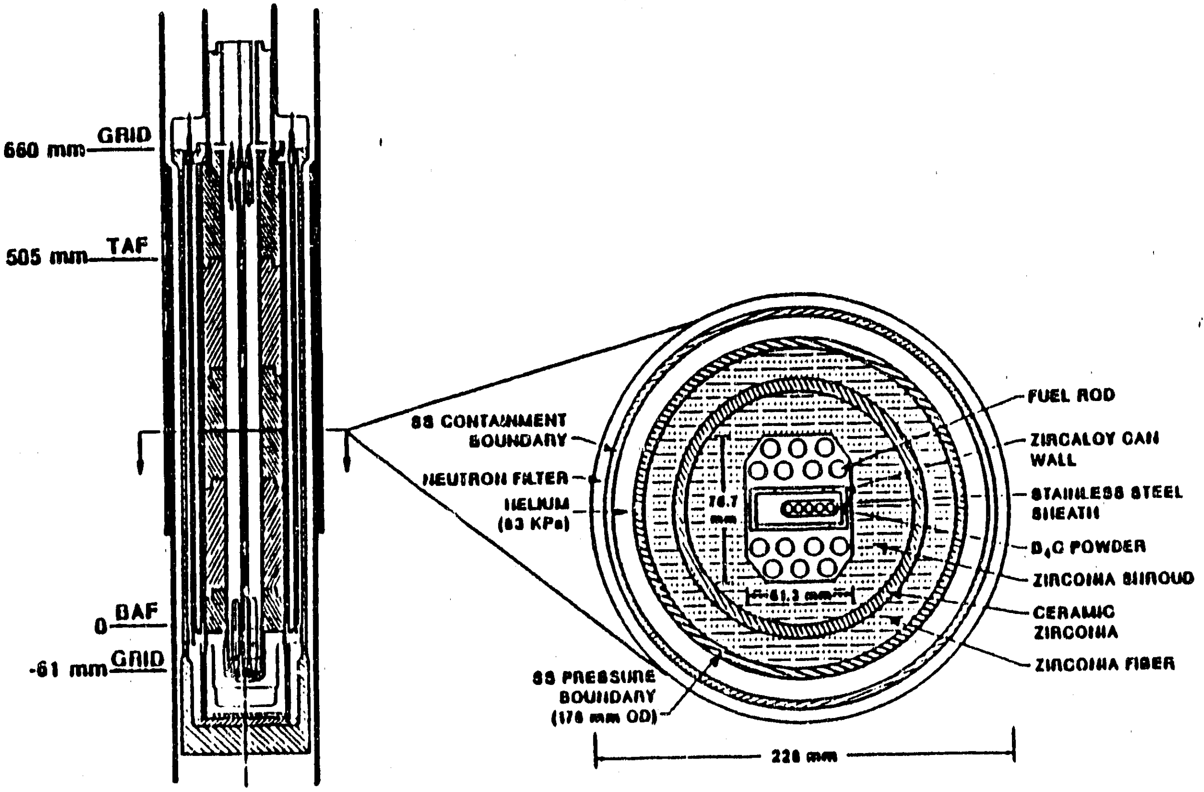


Fig. 1. DF-4 experimental geometry.

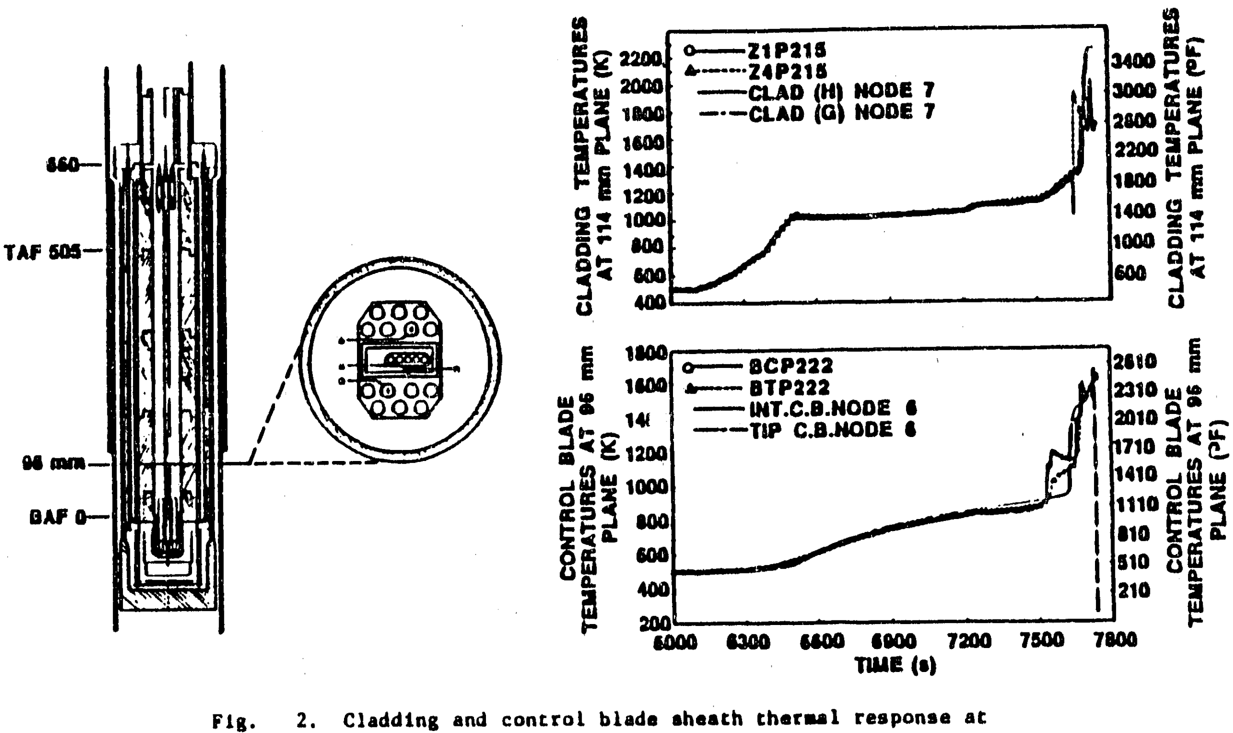


Fig. 2. Cladding and control blade sheath thermal response at the 96 mm plane. The circle and triangle figures represent thermocouple readings.

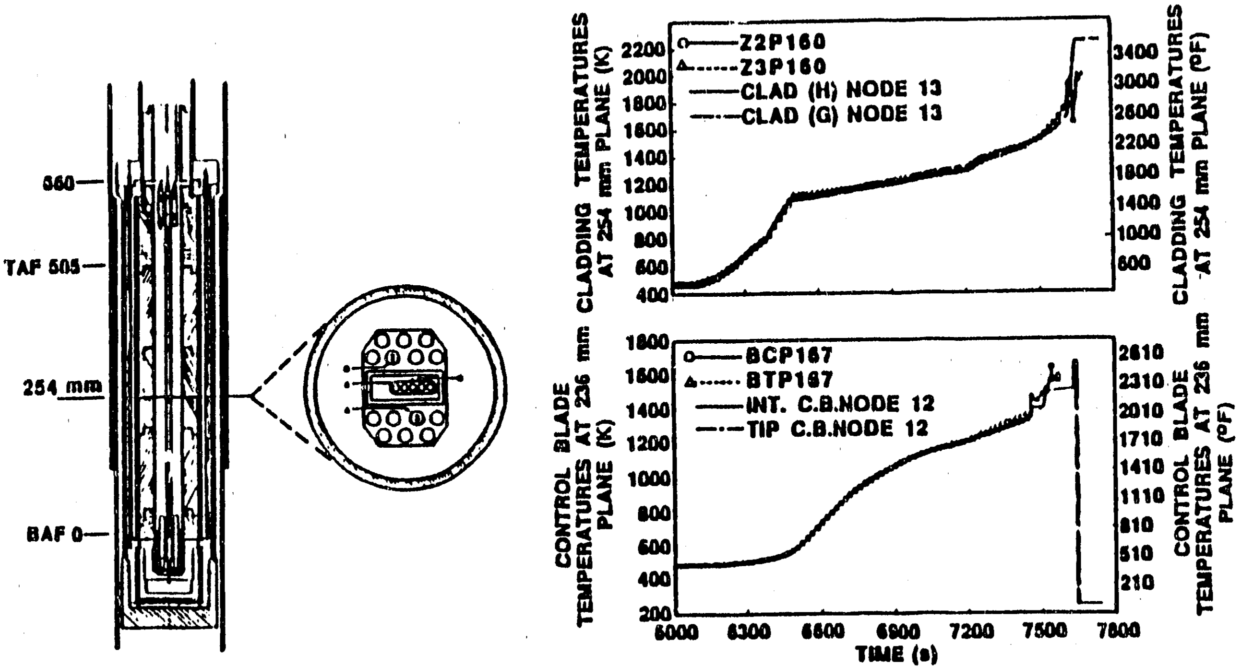


Fig. 3. Cladding and control blade sheath thermal response at the 254 mm plane.

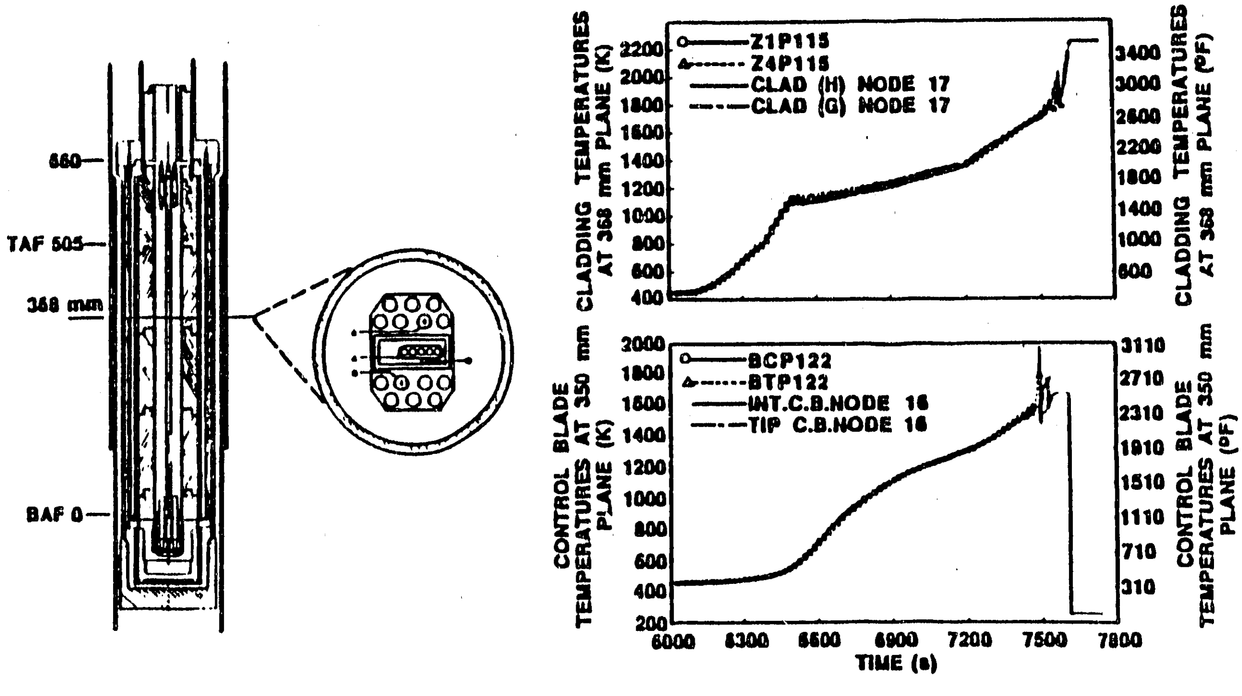
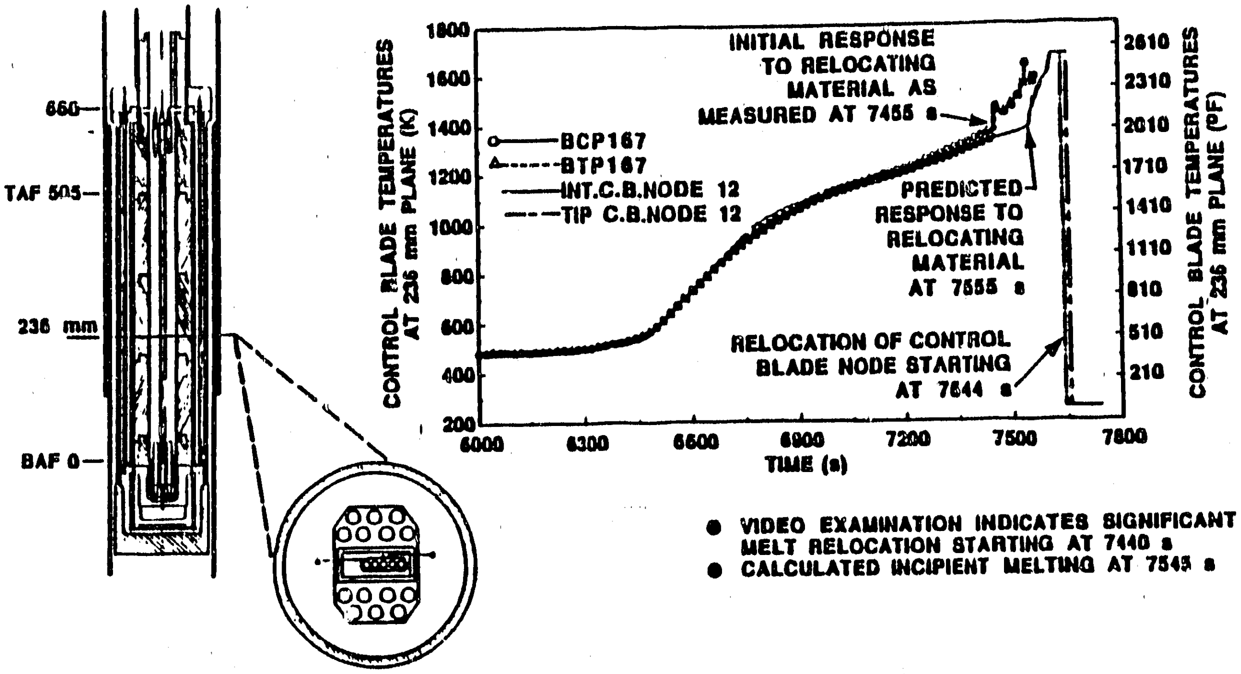


Fig. 4. Cladding and control blade sheath thermal response at the 368 mm plane.



- VIDEO EXAMINATION INDICATES SIGNIFICANT MELT RELOCATION STARTING AT 7440 s
- CALCULATED INCIPIENT MELTING AT 7545 s

Fig. 5. Control blade response at the 236 mm level using a single melting temperature of 1672 K for the control blade structure.

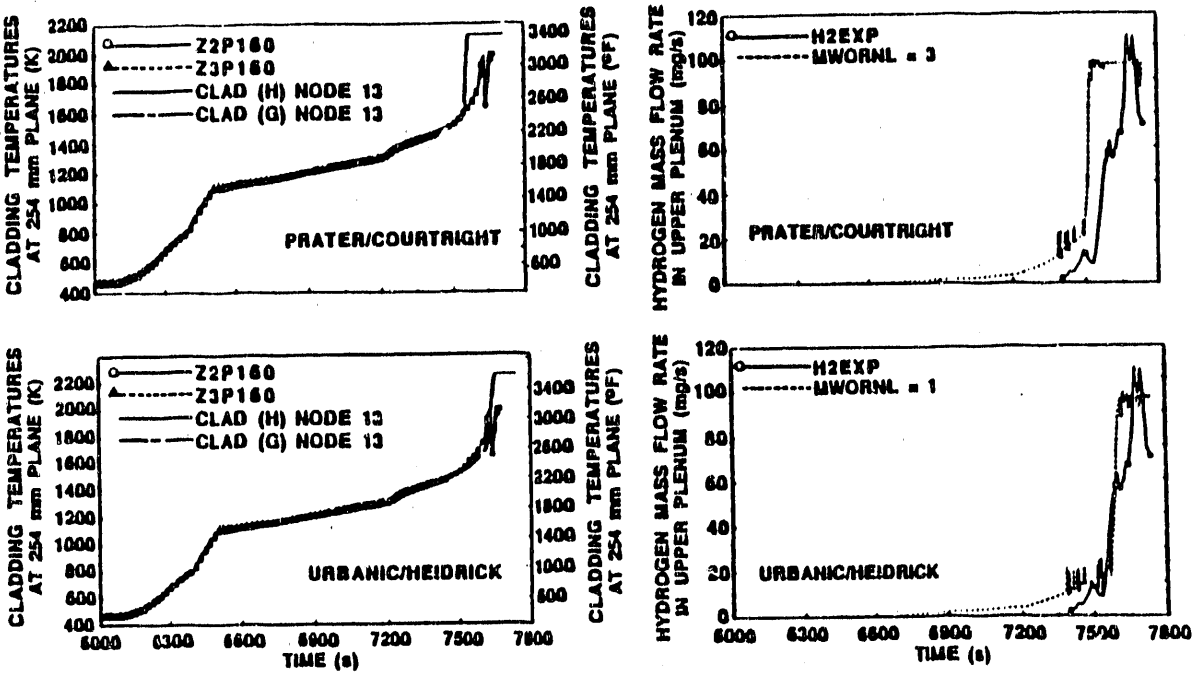


Fig. 6. Comparison of calculated/experimental results using Prater/Courtright and Urbanic/Heidrick zirconium/steam oxidation kinetics.

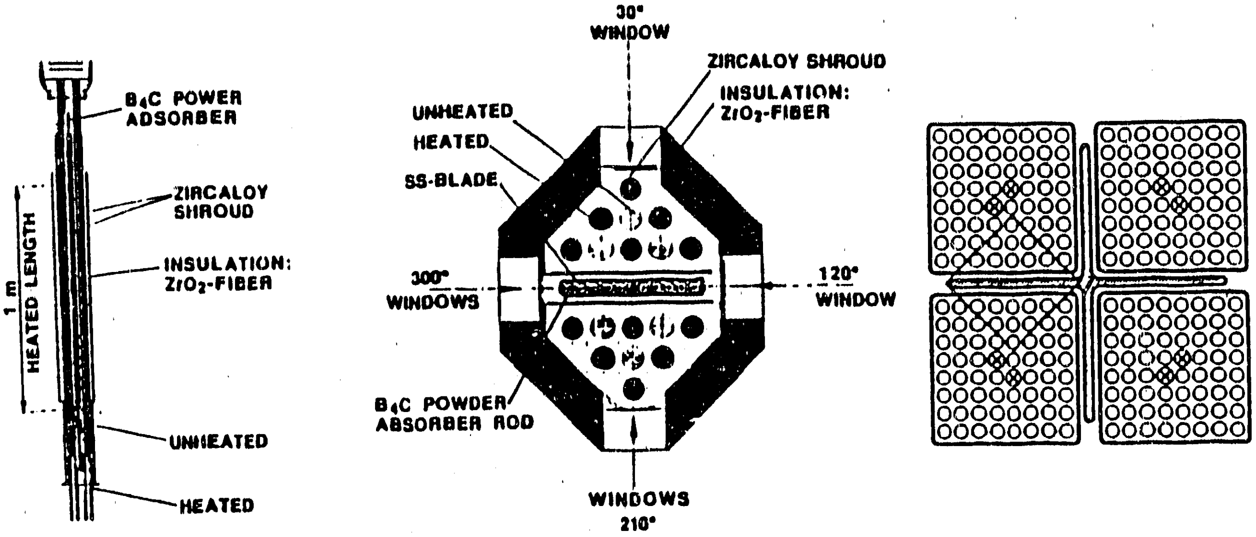
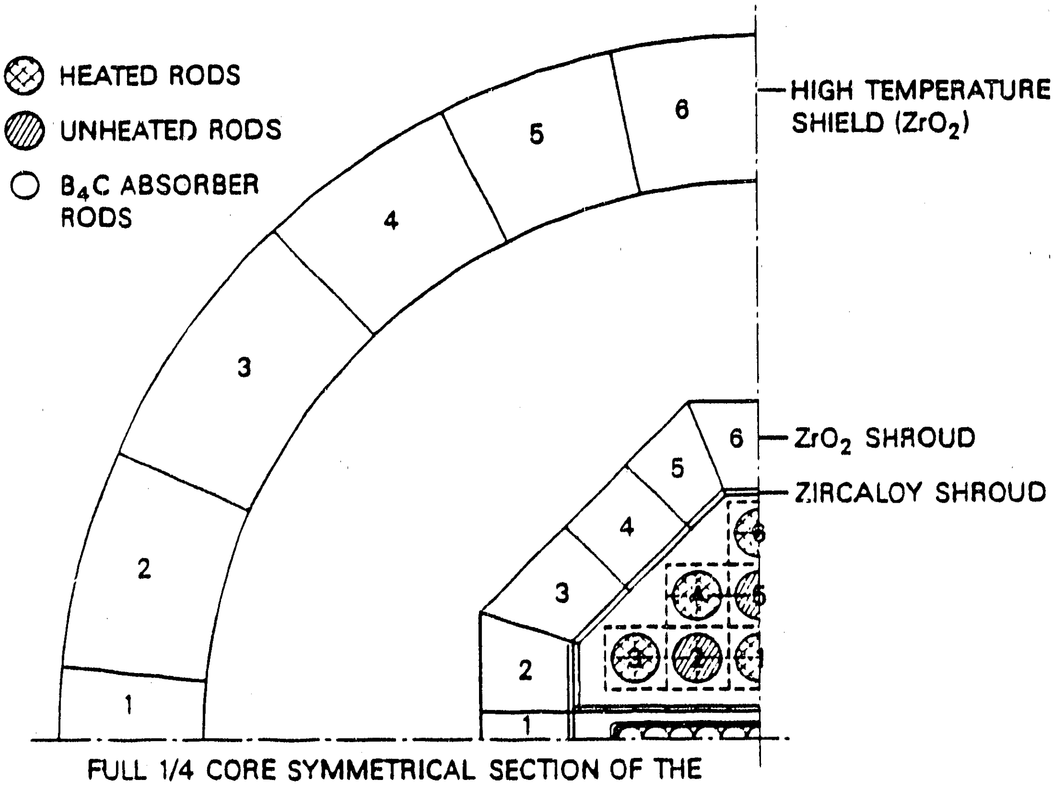


Fig. 7 . CORA BWR experimental cross-sections and the portion of a BWR core unit cell that is represented in the experiment.



FULL 1/4 CORE SYMMETRICAL SECTION OF THE CORA-16 CROSS-SECTION

Fig. 8 . CORA/BWR code representation of a CORA BWR test section.

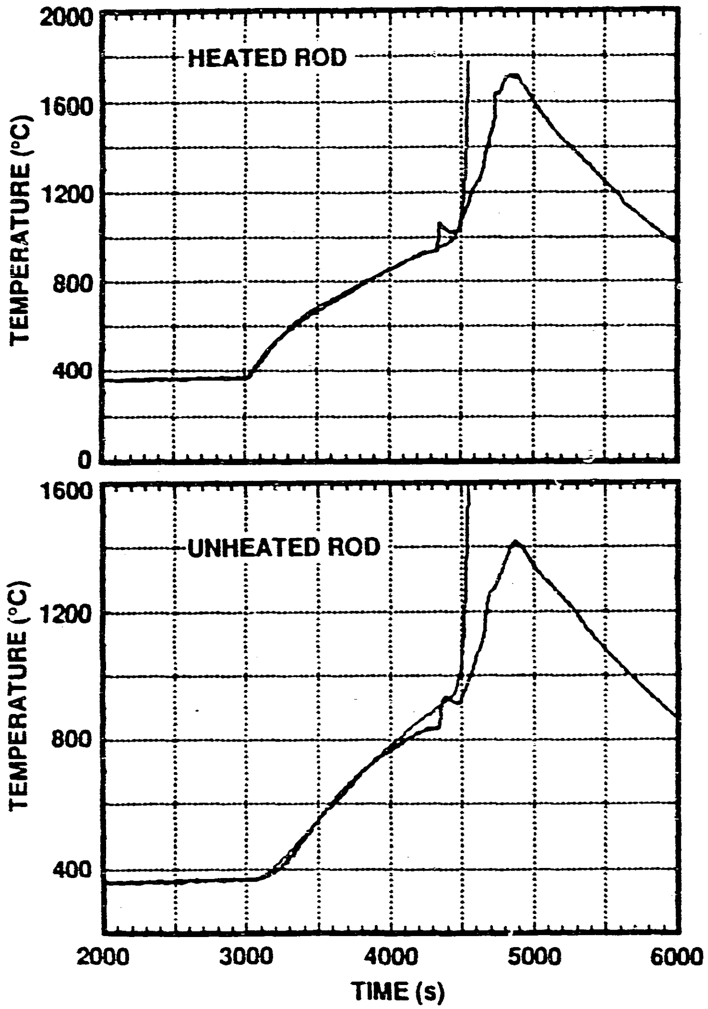


Fig. 9. CORA-16 structural thermal response at the 150 mm plane (heated and unheated rods).

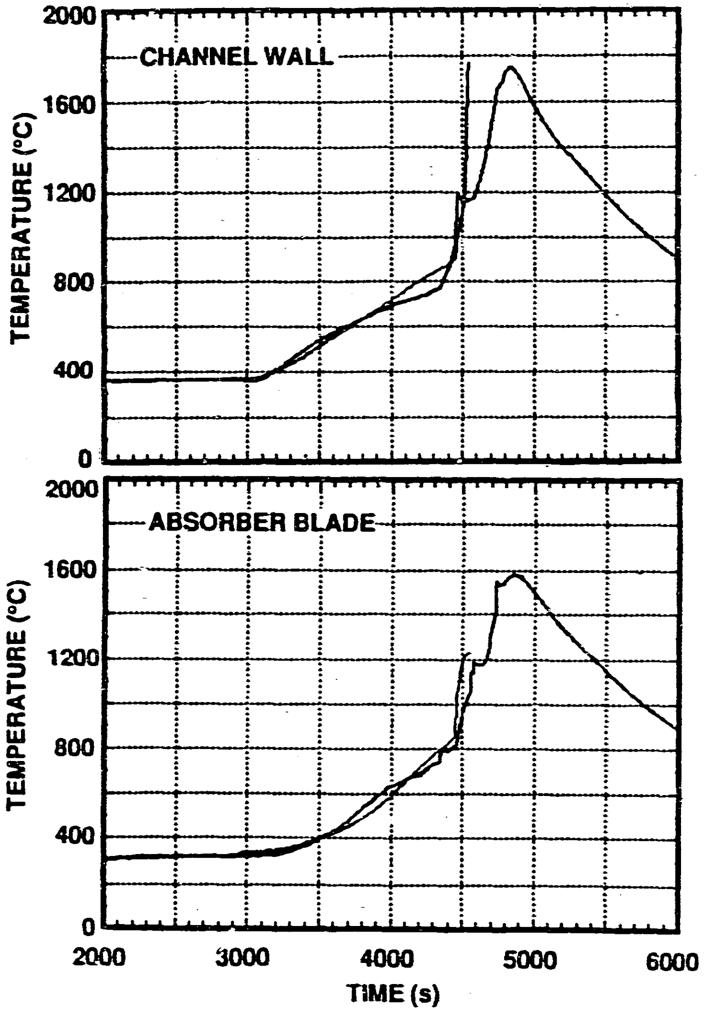


Fig. 10. CORA-16 structural thermal response at the 150 mm plane (channel wall and absorber blade).

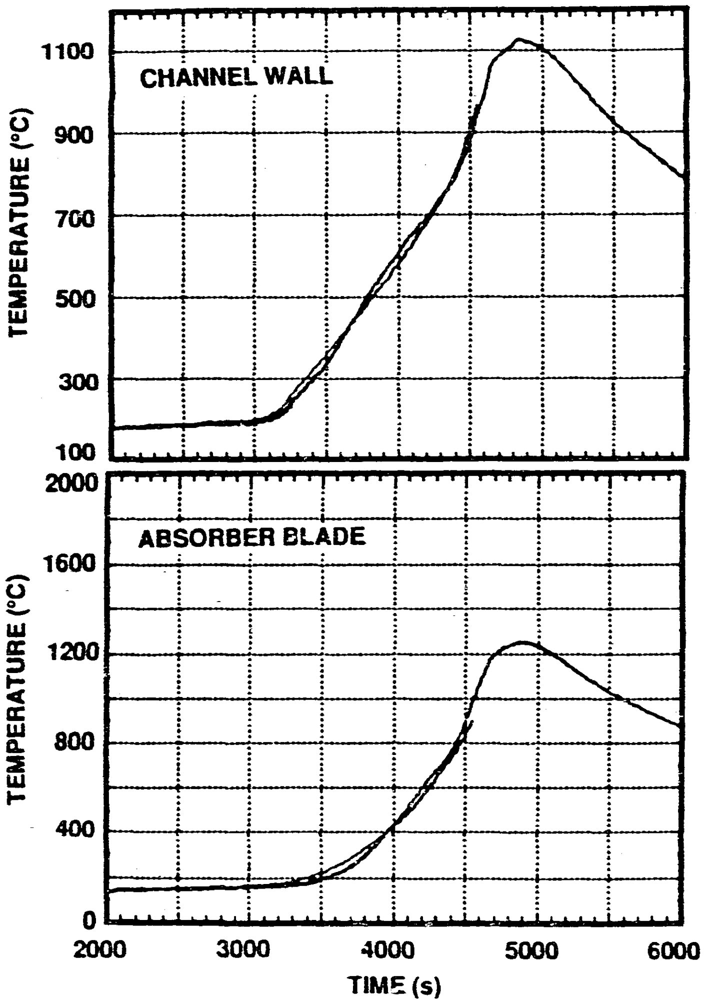


Fig. 11. CORA-16 structural thermal response at the 1150 mm plane (above the active heated length).

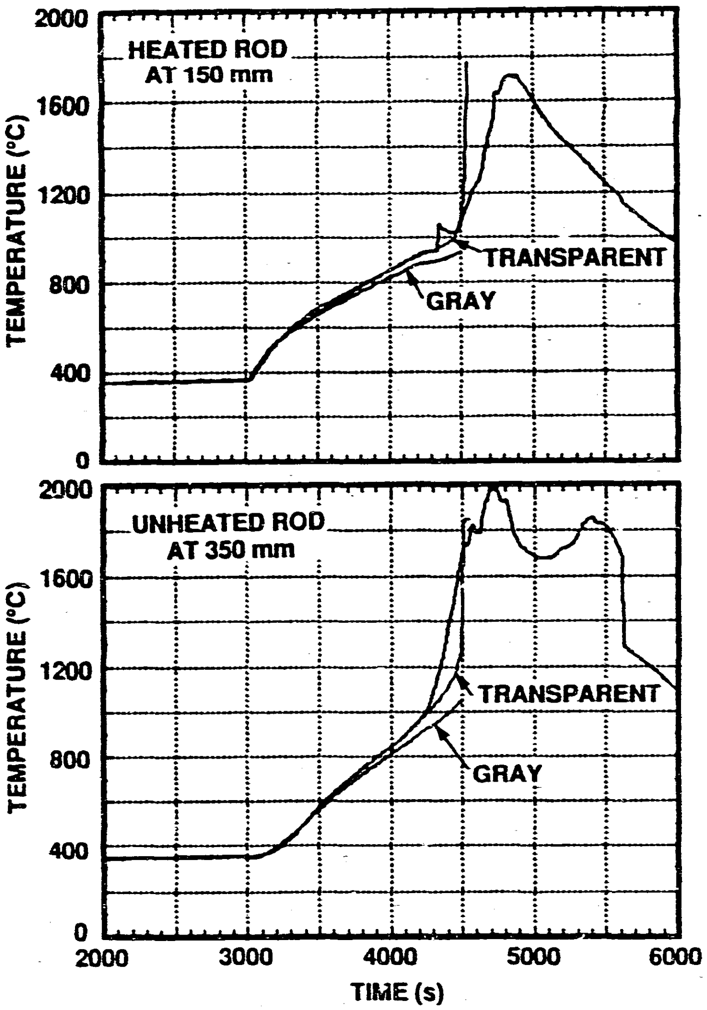


Fig. 12. CORA-16 predicted structural thermal response using transparent and gray gas models.

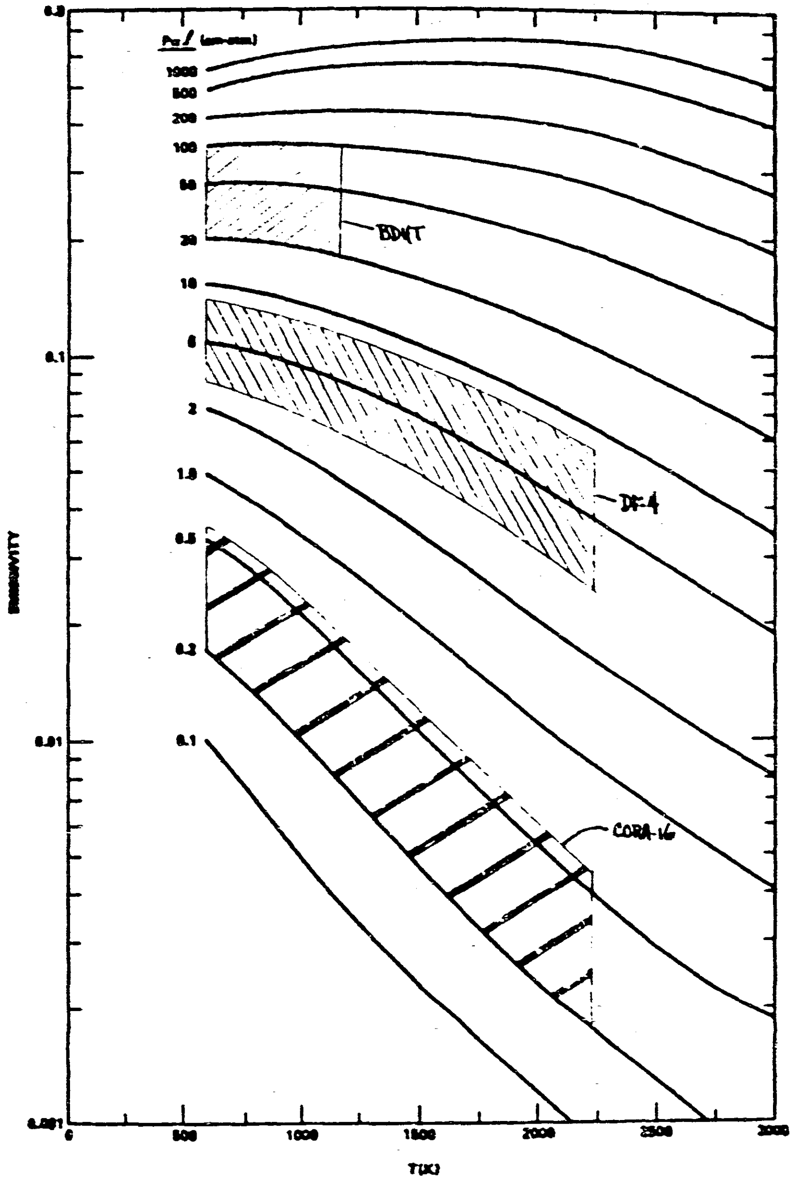


Figure 13. Total emissivity for water vapor.

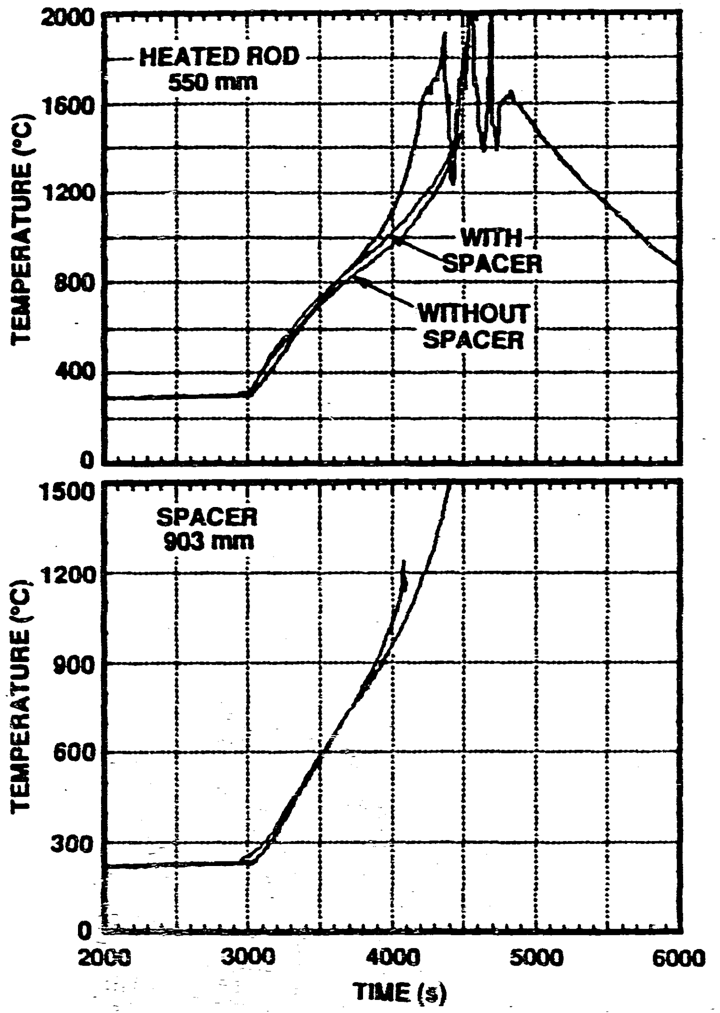


Fig. 14. CORA-16 predicted structural thermal response with and without explicit spacer models.

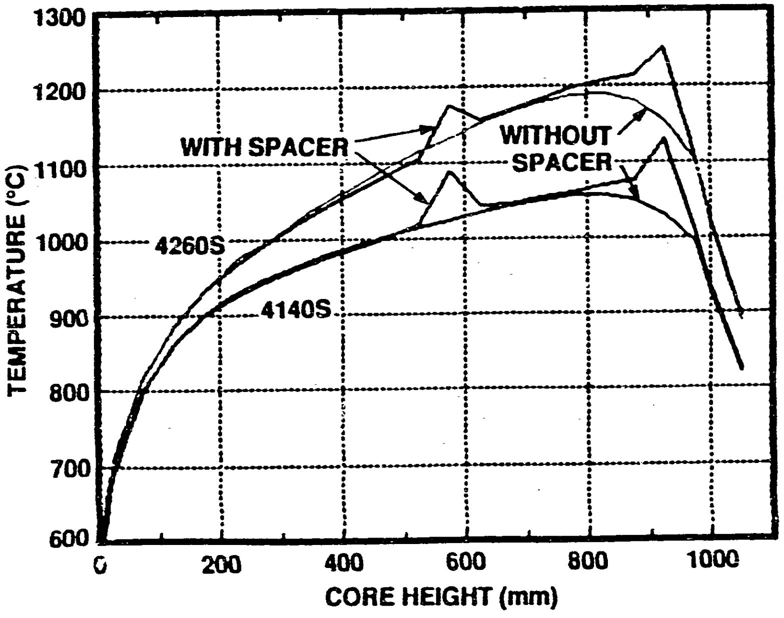


Fig. 15. Predicted axial cladding temperatures with and without spacer models.

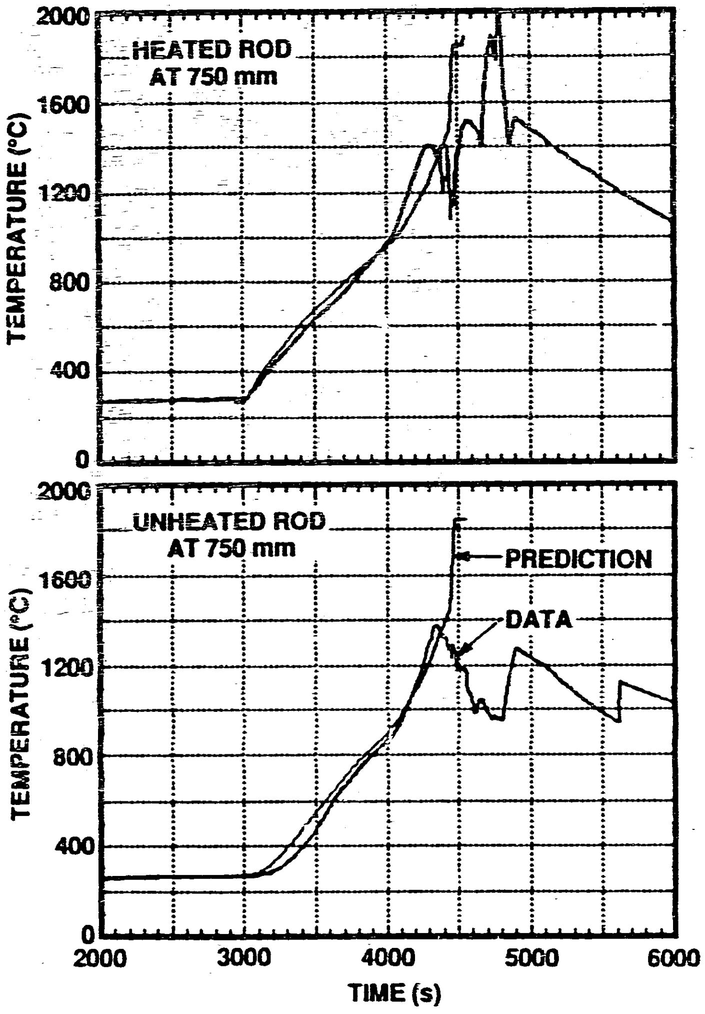


Fig. 16. CORA-16 thermocouple response at the 750 mm plane indicates material relocation/interaction.

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