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October 1980

FEASIBILITY ANALYSIS FOR THE SEVERE FUEL DAMAGE TEST PROGRAM IN THE POWER BURST FACILITY

R. W. Garner
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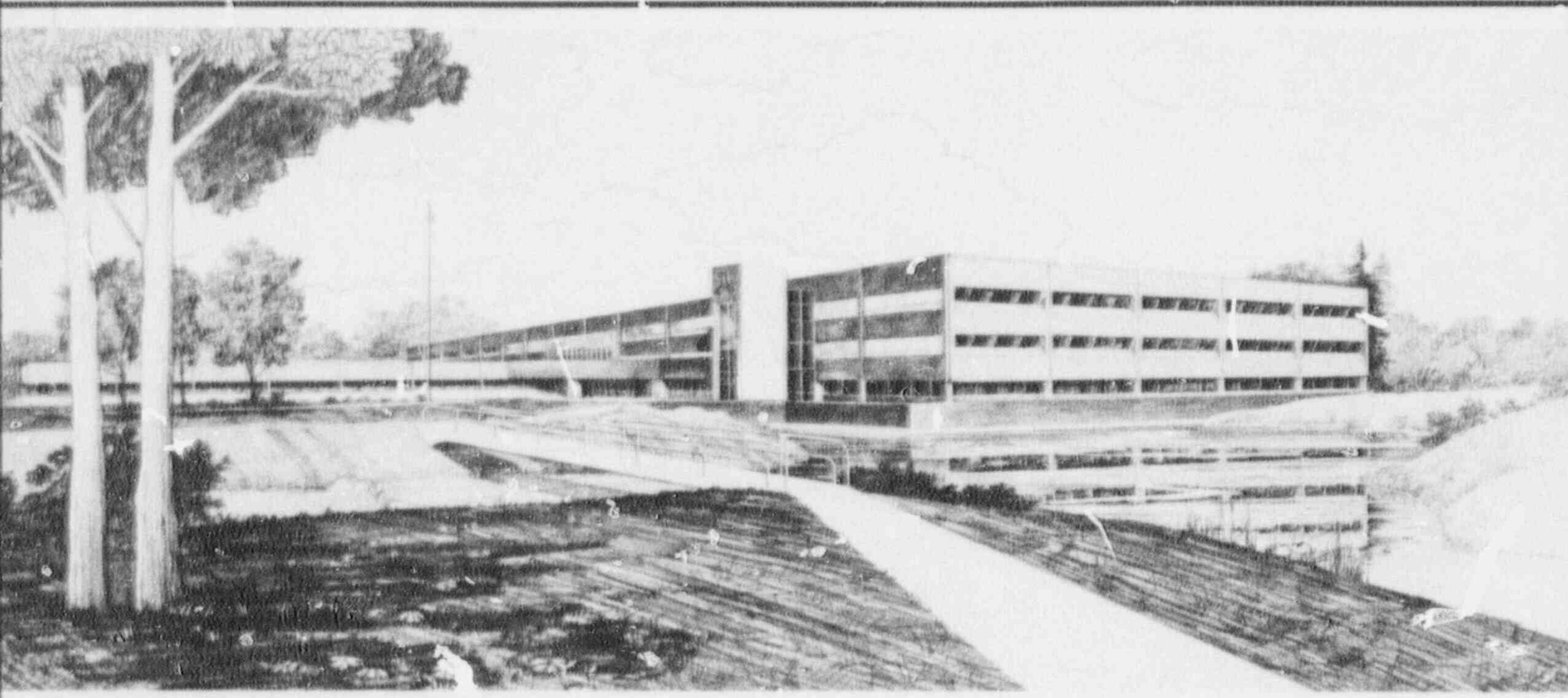
NRC Research and Technical Assistance Report

U.S. Department of Energy

Idaho Operations Office • Idaho National Engineering Laboratory



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1. INTRODUCTION

The United States Nuclear Regulatory Commission (NRC) has initiated a severe fuel damage assessment program to be performed in the Power Burst Facility (PBF) at the Idaho National Engineering Laboratory. This program is designed to evaluate fuel rod and core response during accidents more severe than design basis accidents. The proposed program[1] would eventually extend the data obtained from ongoing PBF programs[2-5] to conditions of gross fuel melting and possibly interaction of molten fuel with support structures and the pressure vessel. The initial phase of the test program, termed the PBF Severe Fuel Damage (SFD) Tests, will provide data to evaluate fuel rod and bundle behavior under system conditions which will expose preconditioned test fuel rods to a slow heatup to peak cladding temperatures of at least 2300 K, resulting in cladding ballooning and rupture in a manner similar to that which could occur in a commercial power reactor during a small break LOCA. Subsequent phases are intended to extend the experimental program to provide fuel behavior data at temperatures in the range from 2300 to about 3100 K.

The tests will be initiated from a low power, steady state operating condition with forced convection cooling. The coolant flow rate will be reduced to a predetermined value that will result in boiloff of the coolant and dryout in the upper regions of the test cluster. The tests will be terminated by either slow cooling or quenching of the fuel. Variations in test rod power during the temperature transient may be required to provide the desired fuel rod thermal-mechanical response.

This report describes the results of scoping calculations performed to evaluate the feasibility of performing the desired severe fuel damage experiments in the PBF. The feasibility analyses were performed with the computer code TRAC-BD0, which is a preliminary BWR version of the TRAC-P1A code that was developed for the analysis of LOCA transients in PWRs. Section 2 provides a brief description of the sequence of events that is expected to occur during a small break LOCA. Specific objectives and test conditions and conduct for the proposed severe fuel damage tests in PBF are explained in Section 3. Section 4 describes the PBF test facility and a conceptual test configuration. Section 5 describes the TRAC-BDO computer code and the models used in applying the code to the PBF experiments. The results of the calculations are described and interpreted in Section 6, and conclusions regarding the feasibility of performing the desired severe fuel damage tests in the PBF are presented in Section 7.

2. EXPECTED SYSTEM THERMAL-HYDRAULIC AND FUEL ROD RESPONSE DURING A SMALL BREAK LOCA TRANSIENT

During a small break LOCA transient, the system conditions which result in fuel damage can be characterized by a slow depressurization and reduced core flow. If the core flow rate decreases sufficiently, the core may uncover by boiling. If cladding temperatures increase sufficiently because of the reduced cladding surface heat transfer and the system pressure drops below the fuel rod internal pressure, the zircaloy cladding will balloon and rupture. The cladding will also react with the coolant at temperatures above about 1200 K and become embrittled with time. At high temperatures the oxygen stabilized alpha-zircaloy will melt and can dissolve a significant fraction of the UO_2. Resolidification of the liquified fuel rod material may cause extensive blockages of coolant subchannels. Quenching of the embrittled cladding will result in fragmentation of the fuel rods thus creating a large rubble bed composed primarily of slag (previously molten material), and fragmented ZrO_2 and UO_2. Fission products will be released to the system when the cladding is ruptured and additional fission product release can occur from dissolution of UO_2, and also from pellet fragmentation and desintering.

In general, the time involved in the complete scenario for a small break LOCA is much longer than for a large break LOCA where depressurization of the system is usually completed within approximately 35 to 40 s. Also the phenomena related to coolant behavior and heat transfer from the fuel to coolant are more complex, involving long periods of convection cooling, boildown of the coolant involving both convection cooling in the subcooled regions and convection and radiation heat transfer in the dryout regions, and finally a return to convection cooling as the reflood and quench phases are completed. Due to the long periods of time at high temperataures in a steam atmosphere, the phenomena of fuel cladding ballooning, rupture, oxidation, and embrittlement are all very important in determining the thermal and mechanical response of the fuel rods.

3. OBJECTIVES AND PLANNED PBF SEVERE FUEL DAMAGE TEST PROGRAM

The PBF Severe Fuel Damage Test Program will consist of a series of highly controlled and instrumented tests with nuclear fuel rods during small break loss-of-coolant accident conditions. The tests will progress from the initial reduction in coolant flow and initiation of boiling, through cladding melting and redistribution of liquified fuel rod material, followed by quench and fuel rod fragmentation. The hydrodynamic and heat transfer characteristics of the rubble pile will also be examined.

The primary experimental objectives for the program are:

1. To examine the primary fuel rod damage mechanisms and the controlling processes during system conditions of decreased pressure and steam cooling, and

2. Examine the hydrodynamic and heat transfer characteristics of a rubble pile formed from previously molten and fragmented fuel rods.

Detailed test objectives have been defined to ensure that the more general primary experiment objectives are attained. They are:

1. Confirm existing zircaloy cladding oxidation correlations at temperatures above 1773 K, and the embrittlement and fuel rod fragmentation criteria at quench temperatures.

2. Characterize fuel rod fragmentation and UO_2 desintering when quenched.

3. Determine the extent of UO_2 dissolution by molten zircaloy, and characterize the redistribution and solidification of liquified fuel rod material.

4. Monitor fission product release during heatup and evaluate the effect of UO_2 dissolution and fragmentation on fission product release.

5. Evaluate the hyrodynamic and heat transfer characteristics of the rubble bed with respect to fuel rod fragment size and coolant flow rate.

To accomplish the identified objectives, the PBF Severe Fuel Damage Test program is initially planned to consist of five tests; a scoping test (SFD-ST), followed by four additional tests (SFD-1, SFD-2, SFD-3, SFD-4), each to be conducted in a manner to meet specific objectives. To ensure melting of the metallic zircaloy, a peak fuel rod cladding temperature of at least 2300 K will be attained. Different rates of cladding oxidation and embrittlement will be attained prior to melting the zircaloy by varying the cladding temperature rise rates. The time at peak temperature will be varied to attain the desired degree of cladding oxidation. Cladding cooldown will be either slow, by radiation to a flow shroud, or fast, by inserting quench water. The slow cooldown will preserve the test bundle for posttest examination, and quench will result in fragmentation of the test rods. The postquench flow rate will be varied to provide information on the hydrodynamic and heat transfer characteristics of the test bundle rubble pile. Cladding ballooning and rupture will be enhanced by prepressurizing the test rods to approximately 3 MPa. Preliminary values of the controlling variables and the specific objectives for each test are described in Table 1.

TABLE 1. PLANNED PBF SEVERE FUEL DAMAGE TEST CONDITIONS AND OBJECTIVES

Test	Rod Pressure (MPa)	Peak Cladding Temperature (K)	Cladding Temperature Rise rate (K/s)	Test Termination	Objectives
SFD-ST	3	2300	0.5	Slow cool-down	Characterize fuel rod damage resulting from ballooning and rupture followed by cladding oxidation and hydrogen uptake during a severe core damage test
SFD-1	3	2300	4.0	Slow cool-down	Characterize fuel rod and bundle damage resulting from cladding melting and simultaneous dissolution, redistribution, and solidification of the fuel
SFD-2	3	2300	0.5	Quench	Characterize the fragmentation of ballooned and embrittled fuel rods and assess rubble bed coolability
SFD-3	3	2300	4.0	Quench	Assess the coolability of a fragmented and partially liquefied fuel rod bundle
SFD-4	3	2300	See objectives	Quench	Simulate the TMI-2 transient and characterize fuel rod damage and rubble bed coolability

4. THE PBF TEST FACILITY AND FEASIBILITY DESIGN FOR THE SEVERE FUEL DAMAGE TEST CONFIGURATION

The Power Burst Facility is an unpressurized, light water moderated and cooled nuclear reactor core with a hole in the center of the core designed to accommodate a reentrant in-pile tube. The reactor core is approximately 1 m in diameter and has an active fuel length (height) of 0.91 m.

The in-pile tube is designed to accommodate single test rods or clusters of test rods with active fuel lengths of 0.91 m, and to provide a wide range of coolant conditions of inlet temperature, pressure, and flow rate. The in-pile tube consists of a right circular cylinder of 19.05 mm thick Inconel with an inside diameter of 154.94 mm and an outside diameter of 193.04 mm. A right circular cylinder flow tube is positioned symmetrically in the in-pile tube to provide a downcomer for the inlet flow to the in-pile tube and an outer boundary for test assemblies. Cross section views of the proposed test train in the in-pile tube are shown in Figures 1a, 1b, and 1c. During normal operation of the in-pile tube loop, coolant enters the inlet, proceeds down the downcomer to the lower plenum region, reverses direction and proceeds up the flow tube through the test space and eventually out the outlet.

In order to attain the high fuel rod cladding temperatures (~2300 K) desired for the severe fuel damage tests, the test rods will be surrounded by an insulating shroud. This shroud must also be fit within the flow tube. Test rod clusters of 16 (17 x 17 PWR design) fuel rods on a square array (4 x 4), and 25 (17 x 17 PWR design) rods on a square array (5 x 5) were evaluated, each with a square insulating shroud of ZrO_2 with wall thicknesses of 10 mm and 15 mm. Analyses were also performed for a

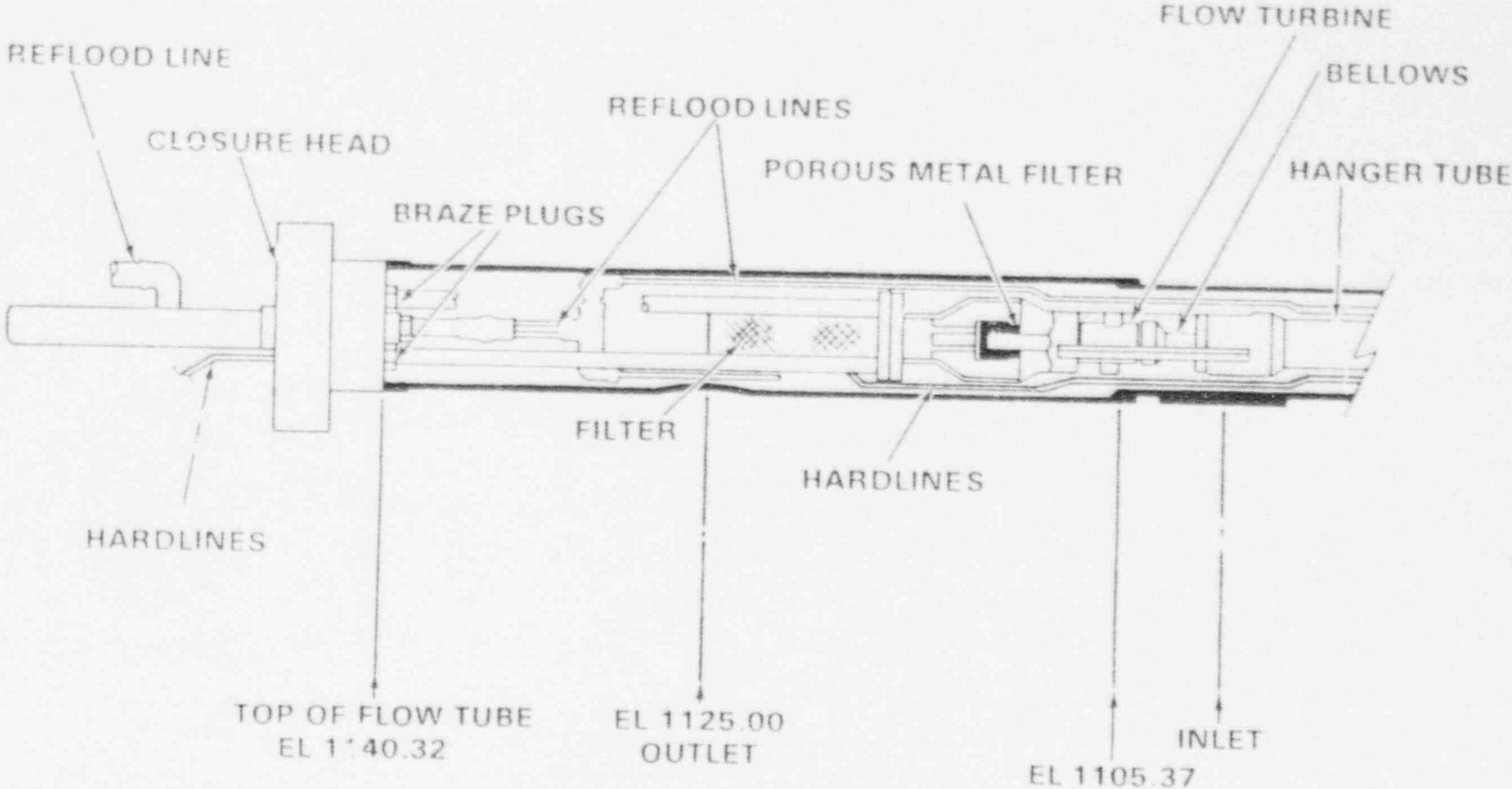

Figure 1a. Cross-section of top portion of proposed severe fuel damage test train in the PBF in-pile tube.

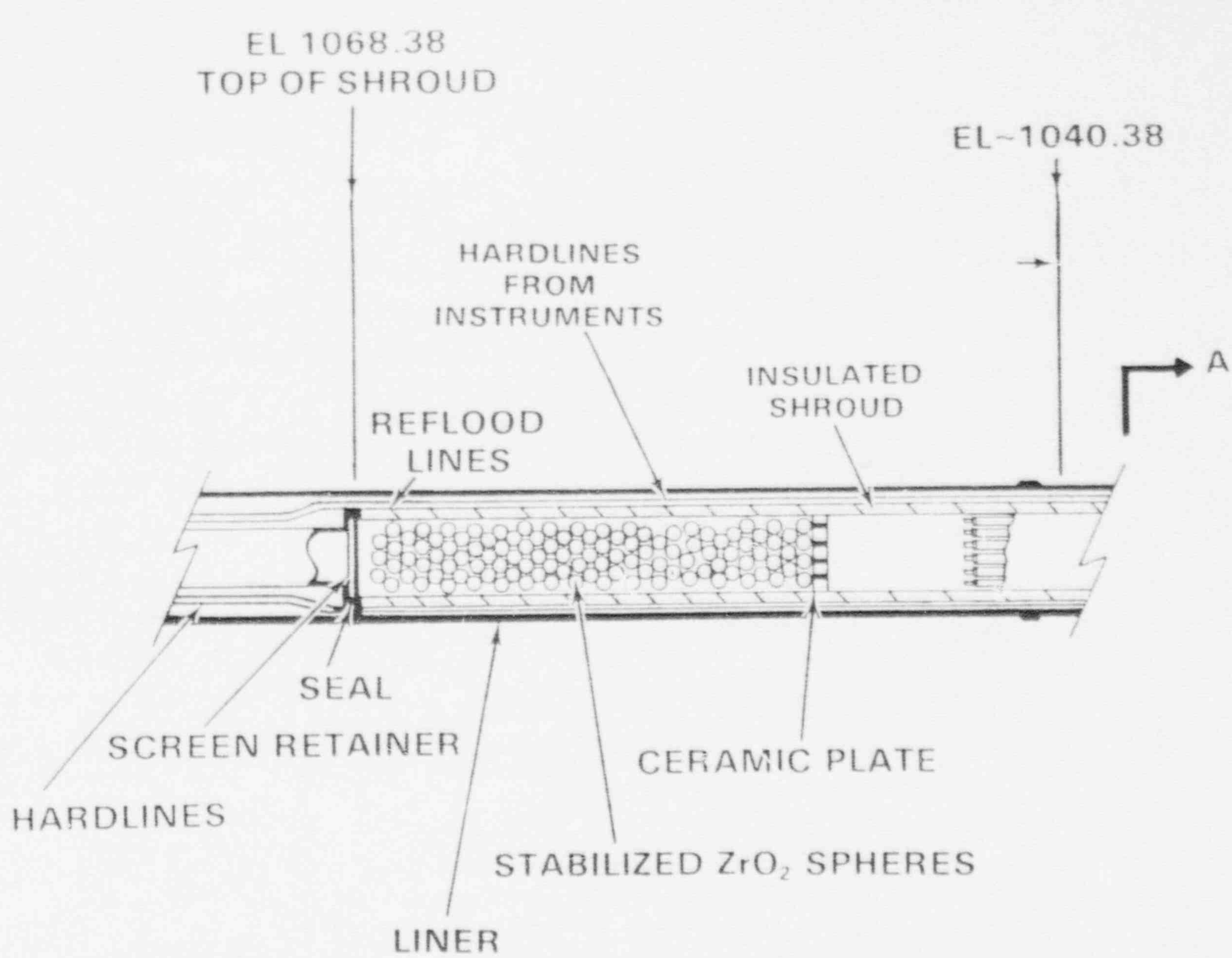

Figure 1b. Cross-section of middle portion of proposed severe fuel damage test train in the PBF in-pile tube.

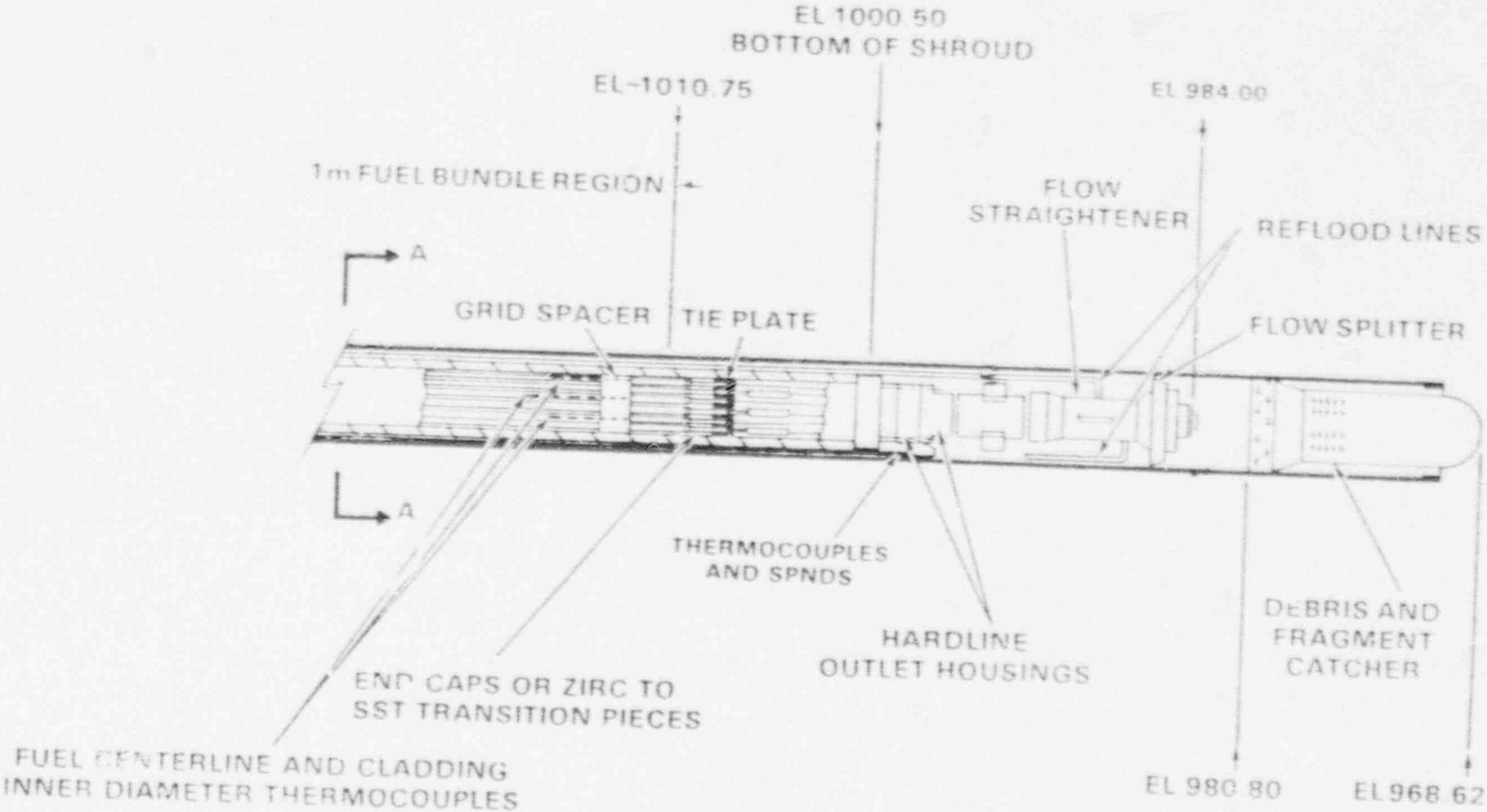

Figure 1c. Cross-section of lower portion of proposed severe fuel damage test train in the PBF in-pile tube.

5 x 5 array with an insulating shroud consisting of ZrO_2 sandwiched between thin slabs of Zirconium with an overall thickness of 16 mm. Figure 2 shows a representative cross section of the test cluster and ZrO_2 insulating shroud for a 5 x 5 square array of PWR design test rods.

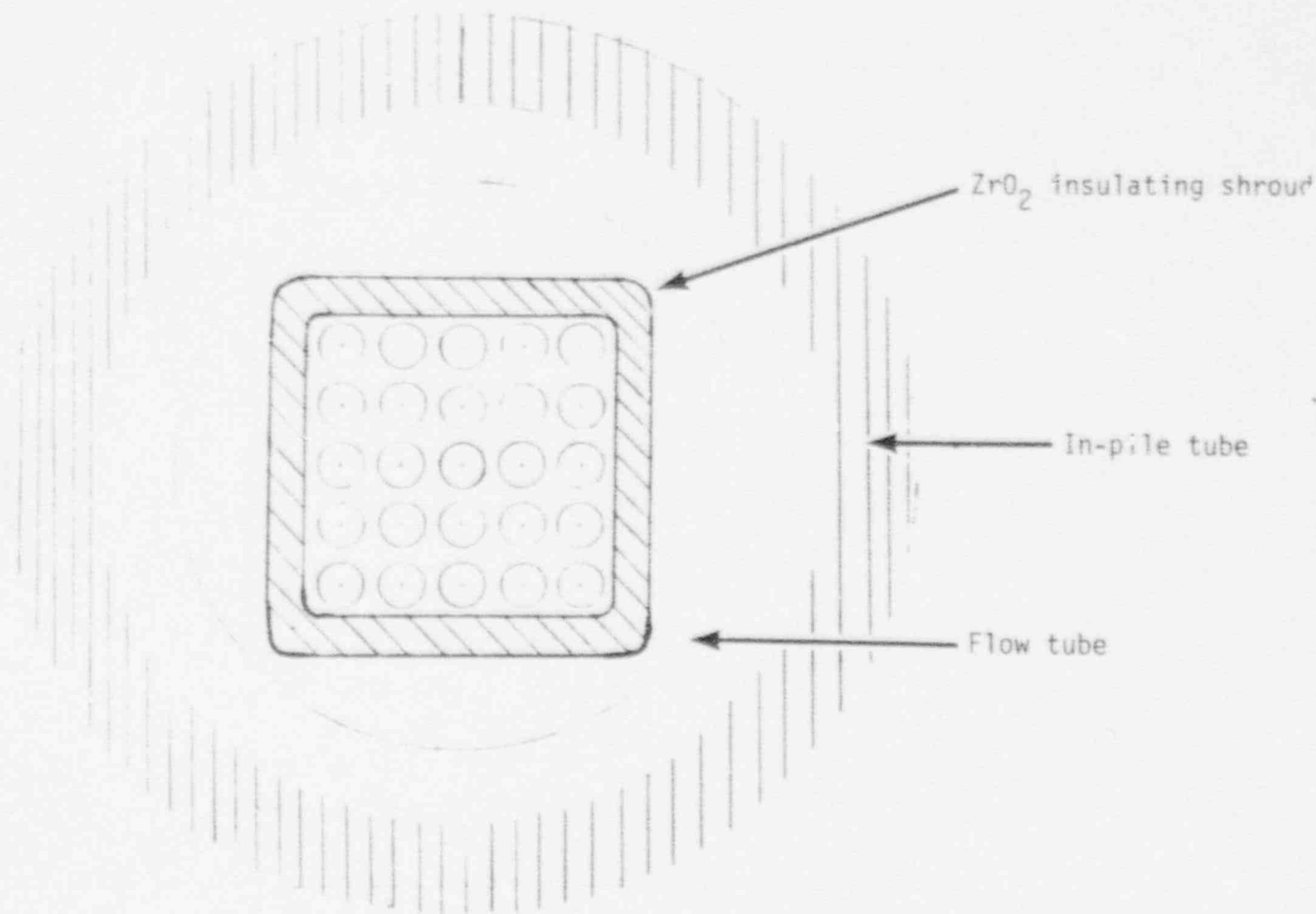

Figure 2. Cross-sectional view of 25-rod cluster of fuel rods with an insulating shroud of ZrO_2 in the PBF in-pile tube.

5. THE TRAC-BDO COMPUTER CODE AND MODEL FOR APPLICATION TO THE PBF SEVERE FUEL DAMAGE TESTS

EG&G, Idaho is in the process of developing a capability for analysis of LOCA transients in BWR systems in the TRAC code. A basic BWR capability, TRAC-BDO, has been developed from TRAC-P1A, the basic TRAC code for analysis of LOCA transients in PWR systems.

5.1 General Description of TRAC

The Transient Reactor Analysis Code (TRAC) is an advanced best-estimate systems code for analyzing accidents in LWRs. It is being developed at the Los Alamos Scientific Laboratory (LASL) and at the Idaho National Engineering Laboratory (INEL) under the sponsorship of the Reactor Safety Research Division of the U.S. Nuclear Regulatory Commission.

TRAC-P1, completed in December 1977, was the first publicly released version and is described in the Los Alamos report LA-7279-MS. TRAC-P1 was designed primarily for the analysis of large break loss-of-coolant accidents (LOCAs) in pressurized water reactors (PWRs). TRAC-P1A is an improved version of TRAC-P1.

5.1.1 TRAC Characteristics

Some of the distinguishing characteristics of TRAC are summarized below. All of these characteristics reflect the state-of-the-art in the various areas.

1. Multidimensional Fluid Dynamics

Although the flow within the ex-vessel components is treated in one-dimension, a full three-dimensional (r,θ,z) flow calculation is used within the reactor vessel.

2. Nonhomogeneous, Nonequilibrium Modeling

A full two-fluid (six-equation) hydrodynamics approach is used to describe the steam-water flow within the reactor vessel, thereby allowing such important phenomena as countercurrent flow to be treated explicitly. The flow in the one-dimensional loop components is descri[illegible] using a five-equation drift flux model, which differs from the standard four-equation drift-flux approach by the addition of a separate vapor energy equation.

3. Flow-Regime-Dependent Consititutive Equation Package

The thermal-hydraulic equations in TRAC describe the transfer of mass, energy, and momentum between the steam-water phases, and the interaction of these phases with the system structure. Since the nature of these interactions is dependent on the flow topology, a sophisticated flow-regime-dependent constitutive equation package has been incorporated into the code.

4. Consistent Analysis of Entire Accident Sequences

An important feature of TRAC is the ability to address entire accident sequences, including computation of initial conditions, with a consistent and continuous calculation. For example, the code models the blowdown, refill, and reflood phases of a LOCA. This eliminates the necessity of synthesizing several calculations performed with different codes to complete the analysis of a given accident.

5. Component and Functional Modularity

TRAC is completely modular by component. The component modules are assembled through input data to model virtually any system design or experimental configuration. Modules are available to model accumulators, pipes, pressurizers, pumps, steam generators, tees, valves, and vessels with associated internals.

5.2 General Description of TRAC-BDO Code

TRAC-BDO is a preliminary version of TRAC developed at INEL for analysis of LOCAs in boiling water reactors (BWRs). In addition to accounting for other design differences between BWRs and PWRs, TRAC-BDO takes into consideration the important influence of the fuel canister walls of BWRs on radiation heat transfer during a LOCA.

A set of models describing certain BWR components (jet pump, separator, and dryer) and important hydrodynamic phenomena (level swell and countercurrent flow) has been implemented into TRAC-BDO, and an important new TRAC component called CHAN (for channel) has been developed to enable realistic modeling of BWR core heat transfer and coolant flow. The CHAN component is an extension of the standard TRAC-code PIPE component and simultes a BWR rod bundle and canister assembly. CHAN components are used to represent all fuel channels of a BWR and are connected across the usual core region of a conventional VESSEL component. Three-dimensional core flow in the bypass region between channels is calculated with the usual VESSEL hydrodynamics. Convective heat transfer between channels and bypass coolant is also modeled by the CHAN component. In a BWR, each CHAN component represents a large number of actual fuel rod bundles, and the flow through each CHAN component is assumed to be one-dimensional. Within each CHAN component, radiation and conduction heat transfer are calculated for a number of rod groups specified by the user.

The total heat transfer within a CHAN component is determined by calculating the heat transfer from an average rod bundle and multiplying by the number of bundles lumped together in the CHAN component. The heat transfer modes provided by each CHAN component are:

1. Conduction heat transfer in the fuel rods and the channel wall.

2. Convective heat transfer from the fuel rods and channel wall during blowdown and reflood.

3. Radiation heat transfer from surface to surface, surface to steam, and surface to water droplets.

Existing TRAC-P1A models are used for the conductive and convective heat transfer modes. For the radiation heat transfer mode, a diffuse gray body model with steam and droplet participation was developed and is similar to the models used in the NORCOOL[8] and MOXY-SCORE[9] models. The major differences between the TRAC-BDO radiation model and the NORCOOL and MOXY-SCORE models are in the methods for calculating emissivities and absorptivities of steam and water droplets.

In TRAC-BDO, the hydrodynamics model in all one-dimensional components (PIPES, VALVES, CHAN, etc.) is the same as is used in the one-dimensional components in TRAC-P1A, and is a drift-flux model involving the velocity of the mixture of vapor and water ($v_{mixture}$) and the relative velocity between vapor and liquid ($v_{relative}$). In this model, $v_{relative}$ is calculated from the Zuber-Findlay[10] or Ishii[11] correlations, depending on the specific flow regime (bubbly, slug, churn-turbulent or annular) involved. The system of partial differential equations describing the two-phase flow and heat transfer are solved by finite differences. The heat transfer equations are treated as one-dimensional using a semi-implicit differencing technique. The fluid dynamics equations utilize either semi- or fully-implicit differencing under user control. The finite difference equations for hydrodynamic phenomena form a system of coupled, nonlinear equations which are solved by a Newton-Raphson iteration procedure. The specific variables calculated and provided as printout by TRAC-BDO, along with the identifiers and units for each variable, are described in Appendix A.

5.3 TRAC-BDO MODEL FOR THE PBF SEVERE FUEL DAMAGE TEST CONFIGURATION

Application of TRAC-BDO to the specific proposed PBF Severe Fuel Damage tests has been implemented by considering that a single CHAN component represents the test cluster and associated geometries, and by specifying the necessary thermal-hydraulic boundary conditions, such as inlet coolant flow rate, temperature, and pressure, test rod power densities, and insulating shroud outside wall temperature, etc.

The geometrical configuration assumed for the feasibility study is shown in Figure 2. In addition, in the model the 0.91 m axial length of the active region of the fuel rods was divided into six axial levels. At each axial level, the power distribution across the cluster was assumed to be uniform. Figure 3 shows the dimensions of each axial level and the corresponding axial power distribution used in the model. Symmetrical rods were grouped together at each axial level in groups 1 through 6, as shown in Figure 4.

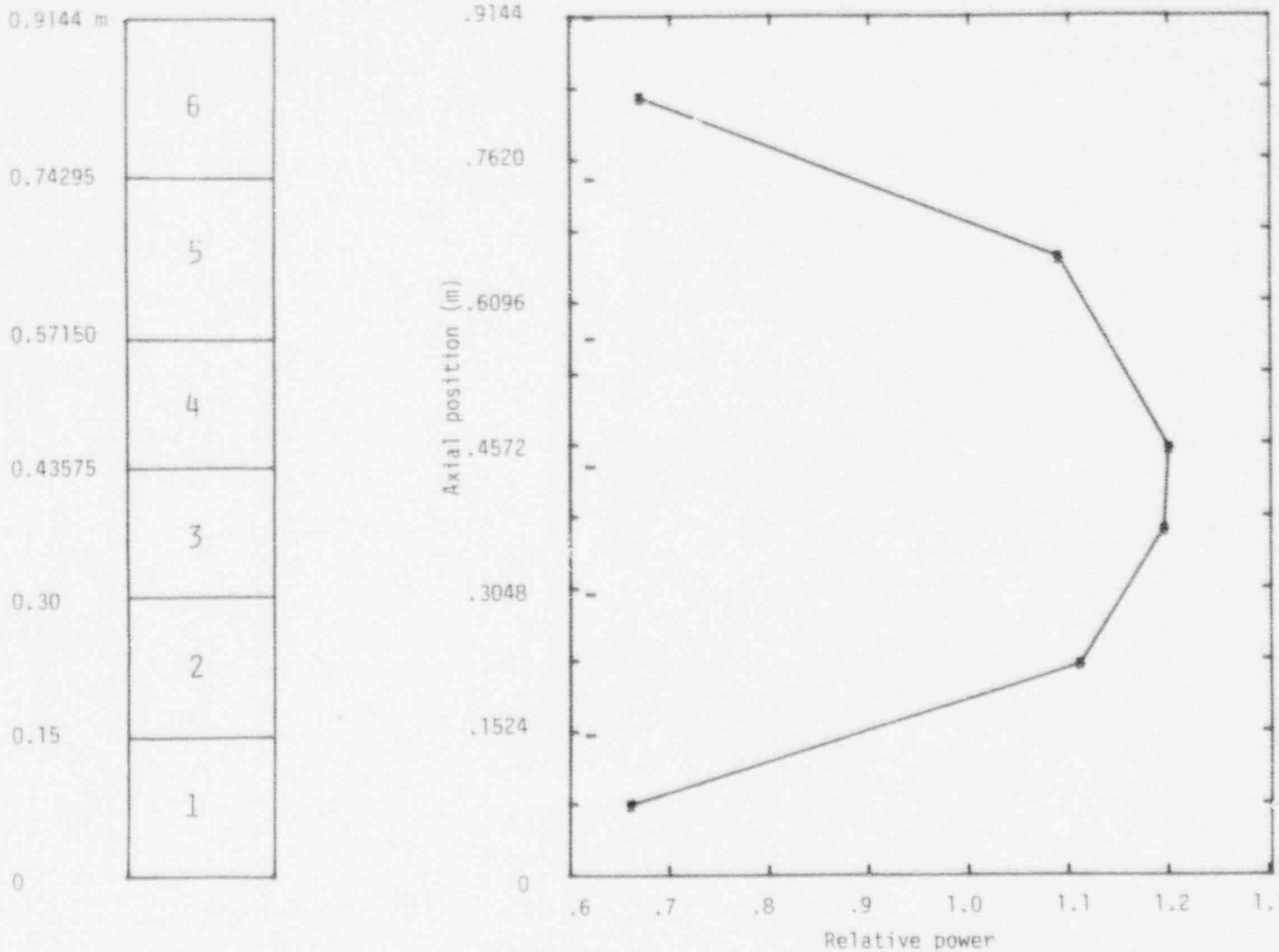

Figure 3. Relative axial power distribution and axial level dimensions used in the PBF severe fuel damage test scoping calculations.

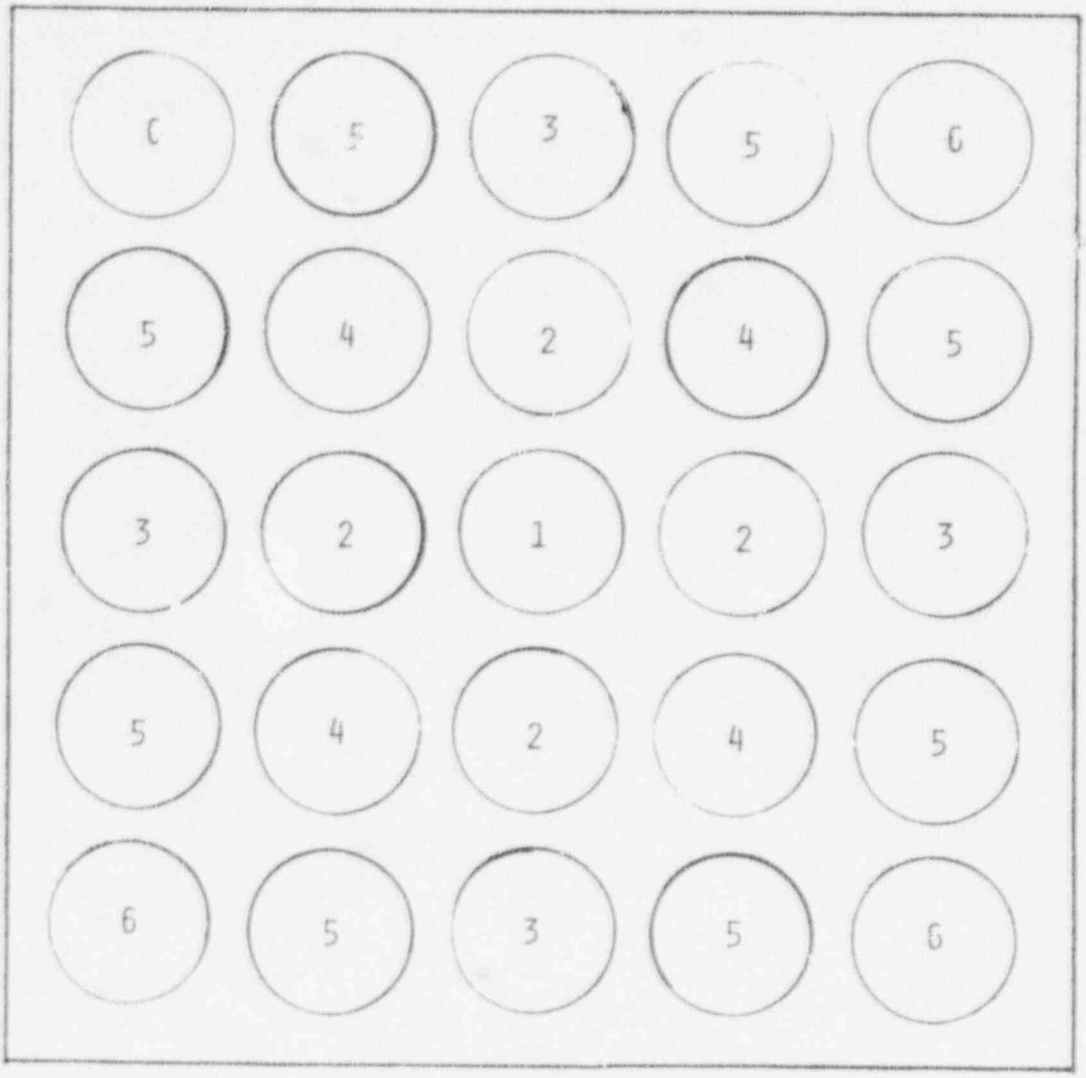

Figure 4. Fuel rod group in the 5 x 5 severe fuel damage test cluster scoping calculations.

6. TRAC-BDO CALCULATION FOR THE PROPOSED PBF SEVERE FUEL DAMAGE TESTS

Two primary concerns regarding performance of the proposed severe fuel damage tests in PBF are (a) the fission heat requirements to compensate for radiative and convective heat transfer from the test bundle when cladding temperatures are approximately 2300 K and (b) the relatively small size of the test bundle. Fuel centerline melting could occur if the required fission heat were approximately 26 kW/m or greater, and the relatively large radial temperature gradients within a test rod at such power levels are undesirable. Both of these conditions are atypical of the actual state of the fuel rods in a commercial reactor during a hypothetical small break LOCA.

The relatively small size of the test bundle (either 4 x 4 or 5 x 5) could severely restrict the validity of the results if the tests were primarily intended to be demonstrative. However, the tests are intended to examine, phenomenologically, the fuel rod damage mechanisms that occur during system conditions of decreased pressure and steam cooling, and the hydrodynamic and heat transfer characteristics of the rubble bed formed during quench. If the tests are structured and performed correctly, and properly instrumented, the desired data should be obtained.

Two types of calculations were performed for evaluating the feasibility of performing small break LOCA tests in the PBF. The first type simulated the condition that a small break had occurred and the coolant had boiled away to the extent that only a low flow rate steam atmosphere existed in the fueled region. The purpose of the TRAC-BDO calculations was to determine the steady state cladding temperatures as a function of fuel rod linear power, with the specific objective of identifying the test rod linear power and the steam mass flow rate required to attain steady state cladding temperatures of approximately 2300 K. These calculations were performed for both the 16-rod and 25-rod test clusters.

The second type of calculation simulated the reduction in coolant flow and the transient boiloff following the initiation of a small break LOCA. The objective of these calculations was again to determine the test rod power and coolant mass flow rates required to attain steady state cladding temperatures of approximately 2300 K, and to obtain some insight as to the transient response of the system during the boildown process. Boildown calculations were performed for the 25-rod cluster only.

For the first type of calculation steady state conditions were desired and computer time and costs were minimized by initializing cladding surface temperatures at a higher temperature than inlet conditions. Although this technique provided the appropriate values of the variables at steady state conditions, the transient response of the pertinent variables is not necessarily correct. The representative transient response was obtained with the second type calculation involving the calculated boil-off and heatup.

The system boundary conditions used in the calculations, and the results of the calculations, are described below for the steam atmosphere test conditions (Section 6.1), and the steam/water test conditions (Section 6.2), respectively.

6.1 Steam Atmosphere Test Conditions

For the system thermal-hydraulic conditions of 7 MPa pressure and 559 K inlet steam temperature, calculations were performed to scope the effects of steam flow rate, test rod power, and insulator shroud thickness on the steady state peak cladding temperature for the 25-rod cluster of test rods. Steam flow rates used were 0.3 m/s and 0.6 m/s. The test rod peak power densities used were 1.2 kW/ft (3.94 kW/m) and 2.0 kW/ft (6.56 kW/m), and the insulator shroud thicknesses used were 10 mm and 15 mm. Other boundary conditions included: a uniform radial power distribution in the cluster at each axial level, a coolant temperature at the outside of the insulator shroud of 537 K, an outside shroud wall heat transfer coefficient of 10^5 W/m^2•K, and a ZrO_2 (insulator shroud) thermal conductivity of 2.49 W/m•K. Figure 5 shows the predicted effect of test rod peak power density on the steady state peak cladding temperature for a steam flow rate of 0.3 m/s and an insulator thickness of 15 mm. The indicated steady state peak cladding temperature of approximately 2360 K at 600 s occurs on the center rod (Group 1) at axial level 5 (0.5715 to 0.7429 m above the bottom of the fuel stack in the fuel rods).

The effect of steam flow rate on the steady state peak cladding temperature is shown in Figure 6 for steam flow rates of 0.3 m/s and 0.6 m/s at a test rod peak power of 2.0 kW/ft and for an insulating shroud thickness of 15 mm. Comparing Figures 5 and 6 shows that decreasing the power from 2.0 to 1.2 kW/ft has about the same effect on cladding temperature (ΔT approximately 525 K) as increasing the steam flow rate from 0.3 to 0.6 m/s.

Figure 7 shows the effect of insulator shroud thickness on the steady state peak cladding temperature for a change from 15 mm to 10 mm at a peak power of 2.0 kW/ft and a steam flow rate of 0.3 m/s. The predictions indicate that peak cladding temperature is not strongly affected by shroud thickness, in the range investigated, and the change in temperature, ΔT, is only about 100 K.

The results shown in Figures 5, 6, and 7 indicate that the desired peak cladding temperature of approximately 2300 K can be attained in the PBF test configuration of a 5 x 5 cluster of 17 x 17 PWR design test rods by using an insulating shroud of ZrO_2 of 15 mm thickness, an inlet steam flow rate of 0.3 m/s, and a peak rod power as low as 2.0 kW/ft.

Scoping calculations were also performed for a 16-rod (4 x 4) test cluster and the same boundary conditions as for the 25-rod calculations. For a 4 x 4 cluster with a 15 mm thick Z_rO_2 insulating shroud and an inlet steam flow rate of 0.3 m/s, a test rod peak power of at least 2.35 kW/ft (5.6 kW/m) would be required to attain peak cladding temperatures of approximately 2300 K.

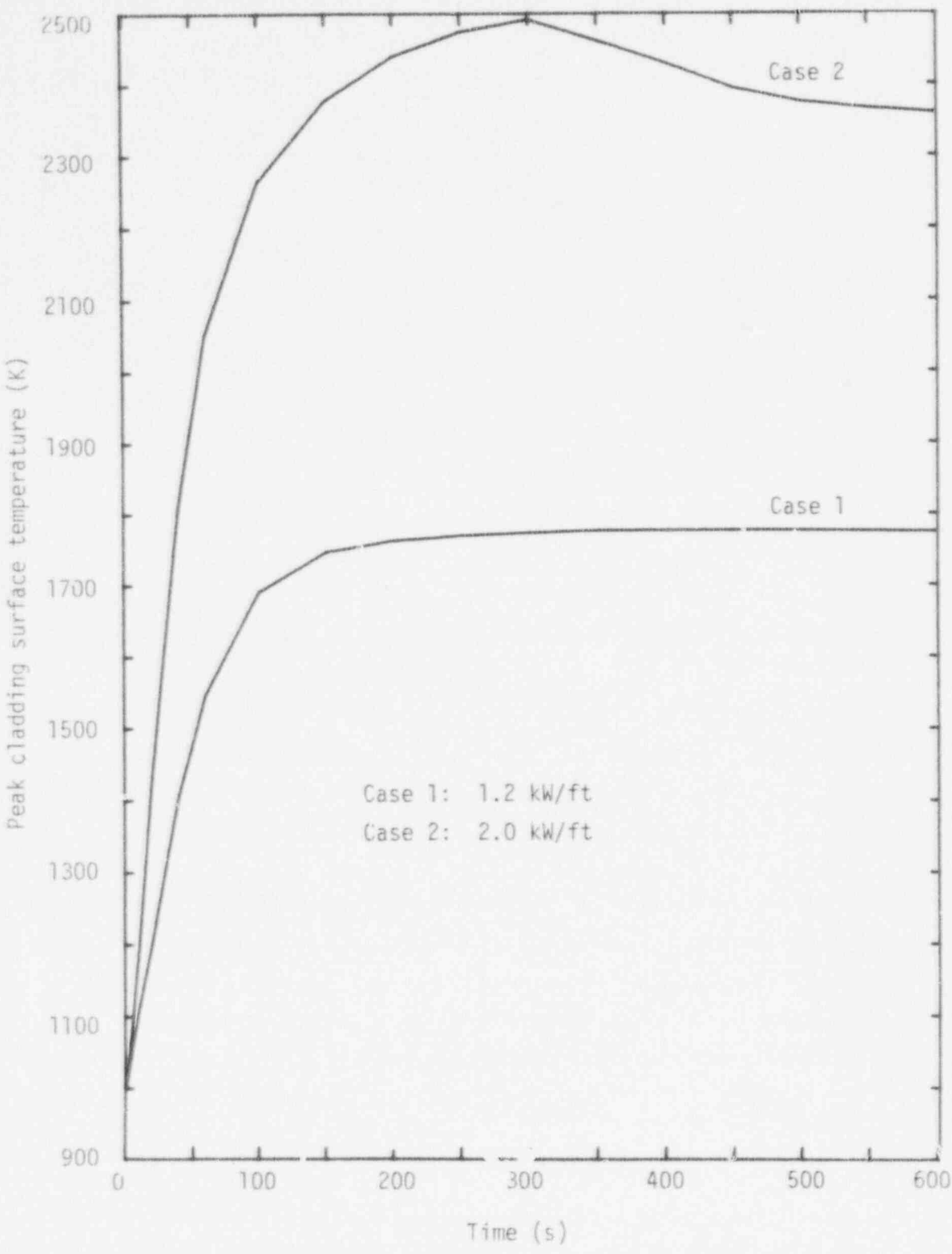

Figure 5. Calculated peak cladding surface temperature as a function of time showing the effect of test rod peak power on the steady state temperature for an insulator thickness of 15mm ZrO_2 and a steam flow rate of 0.3 m/s.

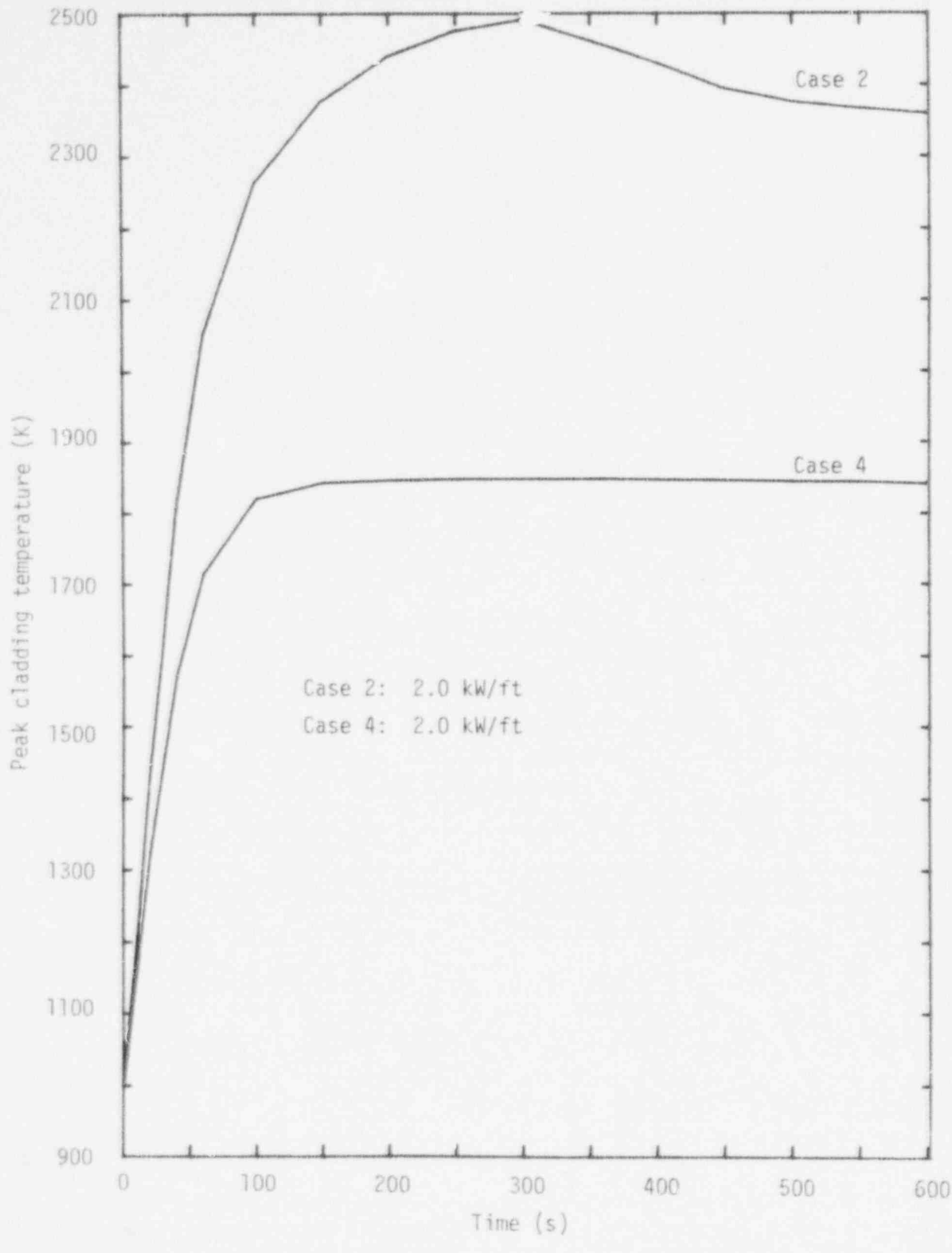

Figure 6. Calculated peak cladding surface temperature as a function of time showing the effect of steam flow rate on the steady state temperature for a test rod peak power of 2.0kW/ft and a shroud thickness of 15mm ZrO_2.

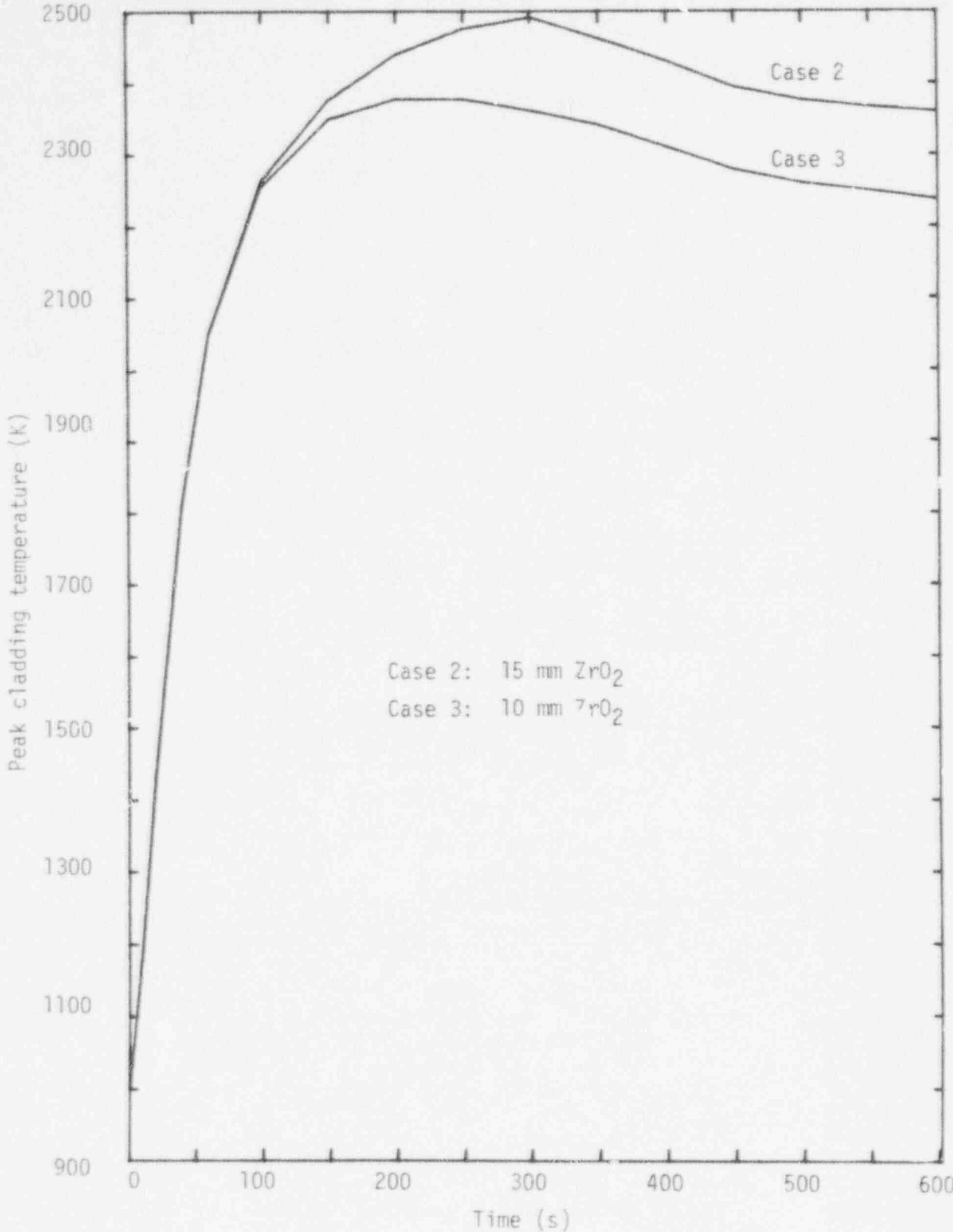

Figure 7. Calculated peak cladding surface temperature as a function of time showing the effect of insulator shroud thickness on the steady state temperature for a test rod peak power of 2.0 kW/ft and a steam flow rate of 0.3 m/s.

As already discussed, the calculational results described in Figures 5 through 7 are scoping in nature and were performed simply to determine relative ranges of power, flow rate, cluster size, etc., that would be required to perform the desired experiments in the PBF. Once these initial calculations were complete, several features of the model were changed to be more representative of the specific conditions that will exist in the PBF test space and existed in the Three Mile Island (TMI-2) incident. In particular, the assumption of a uniform radial power distribution is not representative of the expected radial distribution in a PBF test cluster unless a special distribution of fuel enrichments were used. For a 25-rod cluster of test rods with the same enrichments, a more realistic radial power distribution would be as shown in the table below for each identified group of rods:

Rod Group	1	2	3	4	5	6
Relative Power	0.31	0.62	1.23	0.62	1.23	1.23

The estimated steam flow rate in TMI-2 was about 2.5 lb/hr•rod, which corresponds to a steam flow rate in the 25-rod PBF cluster of 0.1 m/s rather than the 0.3 m/s used in the scoping calculations.

Also ZrO_2 is a ceramic and is thus a brittle material. To provide structural stability, the insulating shroud design was changed slightly by reducing the ZrO_2 thickness to 11.31 mm and adding a layer of Zr on the inside of 1.54 mm thickness and a layer on the outside of 3.18 mm thickness. The revised shroud design is shown in Figure 8. This increased the overall thickness of the shroud from 15 mm to 16 mm, but the effective thermal conductivity of the shroud was increased to 3.5 $W/_{m \cdot K}$ (from 2.5 $W/_{m \cdot K}$ for ZrO_2 alone).

Calculations were performed with the identified design changes. The responses of the test rod cladding temperature, steam temperature, and insulating shroud inside surface temperature at the elevation of peak cladding temperature are shown as functions of time in Figure 9. Rod to

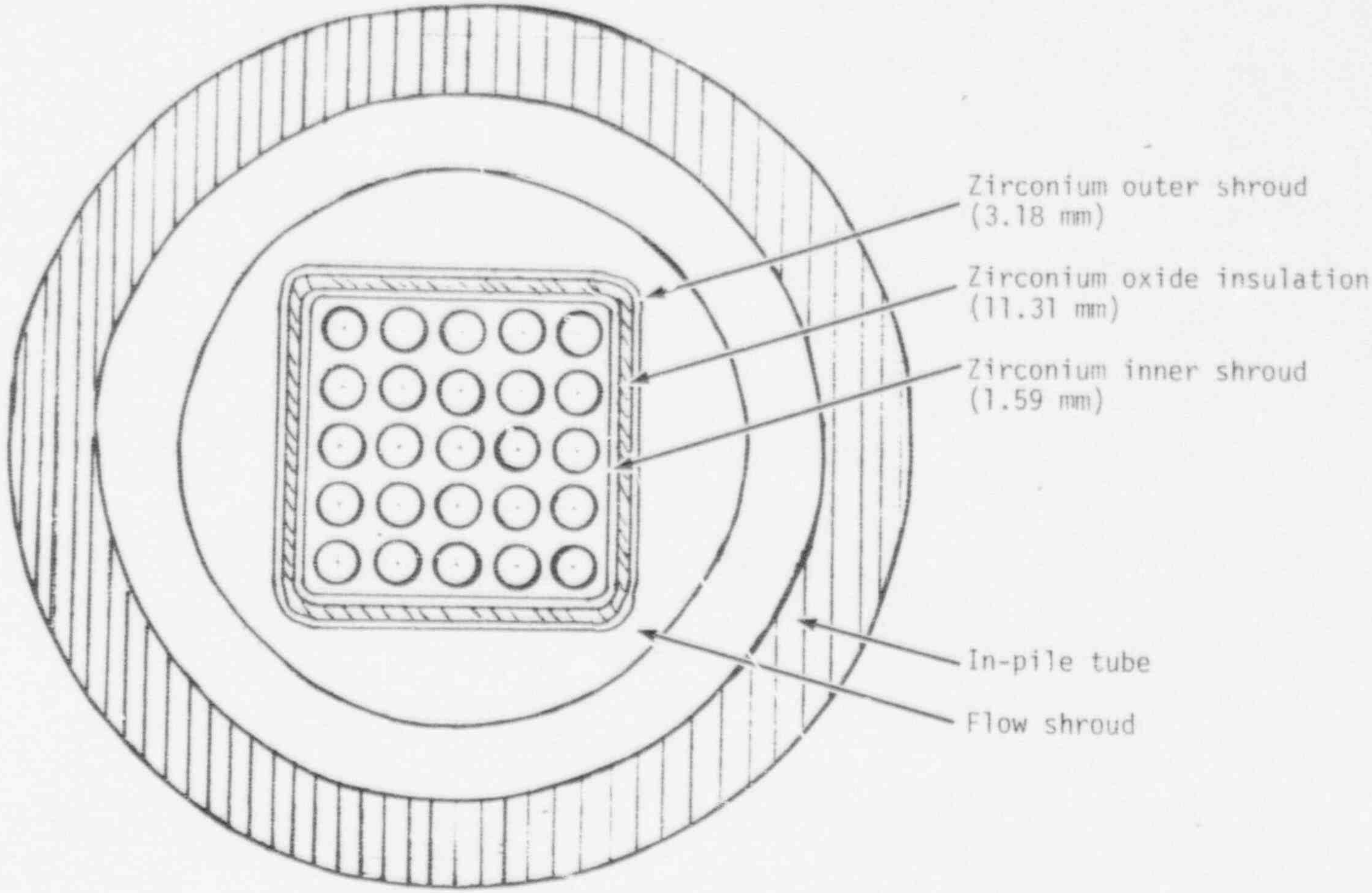

Figure 8. Proposed revised insulating shroud design showing ZrO_2 sandwiched between two layers of zirconium.

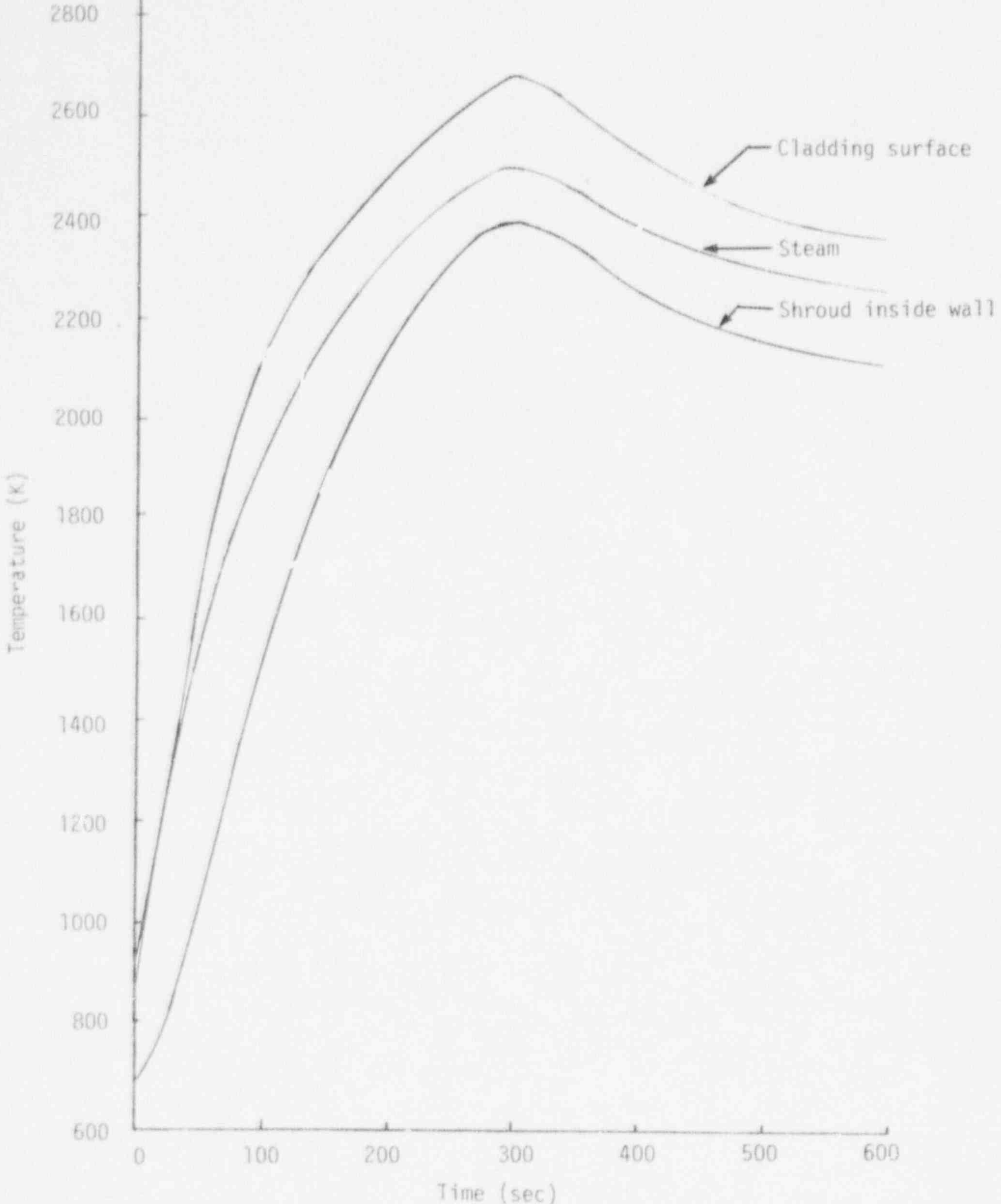

Figure 9. Calculated temperatures as functions of time for the cladding surface, steam, and shroud inside surface at the axial level of the peak cladding surface temperature for a rod power of 1.44 kW/ft and a steam flow rate of 0.1 m/s.

rod temperature differences within the bundle are shown in Figure 10. The relatively small differences are indicative of the insulating effect of the shroud and the uniformity of temperature throughout the test cluster at equilibrium conditions. The one-dimensional formulation of the rod bundle hydrodynamics also tends to minimize rod to rod radial temperature differences because it cannot account for temperature differences within the steam that would naturally develop from the center of the cluster to the shroud.

Close examination of the thermal response of the cladding, steam, shroud inside surface, the metal-water reaction, and the modes of heat transfer as functions of both elevation in the cluster and time at temperature provide insight for understanding the thermal response of the cluster. Figure 11 shows the relative magnitudes of the convection and radiation heat fluxes of the group-2 rods and the shroud inside surface at axial level-5 as functions of time. Also shown is the time response of fuel rod energy contributed by the metal-water reaction of the cladding with the steam. Because of the low steam flow rate (0.1 m/s), heat transfer from the fuel rods to the steam is primarily by the radiative mode, with the steam radiating to the shroud inside wall. With the very high rate of cladding temperature rise shown in Figure 9, the metal-water reaction became significant as early as about 20 s, and by 300 s oxidation of the cladding had essentially consumed the entire thickness of the cladding and the energy addition from the metal-water reaction was terminated by 350 s. Completion of the metal-water reaction significantly reduced the rate of energy generation in the rods and the cladding temperature peaked out at 300 s and subsequently stabilized at the level (2400 K) consistent with the rate of energy added by the nuclear fuel and removed by, primarily, radiation to the steam.

Figure 12 shows the axial distribution of the rod power in the PBF test space and the associated variations in the fuel rod convective and radiative heat fluxes, the metal-water reaction heat flux, and the thickness of the cladding that has reacted to form zirconium oxide, at the

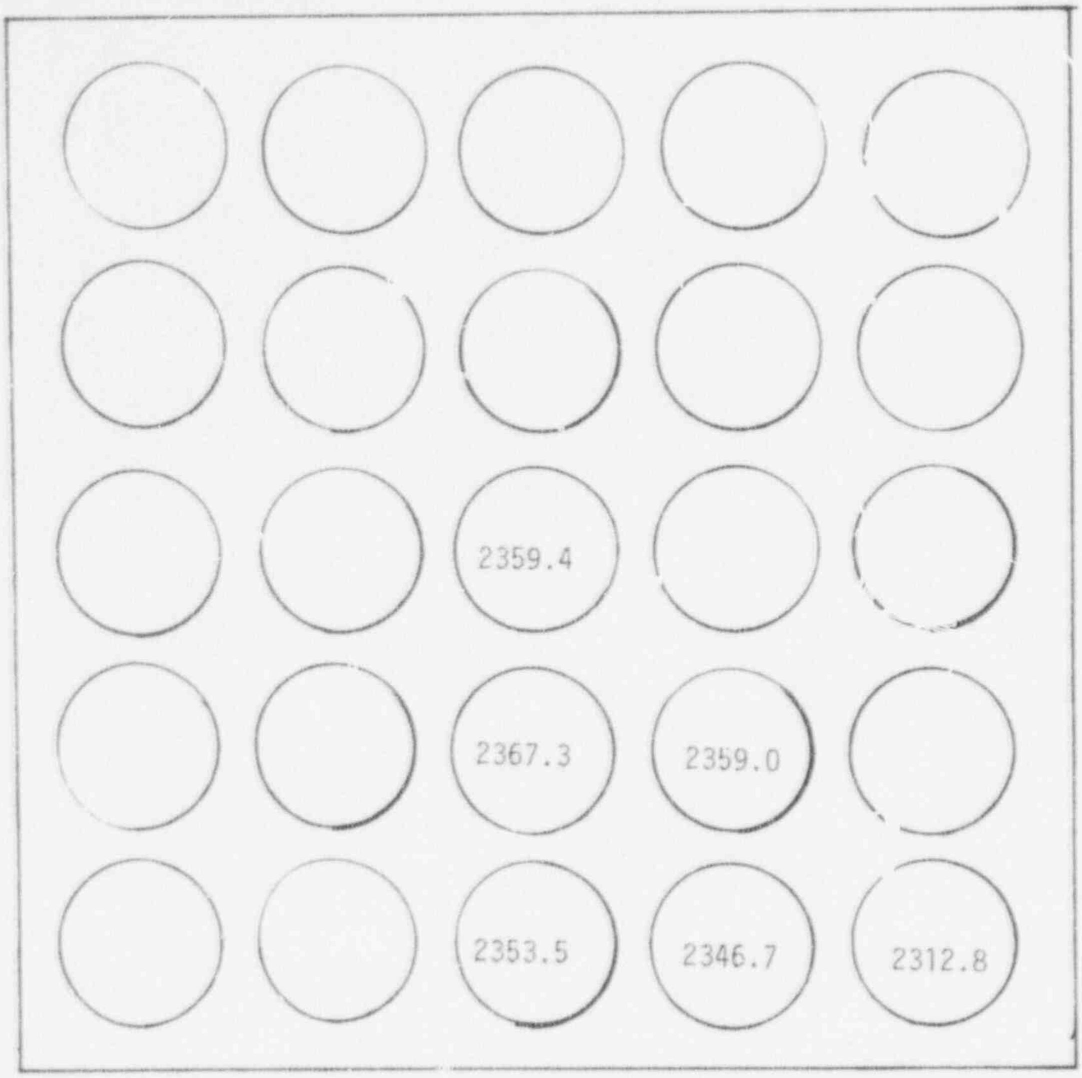

Figure 10. Calculated distribution of rod cladding surface temperatures at the axial level of the peak cladding surface temperature for a rod power of 1.44 kW/ft and a steam flow rate of 0.1 m/s.

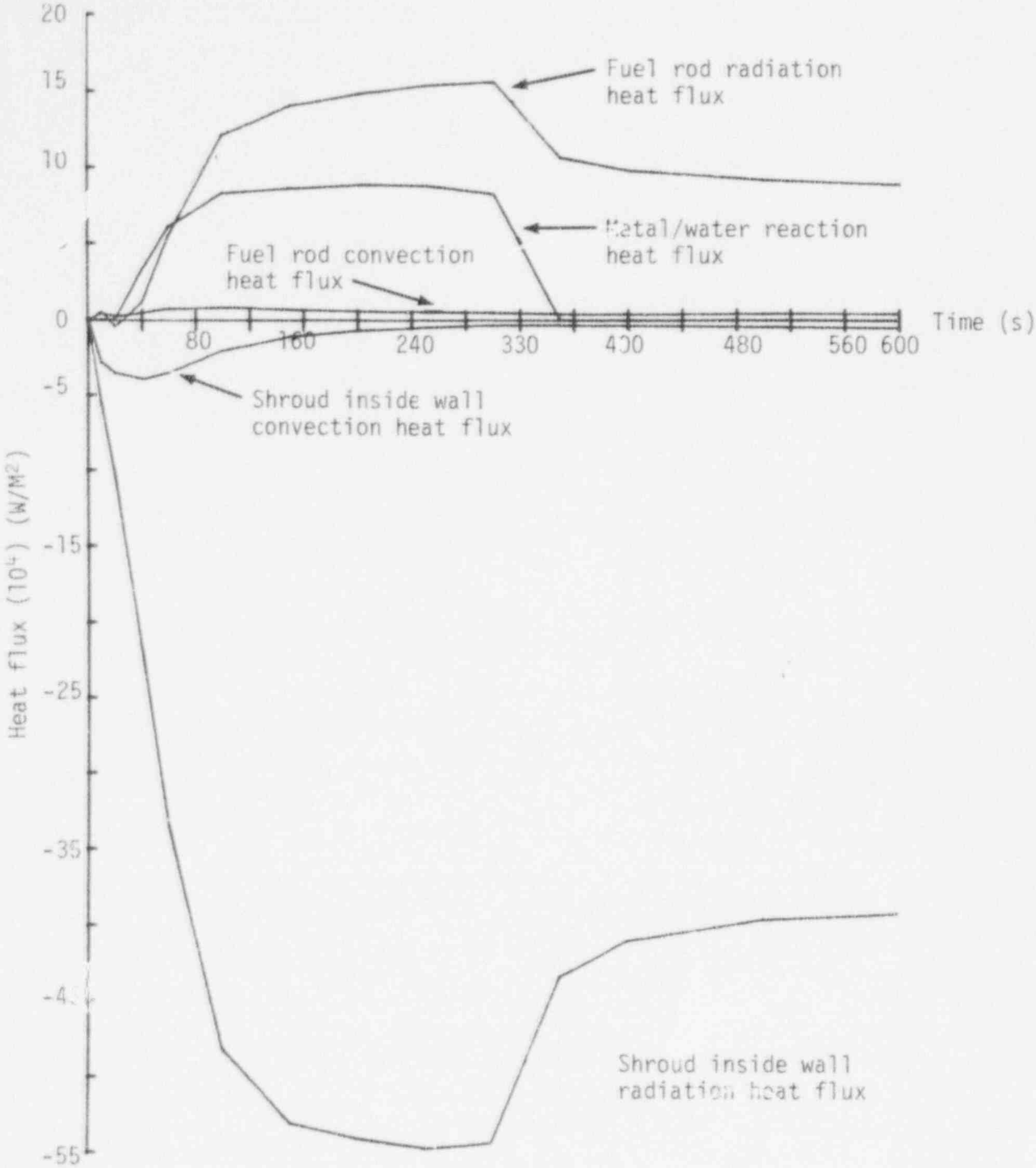

Figure 11. Calculated convection and radiation heat fluxes as functions of time at the axial level of the peak cladding surface temperature for a test rod power of 1.44 kW/ft and a steam flow rate of 0.1 m/s.

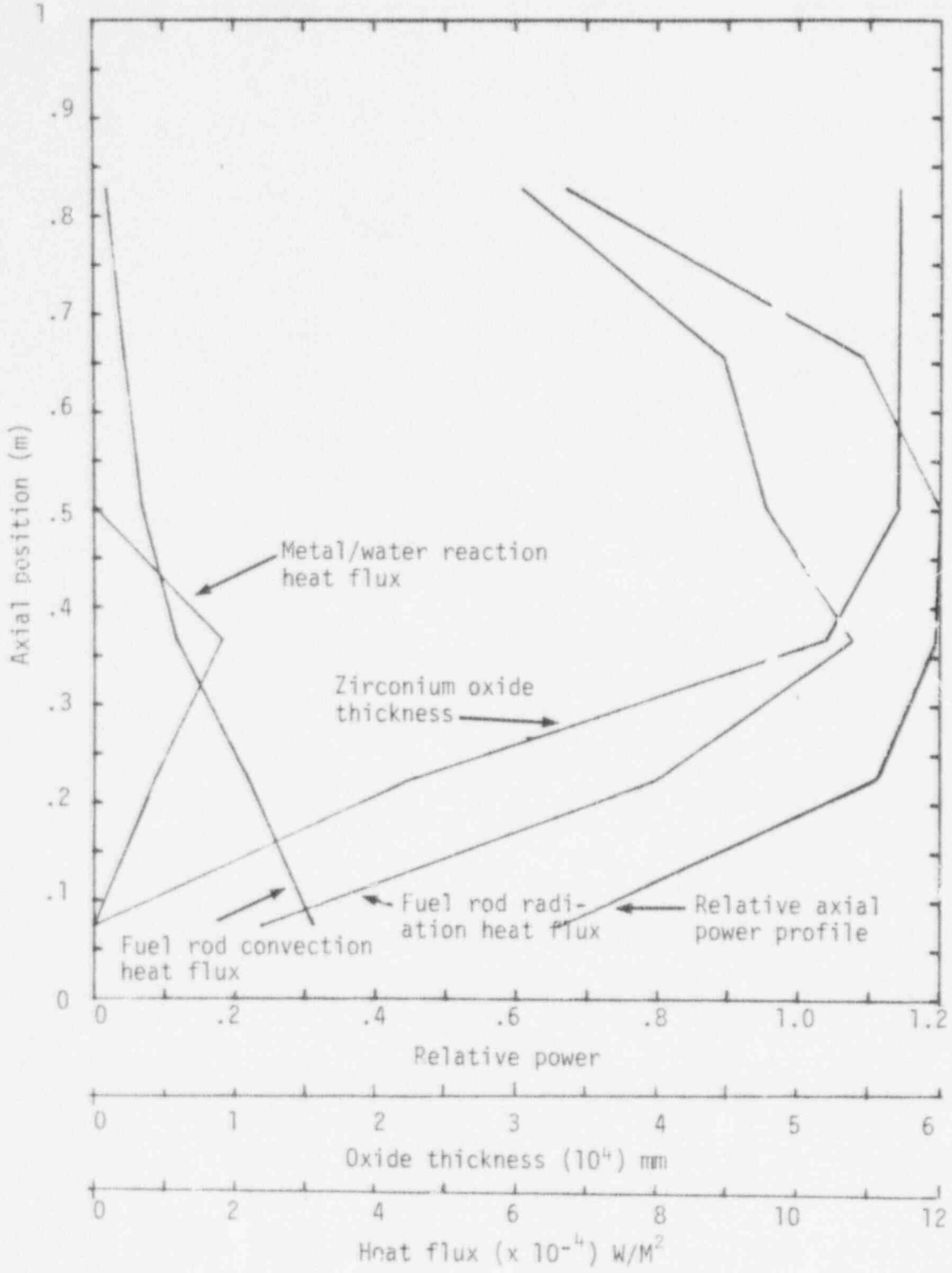

Figure 12. Comparisons of calculated axial variations in the rod convection, radiation, and metal/water reaction heat fluxes and the zirconium oxide layer thickness with the relative axial power distribution for a rod power of 1.44 kW/m and a steam flow rate of [illegible].1 m/s.

time of 600 s. Figure 12 shows that because of the low steam velocity and the high rod temperatures, radiation heat transfer from the fuel rods to the steam is far greater than convection heat transfer, even at the bottom of the fuel rod, and that convection heat transfer is very low above the axial midpoint. In addition, the metal-water reaction completely oxidizes the cladding above the axial midpoint, which significantly reduces the radiation heat flux, which is also reduced above the axial midpoint because the power profile is reduced.

The effects of the axial power distribution and the axial variation in cladding oxidation on the cladding, steam, and shroud inside surface temperatures as functions of axial position are shown in Figure 13. All three temperatures show a reduced rate of increase, and then a decrese, above the axial midplane, reflecting the decrease in the axial power profile. With the rod to steam convective heat transfer decreasing to a very low value above the axial midpoint, but the rod to steam radiative heat transfer remaining high, the steam temperature in the top of the cluster increases and is practically as hot as the cladding surface when it exits the cluster.

6.2 Steam/Water Test Conditions

In the event a small break LOCA occurs, the system conditions will not initially be steam, as described above in the scoping calculations, but will be normal operating conditions with a slow decrease in coolant flow rate, heating up of the coolant until boiloff occurs, and eventually arriving at the steam atmosphere conditions. The scoping calculations decribed above identified the minimum test rod power density (1.44 kW/ft, 4.72 kW/m) and steam flow rate (0.1 m/s) required to attain equilibrium peak cladding surface temperatures in the range of 2300 K with a 16 mm thick Zr, ZrO_2, Zr insulating shroud and a specific radial power distribution in the test rods.

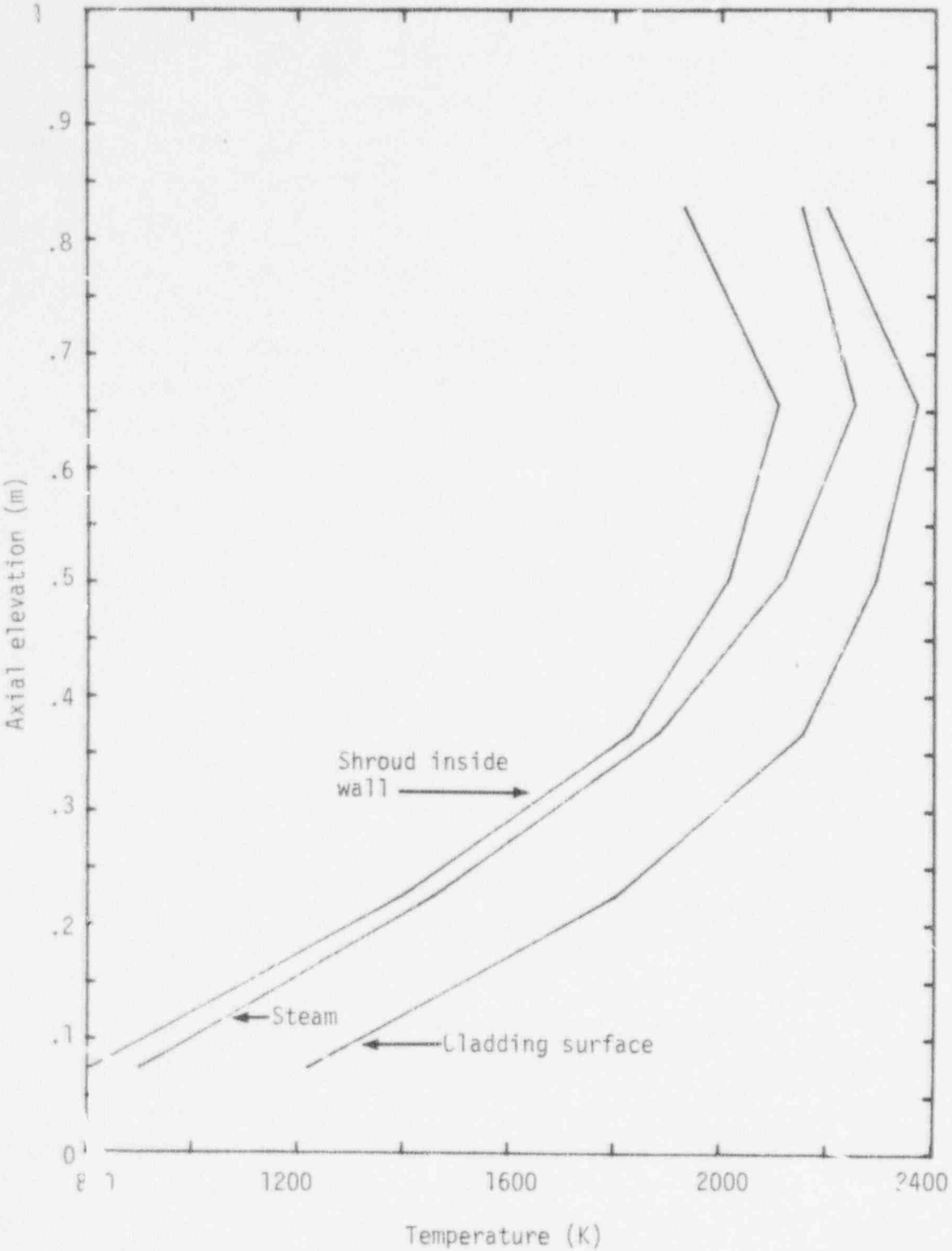

Figure 13. Calculated axial variations in cladding surface, steam, and shroud inside surface temperatures showing effect of axial power distribution and axial variation in cladding oxidation.

Calculations were also performed with the same test cluster initially full of subcooled liquid at an inlet temperature of 550 K, 0.72 m/s subcooled water flow rate and 7.0 MPa pressure. To simulate the boiloff sequence, the test rod power level was increased to 1.6 kW/ft (5.25 kW/m) to provide the additional energy to vaporize saturated water, and the inlet subcooled water flow rate was held constant at 0.72 m/s from time zero to 50 s, at which time the flow rate was reduced to 0.0048 m/s, within one second, and held constant for the duration of the calculation (700 s). The mass flow rate corresponding to 0.0048 m/s of subcooled water at 550 K is equivalent to a steam mass flow rate of 0.1 m/s.

During the boiloff process, single phase liquid coolant will enter the bundle at a low mass flow rate, vaporize while cooling the bundle, and exit the bundle as vapor when equilibrium is reached. The rod to coolant heat transfer processes within the bundle under these stabilized thermal-hydraulic conditions are shown schematically in Figure 14. At the bottom of the bundle, heat transfer is by single phase convection to the water. Nucleate boiling will commence when the water saturation temperature is reached, and then the heat transfer will change to forced convection boiling as the coolant quality increases and the flow regime changes from bubbly or slug to annular. Gradually, as the quality continues to increase, the fuel rod surface dries out and the heat transfer mode is dispersed-flow boiling with radiation to the vapor and water droplets. The radiation component is not turned on in TRAC until the quality is at least 0.8. (This value is a user option.) When the quality is calculated to be 1.0, heat transfer is by forced convection and radiation to the steam. Between a quality of 0.96 and 1.0, the TRAC heat transfer routine linearly interpolates between the dispersed-flow film boiling and forced convection to vapor heat transfer modes.

A frothy mixture of flowing steam and entrained liquid will probably exist over a considerable axial length of the test bundle. For these conditions, it is not feasible to identify a distinct liquid/vapor interface. However, a "collapsed water level" can be defined, which is determined by calculating the amount of liquid present within each axial

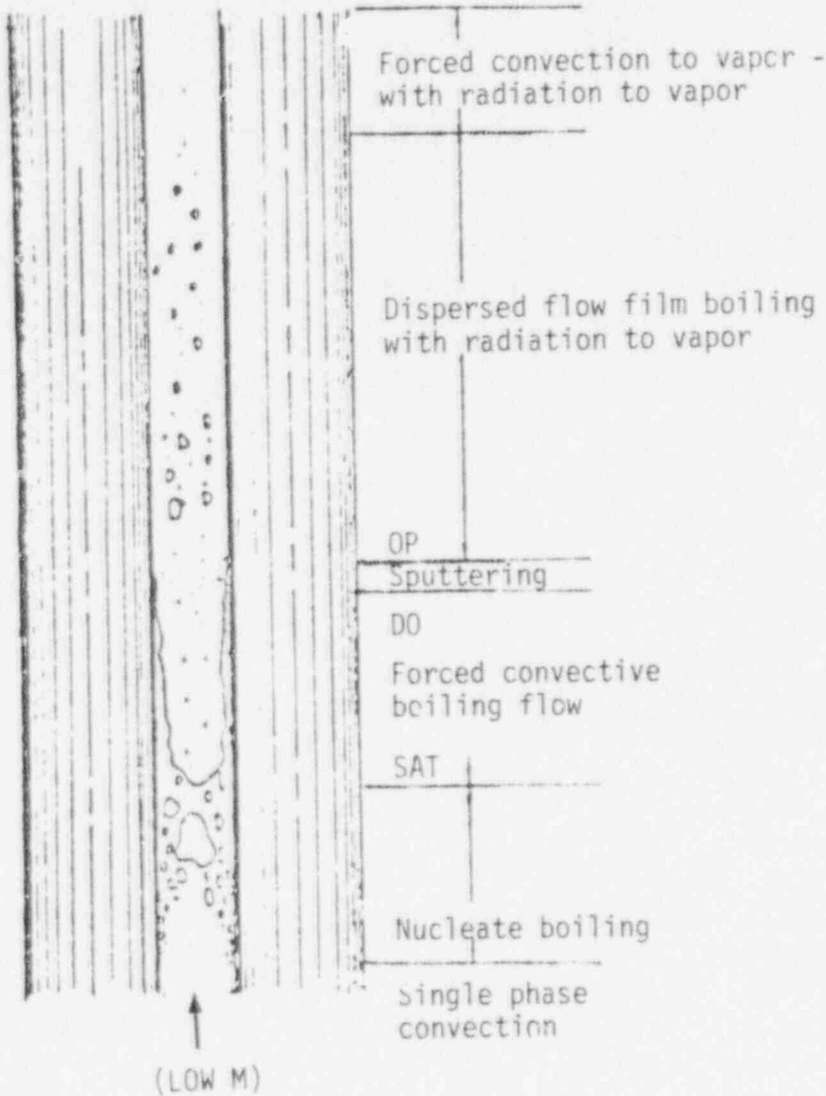

Figure 14. Heat transfer regimes modeled by TRAC-BDO at different axial levels with low inlet coolant flow rate.

level, and artificially summing the liquid to an equivalent "collapsed" level. The formula that was used is

$$\text{collapsed water level} = \sum_{i=1}^{n} (1 - \alpha_i) \frac{V_i}{A}$$

where α_i = vapor fraction in axial level, i

n = number of axial levels

V_i = volume of axial level, i

A = Cross-sectional flow area for each axial level.

The collapsed water level and the test rod peak cladding temperature are strong functions of the test rod peak power density and the coolant mass flow rate. Figure 15 shows the calculated collapsed water level as a function of time. In the peak power region, and above, the water very quickly vaporizes and the water level drops to about 0.375 m from the bottom of the fuel. The collapsed liquid level remains at about this level with a steady inlet flow rate of 0.72 m/s subcooled water. When the flow rate is reduced to 0.0048 m/s (from time t = 50 s to t = 51 s), the collapsed liquid level immediately drops to essentially the bottom of the fuel, i.e., the inlet to the test cluster.

The responses of the test rod cladding temperature, steam temperature, and insulating shroud inside surface temperature at the elevation of peak cladding temperature are shown as functions of time in Figure 16. The cladding and steam temperatures show the effect of the high initial coolant flow rate by remaining at the initial temperature until 50 s, at which time the flow rate is reduced to the low rate of 0.0048 m/s and the temperatures of the cladding and steam immediately begin to increase at extremely high rates. The shroud temperature is initially higher than the cladding and steam temperatures (at time zero) because of an initialization transient, but very quickly reduces to realistic values and then lags slightly in

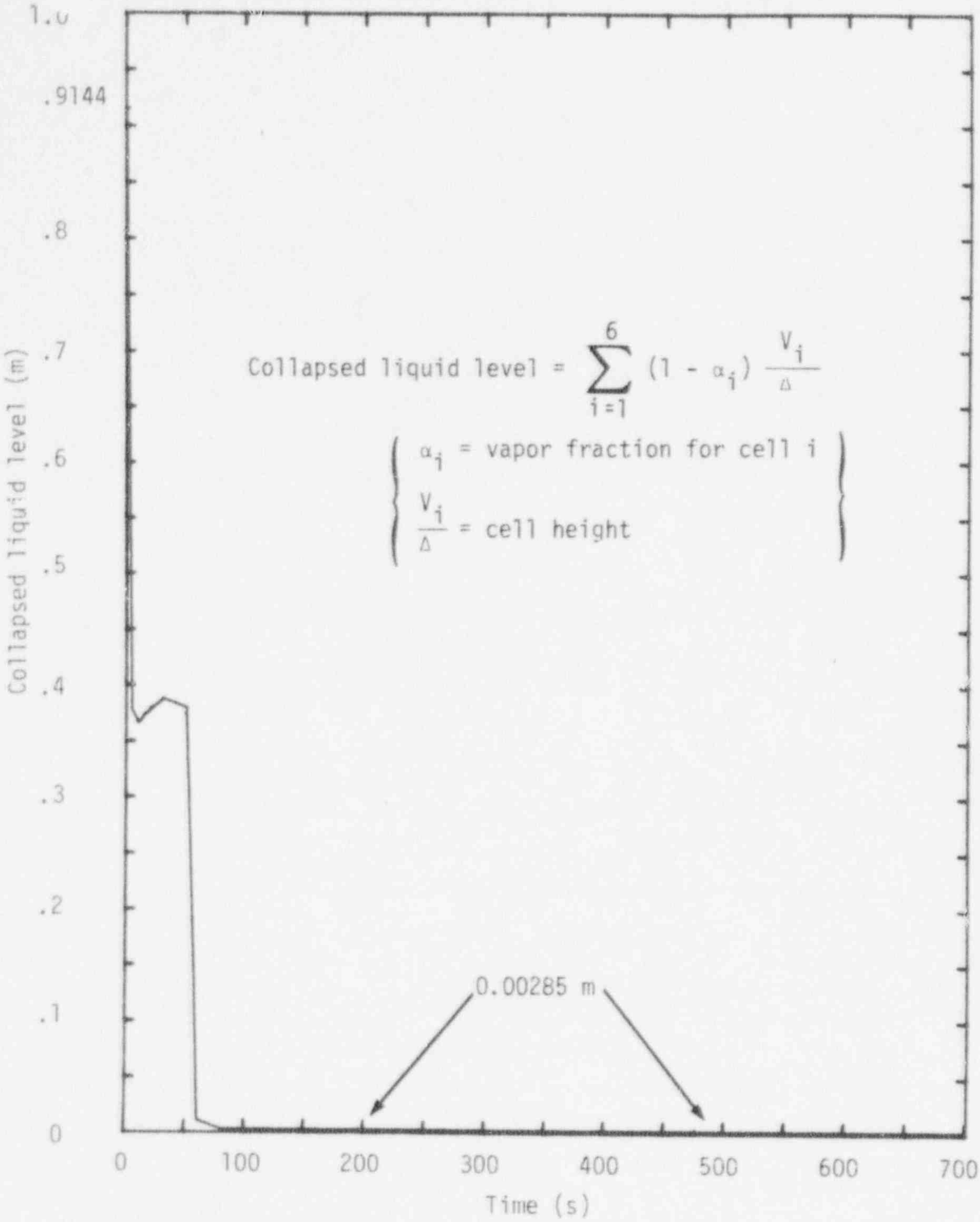

Figure 15. Calculated collapsed water level as a function of time for a rod power of 1.6 kW/ft and an initial inlet flow rate of 0.72 m/s, reducing to 0.0048 m/s at 51 s.

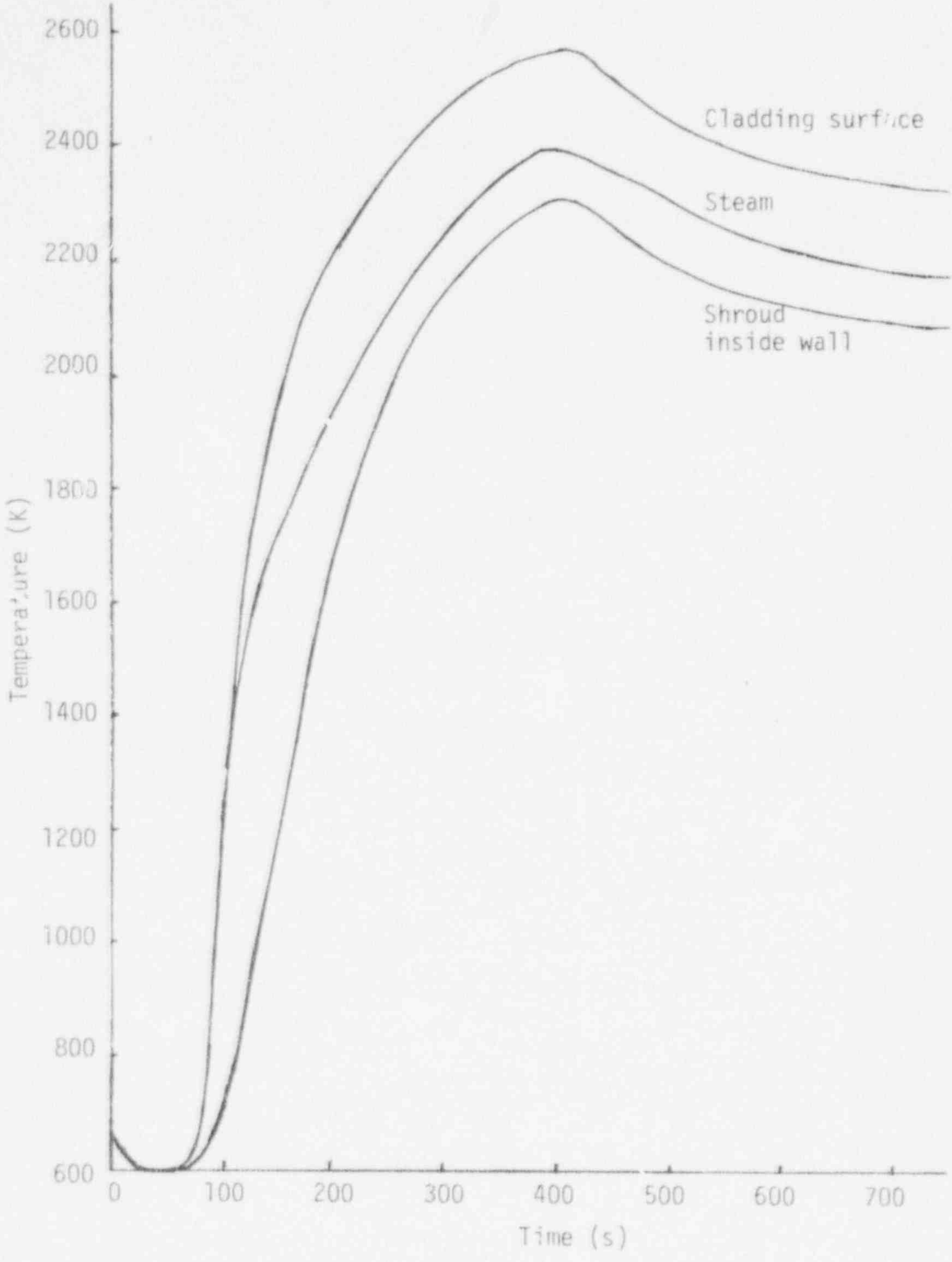

Figure 16. Calculated temperatures as function of time for the cladding surface, steam and shroud inside surface at the axial level of the peak cladding surface temperature for a rod power of 1.6 kW/ft and a steam flow rate of 0.72 m/s reducing to 0.0048 m/s.

time but also increases very rapidly after about 100 s. At 400 s the energy added by the metal-water starts decreasing rapidly and the temperatures of the cladding, steam, and shroud inside wall also decline, reaching steady state values by 700 s when the calculations were terminated.

The all-steam calculations resulted in peak temperatures approximately 100 K higher than the more realistic boiloff calculations, but the time to reach peak temperatures, after boiloff had occurred, was the same (300 s) in both cases.

During the time from zero to 50 s, at axial level 5, the location of peak cladding temperature, the primary mode of heat transfer is by convection to the steam, as shown in Figure 17. When the inlet flow rate is reduced to 0.0048 m/s, the convective heat flux is quickly reduced to zero and the primary mode of heat transfer becomes radiation to the steam. The initial positive convection heat flux from the shroud inside wall is due to the initialization transient described earlier and does not affect the response following the boiloff. During the boiloff process, the metal-water reaction is delayed until 100 s. From 100 s until the end of the calculation at 700 s, the response of the cluster is essentially identical to the calculated response described in Section 6.1 for the steam atmosphere conditions, as shown in Figures 17, 18, and 19.

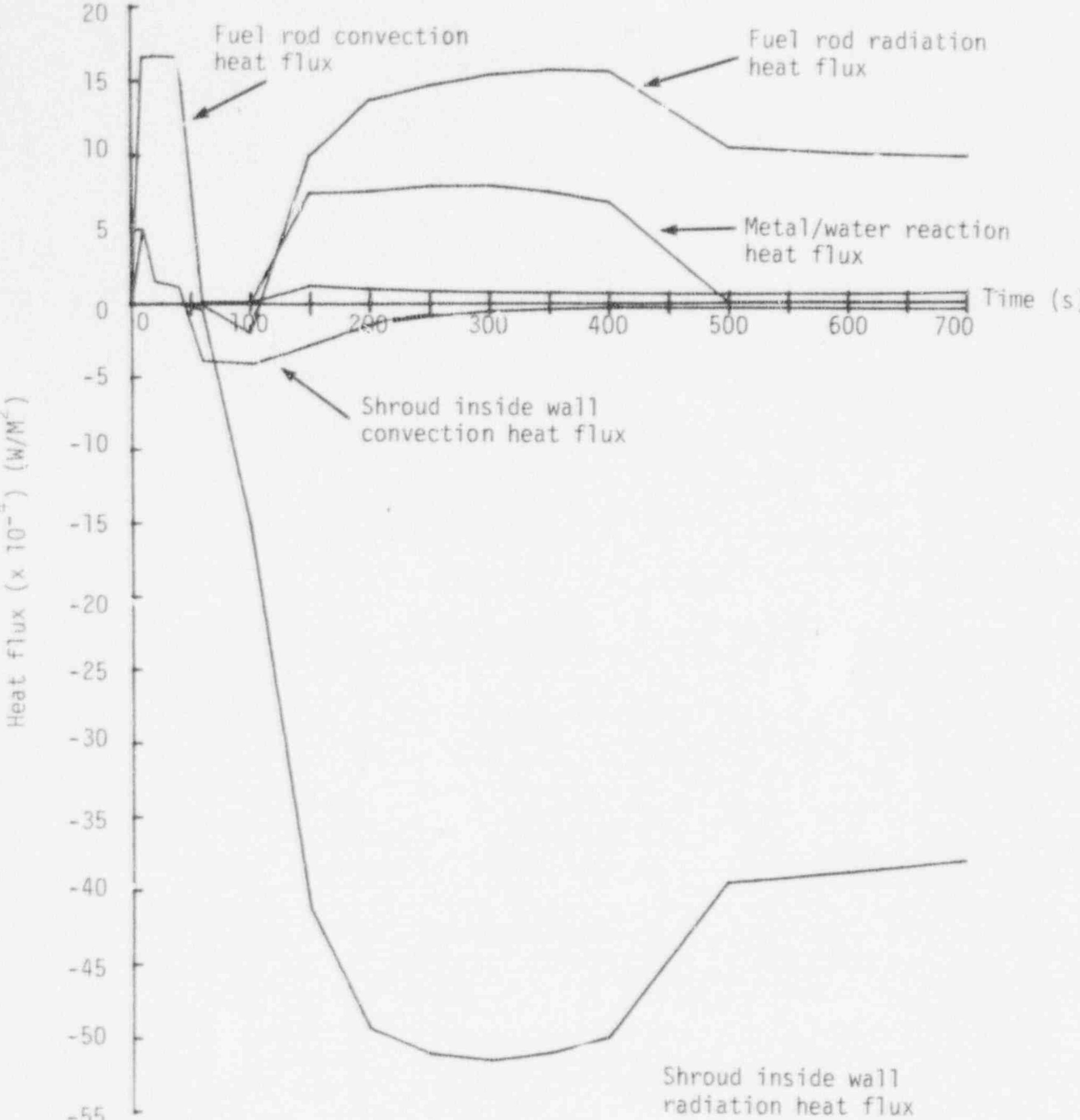

Figure 17. Calculated convection and radiation heat fluxes as functions of time at the axial level of the peak cladding surface temperature for a test rod power of 1.6 kW/ft and a steam flow rate of 0.72 m/s reducing to 0.0048 m/s.

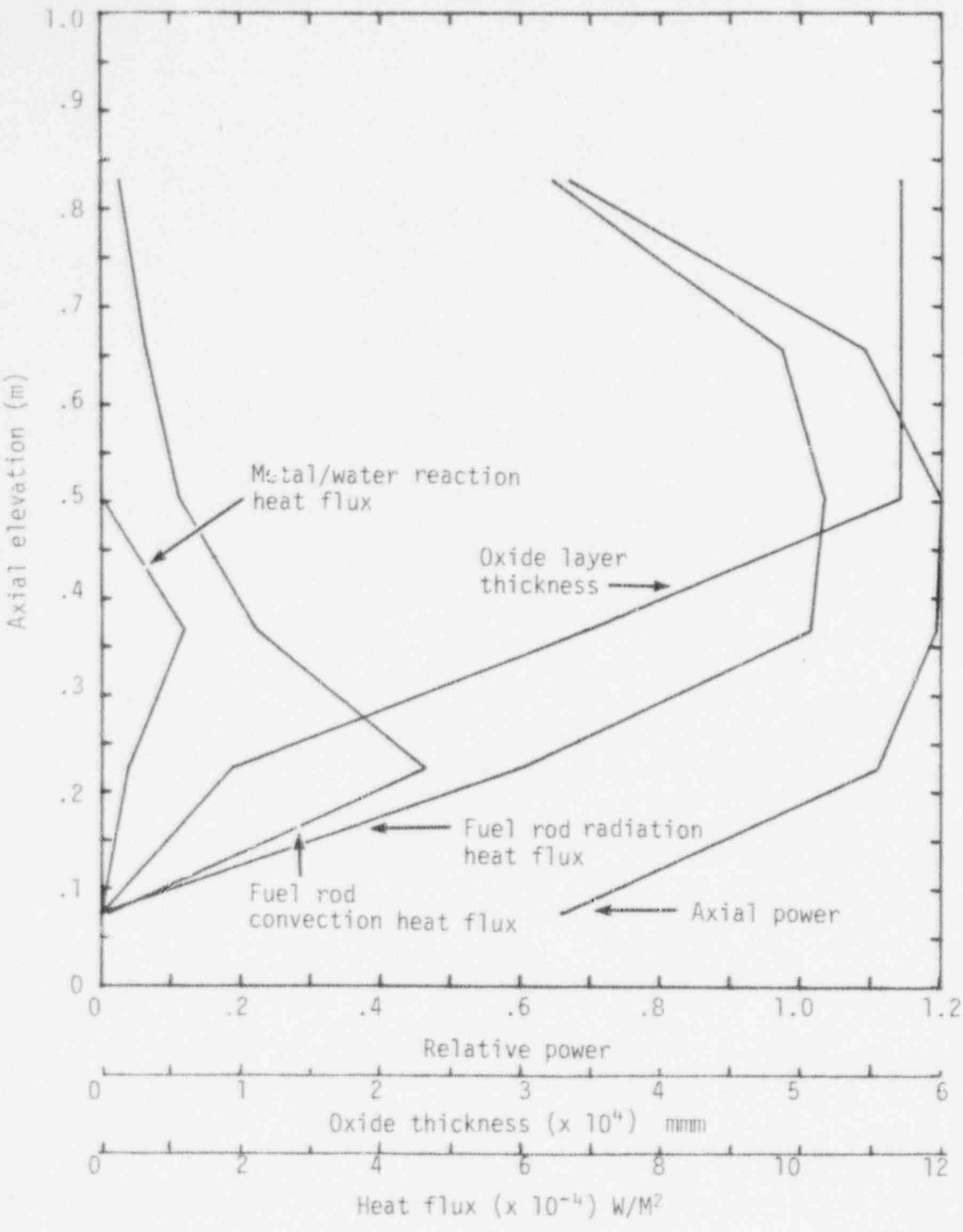

Figure 18. Comparisons of calculated axial variations in the rod convection, radiation, and metal/water reaction heat fluxes and the zirconium oxide layer thickness with the relative axial power distribution for a rod power of 1.6 kW/ft and a steam flow rate of 0.72 m/s reducing to 0.0048 m/s.

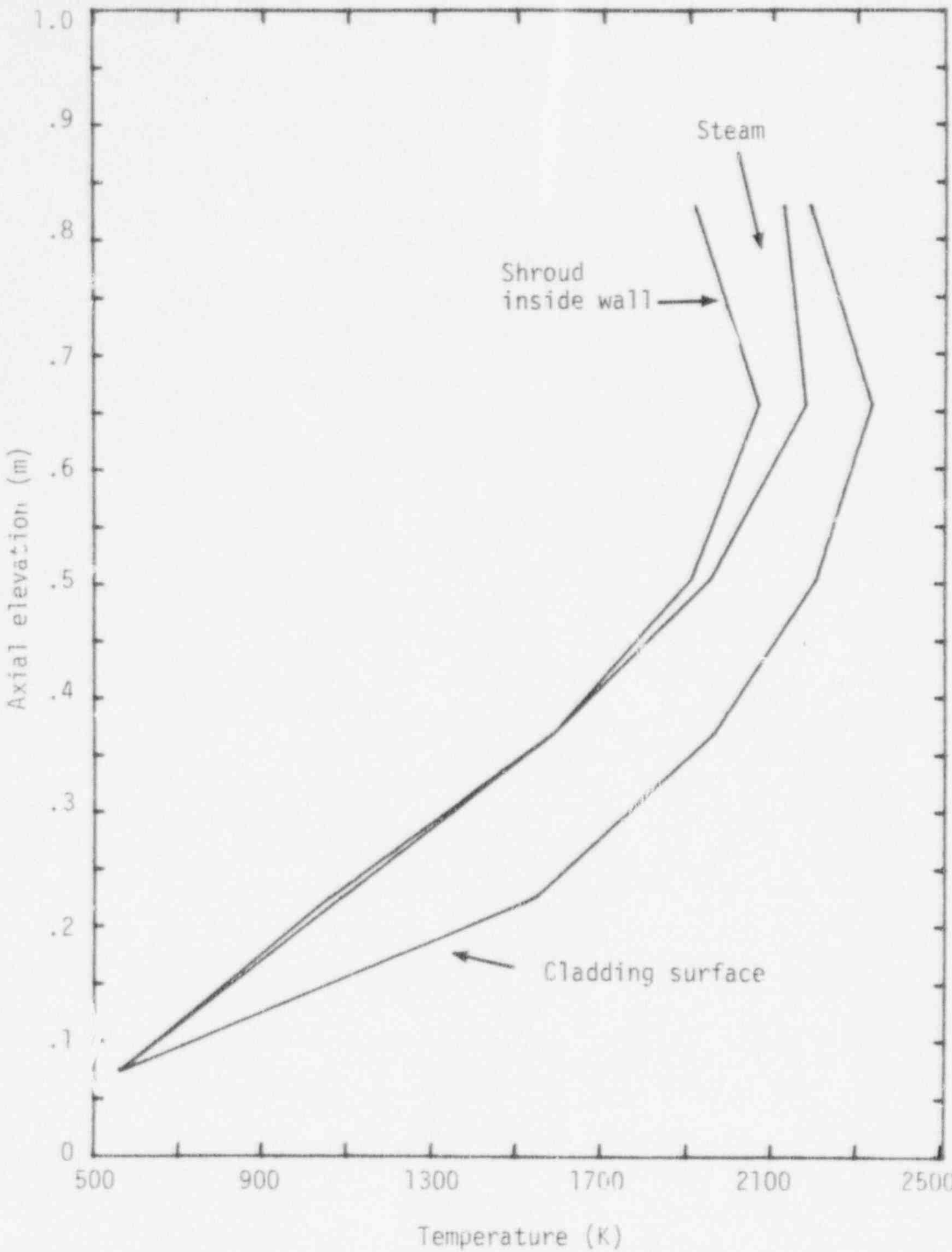

Figure 19. Calculated axial variations in cladding surface, steam, and shroud inside surface temperatures showing effect of axial power distribution and axial variation in cladding oxidation.

7. CONCLUSIONS

The TRAC-BDO analyses described in this report indicate that the desired conditions (2300 K peak cladding temperature in a steam environment) for performing small break LOCA/severe fuel damage experiments in the PBF test loop can be attained with either a 4 x 4 or a 5 x 5 cluster of 17 x 17 PWR design test rods. With a 5 x 5 cluster, test rod peak power densities as low as 1.6 kW/ft (5.25 kW/m) appear to be adequate, and the desired steam atmosphere can be achieved by reducing the inlet coolant flow to approximately 0.0048 m/s at the inlet to the test cluster.

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