REACTIVITY INITIATED ACCIDENT TEST SERIES TEST RIA 1-4
EXPERIMENT PREDICTIONS

S. K. Fukuda
Z. R. Martinson

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U.S. Department of Energy

Idaho Operations Office • Idaho National Engineering Laboratory

This is an informal report intended for use as a preliminary or working document

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The RELAP and FRAP-T model and code updates utilized for the RIA test predictions contained in this report have been reviewed by the EG&G PBF Pretest Prediction Consistency Committee.

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SUMMARY

The results of the pretest analyses for Test RIA 1-4 are presented. Test RIA 1-4 consists of a 3x3 array of previously irradiated MAPI fuel rods. The rods have 5.7% enriched UO_2 fuel in zircaloy-4 cladding with an average burnup of 5300 MWd/t. The objective for Test RIA 1-4 is to provide information regarding loss-of-coolable fuel rod geometry following RIA event for a radial-average peak fuel enthalpy equivalent to the present licensing criteria of 1172 J/g (280 cal/g UO_2). Radial averaged peak fuel enthalpies of 1172 J/g (280 cal/g) 1077 J/g (257 cal/g), and 978 J/g (234 cal/g) for the corner, side, and center fuel rods, respectively, are planned to be achieved during a 2.7 ms reactor period power burst.

The results of the FRAP-T5 analyses indicate that all nine rods will fail within 26 ms from the start of the power burst due to pellet-cladding mechanical interaction. All of the rods will undergo partial fuel melting. All rods will operate under extended film boiling (>30 sec) conditions and about 70% of the cladding length is expected to be molten. Approximately 15% of the cladding thickness will be oxidized. Fuel swelling due to fission gas release and melting combined with fuel and cladding fragmentation, will probably produce a complete coolant flow blockage within the flow shroud.

I. INTRODUCTION

This document presents the pretest analysis for the Reactivity Initiated Accident (RIA) Test RIA 1-4. The test will be conducted in the Power Burst Facility (PBF) at the Idaho National Engineering Laboratory as part of the Nuclear Regulatory Commission's Fuel Behavior Program¹. The objectives of the RIA test series are to: (a) provide fuel performance data during a RIA event for model development and assessment purposes, and (b) provide data for evaluation and possible revision of the current nuclear reactor licensing criteria regarding hypothetical control rod drop or ejection accidents in commercial nuclear power plants. The specific objective for Test RIA 1-4 is to provide information regarding loss-of-coolable fuel rod geometry following a RIA event for a peak fuel enthalpy equivalent to the present licensing criteria of 1172 J/g (280 cal/g).

The most severe RIA is the postulated Boiling Water Reactor (BWR) control rod drop during reactor startup. Therefore the test will be conducted at BWR hot startup coolant conditions (538 K, 6.45 MPa, 0.76 l/sec). The test assembly will consist of a 3x3 array of preirradiated fuel rods. The nine fuel rods contain 5.7% enriched UO_2 in zircaloy-4 cladding irradiated to an average burnup of 5300 MWd/t.

The test sequence will begin with steady state power operation to condition the fuel, establish a short-lived fission product inventory, and calibrate the calorimetric measurements and core power chambers. The test train will be removed from the in-pile tube (IPT) to replace the flux wires and one of the fuel rods. Several trial bursts will be made without the test train in the IPT to verify the reactor period required to provide a radially averaged peak fuel enthalpy of 1172 J/g (280 cal/g) at the axial peak in the corner rod, 1077 J/g (257 cal/g) in the side rods, and 978 J/g (234 cal/g) in the center rod. The test train will be reinserted into the IPT and the actual power burst conducted.

Section 2 of this document gives a brief description of the test hardware and instrumentation. Section 3 describes the test sequence. Section 4 describes the pre-test analysis of the fuel rod response. Section 5 presents the conclusions of the pretest analysis.

II. EXPERIMENT HARDWARE

A description of the test hardware and instrumentation is provided in this section.

1. Test Hardware

Test RIA 1-4 will consist of a 9 rod bundle of preirradiated MAPI^a fuel rods. The rod bundle will be contained in a zircaloy flow shroud. Four axially spaced grid spacers at 15, 320, 625, and 930 mm above the bottom of the fuel stack form a 3 x 3 fuel rod array with a rod-to-rod pitch of 14.3 mm inside the flow shroud. A support structure will center the flow shroud inside the PBF in-pile tube.

A total of 10 preirradiated fuel rods will be used in Test RIA 1-4. After preconditioning and power calibration has been completed, one rod will be replaced for the power burst. Table 1 presents the nominal design characteristics for the fuel rods and Table 2 presents the specifications for the individual fuel rods. None of the fuel rods will be opened prior to testing.

2. Test Instrumentation

The instrumentation for Test RIA 1-4 will provide information on fuel rod energy deposition, the coolant pressure pulse generated during the power burst, and time in film boiling. Fuel rod and test train instrumentation are described in Reference 2. Figures 1 and 2 show the relative location of the instrumentation to be used in Test RIA 1-4.

a. MAPI is an acronym for Mitsubishi Atomic Power Industries.

TABLE 1

NOMINAL DESIGN CHARACTERISTICS FOR TEST RIA 1-4 FUEL RODS

Characteristic ^a	MAPI
Fuel	
Material	UO ₂
Pellet OD (mm)	8.59
Pellet length (mm)	15.2
Pellet enrichment (wt%)	5.7
²³⁵ U weight (g/rod)	27.0
Density (% TD) ^b	94
Fuel stack length (m)	0.914
End configuration	dished
Cladding	
Material	zircaloy-4
Tube OD (mm)	9.99
Tube wall thickness (mm minimum)	0.572
Yield strength (MPa)	570
Ultimate strength (MPa)	700
Fuel Rod	
Gas plenum length (mm)	45.7
Insulator pellets	none
Backfill gas composition	c
Backfill pressure (MPa)	0.103
a. Data are preirradiation values. b. Theoretical density (TD) of UO ₂ (10.97 g/cm ³). c. See appendix C for gas composition.	

TABLE 2
SPECIFICATIONS FOR INDIVIDUAL FUEL RODS FOR TEST RIA 1-4

<u>Rod Number</u>	<u>Original Westinghouse Number</u>	<u>Average Burnup (MWd/t)</u>
804-1	M-55	6090
804-2	M-30	4440
804-3	M-34	5500
804-4	M-29	4950
804-5	M-62	5450
804-6	M-38	5050
804-7	M-41	5930
804-8	M-40	4720
804-9	M-61	5650
804-10	M-32	4390

Figure 1. Flow shroud and fuel rod instrumentation

Figure 2. Test train instrumentation

III. EXPERIMENT CONDUCT

Test RIA 1-4 will consist of three primary test phases; the fuel rod power calibration and preconditioning phase, the trial power burst check with the test train removed, and the test power burst transient.

The rod power calibration and preconditioning phase will calibrate the thermal-hydraulically determined fuel rod power with reactor power and self-powered neutron detectors (SPNDs). Fuel relocation and a buildup of short-lived fission products will occur during this phase.

After completion of the calibration and preconditioning phase, the test loop will be cooled and the test train removed from the IPT. The flux wires mounted on the flow shroud will be replaced and fuel Rod 804-10 will be replaced by fuel Rod 804-2.

The second phase will consist of two or more trial power bursts with the test train out of the IPT to provide information on the reactor period required to provide the desired test power burst. The reactor period for the test power burst will be chosen such that a predicted radially averaged fuel enthalpy of 1172 J/g (280 cal/g) is achieved at the axial power peak of the corner fuel rods.

After replacing the test train in the IPT, the final test phase will consist of a power burst starting from BWR hot startup conditions of 538 K, 6.45 MPa, and a fuel bundle shroud coolant flow of 0.766 l/s.

IV. EXPERIMENT PREDICTIONS

The experiment predictions for Test RIA 1-4 are described below. The predictions are based on the analysis of reactor physics, thermal-hydraulic response, and fuel rod steady state and transient response expected during the test. The computer codes along with the computer code configuration control numbers used in the analysis are listed in Appendix A.

1. Reactor Physics Analysis

A physics analysis for Test RIA 1-4 was performed to determine: (a) fuel enrichments necessary to provide the most balanced energy depositions in the test rod array, (b) the $R-\theta$ power distributions in the test rod bundle for thermal-hydraulic and fuel rod analysis, and (c) the core energy release required to attain the desired fuel enthalpy in the test.

A summary of the calculational methods used in the physics analysis is as follows. The computer code COMBINE was used to collapse the ENDF/B IV library to provide 32 group, and 4 group, constants for the test space. These constants were used by the S_N code SCAMP to generate the neutron currents entering the test space. The Monte Carlo code, RAFFLE, used the neutron currents provided by SCAMP to determine the neutron flux and the absorption, fission and production of neutrons in the test space. The GAMSOR code was then used to obtain the prompt γ sources in the test region. The nuclide buildup and depletion code, SINBAD, was used to determine the γ and delayed β sources for the test space. Finally, the shielding code, QUADBS, was used to determine the spatial distribution of the γ energy deposition.

The results of the physics calculations indicate that of the enrichments available (5.7, 9.5, and 12%) in the preirradiated fuel rods, the most uniform distribution of rod-to-rod energy deposition would occur using all 5.7% enriched fuel rods. The calculated energy

deposition distributions within the corner, center and side fuel rods are shown in Figure 3. Relative rod-averaged powers of 0.88, 0.97, and 1.06 were calculated for the center, side and corner rods, respectively. As expected, the relative powers are higher on the outer, higher neutron flux regions of the bundle. As shown in the figure, there exists a power gradient across the side and corner rods. The power gradient is caused by depression of the neutron flux, due to neutron absorption across the diameter of the peripheral rods. The center rod, which is irradiated at a lower but relatively uniform neutron flux than the peripheral rods, has a relatively flat circumferential power distribution, similar to that of previous single-rod tests. For the side rods, the maximum pellet surface peaking factor is about 14% higher than the minimum pellet surface peaking factor. For the corner rods, the maximum pellet surface peaking factor is about 22% higher than the minimum pellet surface peaking factor.

The calculated figures of merit (FOM, ratio of test rod linear power to core power) for an average rod in the bundle are listed in Table 3. The calculated figure of merit changes slightly during the preconditioning due to control rod position differences and changes in the energy per fission as a function of time. The average energy per fission for the test rods and the average PBF core fuel rod, caused by various fission energy sources during preconditioning and the power burst, are listed in Table 4. The PBF core power and test rod FOM's as a function of control rod position are shown in Figure 4. For the control rod position estimated to be required for the power burst (0.591 m withdrawn), the steady state figure of merit for the axial peak of the average test rod will be 1.45 kW/m-MW and 0.59 cal/g per MW·s during the test power burst. Based on the relative rod power peaking factors of 0.88, 0.97, and 1.06 for the center, side, and corner rods, the steady state figure of merits for the axial peak of the test rods will be 1.28, 1.41, and 1.54 kW/m-MW for the center, side, and corner rods respectively. The power burst figure-of-merit values are 0.516, 0.565, and 0.624 cal/g per MW·s of core energy release for the center, side, and corner rods, respectively.

Figure 3. Relative power distribution for Test RIA 1-4.

Note: Values underlined are the radial averages for the sector. Values given in boxes are ratio of rod power to average rod power in test.

TABLE 3
ANTICIPATED POWER HISTORY DURING PRECONDITIONING

Approximate Axial Peak Power For Average Rod (kW/m)	Elapsed Time (min.)	MeV/fission in Test	MeV/fission in PBF Core	Average Rod FOM (kW/m·MW)
0.1	6.5			
10.0	14.5	185.1	196.1	1.47
0.1	19.5			
20.0	20.	185.5	196.2	1.44
0.1	25.			
30.0	20.	185.6	196.7	1.40
0.1	25.			
40.0	20.	185.7	196.8	1.38
0.1	20.			
5.0	15.	190.5	202.6	1.50
0.1	20.			
15.0	20.	186.2	197.5	1.45
0.1	25.			
25.0	20.	185.8	197.0	1.41
0.1	25.			
35.0	20.	185.8	197.0	1.39
0.1	20.			
10.0	15.	187.8	199.4	1.47
0.1	20.			
20.0	20.	186.2	197.4	1.43
0.1	25.			
30.0	20.	185.9	197.1	1.40
0.1	25.			
40.0	20.	185.8	197.0	1.38
0.1	20.			
0.	2880.			
Power Burst		176.8	186.1	(a)

a. 2.45 J/g· per MJ core energy release.

TABLE 4

TOTAL ENERGY PER FISSION 9-ROD RIA 1-4 TEST

Energy-per-Fission During Preconditioning, $\frac{\text{MeV}}{\text{fission}}$						
<u>Rod</u>	<u>Fission Fragments</u>	<u>Betas</u>	<u>Gammas & Neutrons</u>	<u>Fuel Rods</u>	<u>Water & $\frac{1}{2}$ Shroud</u>	<u>Total</u>
Center	167.9	6.2	14.2	185.3	3.1	188.3
Corner	167.9	6.2	11.3	182.9	2.5	185.4
Side	167.9	6.2	12.6	184.0	2.8	186.8
Average	167.9	6.2	12.1	183.6	2.7	186.2
PBF Core	167.9	6.2	23.4			197.5

Energy-per-Fission During Burst*, $\frac{\text{MeV}}{\text{fission}}$					
<u>Rod</u>	<u>Fission Fragments</u>	<u>Gammas & Neutrons</u>	<u>Fuel Rods</u>	<u>Water & $\frac{1}{2}$ Shroud</u>	<u>Total</u>
Center	167.9	10.4	175.8	2.5	178.3
Corner	167.9	8.2	174.1	2.1	176.1
Side	167.9	9.2	174.9	2.3	177.1
Average	167.9	8.9	174.6	2.2	176.8
PBF Core	167.9	18.2			186.1

* Contributions by delayed gammas and betas are negligible (less than 0.1 MeV/fission for a burst of 30-40 ms full width at half-maximum).

Figure 4. PBF core power and RIA 1-4 Test FOM as functions of control rod position.

To achieve the test objective of a peak fuel enthalpy of 1172 J/g (280 cal/g) at the axial peak in the corner rod, the analyses discussed in the following sections indicates that a total power burst fuel energy of about 1570 J/g (375 cal/g) will be required for the corner rod. Based on previous RIA test results, a fuel energy of about 305 cal/g up to the time of control rod scram will be required to provide a total power burst energy of 375 cal/g for the corner rod. This amount of test rod power burst energy is equivalent to a PBF core power burst energy of 460 MW s. According to previous RIA and PBF lead rod test data, a power burst with a reactor period of 2.7 ms, a reactivity insertion of 2.26\$, and a peak power of 37,000 MW will be required.

2. Thermal-Hydraulic Analysis

Thermal-hydraulic analyses using RELAP-IV/MOD6 and COBRA IV were performed to provide the boundary conditions for the FRAP-T5 analysis. A RELAP-IV calculation was performed to determine the pressure, flow, enthalpy, and surface heat flux histories in the test section. The inlet and outlet mass flows as calculated by RELAP are presented in Figure 5 and the surface temperatures of each of the heat slab segments are shown in Figure 6. The results from RELAP-IV were then used as inlet and outlet boundary conditions for the COBRA-IV calculations. The COBRA IV results provided the thermal-hydraulic boundary conditions for each rod analyzed in the FRAP-T5 calculations. The COBRA IV calculated relative mass flux (local mass flux/local bundle averaged mass flux) through the shroud prior to the initiation of the power burst is shown in Figure 7. As shown in the figure, the flow redistributes toward the center channel as the coolant moves upward through the shroud. This is probably due to the larger pressure drop in the side and corner channels caused by the shroud walls. The COBRA IV calculated relative mass flux through the shroud at 29 s after the initiation of the power burst is shown in Figure 8. Again the flow is redistributed toward the center channel

Fig. 5 RELAP-IV inlet and outlet mass flow for Test RIA 1-4.

Fig. 6 RELAP-IV heat slab surface temperatures for Test RIA 1-4.

Figure 7. COBRA relative mass fluxes for the center, side, corner channels at $t=0$ s.

Figure 8. COBRA relative mass fluxes for the center, side, corner channels at $t=29$ s.

but to a lesser extent than at 0 s, due to the two-phase mixture exiting the shroud at 29 s (exit quality is 0.13). A brief summary of the models and assumptions used in the thermal-hydraulic calculations is given in Appendix B.

The results of the RELAP-IV and COBRA-IV calculations indicate that flow reversal will occur during the power burst due to rapid heat transfer into the coolant. Flow reversal is also predicted to occur at 11.2, 22.8, and 24.5 seconds into the transient, but these flow reversals will probably not occur, since these times only correspond to the quench times of 3 of the 5 axial heat slab segments used to model the fuel bundle in RELAP-IV. Four of the five segments are predicted by RELAP-IV to quench within the first 30 seconds. The center heat slab segment is predicted to reach the cladding melting temperature at less than 1.5 seconds into the transient, and remain molten for the remainder of the 30 seconds analyzed by RELAP-IV.

3. Fuel Rod Behavior Analysis

The steady state and transient fuel rod response for Test RIA 1-4 was calculated using the FRAP-T4 and FRAP-T5 codes. The FRAP-T4 and FRAP-T5 codes couple the effects of mechanical, thermal, internal gas, and material properties to determine the response of the fuel rod during transient operation.

3.1 Preconditioning Operation

The minimum departure from nucleate boiling (DNB) ratios were calculated for the center, side, and corner rods during the preconditioning phase for an axial peak power level of 40 kW/m. The minimum DNB ratios were 3.21, 3.16, and 3.17 for the center, side, and corner rods respectively. The FRAP-T5 code was used to calculate the DNB ratio of the corner rod because of temperature-deformation convergence problems encountered in FRAP-T4.

3.2 FRAP-T5 Calculations for the Corner Rod

The results of the FRAP-T5 calculations for the corner rod are presented below. The power burst shape used in the calculations was obtained from the PBF Lead Rod Test No. 9 which had a 2.5 ms period, near that predicted to be required for Test RIA 1-4. The power burst shape had a width at half-maximum power of 9.4 ms. The linear power was scaled to produce a radial averaged, peak axial fuel enthalpy of 1180 J/g (282 cal/g) for the burst. In addition to the burst energy, the energy after scram was included. The energy after scram was obtained from previous RIA test data and scaled to produce 26% of the calculated burst energy deposition. The thermal-hydraulic boundary conditions used in FRAP-T5 were obtained from the corner channel conditions calculated by COBRA-IV. A summary of the input conditions used for the FRAP-T5 calculations is given in Appendix C. All times given in this report are referenced to the initiation of the power burst. The time of peak power was 24 ms after the initiation of the power burst. All enthalpy values given in this report use 300K as the reference temperature.

Figure 9 shows the local fuel rod power and calculated fuel centerline, fuel surface and cladding surface temperatures at the axial node located 0.432 m from the bottom of the fuel stack (BOF) of the corner rod during the first 0.15 seconds of the transient. Figure 10 compares the same variables as shown in Figure 9 over the entire 30 second calculation time, and indicates that the cladding surface and fuel centerline are predicted to be molten for about 6 and 13 s, respectively. The predicted fuel and cladding axial elongation during the transient are shown in Figures 11 and 12 for the corner rod. Figure 13 shows the radial temperature profile at the axial peak location for the corner rod at 0.026 s (rod failure), 0.042 s (time of peak fuel enthalpy), 0.194 s and 29.2 seconds. Figures 14 and 15 show the energy distribution of the corner rod at the axial peak location during the first 170 ms of the transient. A plot of the power burst

Figure 9. Local fuel rod power and calculated fuel centerline, fuel surface, cladding surface temperatures at the 0.432 m BOF position of the corner rod (0-0.15 sec).

Figure 10. Fuel centerline, fuel surface, cladding surface temperatures at the 0.432 m BOF position of the corner rod (0-30 sec).

Figure 11. Fuel and cladding axial elongation for the corner rod (0-0.25 seconds).

Figure 12. Fuel and cladding axial elongation for the corner rod (0-30 seconds).

Figure 13. Radial temperature profile for the corner rod at 6.026, 0.042, 0.194, 29.2 seconds.

Figure 14. Energy distribution of the corner rod.

Figure 15. Energy distribution of the center rod.

is also shown on these figures for reference. At the beginning of the burst, the stored energy of the rod is due to the initial temperature of 538 K. Figure 13 shows that during the burst, almost all the energy is deposited into the fuel. Figure 14 shows that the fuel starts to transfer a noticeable amount of energy to the cladding during the burst and that the cladding does not begin to transfer heat to the coolant until well after the power burst.

The energy deposition of the burst is nearly adiabatic. The radially averaged fuel temperature starts at 538 K with a flat profile and increases to 2320 K with the temperature profile peaked near the pellet surface during the first 25 ms of the burst as shown in Figure 11. In contrast, the average cladding temperature increases by only 91 K, from 538 to 629 K, within the same time period. The high fuel temperature causes the fuel to expand faster than the cladding and at 25 ms the structural gas gap closes over the central 40% of the rod (0.23 to 0.63 m BOF). The pellet-cladding interface pressure increases from zero to 55 MPa as the structural gas gap closes. Cladding failure (failure probability >50%) occurs 1 ms later (26 ms) over the upper 75% of the rod. The maximum interface pressure at time of failure is 71 MPa and the maximum permanent hoop strain is 0.2%. The radially averaged fuel enthalpy at the axial power peak is 765 J/g (183 cal/g) at the time of failure.

The time of failure (26 ms) occurs 2 ms after the time of peak power when only 49% of the burst energy has been deposited in the rod. Of the energy deposited in the rod to time of failure, only 1.2% of the burst energy has been transferred to the cladding at the failure location. The fuel temperature continues to increase as the rest of the burst energy is deposited in the fuel. At 29 ms, the fuel begins to melt (3098 K) just inside of the fuel surface in the region around the axial power peak (0.32-0.55 m BOF). The fuel reaches a radial averaged peak fuel enthalpy of 1178 J/g (282 cal/g) and a local peak fuel enthalpy of 1417 J/g (339 cal/g) at 42 ms. The fuel

contains about 97% of the energy deposited at 42 ms, the cladding has about 3% of the energy and the coolant has less than 1% of the deposited energy at 42 ms. The melt region grows axially and then radially inward until approximately 10% of the total fuel volume is molten at 144 ms into the transient. The molten region reestablishes a normal temperature profile (i.e. heat flux radially outward) 3.9 seconds after the start of the transient while 15% of the fuel is still molten. All of the fuel has resolidified at 5.9 seconds.

The cladding surface temperatures remain relatively constant for the first 25 ms but rapidly rise once the structural gas gap closes. Film boiling begins (cladding surface temperatures > 600 K) 6 ms after the gap closes (31 ms into the transient), reaches the α -recrystallization temperature (920 K) 20 ms after gap closure (45 ms into the transient), the α - β transition temperature (1105 K) 32 ms after gap closure (57 ms into the transient), and the β -phase transition temperature (1245 K) 45 ms after gap closure (70 ms into the transient). The inflection in the cladding surface temperature curve of Figure 9 is due to the heat of phase transformation and change in specific heat associated with the α - β transition.

Gap closure rapidly extends axially until the central 83% of the rod has a closed gap at 32 ms. At 58 ms into the transient, the entire rod length is in film boiling. The cladding continues to heat up and at 179 ms the cladding begins to melt on the inside surface at the fuel midplane (0.37 m BOF). The melt region expands outward and at 0.25 seconds, the 0.432 m BOF node reaches cladding melt-through. The region of cladding melt-through expands axially until the central 83% of the cladding is molten at 1.92 seconds into the transient. The molten cladding region starts to decrease at 2.92 s and has completely resolidified at 14 s into the transient.

The rod remains in film boiling beyond the 30 seconds analyzed by FRAP-T5. The extended period of time the rod is in film boiling

causes oxidation of the cladding and energy generation from the metal-water reaction. The oxide layer thickness increases rapidly during the first second of the transient and then gradually reaches 15% of the original cladding thickness at 29 s. An oxygen stabilized α layer representing 25% of the original cladding thickness is also formed by 29 sec.

3.3 FRAP-T5 Calculations for the Center Rod

The shape of the power burst used in the FRAP-T5 calculations for the center rod was identical to the shape used for the corner rod, but the linear powers were scaled according to the calculated power distribution for the 9 rod bundle. The thermal-hydraulic boundary conditions were obtained from the center channel conditions calculated by COBRA-IV.

The local fuel rod power, cladding surface, fuel surface, and fuel centerline temperatures at the 0.432 m BOF location are presented in Figures 16 and 17. The axial elongation of the fuel and cladding are plotted in Figures 18 and 19.

The response of the center fuel rod is as follows. The rapid heat up of the fuel during the first 26 ms of the transient while the cladding temperatures remain relatively constant causes the structural gas gap to close over 35% of the rod. The pellet-cladding interface pressure rises rapidly once the gap is closed causing failure due to overstress at 27 ms over the upper 65% of the rod. Gap closure advances rapidly along the axial length of the rod and at 36 ms, the gap is closed over 74% of the rod. Once the gap is closed the pellet-cladding heat transfer increases causing the cladding to heat up rapidly. The cladding inside surface reaches the α - β transition temperature (1120 K) 5 ms after gap closure. Film boiling over the center third of the rod begins at 32 ms and energy released from the

Figure 16. Local fuel rod power and calculated fuel centerline, fuel surface, cladding surface temperatures at the 0.432 m BOF position of the center rod (0-0.15 sec).

Figure 17. Fuel centerline, fuel surface, cladding surface temperatures at the 0.432 m BOF position of the center rod (0-30 sec).

Figure 18. Fuel and cladding axial elongation for the center rod (0-0.25 sec).

Figure 19. Fuel and cladding axial elongation for the center rod (0-30 sec).

metal-water reaction begins. By 139 ms film boiling extends over the entire length of the rod.

A radially averaged peak fuel enthalpy of 978 J/g (234 cal/g) and a local peak fuel enthalpy of 1118 J/g (267 cal/g) at the axial peak is attained at 42 ms. The peak fuel enthalpy represents about 96% of the adiabatic energy deposition, the cladding about 4%, and the coolant less than 0.2% of the deposited energy.

Fuel melting begins at 37 ms just inside the pellet surface at the axial peak. The fuel completely resolidifies at 70 ms and reestablishes a normal temperature profile (heat transfer radially outward) at 2 seconds.

Cladding melting begins at 0.28 seconds into the transient and starts on the inside surface at the axial peak location. Complete cladding melt-through occurs at the axial peak (0.432 m BOF) at 0.36 seconds. The maximum extent of molten cladding occurs at 2.1 seconds when 66% of the rod has cladding melt-through. The cladding resolidifies at 12 seconds.

At the end of the 29 second period analyzed by FRAP-T5, 92% of the rod remains in film boiling. The maximum cladding thickness oxidized is located below the axial peak (0.37 m BOF) and represents 13% of the original cladding thickness. The oxygen stabilized α layer thickness represents 23% of the original cladding thickness at 29 sec.

3.4 FRAP-T5 Calculations for the Side Fuel Rod. The FRAP-T5 calculations for a side fuel rod used the same power burst shape as used for the corner and center rods. The power history was scaled according to the reactor physics calculations. Thermal-hydraulic conditions were provided by the side channel conditions calculated by COBRA.

The local rod power and calculated fuel centerline, fuel surface and cladding surface temperatures at the 0.432 m BOF position of the side rod are plotted in Figures 20 and 21. The fuel and cladding elongation of the side rod during the transient are shown in Figures 22 and 23.

The calculated response for the side fuel rod is very similar to the responses of the corner and center rods. The rapid heating of the fuel causes the gap to close over 34% of the rod at 25 ms. Failure (failure probability >50%) due to overstress occurs 1 ms later along the upper 65% of the fuel rod with a peak pellet-cladding interface pressure of 71 MPa. Gap closure rapidly extends along the length of the rod until 74% of the rod has a closed gap at 30 ms. Once the gap is closed, the cladding rapidly heats up and by 33 ms film boiling covers the central 52% of the rod. Fuel melting starts at the axial peak location just inside the pellet surface at 31 ms.

A radially averaged peak fuel enthalpy of 1077 J/g (257 cal/g) with a local peak fuel enthalpy of 1247 J/g (298 cal/g) was calculated to occur at 42 ms. At the time of peak fuel enthalpy, the fuel contains about 96% of the energy deposited, the cladding has about 4% and the coolant has less than 0.2% of the deposited energy. At the time of peak fuel enthalpy, 82% of the rod is in film boiling and 2% of the fuel is molten.

The cladding begins to melt on the inside surface at 0.22 seconds, at which time 2% of the fuel is molten. At 0.32 seconds, cladding melt-through occurs at the axial peak location while 1% of the fuel is still molten. The cladding melt through region expands axially until a maximum of 66% of the rod cladding has melt-through at 1.1 seconds.

The molten fuel region continues to shrink and at 2.1 seconds a normal temperature profile (i.e. heat transfer radially outward)

Figure 20. Local fuel rod power and calculated fuel centerline, fuel surface, cladding surface temperatures at the 0.432 m BOF position of the side rod (0-0.15 sec).

Figure 21. Fuel centerline, fuel surface, cladding surface temperatures at the 0.432 m BOF position of the side rod (0-30 sec).

Figure 22. Fuel and cladding axial elongation for the side rod (0-0.25 sec).

Figure 23. Fuel and cladding axial elongation for the side rod (0-30 sec).

reappears. At 3.1 seconds the fuel is completely resolidified. The outer surface of the cladding resolidifies at 8.1 seconds and all of the cladding resolidifies by 13 seconds.

The upper 92% of the rod remains in film boiling beyond the 29 seconds analyzed by FRAP-T5. The extended time in film boiling results in an oxide layer formation that is 14% of the original cladding thickness and an oxygen stabilized α layer that is 24% of the original cladding thickness.

V. DISCUSSION

The FRAP-T5 analyses described in the previous section indicate that the fuel rods will undergo severe stress loadings during the test. A summary of the FRAP-T5 results for Test RIA 1-4 is given in Table 5. This section will discuss the possible consequences of such stress loadings with respect to maintaining a coolable rod geometry in light of the results from Test RIA 1-1.³

The two irradiated fuel rods of Test RIA 1-1 with burnups of about 4600 MWd/t were identical to Test RIA 1-4 fuel rods. Each of the four rods in Test RIA 1-1 were enclosed in separate flow shrouds. A peak fuel enthalpy of 285 cal/g (local peak fuel enthalpy of 335 cal/g) resulted in complete flow blockage for the two preirradiated rods. The blockage was located in the central 20-30% of the flow shroud and was composed of fine fragments of deformed cladding and molten or nearly molten fuel that had swelled about 180% in area. The cladding had azimuthal cladding thickness variations and oxide layer thickness that indicated an accelerated oxidation rate.

The rods in Test RIA 1-4 will have burnups of approximately 5300 MWd/t with most of the fission gas atoms still held in the fuel matrix. The side and corner rods in Test RIA 1-4 will have large regions of fuel in a molten or near-molten state (local peak fuel enthalpy of 298 and 339 cal/g, respectively) for up to 5 seconds after the power burst causing extreme fuel swelling (50-180%) in the side and corner rods. The center rod will not have as large a molten region of fuel (local peak fuel enthalpy of 267 cal/g) as in the side and corner rods and is not expected to have extreme fuel swelling. All 9 rods are predicted to fail during the burst from fuel-cladding mechanical interaction due to the rapid heat up and expansion of the hot fuel against the cold cladding. After a failure crack has formed, the thermal expansion and fission gas swelling of the fuel against the breached cladding will probably cause the cladding to split open. The Test RIA 1-1 showed this type of cladding deformation but was limited

TABLE 5
SUMMARY OF FRAP-T5 RESULTS FOR TEST RIA 1-4

Rod Position	Radially Averaged Peak Fuel Enthalpy (cal/g)	Local Peak Fuel Enthalpy (cal/g)	Maximum Fuel Temperature (K)	Maximum Cladding Surface Temperature		Maximum Fuel Stack Elongation (m)	Maximum Cladding Elongation (m)
				At 0.43 m (K)	At 0.79 m (K)		
Center	234	267	3096	2098	1718	0.029	0.007
Side	257	298	3097	2098	1890	0.031	0.007
Corner	282	339	3142	2098	2098	0.032	0.008

deformation but was limited by the size of the shroud tube. Although Test RIA 1-4 has a higher coolant-to-fuel area ratio than Test RIA 1-1, it is probable that the cladding will continue opening until it contacts the adjacent rods.

As the hot fuel comes into direct contact with the coolant, the fuel will fragment or froth as the fuel is expelled from the deformed rod. The fuel coolant contact will also cause the fuel to cool down faster and undergo less fuel melting than predicted by FRAP-T.

All 9 rods will have severe cladding-oxidation and melting along with fuel fragmentation. The fuel and cladding fragments will probably collect near the grid spacers or the deformed fuel region to cause 80-100% flow blockage within the first 5 seconds of the test.

The cluster geometry of Test RIA 1-4 produces a flux tilt in the corner and side rods of 22% and 14% respectively. The circumferential temperature gradient in the corner and side rods produced by the flux tilt may cause the rods to bow between the two central grid spacers located at 0.320 and 0.625 m BOF.

A coolant pressure pulse of 2 to 3 MPa, due to direct γ and neutron heating of the coolant is expected to occur in Test RIA 1-4, similar to the 1.3 MPa coolant pressure pulse observed in Test RIA 1-1.

VI. REFERENCES

1. United States Nuclear Regulatory Commission, Reactor Safety Research Program, Description of Current and Planned Reactor Safety Research Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, NUREG-75/058 (June 1975).
2. C. J. Stanley, Z. R. Martinson, Reactivity Initiated Accident Test Series Test RIA 1-4 Experiment Operating Specification, TFBP-TR-323, June 1979.
3. S. L. Seiffert et. al., "Fuel Rod Behavior During a Reactivity Initiated Accident At Energy Depositions Near 280 cal/g UO₂," Proceedings of International Colloquium on Irradiation tests for Reactor Safety Programmes, Petten, the Netherlands, June 25-28, 1979.

APPENDIX A

CODE CONFIGURATION

1. F. J. Wheeler, User's Manual for the RAFL/4 General Purpose Monte Carlo Code, RE-P-78-065 (July 1978) CCCM Number H00941IB.
2. R. S. Marsden, SCAMP Code Input Description, MARS-15-68 May 17, 1968 CCCM Number H00029IB.
3. RELAP4/MOD6: A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems User's Manual, PG-R-77-06, March 1977, CCCM Number C00010006.
4. C. L. Wheeler, et al., COBRA IV-I: An Interim Version of COBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Codes, BNWL-1962, March 1976, CCCM Number H00324IB.
5. L. J. Siefken et al., FRAP-T4: A Computer Code for the Transient Analysis of Oxide Fuel Rods, CDAP-TR-78-027, July 1978, CCCM Number H00389IB.
6. L. J. Siefken et al., FRAP-T5: A Computer Code for Transient Analysis of Oxide Fuel Rods, NUREG/CR-0840, June 1979, CCCM NUMBER H001183B.

APPENDIX B

THERMAL-HYDRAULIC MODELS FOR TEST RIA 1-4

RELAP4/MOD6 UPDATE4 was used to perform the thermal hydraulic analysis of the RIA 1-4 test train. The average rod surface heat flux and average thermal hydraulic conditions at the test inlet and outlet obtained from RELAP4 were then input into COBRA IV so that the local subchannel conditions could be determined.

1. RELAP4 MODEL

The geometric model used by RELAP4 to analyze the RIA 1-4 9 rod experiment is shown in Figure B-1, control volumes 19-23 refer to the test section and correspond to the five axial heat slabs used to model the rod cluster. Table B-1 describes the hydraulic volumes used by RELAP to model the RIA 1-4 in-pile tube.

The heat transfer correlations used in the analysis consisted of the set of MOD6 blowdown correlations, the modified Tong-Young transition boiling correlation, the greater of the new Groeneveld or Condie-Bengston III film boiling correlation and a DNB correlation consisting of Tong (W-3) for the high flow subcooled state, Hsu and Beckner for the high flow saturated state and modified Zuber for low flow conditions. Choking phenomenon was allowed to occur only at junctions within the flow shroud. The Henry Fauske/Homogeneous Equilibrium Model was used to determine critical flow. A vertical phase slip model was used for the vertically stacked volumes in the downcomer and bypass regions of the in-pile tube.

An axial average energy insertion of 774 J/g (185 cal/g) was specified as the fuel rod power transient input for RELAP4. The power profile used was obtained from PBF Lead Rod Test #10 with the power peak occurring at 21.9 ms. In addition to the fuel rod power transient, direct moderation heating was modeled by depositing a total

Figure B-1. RELAP-IV Model Used in PBF-RIA 1-4 Analysis

TABLE B-1

RELAP CONTROL VOLUME DATA USED IN RIA 1-4 ANALYSIS

Volume Number	Physical Description	Volume FT**3	Flow Area FT**2	Steady-State Pressure, PSI	Steady-State Temperature, F	Volume Height, FT	Diameter FT
1	Pump	.4106E+00	.7170E-01 Piping	1133.0000	508.00	1.0830	.3022
2	Piping & Line Heaters	.3643E+01	.7170E-01 P	1321.5500	508.00	8.0000	.3022
3	Piping	.1111E+01	.7170E-01	936.4500	508.00	4.5830	.3022
4	Piping	.7170E-01	.7170E-01	934.9470	508.00	.3022	.3022
5	Piping	.7120E+00	.7170E-01	934.8120	508.00	2.2460	.3022
6	Piping	.7110E+00	.7170E-01	934.0520	508.00	2.2460	.3022
7	Piping	.1793E+00	.7170E-01	933.8950	508.00	.3022	.3022
8	Piping	.3610E-01	.5420E-01	933.8920	508.00	.7230	.2627
9	Thermal Swell Accumulator	.1100E+02	.1375E+01	20.0000	-1.00	1.3230	1.3230
10	Acoustic Filter	.1284E+02	.1070E+01	933.8920	508.00	1.1670	1.1670
11	Piping	.5291E+00	.7170E-01	933.8410	508.00	.3022	.3022
12	Downcomer	.3770E+00	.6900E-01	934.6420	508.00	5.4700	.0950
13	Downcomer	.2070E+00	.6900E-01	935.2660	508.00	3.0000	.0950
14	Downcomer	.2000E+00	.6900E-01	935.9120	508.00	2.9000	.0950
15	Lower Plenum	.4900E-01	.3500E-01	936.1590	508.00	.7100	.0883
16	CatchBasket	.7570E-01	.7570E-01	936.1340	508.00	1.1320	.2670
17	Lumped Vol. (T. M & F. S.)	.6800E-02	.5300E-02	935.2610	508.00	1.2800	.0874
18	Fuel Shroud Entrance	.9100E-02	.1960E-01	935.0200	508.00	.4700	.0664
19	Test Fuel Region	.5000E-02	.1500E-01	934.8000	508.00	.3333	.0393
20	Test	.1000E-01	.1500E-01	934.6260	508.00	.6667	.0393
21	Test	.1000E-01	.1500E-01	934.3950	508.00	.6667	.0393
22	Test	.1000E-01	.1500E-01	934.1640	508.00	.6667	.0393
23	Test	.1000E-01	.1500E-01	933.9330	508.00	.6666	.0393
24	Lumped Volume	.1840E-01	.2150E-01	933.6780	508.00	.8583	.0667
25		.1650E-01	.2320E-01	933.4130	508.00	.7100	.1353
26	Bypass Region	.1930E-01	.9260E-01	934.4470	508.00	.2083	.1930
27	Bypass Region	.1110E+00	.9830E-01	934.2280	508.00	1.1300	.2125
28	Bypass Region	.1902E+00	.9260E-01	933.6900	508.00	2.0500	.1859
29	Bypass Region	.1902E+00	.9260E-01	932.9970	508.00	2.0500	.1859
30	Flow Recombination	.7900E-01	.9210E-01	932.5100	508.00	.8500	.1925
31	IPT Internals	.9300E-01	.1139E+00	932.2280	508.00	.8200	.2758
32	IPT Internals	.1898E+00	.1139E+00	931.8070	508.00	1.6700	.2758
33	IPT Internals	.1874E+00	.1139E+00	931.2450	508.00	1.6500	.2758
34	IPT Internals	.9610E-01	.1104E+00	930.8170	508.00	.8700	.3750
35	IPT Internals	.6090E-01	.6210E-01	930.7820	508.00	.9800	.0938

TABLE B-1 (Continued)

Volume Number	Physical Description	Volume FT**3	Flow Area FT**2	Steady-State Pressure, PSI	Steady-State Temperature, F	Volume Height, FT	Diameter FT
36	IPT Internals	.5260E-01	.7270E-01	930.4950	508.00	.7180	.1369
37	Piping	.5291E+00	.7170E-01	928.8370	508.00	.3022	.3022
38	Acoustic Filter	.1284E+02	.1070E+01	928.8360	508.00	1.1670	1.1670
39	Piping	.1793E+00	.7170E-01	928.5360	508.00	.3022	.3022
40	Piping	.3610E-01	.5420E-01	928.5360	508.00	.8850	.2627
41	Thermal Swell Accumulator	.1100E+02	.1375E+01	20.0000	-1.00	1.3230	1.3230
42	Piping	.2200E+00	.7170E-01	928.5350	508.00	.9933	.3022
43	Piping	.9019E+00	.7170E-01	928.9480	508.00	4.7420	.3022
44	Piping	.3169E+00	.7170E-01	928.9470	508.00	1.4170	.3022
45	Piping	.1029E+01	.7170E-01	928.9400	508.00	1.4170	.3022
46	Strainer	.1570E+01	.7857E+00	928.7170	508.00	3.0000	.7500
47	Piping	.3140E+00	.7170E-01	928.7210	508.00	.3022	.3022
48	Piping	.5020E-01	.7170E-01	928.1020	508.00	.3022	.3022
49	Piping	.1076E+00	.7170E-01	928.0860	508.00	.3022	.3022
50	Piping	.1936E+00	.7170E-01	918.1520	508.00	.3022	.3022
51	Piping	.6768E+00	.7170E-01	918.1160	508.00	7.0840	.3022
52	Piping (Bypass)	.7170E-01	.7170E-01	932.7630	508.00	.3022	.3022
53	Piping (Bypass)	.9341E+00	.7170E-01	928.5320	508.00	2.0830	.3022
54	Piping (Bypass)	.1793E+00	.7170E-01	928.1090	508.00	.4167	.3022
55	Piping (Bypass)	.3645E-01	.9760E-02	927.6660	508.00	2.7920	.1115
56	Piping (Hx)	.2336E+00	.9760E-02	920.9880	508.00	8.0830	.1115
57	Heat Exchanger or (Small)	.5520E-01	.2760E-01	920.8900	508.00	.1874	.1874
58	Piping (Hx)	.1484E+00	.9760E-02	919.0490	508.00	10.6900	.1115
59	Piping (Hx)	.1776E+00	.3760E-01	918.1000	508.00	.1408	.2187
60	Piping (Hx)	.3549E+00	.3760E-01	918.1100	508.00	2.8340	.2187
61	Pressurizer Piping	.6010E-01	.3600E-02	920.9700	508.00	10.0400	.0679
62	Pressurizer Piping	.9990E-02	.3600E-02	920.9700	519.00	2.7750	.0679
63	Pressurizer	.4177E+02	.3978E+01	919.0000	-1.00	10.5000	2.2500

LISTING OF RELAP4 INPUT FOR RIA 1-4

[illegible]

TABLE B-II (Continued)

[illegible]

TABLE B-II (Continued)

[illegible]

TABLE B-II (Continued)

[illegible]

TABLE B-II (Continued)

45	* HEAT EXCHANGER DATA CARD	
46	* TEST WATER VOLUMES 19 - 23 *	
47	2101000 -14 2 19 0. -1.0 .005 -1.00+1 .C1 -2.59+1	
48	2101001 .C15 -1.81+2 .0175 -4.85+2 .02 -1.47+3 .C21 -1.08+3	
49	2101002 .C219 -1.12+3 .0225 -1.10+3 .025 -7.60+2 .03 -1.63+2	
50	2101003 .C467 -1.23+1 .05 0.0 50.0 0.0	
51	2102000 -14 2 20 0. -1.0 .005 -3.72+1 .01 -9.60+1	
52	2102001 .C15 -6.72+2 .0175 -1.80+3 .02 -3.53+3 .C21 -4.00+3	
53	2102002 .C219 -4.16+3 .0225 -4.10+3 .025 -2.82+3 .03 -6.04+2	
54	2102003 .C467 -4.57+1 .05 0.0 50.0 0.0	
55	2103000 -14 2 21 0. -1.0 .005 -4.72+1 .C1 -1.22+2	
56	2103001 .C15 -8.54+2 .0175 -2.28+3 .02 -4.47+3 .C21 -5.07+3	
57	2103002 .C219 -5.25+3 .0225 -5.19+3 .025 -3.60+3 .03 -7.66+2	
58	2103003 .C467 -5.82+1 .05 0.0 50.0 0.0	
59	2104000 -14 2 22 0. -1.0 .005 -4.10+1 .C1 -1.05+2	
60	2104001 .C15 -7.38+2 .0175 -1.97+3 .02 -3.88+3 .C21 -4.38+3	
61	2104002 .C219 -4.57+3 .0225 -4.47+3 .025 -3.10+3 .03 -6.63+2	
62	2104003 .C467 -5.03+1 .05 0.0 50.0 0.0	
63	2105000 -14 2 23 0. -1.0 .005 -2.34+1 .C1 -6.04+1	
64	2105001 .C15 -4.22+2 .0175 -1.13+3 .02 -2.22+3 .C21 -2.51+3	
65	2105002 .C219 -2.60+3 .0225 -2.56+3 .025 -1.77+3 .03 -3.78+2	
66	2105003 .C467 -2.87+1 .05 0.0 50.0 0.0	
67	* BYPASS VOLUMES 28 ANC 29 *	
68	2106000 -14 2 28 0. -1.0 .005 -4.37+2 .01 -1.12+3	
69	2106001 .C15 -7.87+3 .0175 -2.11+4 .02 -4.14+4 .C21 -4.69+4	
70	2106002 .C219 -4.36+4 .0225 -4.80+4 .025 -3.32+4 .03 -7.07+3	
71	2106003 .C467 -5.36+2 .05 0.0 50.0 0.0	
72	2107000 -14 2 29 0. -1.0 .005 -5.41+2 .C1 -1.39+3	
73	2107001 .C15 -9.77+3 .0175 -2.62+4 .02 -5.13+4 .C21 -5.81+4	
74	2107002 .C219 -6.02+4 .0225 -5.94+4 .025 -4.10+4 .C3 -8.75+3	
75	2107003 .C467 -7.36+2 .05 0.0 50.0 0.0	
76	* COMMER VOLUME 13 *	
77	2108000 -14 2 13 0. -1.0 .005 -7.31+2 .01 -1.88+3	
78	2108001 .C15 -1.32+4 .0175 -3.54+4 .02 -6.93+4 .C21 -7.83+4	
79	2108002 .C219 -8.13+4 .0225 -8.00+4 .025 -5.53+4 .03 -1.18+4	
80	2108003 .C467 -8.99+2 .05 0.0 50.0 0.0	
81	.	

of 0.585 MJ into the test section coolant and the core axial regions of the test bypass and downcomer water. The RELAP4 input for the analysis is listed in Table B-2.

2. COBRA IV Model

The COBRA IV code was used to obtain the subchannel conditions in the flow shroud. The calculations were done using a modified version of COBRA-IV-I-FIX11. The modifications involved tape input from RELAP and tape output in a FRAP-T4 compatible form and are listed in Table B-3. Table B-4 lists the COBRA-IV input.

The COBRA IV model consisted of a 1/4-core section of the RIA 1-4 9 rod assembly and is shown in Figure B-2. The axial profile was obtained from Test PCM 8-15 with a peak-to-average ratio of 1.34. The RELAP4 generated average surface heat flux were input into COBRA as a function of time.

TABLE B-III
COBRA IV UPDATE LISTING

```

MXYCEGG,P1,T77,STMFZ,SN.
ACCCUNT,3510,4218F1602,EGG.
LABEL,FIX11,R,L=CCBFIX11.
STAGE,FIX11,E,PE,VSN=A42042.
CCPYP,FIX11,CCBRA4.
CCPYP,FIX11,COMDECK.
CCPYP,FIX11,SPECSET.
REWIND,SPECSET.
RETURN,FIX11.
UPDATE,F,P=SPECSET,C=CSPECS,I=NULL.
RETURN,SPECSET.
FTN,I=CSPECS,R=3,B=SPECS.
RETURN,CSPECS.
UPDATE,F,P=COMDECK,S=TAPE10,L=F.
SPECS,,TAPE1.
REWIND,TAPE1.
UPDATE,F,E,P=CCBRA4,N=CLDCCB,C=0.
RETURN,CCBRA4.
UPDATE,F,E,P=OLDCCB,N=NEWCCB,C=0.
RETURN,CLCCB.
REQUEST,NEWPL,*PF.
UPDATE,F,P=NEWCCB,N,W.
CATALCG,NEWPL,CCBRAUD,IC=MXY,RP=999,AC=CCC78.
STAGE,TAPE4,PE,E,VSN=A31227.
STAGE,TAPE51,NT,PE,N,PCST.
STAGE,TAPE52,NT,PE,N,POST.
STAGE,TAPE53,NT,PE,N,PCST.
RFL,16CCCC.
FTN,I=CCMPLE,CPT=2,R=3,B=CCBRA.
REDUCE.
  GO
  *ID RIASPEC
  *D SPEC19.10
    = IMP(10),JMP(10),IQ,IH,IG,IFINP,IFCLTH,K10,
    GO
    LCM
    MP=3CC
    MC=3
    MG=3
    ML=1
    MX=13
    MN=4
    MT=1
    MR=3
    ME=13
    MZ=1
    MK=1

```

...JCL

...Redimensioning
via Specset

TABLE B-III (Continued)

```

MA=1
MW=1
MY=1
MC=5
MI=5
MS=1
CC
*PD SPECS.SPEC27
CC
*ICENT RIA14
*D RESTR1.19
      I8=10
*I CCBRA.59
      XT=2.
      XTIME=2.
      XTIM=2.
*I FIX2.51
      IF(ETIME.LT.XTIM)GO TO 650
      XTIM=XTIME*XT
      CALL DUMPIT(NTSAVE)
      XT=XT+1.
*I DUMPIT.18
      WRITE(I3,740)ETIME
740 FORMAT (//,"COMMON DUMPED TO TAPE8 :FOR ETIME=",F7.3)
*C CHFCCR.23
      QE=AMIN1(.999,AMAX1(0.,(H(I,J)-HF)/HFG))
*I CHFCCR.27
      FLOWSA=FLOW
*C CHFCCR.81
      IF(ITEST.EC.I)CHF1=(CHF1-9.E4)*FLOWSA/2.E5+9.E4
*D CCBRA.3
      +TAPE17,TAPE9,TAPE10,TAPE4,TAPE51,TAPE52,TAPE53,TAPE61,
      +TAPE62,TAPE63)
*I CCBRA.41
      CALL ECHC
*I CCBRA.65
      DO 60 J=1,3
      JIN=60+J
      ICUT=50+J
10 READ(JIN)ATIME
   IF(ECF(JIN))60,20
20 READ(JIN)A1,A2,A3
   WRITE(ICUT)ATIME
   WRITE(ICUT)A1,A2,A3
      DO 50 I=1,13
      REAC(JIN)A1,A2,A3,A4,A5,A6
      WRITE(ICUT)A1,A2,A3,A4,A5,A6

```

...UPd's

TABLE B-III (Continued)

```

50 CONTINUE
   REAC(JIN)A1,A2,A3
   WRITE(IOUT)A1,A2,A3
   GC TC 10
60 CONTINUE
*D CHF1.19
*D CHF1.20
   CHF1=(.84-XX)/.C036
   IF(XX.GE.0.839)CHF1=.C2
   IF(XX.GE.0.83)PRINT*," ***FROM CHF1, XX=",XX
*I CHF1.22
   IF(FAXIAL.LE.2.C) GC TC 140
*C RESULT.44,45
   IF (KT.LT.ANKIPT.ANC.ETIME.LT.TTIME) GC TC 162
*I RESULT.148
162 CC 166 L1=1,3
   LU=L1+50
   WRITE(LU)ETIME
   CC 166 L2=1,NCXP1
   ELE1=X(L2)-.125
   ELE2=X(L2)+.125
   CUTPUT(2)=PREF+P(L1,L2)/144.
   CUTPUT(3)=F(L1,L2)
   CUTPUT(4)=TF
   VIN=F(L1,L2)
   IF(F(L1,L2).LT.FF)
*      CALL CURVE (CUTPUT(4),VIN,TT,FFF,NPRCP,IERRCR,1)
   IF(F(L1,L2).GT.FG)
*      CALL CURVE (CUTPUT(4),VIN,TPCLT,HGPT,NPRCP,IERRCR,1) 33
   IF(IERRCR.GT.1) GC TC 730
   CUTPUT(5)=F(L1,L2)*.C036/A(L1,L2)
   IF(L2.NE.2.AND.L2.NE.13)GO TO 163
   FACT=0.75
   ELEV1=0.
   ELEV2=.125
   IF(L2.EQ.2)GC TC 167
   FACT=0.25
   ELEV1=2.875
   ELEV2=3.C
167 P1=PREF+(P(L1,L2)-FACT*(P(L1,L2)-P(L1,L2-1)))/144.
   FH1=F(L1,L2)-FACT*(F(L1,L2)-F(L1,L2-1))
   T1=CUTPUT(4)-FACT*(CUTPUT(4)-TCLD)
   F1=(F(L1,L2)-FACT*(F(L1,L2)-F(L1,L2-1)))*.C036/A(L1,L2)
   WRITE(LU)ELEV1,ELEV2,P1,FH1,T1,F1
163 TCLC=CUTPUT(4)
   IF(L2.EQ.1.CR.L2.EQ.NCXP1) GO TO 164
   WRITE(LU) ELE1,ELE2,(CUTPUT(LLP),LLP=2,5)
   GC TC 166

```

TABLE B-III (Continued)

```

164 WRITE(LU) (CUTPLT(LLP),LLP=2,4)
166 CONTINUE
    IF (KT.LT.NSKIPT.AND.ETIME.LT.TTIME) GO TO 520
    KT=C
*I SETUP.4
    DIMENSION ACATA(5000)
    EQUIVALENCE (IA,ACATA(1))
*D SETUP.42
    100 READ (I2,1700) KASE,J1,TMIN,TMAX,TEXT
*C SETUP.994
    1700 FORMAT(2I5,2F5.0,15A4)
*D SETUP.956
*C SETUP.957
    WRITE(I3,2828) (YP(I),FP(I),FG(I),FH(I),FHX(I),FG(I),I=1,NP)
*D SETUP.963
*C SETUP.964
*D SETUP.965
    1580 CONTINUE
    1590 CONTINUE
*I FIX6.32
    2828 FORMAT(1X,"FORCING FUNCTIONS (NORMALIZED) OBTAINED FROM RELAP4",
    1" CALCULATION",/1X,"--TIME(S)-----PRESSURE-----INLET FLOW-----",
    2"INLET ENTH-----EXIT ENTH-----SURFACE HEAT FLUX-----",/
    3(1X,F10.4,5(3X,F10.4)))
*C SETUP.695,701
    1070 I=0
        NVCL=63
        NJUN=66
        NSLB=5
        NP23=23+41+C*NVCL
        NF18=18+41+24*NVCL
        NF18=18+41+24*NVCL+1*NJUN
        NF23=NF18-18+23
        NC23=23+41+4*NVCL
        NPAR=41+24*NVCL+16*NJUN+5*NSLB
        NS1=1+NPAR
        NS2=2+NPAR
        NS3=3+NPAR
        NS4=4+NPAR
        NS5=5+NPAR
        IPLCT=4HPLCT
    30  BUFFER IN (4,1) (ACATA(1),ACATA(5000))
        U4=UNIT(4)
        IF(U4.GE.0.) GO TO 180
    40  IF(IA.NE.IPLCT) GC TO 30
        IF(TMIN.GT.ACATA(39))GC TO 30

```

TABLE B-III (Continued)

```

IF(TMAX.LT.ACATA(39))GO TO 180
I=I+1
NP=I
NF=I
NG=I
NQ=I
NFX=I
YP(I)=ACATA(39)
YF(I)=ACATA(39)
YG(I)=ACATA(39)
YQ(I)=ACATA(39)
YFX(I)=ACATA(39)
FP(I)=(ACATA(NP23)-(ACATA(NC23)*.3333/144.))/933.8
FH(I)=ACATA(NF18)/497.4
FHX(I)=ACATA(NH23)/497.4
FG(I)=(ACATA(NF18)*360./0.015)/310320.
FQ(I)=ACATA(NS1)*.3333/3.0+.6666/3.0*
* (ACATA(NS2)+ACATA(NS3)+ACATA(NS4)+ACATA(NS5))
FQ(I)=FQ(I)/267.8
GO TO 30
*ACDFILE INPUT,CUMFIT
*OK ECHC
SUBROUTINE ECHC
C*****
C*****
C CCDC BY R.C. GLENZLER - OCTOBER 1976
C*****
C DIMENSION DATA (10)
C DATA NIT,NCT/5,6/
C*****
C*****
10 FCRMAT (1H1, 9X,"C" 1 2 3 4
15 6 7 8"/,10X,"12345678901234567890123456
2789012345678901234567890123456789012345678901234567890",/)
11 FCRMAT (1H ,14,5X,10A8)
12 FCRMAT (1H0, 9X,"C" 1 2 3 4
15 6 7 8"/,10X,"123456789012345678901234567890123456
2789012345678901234567890123456789012345678901234567890",/,1H1, 9X,
3"0 1 2 3 4 5 6
4 7 8"/,10X,"1234567890123456789012345678901234567890123456789012
5345678901234567890123456789012345678901234567890",/)
13 FCRMAT (1H0, 9X,"C" 1 2 3 4
15 6 7 8"/,10X,"123456789012345678901234567890123456
2789012345678901234567890123456789012345678901234567890")
20 FCRMAT (10A8)

```

TABLE B-III (Continued)

```

C*****a*****
C*****
      ILINE = C
      LINE = 0
      WRITE (NCT,10)
100 REAC (NIT,20) DATA
      IF (EOF(NIT)) 120,115
115 ILINE = ILINE + 1
      WRITE (NCT,11) ILINE,DATA
      LINE = LINE + 1
      IF (LINE.EQ.55) GC TC 110
      GC TC 100
110 WRITE (NCT,12)
      LINE = 0
      GC TC 100
120 WRITE (NCT,13)
      DO 150 I=1,ILINE
      BACKSPACE NIT
150 CONTINUE
      RETURN
      END
00
*AF TAPE1,YANK$$$

```

TABLE B-IV
LISTING OF COBRA IV INPUT

```

1      TRIAL-4 TEST: MOCEL 1/8 BUNDLE; 3RCDS, 3 CHANNELS.
1      36
.0886 32. .016022 3302.4 .0003 1075.5 4.22 .329 .0
0.50 79.59 0.016071 641.5 47.623 1096.3 2.063 0.395 0.
.9492 100. .01613 350.4 67.999 1105.1 1.64 .363 .0
3.718 150. .016343 97.07 117.95 1126.1 1.06 .381 .0
12.5 204. .016663 31.18 172.06 1147.5 .709 .392 .0
25. 240. .016927 16.301 208.52 1160.6 .576 .396 .0
50. 281. .017274 8.514 250.2 1174.1 .482 .397 .0
100. 327.8 0.01774 4.4310 298.5 1187.2 0.405 0.394 0.
200. 381.8 0.01839 2.28728 355.5 1198.3 0.336 0.386 0.
300. 417.4 0.01889 1.54274 394.0 1202.9 0.302 0.378 0.
340. 429.0 0.01908 1.36405 406.8 1203.8 0.294 0.375 0.
400. 444.6 0.01934 1.16095 424.2 1204.6 0.284 0.371 0.
440. 454.0 0.01950 1.05535 434.8 1204.8 0.278 0.367 0.
500. 467.0 0.01975 0.92762 449.5 1204.7 0.270 0.362 0.
560. 478.8 0.01998 0.82637 463.1 1204.2 0.263 0.357 0.
600. 486.2 0.02013 0.76975 471.7 1203.7 0.258 0.355 0.
640.493.2 0.02029 0.71995 479.9 1203.0 0.260 0.3552
720.506.2 0.02058 0.63639 495.4 1201.4 0.252 0.3462
800.518.2 0.02087 0.56896 509.8 1199.4 0.245 0.3405
840.523.9 0.02101 0.53988 516.7 1198.2 0.242 0.3377
920.534.6 0.02130 0.48901 530.0 1195.7 0.237 0.3325
1000.544.6 0.02159 0.44596 542.4 1192.9 0.232 0.3275
1120.558.5 0.02203 0.39244 560.5 1188.2 0.225 0.3198
1280.575.4 0.02262 0.33603 582.9 1181.2 0.217 0.3086
1400.587.1 0.02307 0.30178 598.8 1175.3 0.216 0.3055
1500.596.2 0.02346 0.27719 611.7 1170.1 0.215 0.3005
1600.604.9 0.02387 0.25545 624.2 1164.5 0.213 0.2960
1700.613.1 0.02428 0.23607 636.5 1158.6 0.212 0.2910
1800.621.0 0.02472 0.21861 648.5 1152.3 0.210 0.2870
1900.628.6 0.02517 0.20278 660.4 1145.6 0.209 0.2830
2000.635.8 0.02565 0.18831 672.1 1138.3 0.209 0.2787
2100.642.8 0.02615 0.17501 683.8 1130.5 0.208 0.2745
2200.649.5 0.02669 0.16272 695.5 1122.2 0.207 0.2707
2300.655.9 0.02727 0.15133 707.2 1113.2 0.207 0.2665
2400.662.1 0.02790 0.14076 719.0 1103.7 0.206 0.2630
2600.673.9 .02938 .12110 744.5 1082.0 .205 .2550
2      1      1      1
.2 -.2 0. .11 -.2 0. .25 -.2 0.
3      6
0.0 .29 .222 1.05 .444 1.34 .667 1.16 .778 .889 1.0 .32
4      3      3
1 1.0205.4068.1543 2 .102
1 2.1074 1.18.6173 3 .170
1 3.0978.9260.9260

```


TABLE B-IV (Continued)

	8	3	3	0	0	2	0	1	0	0	0
0	1	.393	1.063	1.1339	2.2519	3.1143					
0	2	.393	.968	2.2614	3.2386						
0	3	.393	.875	3 .125							
	9	1	15	3	5						
36.	1.1	.01	.001	.001	.5 .395	0.	0.	0.	.25	1.6	1.
	12	5400	20	20	5						
	.12	.05	.001	.0001		.2	.8				
300	50	0	0								
0.	.05	.025	.005	.045	.05	30.0	.05				
10	0	0	0								
.05											
11	0	0	250	250	250	250		250			
	933.8	.310320		497.4	.0002678			497.4			
12	3	0	0	3	0	2					

00
17

	Coolant channel (circled)		
	1	2	3
Flow area (in ²)	0.0205	0.1074	0.0978
Wetted perimeter (in)	0.4068	1.18	0.9260
Heated perimeter (in)	0.1543	0.6173	0.9260

Figure B-2. COBRA-IV geometric model of Test RIA 1-4.

APPENDIX C

The input variables for the FRAP-T5 analysis of the Test RIA 1-4 rods are given below. Table C-1 lists the radial power profiles for the center, side, and corner rods. Table C-2 presents the axial power profile used for all 3 rods. Table C-3 presents the fuel rod data for the RIA 1-4 rods. Table C-4 lists the various input parameters, correlations, and modeling options used in the FRAP-T5 calculations.

TABLE C-1

RADIAL POWER PROFILES FOR TEST RIA 1-4 RODS

Outer Radius of Mesh (mm)	Normalized Power		
	Center Rod	Side Rod	Corner Rod
1.9199	0.8513	0.8559	0.8256
2.7152	0.9227	0.9251	0.9085
3.1838	0.9874	0.9879	0.9845
3.5918	1.0449	1.0436	1.0522
3.8398	1.0940	1.0911	1.1106
4.0727	1.1343	1.1300	1.1585
4.1843	1.1650	1.1597	1.1952
4.2930	1.1858	1.1797	1.2200

TABLE C-2

AXIAL POWER PROFILE FOR TEST RIA 1-4 RODS

Axial Node Mid-Plane Elevation (m)	Normalized Power
0.0254	0.3971
0.0762	0.5074
0.1270	0.6250
0.1778	0.7574
0.2286	0.8309
0.2794	0.9118
0.3302	0.9559
0.3810	0.9853
0.4318	1.0000
0.4826	0.9706
0.5334	0.9412
0.5842	0.8824
0.6350	0.8088
0.6858	0.7353
0.7366	0.6397
0.7874	0.5294
0.8382	0.4265
0.8890	0.3235

TABLE C-3
FUEL ROD DATA FOR TEST RIA 1-4

<u>Fuel Pellet</u>	Center Rod ^a	Side Rod	Corner Rod ^a
Diameter (m)		0.00859	
Height (m)		0.0152	
Stack Length (m)		0.9144	
Density (Kg/m ²)		10304.	
Shoulder Radius (m)		0.003747	
Dish Depth (m)		0.000343	
Dish Volume (m ³)		1.5163E-08	
RMS Roughness (m)		2.12	
Burnup (MWd/t)	5450	4790	5790
<u>Cladding</u>			
Outer Diameter (m)	0.009968	0.009972	0.009982
Inner Diameter (m)		0.008750	
RMS Roughness (m)		1.14	
<u>Plenum</u>			
Plenum Volume (m ³)		2.751E-06	
Gas Pressure (N/m ²)		1.0342E+05	
Gas Composition (%)		76.6% He	
		10.1% Ar	
		1.11% Kr	
		6.13% Xe	
		4.7 N ₂	
		0.72 O ₂	
Spring Length (m)		0.0457	
Spring Coil Outside Diameter (m)		0.008	
Spring Wire Diameter (m)		0.00102	
Number of Spring Coils		17	
<u>Flow Shroud</u>			
Coolant Flow Area (m ²)	1.2645E-04	1.49458E-04	1.56067E-04
Hydraulic Diameter (m)	0.016152	0.013102	0.0104009
Equivalent Heated Diameter (m)	0.016152	0.019083	0.019846

a. Unless otherwise given, the values for the center and corner rods are the same as those for side rod

TABLE C-4

OTHER INPUT PARAMETERS, CORRELATIONS, AND MODELING OPTIONS
USED IN THE FRAP-T5 CALCULATIONS

Number of fuel rods = 1

Number of flow channels = 1

Number of axial nodes = 12

Number of radial nodes = 12

Plenum gas temperature model used.

Gas flow model turned on.

Cathcart cladding oxidation model specified.

Fuel deformation model type = 3

(Free thermal expansion fuel deformation model specified.)

Modified Ross and Stoute model for gap conductance used.

Critical heat flux to be multiplied by a factor of 1.0

Probability threshold for fuel rod failure = 0.5

Critical heat flux option = No cold wall or axial power factors

CHF correlation = Barnett/W-3

Post-CHF heat transfer code = Groeneveld Eqn. 5.9