THE BTF-104 EXPERIMENT: AN IN-REACTOR TEST OF FUEL BEHAVIOUR, AND FISSION-PRODUCT RELEASE AND TRANSPORT UNDER LOCA/LOECC CONDITIONS

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ABSTRACT

The primary objective of the BTF-104 experiment was to determine the timing, amount and transport characteristics of fission products released from a previously irradiated CANDU-sized fuel element subjected to a high-temperature transient representative of a Loss-of-Coolant Accident (LOCA) with additional Loss-of-Emergency-Core-Cooling (LOECC). This paper summarizes the results of the BTF-104 experiment.

INTRODUCTION

The objective of the Blowdown Test Facility (BTF) experimental programme is to obtain fission-product release and transport, and fuel behaviour data from in-reactor, all-effects tests under representative accident conditions for benchmarking computer codes used in CANDU® safety and licensing analyses. The BTF is located in the National Research Universal (NRU) reactor at the Chalk River Laboratories of AECL.[1,2] Test assemblies in the BTF are cooled with pressurized water or saturated steam provided by the U-1 loop. An accident sequence test in the BTF is initiated by isolating the BTF in-reactor test section from the U-1 loop, and voiding the coolant in the test section through an instrumented piping system (blowdown line) into a sealed tank (blowdown tank) in the basement of the NRU reactor building. Steam, inert gas and cold water may be used for post-blowdown cooling in the BTF. The blowdown line is instrumented to measure coolant thermalhydraulic parameters and fission-product gamma emissions.

The current BTF research programme consists of 3 experiments, BTF-107, BTF-104 and BTF-105. In the BTF-107 experiment, performed in 1990, a three-element cluster of CANDU-sized fuel elements was subjected to severely degraded cooling conditions resulting in a high-temperature transient.[3,4] A flow blockage developed during the test and the high-temperature

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transient was terminated with a cold water quench. The BTF-104 experiment was conducted in 1993 September under conditions representative of a Loss-Of-Coolant Accident (LOCA) scenario with additional Loss-Of-Emergency-Core-Cooling (LOECC). The target volume-average fuel temperature for the BTF-104 experiment was 1550°C. The BTF-105 experiment is intended to be similar to the BTF-104 experiment but with a higher target fuel temperature. Preliminary results of the BTF-104 experiment were presented previously.[5] This paper summarizes the current understanding of the BTF-104 experiment based on the results of the post-irradiation examination (PIE) and post-test code analyses performed to date.

TEST SEQUENCE

The fuel element used in the BTF-104 experiment was previously irradiated to a burnup of about 132 MWh/kg U at an average linear power of about 48 kW/m. After the pre-irradiation period and 8 years of storage, the fuel element was instrumented and installed inside an instrumented test assembly for re-irradiation in the BTF. The fuel element was surrounded with a thermal shroud to minimize radial heat transfer from the fuel during the high-temperature transient (see Figure 1). Thermocouples were installed at various locations on the test assembly to measure coolant and material temperatures.

The BTF-104 fuel assembly was re-irradiated for 15 days prior to the transient at a fuel element linear power of about 60 kW/m under pressurized water cooling conditions (270°C coolant temperature and 10 MPa pressure) to reestablish a representative inventory of fission products with short half lives ($t_{1/2} \leq 8$ days). Four hours before the transient, the coolant was changed to saturated steam (310°C temperature and 10 MPa pressure) to simplify the thermalhydraulics of the blowdown. The reactor power was maintained at about 75% of full power during this period to limit the maximum fuel and sheath temperatures. The total fuel burnup was about 152 MWh/kg U prior to the test transient.

High-Temperature Transient

The reactor power was reduced to 2% of full power to start the transient. The blowdown (depressurization) sequence was initiated about 30 minutes after the power reduction. When the coolant pressure reached 0.5 MPa, low flows of steam (6 g/s) and inert gas (0.1 L/s of 1% H_2 in He) were established to maintain a limited amount of cooling of the fuel element during the high-temperature transient. The reactor power was increased and held at plateaus of 5, 6, 8, 11 and 12% of full power. Fuel element failure was detected by the gamma monitors on the blowdown line during the plateau at 6% of full reactor power. The maximum volume-averaged fuel temperature during the transient was about 1900°C. The transient was terminated with a reactor trip from about 12% of full reactor power about 33 minutes after the blowdown, because of high thermal shroud temperatures. After the reactor trip and a 20 s period of high-rate (12 L/s) inert gas injection, the steam purge flow was restored and increased to 20 g/s to cool the fuel more rapidly. After about 10 minutes of steam cooling, the steam flow was terminated. A slow flow of inert gas was maintained for 48 hours following the transient.

Measurements of coolant temperatures, assembly temperatures, pressure tube temperatures, coolant pressures, neutron flux and blowdown line gamma-radiation fields were obtained during the test. The thermocouples attached to the fuel sheath and the turbine flowmeters upstream of the fuel element did not provide reliable data during the high-temperature transient.

Eleven gamma-spectrometers collected data on-line during the BTF-104 transient. Gamma emissions from Xe, Kr, I, Cs, Te and Rb isotopes were detected. The noble gases (Kr-85m, Kr-87, Kr-88, Xe-133 and Xe-135) were detected immediately following element failure and during subsequent power ramps in the early portion of the test. Later in the test, some of the noble gas isotopes were difficult to detect because of the high background from deposited iodine. A small amount of iodine (I-131, I-132, I-133, I-134 and I-135) was detected at element failure. The iodine signals increased slowly with time until about 200 s before the reactor trip when there was a rapid increase in temperature and iodine release. The timing of the Rb-88 and Cs-138 releases were similar to those of the iodine isotopes.

Fission-product measurements were obtained with the gamma-spectrometry systems on the BTF for several days after the transient. The BTF-104 fuel assembly was removed from the reactor core and potted in epoxy about 3 days after the transient. After the fuel assembly was removed, the BTF blowdown line was flushed with cold water. Fission-product deposition data were collected both before and after the water flush. About 90% of the Cs, I and Te released from the fuel element was retained in the BTF piping system prior to the water flush. This observation indicates that there is potential for crediting fission-product retention in the primary circuit piping for reducing the releases into containment under similar conditions. Integral fission-product releases based on post-test measurements made on the BTF piping and blowdown tank are presented in Table 1. These integral releases agree well with reatined fission-product measurements made during post-test γ-scans of the fuel. Isotopes with longer half lives had higher fractional releases, indicating that a significant portion of the releases during the BTF-104 transient were from the grain boundary and gap inventories produced during the soak irradiation.

POST-IRRADIATION EXAMINATION

The epoxied BTF-104 fuel assembly was sectioned and examined by gamma tomography.[6] The tomographic examination showed the pellet stack to be mainly intact except for a 5 mm gap about 120 mm down from the top of the fuel element where a fuel pellet had fractured and was displaced laterally.

Metallographic examination showed the fuel sheath to be brittle (see Figure 2). Except for the top and middle bearing pad zones where the sheath was almost intact (see Figure 3), much of the sheath had fractured and fallen away from the element. No previously molten material was found. Extensive columnar grain growth had occurred and the fuel centreline temperature may have approached the melting point of UO_2. Post-test analyses (see below) indicate that these high fuel centreline temperatures likely occurred during the pressurized water and steam soak irradiation periods prior to the transient. Metallography suggests that the fuel sheath temperatures varied between 900 and 1700°C axially along the element. There was no evidence of sheath oxidation along fracture surfaces, indicating that the sheath likely shattered after the high-

temperature portion of the transient.

POST-TEST SIMULATIONS

Post-test computer-code simulations were done to aid in understanding the test results. Calculations of fuel power, thermalhydraulics, fuel behaviour, and fission product release, transport and deposition were performed.

Calorimetry coupled with on-line neutron flux measurements on the outside of the BTF pressure tube and reactor physics simulations with the WIMS and MCNP computer codes were used to determine the fuel power during the transient. The fuel power was calculated to be about 40% higher than previous estimates[5] mainly because of the effect of voiding the BTF test section on the fuel power (see Figure 4). The fission-product inventory of the fuel element was calculated with the ORIGEN code using this power history. The measured releases in Table 1 are expressed as fractions of the inventory calculated with the ORIGEN code.

The CATHENA computer code was used to perform post-test thermalhydraulics simulations of the BTF-104 experiment. The fuel powers derived from the reactor physics analysis, and the measured inlet coolant conditions, blowdown line wall temperatures and blowdown tank pressure were used in the CATHENA calculations. The calculated fuel and sheath temperatures, and fuel-to-sheath heat transfer were refined by performing iterative calculations with the CATHENA and ELOCA codes. Good agreement was obtained between measured and calculated thermal shroud liner temperatures during most of the transient period (see Figure 5). Figure 6 presents the temperature readings from the mid-plane fuel sheath thermocouple along with the calculated mid-plane fuel sheath, coolant and thermal shroud inner liner temperatures. The measured temperature is closest to the calculated thermal shroud inner liner temperature, indicating that this particular thermocouple was not in contact with the fuel sheath during the transient.

Fuel behaviour and fission-product release simulations were performed with the ELOCA code using the fuel powers from the reactor physics analysis and the fuel sheath temperatures from the CATHENA calculations. The area-average fuel temperatures calculated in a detailed twenty-segment ELOCA thermo-mechanical simulation are shown in Figure 7. This calculation indicates that a maximum volume-averaged fuel temperature of 1800-1880°C was attained at the end of the high-temperature transient. This calculation also indicates that a significant axial variation in the fuel temperatures developed during the transient due to reduced sheath strain in the braze heat affected zones near the bearing pads and higher strains towards the downstream end of the element. The results of a five-segment fission-product release simulation with the ELOCA.Mk5S code were in good agreement with the measured integral releases expressed as a fraction of the inventory calculated with the ORIGEN code (see Table 1).

Fission-product transport calculations were performed with the VICTORIA computer code to assist in refining the fission-product release rate estimates based on the downstream γ-spectrometry measurements. The calculated Cs, I and Te deposition patterns were similar in form to the experimental data, but the calculated deposition values were about a factor of 20 higher than the measurements. To improve agreement between the measured and calculated

fission-product deposition, it was necessary to assume that Cs transport characteristics were being governed by chemical species less volatile than CsI and CsOH, perhaps a cesium uranate.

CONCLUSIONS

The timing, amount and transport characteristics of fission products released from a previously irradiated CANDU-sized fuel element subjected to a high-temperature transient representative of a LOCA with LOECC were measured in the BTF-104 experiment. The volume-averaged fuel temperature was somewhat higher than the 1550°C target at the end of the transient, likely between 1800 and 1880°C. Failure of some of the key instruments increased the uncertainty in the fuel temperatures during the transient. Fission-product deposition data from this test indicate the potential for crediting fission-product retention in the primary circuit piping in reducing the releases into containment under similar conditions. Following the transient, the fuel sheath was heavily oxidized and brittle at the downstream end. No previously-molten material was found. The performance of the facility and test assembly during the BTF-104 experiment indicates that tests under more severe conditions could be performed successfully.

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TABLE 1

FISSION-PRODUCT RELEASE
DURING THE BTF-104 EXPERIMENT

Isotope	Measured Release* (%)	Calculated Release** (%)
Kr-85m	10 ± 4	13.7
Kr-85	47 ± 6	-
Kr-87	8 ± 2	7.7
Kr-88	7 ± 2	11.4
Xe-133m	20 ± 2	31.7
Xe-133	23 ± 6	30.7
Xe-135	14 ± 2	17.0
I-131	33 ± 5	39.6
I-133	20 ± 5	23.2
Te-132	2.5 ± 0.7	-
Cs-134	72 ± 6	-
Cs-137	59 ± 5	48.6
Ba-140	1.4 ± 0.2	-

* - Measured fission-product release expressed relative to the element inventory calculated using WIMS/ORIGEN

** - Calculated fission-product releases obtained from a five-segment ELOCA.Mk5S simulation

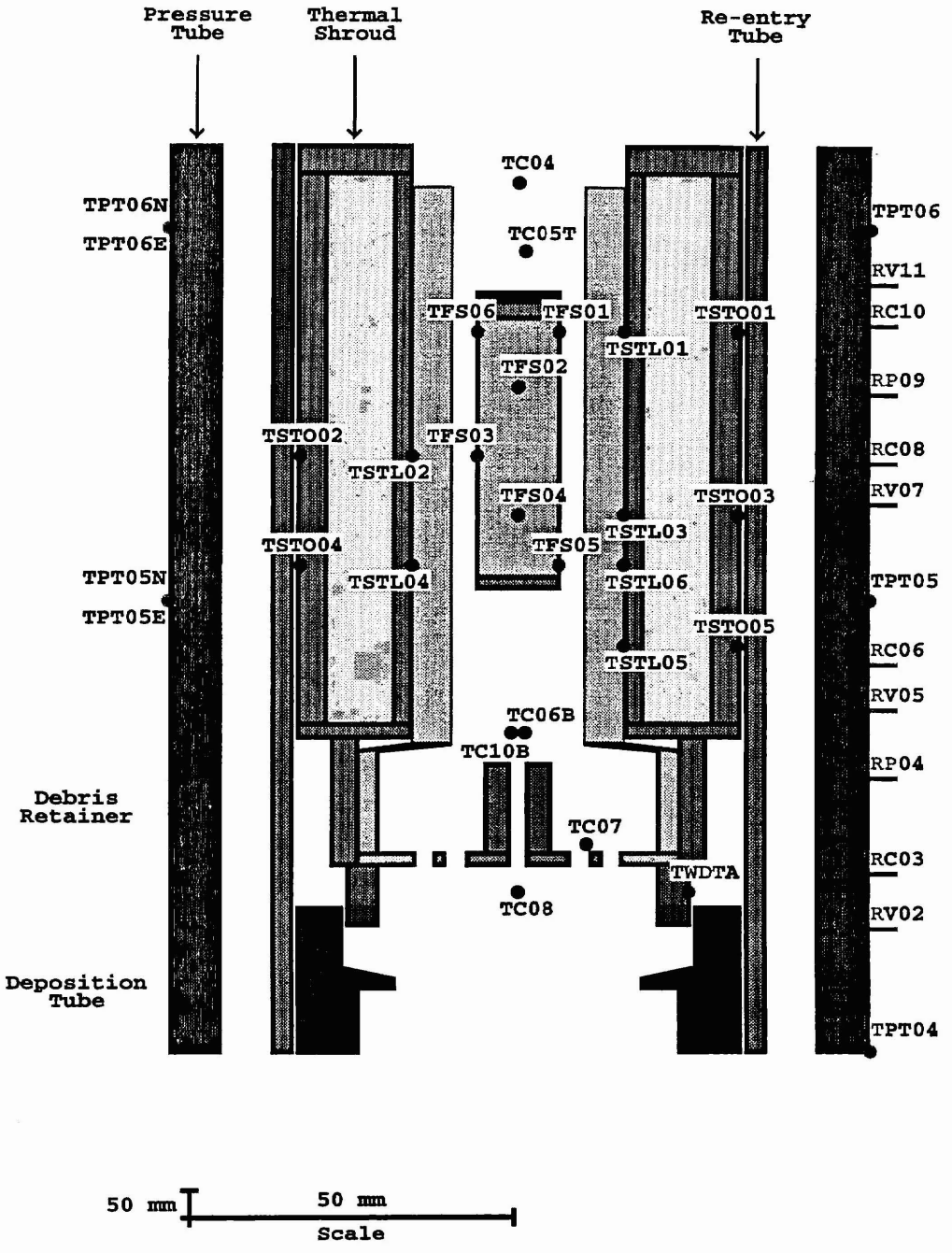

FIGURE 1: LONGITUDINAL CROSS SECTION OF THE BTF-104 FUEL ASSEMBLY

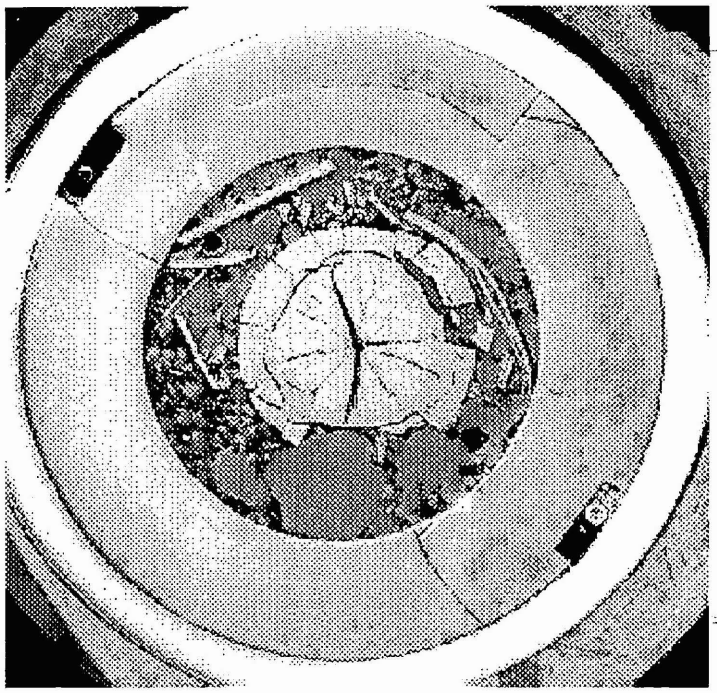

FIGURE 2: POLISHED SECTION OF THE BTF-104 FUEL LOG, ELEVATION 36 mm

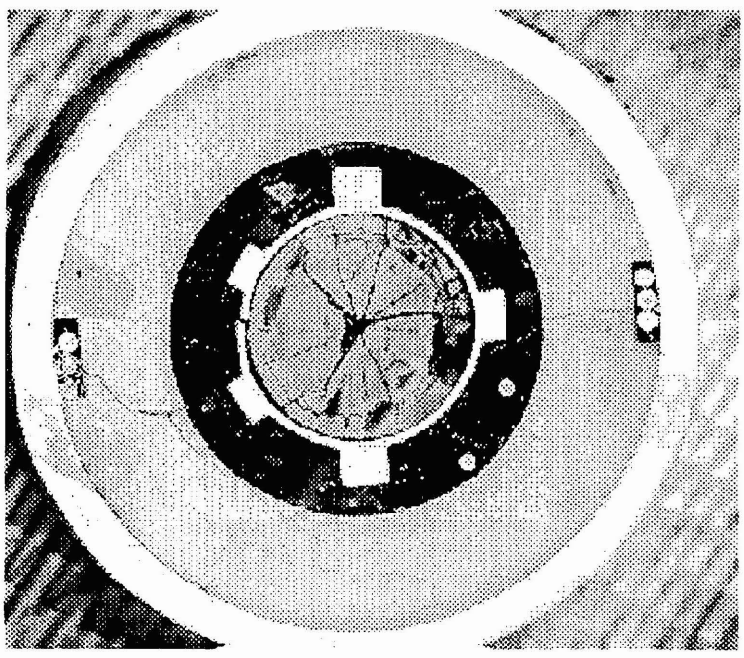

FIGURE 3: POLISHED SECTION OF THE BTF-104 FUEL LOG, ELEVATION 252 mm

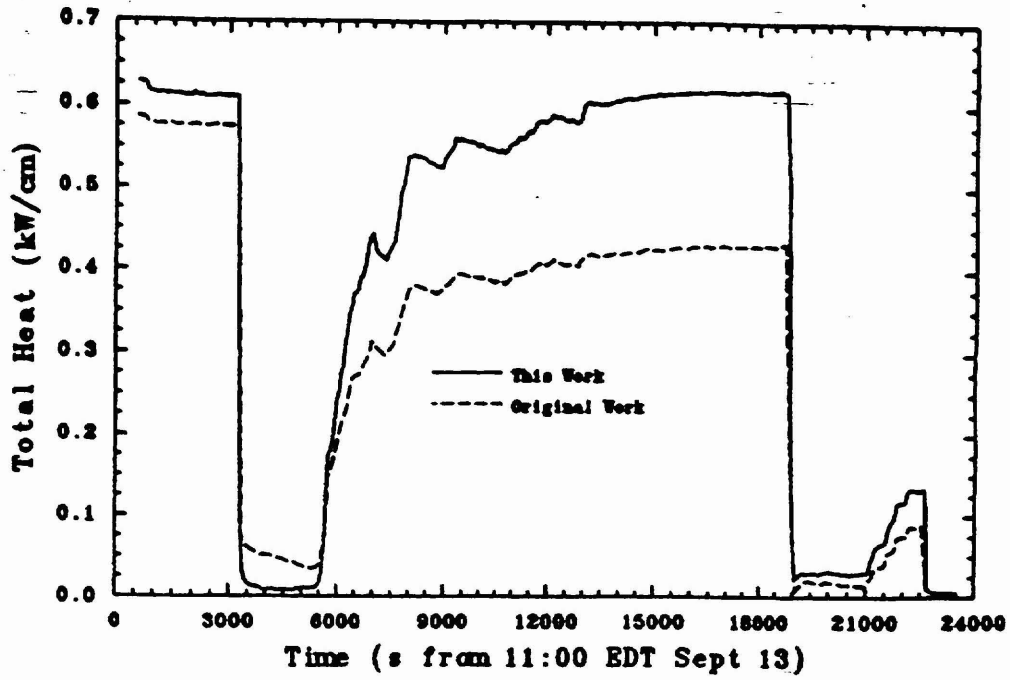

FIGURE 4: FUEL POWER DURING STEAM SOAK AND TRANSIENT PERIODS

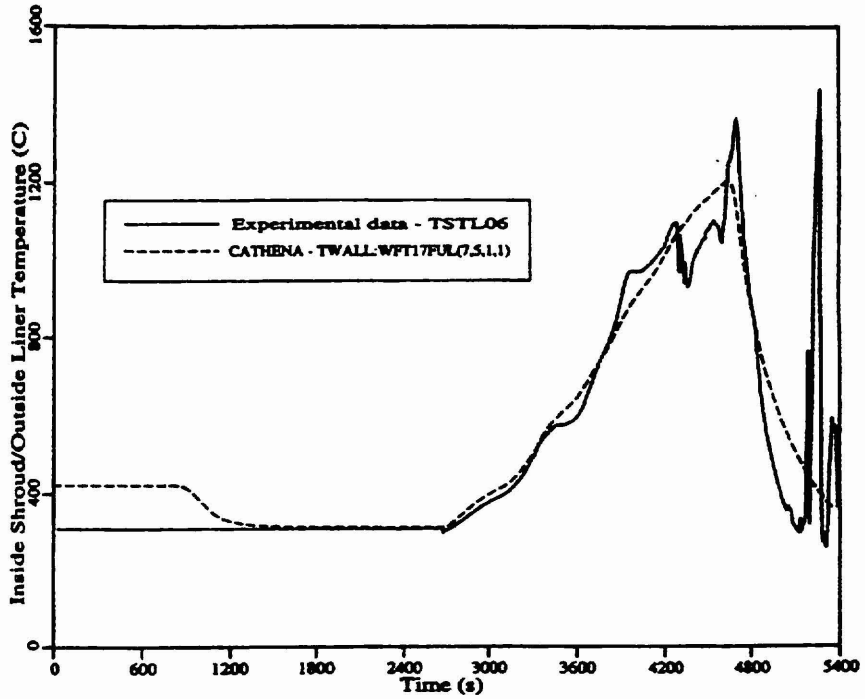

FIGURE 5: MEASURED AND CALCULATED THERMAL SHROUD LINER TEMPERATURES AT THE BOTTOM OF THE FUEL ELEMENT

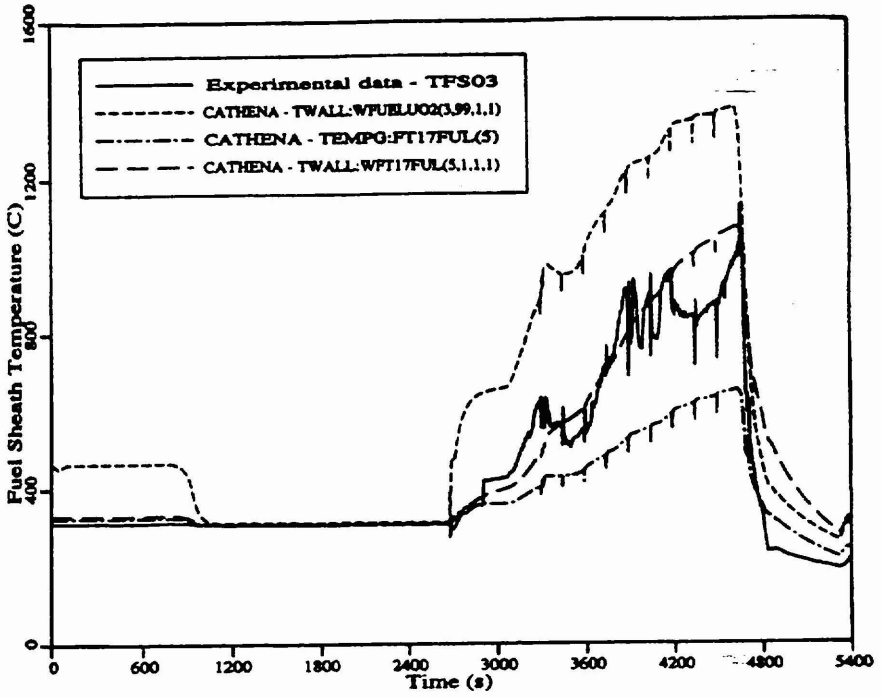

FIGURE 6: MEASURED AND CALCULATED FUEL ELEMENT MID-PLANE SHEATH, LINER AND COOLANT TEMPERATURES

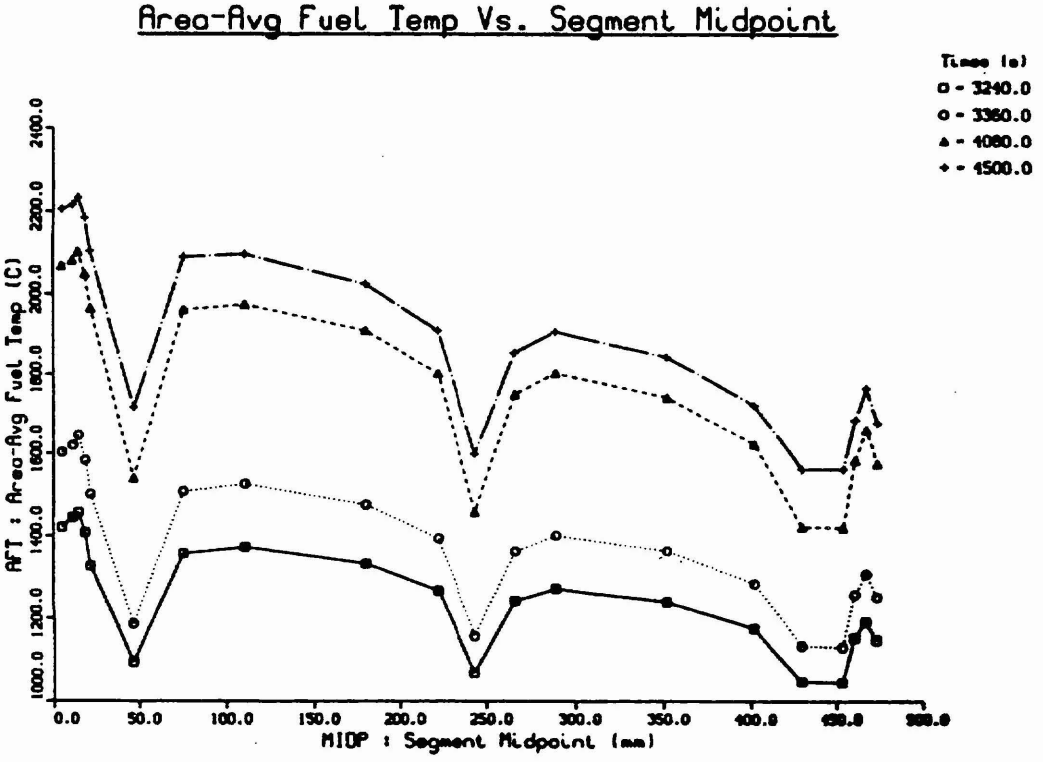

FIGURE 7: CALCULATED AREA-AVERAGE FUEL TEMPERATURES AS A FUNCTION OF SEGMENT MID-POINT DURING THE BTF-104 TRANSIENT