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PNL-6540
UC-520

FULL-LENGTH HIGH-TEMPERATURE
SEVERE FUEL DAMAGE TEST #5

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April 1988 - Completion Date
September 1993 - Publication Date

Prepared for
U. S. Nuclear Regulatory Commission
Under U.S. Department of Energy
Contract DE-AC06-76RLO 1830

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ABSTRACT

This report describes and presents data from a severe fuel damage test that was conducted in the National Research Universal (NRU) reactor at Chalk River Nuclear Laboratories (CRNL), Ontario, Canada. The test, designated FLHT-5, was the fourth in a series of full-length high-temperature (FLHT) tests on light-water reactor fuel. The tests were designed and performed by staff from the U.S. Department of Energy's Pacific Northwest Laboratory (PNL), operated by Battelle Memorial Institute. The test operation and test results are described in this report.

The fuel bundle in the FLHT-5 experiment included 10 unirradiated full-length pressurized-water reactor (PWR) rods, 1 irradiated PWR rod and 1 dummy gamma thermometer. The fuel rods were subjected to a very low coolant flow while operating at low fission power. This caused coolant boilaway, rod dryout and overheating to temperatures above 2600 K, severe fuel rod damage, hydrogen generation, and fission product release. The test assembly and its effluent path were extensively instrumented to record temperatures, pressures, flow rates, hydrogen evolution, and fission product release during the boilaway/heatup transient.

Post-test gamma scanning of the upper plenum indicated significant iodine and cesium release and deposition. Both stack gas activity and on-line gamma spectrometer data indicated significant (~50%) release of noble fission gases. Post-test visual examination of one side of the fuel bundle revealed no massive relocation and flow blockage; however, rundown of molten cladding was evident.

SUMMARY

This report describes and presents data from the full-length high-temperature, number 5, (FLHT-5) severe fuel damage test conducted in the National Research Universal (NRU) Reactor at Chalk River Nuclear Laboratories (CRNL), Ontario, Canada. The objective of the test was to characterize fuel bundle behavior during a severe accident, in particular, fuel temperature history and associated hydrogen generation, melt relocation, and fission product release and transport. The test is described in terms of test train and associated hardware design and instrumentation, test operation, and test results. Key points are summarized below.

TEST TRAIN AND ASSOCIATED HARDWARE DESIGN AND INSTRUMENTATION

The test train included an insulated fuel rod assembly and an insulated 4-m-long plenum above the fuel rods leading to a closure plug. The test train was installed in a pressure tube (within the NRU reactor) that was part of a pressurized-water loop facility that provided bypass water cooling and fuel bundle coolant (makeup) flow.

The fuel bundle consisted of 12 rod positions. It contained 10 unirradiated pressurized-water reactor (PWR)-type, ~4-m-long fuel rods, and one irradiated PWR rod. A stainless steel rod inside a Zircaloy guide tube occupied the twelfth rod position (simulating the gamma thermometer used in FLHT-4).

The fuel bundle effluent passed up through a pressure tube closure plug and across the top of the reactor before descending below the reactor to a catch tank. Shielding and confinement for the top-of-reactor segment of the effluent line were provided by the steam closure cave (SCC) and the effluent control module (ECM), which provided flow and pressure control, housed instruments for effluent sampling, and monitored hydrogen flow and fission product release.

Data from the test train and associated hardware were collected by a computerized data acquisition and control (DACs) system. This system provided test train and effluent system control, in conjunction with the loop control system (LCS) and the reactor control systems (RCS).

The test train instrumentation consisted of approximately 250 individual sensors, including thermocouples (TCs) at numerous axial levels (10- to 20-cm intervals) in the fuel rods, in the shroud, and in the plenum. Pressure transducers (PTs) were located on the unirradiated fuel rods and the shroud cavities. In addition, four devices within the ECM measured hydrogen flow, and numerous instruments recorded temperature and pressure conditions for the effluent and the injection flows.

TEST OPERATION

The FLHT-5 test was planned to be different from the previous FLHT-4 test in ways that would promote Zircaloy relocation, debris formation, and possible flow blockage. Throughout the boilaway transient, the bundle was to be

operated at 30-kw assembly nuclear power versus the 23 kw used in FLHT-4. The higher power would expose more Zircaloy to steam and cause a more rapid boildown and faster heatup after dryout. The result would be that the cladding temperature would increase above the melting temperature (2100 to 2400 K) with relatively less premelting oxidation. This was predicted to enhance rundown and relocation of molten Zircaloy and liquified uranium dioxide, possibly leading to complete flow blockage and cessation of the autocatalytic oxidation reaction. The plan was to continue the transient until the oxidation of the assembly Zircaloy had ceased.

The fuel bundle was preconditioned at a peak power of 670 kW for an integrated power of about 3910 kWh. The assembly was then operated at a 30 kW fission power level and subjected to low coolant flow (1.26 g/s) during the test. The higher bundle power resulted in faster coolant boilaway and rod heatup. Cladding oxidation and temperature escalation (caused by the autocatalytic reaction of Zircaloy with steam) were also achieved more quickly than in previous tests. The high-temperature autocatalytic condition was maintained within the assembly for 45 min and then hydrogen generation ceased due to the consumption of the available Zircaloy cladding. The fission power and makeup flow were continued for another 15 min and were then terminated. The assembly was allowed to cool slowly.

Post-test activities included the spectral gamma scanning of the deposition rod which had been suspended in the plenum above the fuel rods and the axial gamma scanning of the plenum. Several months later with the assembly underwater a visual inspection was accomplished by cutting a 90° window through the insulation, revealing most of the length of two outer rods on one side of the fuel bundle.

TEST RESULTS

Following the uncovering and dryout during the coolant boilaway, the rods heated at a rate of 2 to 5 K/s until peak cladding temperatures of 1700 K were attained, at which time the autocatalytic oxidation reaction resulted in a temperature excursion (at a rate of 10 to 50 K/s) and hydrogen generation. Peak local cladding temperatures are estimated to have exceeded ~2600 K, based on information from thermocouples on the outside of the bundle liner.

The high-temperature oxidation reaction began at the 2.4- to 3.04-m elevation and formed a localized burn front that moved quickly downward as far as the 1.2-m elevation and then steadily upward. The burn front reached the top end caps (3.80 m) and ceased 15 min before the end of the test. The oxidation reaction consumed 75% of the total Zircaloy or almost 100% of the Zircaloy in the path of the burn front. The remaining 25% of the Zircaloy was always below or near the bundle water level. The amount of hydrogen generated was 300 ± 30 g, close to the total conversion of the 1.26-g/s makeup coolant flow within the 45-min high-temperature period. The hydrogen flow fluctuated during the 45-min high-temperature period in response to similar fluctuations (10% to 20% relative) in the bundle coolant flow. The peak hydrogen flow was 190 mg/s, which corresponded to an oxidation power of 28 kW.

Significant noble fission gas release occurred throughout the 45-min high-temperature period as measured by the CRNL stack release monitor and by Idaho National Engineering Laboratory (INEL) and PNL gamma spectrometers at the ECM. Post-test gamma scanning of cesium and iodine deposits on the plenum walls indicated significant iodine/cesium release. The deposition pattern for cesium was similar to that for iodine.

The post-test underwater visual examinations covered the region from the 0.76-m elevation to the 3.30-m elevation. Four distinct areas of damage were observed along this length of the assembly. There was evidence of Zircaloy cladding relocation, but no massive accumulation was found at any specific axial location.

CONCLUSIONS

Several conclusions were drawn from the test results:

- Although local temperatures in the moving burn front rapidly exceeded the Zircaloy melting temperature range of 2100 to 2400 K, and some downward movement of molten metal was detected, the Zircaloy relocation did not result in cessation of the overall oxidation.
- No blockage formed (due to fuel and cladding relocation) that was complete enough to stop the steam flow and thereby abate the Zircaloy-steam reaction.
- The high-temperature and oxidation-rate escalations were coterminous; therefore, almost all the hydrogen generation occurred after Zircaloy melting temperatures were first exceeded.
- Fission gas release rates from the bundle were significant throughout the high-temperature transient, and the release fraction was greater than 50% in those areas traversed by the burn front.

PREFACE

Provided in this preface is general background information on the series of tests conducted by the Pacific Northwest Laboratory (PNL) under the Coolant Boilaway and Damage Progression (CBDP) Program, sponsored by the U.S. Nuclear Regulatory Commission (NRC). The CBDP Program is part of the NRC Severe Fuel Damage and Source Term (SFD/ST) Program (USNRC 1989). Through this program, the NRC is sponsoring a series of well-controlled, highly instrumented separate effects and integral tests. The tests in the CBDP Program have been designated the full-length high-temperature (FLHT) tests. Five FLHT tests on full-length fuel rods are being conducted; four have been completed to date. The FLHT tests are being conducted in the National Research Universal (NRU) reactor at Chalk River, Ontario, Canada. The reactor is operated by Atomic Energy of Canada Limited (AECL) and staff and equipment are provided by the Chalk River Nuclear Laboratories (CRNL).

In the SFD/ST program, the NRC has partnerships and cooperative or bilateral agreements with the U.S. Department of Energy (DOE) and Electric Power Research Institute (EPRI) and with agencies in the following countries: Belgium, Canada, Federal Republic of Germany, Finland, France Italy, Japan, Netherlands, Norway, Taiwan, Republic of Korea, Spain, Sweden, Switzerland, and United Kingdom.

GOAL OF CBDP PROGRAM

The goal of the CBDP Program is to provide experimental results and analyses to the NRC that can be used in assessing the risk of severe accidents in light water reactors (LWRs). The results are used to increase understanding of reactor operations and to then develop strategies and guidelines for preventing and mitigating accidents in LWRs that could result from a severe loss of reactor coolant, referred to as a loss of coolant accident (LOCA). These tests provide prototypic data and analysis on the behavior of reactor fuel rod and control rod materials subjected to conditions simulating a LOCA. In these tests, full-length fuel rods used in commercial LWRs are subjected to severe reductions in the flow of coolant while operating at low heat ratings for progressively longer times. This procedure simulates conditions that would result from a LOCA in an LWR.

In general, the FLHT tests are performed to determine what happens to the fuel when it overheats and melts; this behavior is monitored during the tests to understand the progression of the melt as the fuel heats up. Physical and chemical changes in the fuel bundle are recorded as are releases of fission products and their transport and deposition. The data acquired is used to resolve technical uncertainty about hydrogen generation; the progression of a core melt; amounts and distances of melt relocation; solidification patterns, including blockage of coolant channels; and the release of fission products in-vessel.

The FLHT tests are the only known in-reactor boilaway tests being performed in the world on full-length fuel rods used in pressurized water reactors (PWRs) and boiling water reactors (BWRs). Because full-length rods are used, no axial scaling factors are required when applying the data

acquired from the tests to the operation of LWRs. Both unirradiated and irradiated fuel rods are used in these tests.

COMPUTER MODELS

During these tests, prototypic data are generated and recorded that the NRC can use to develop and evaluate models for assessing the risk of severe accidents in LWRs. In particular, of interest are reactor core models related to the early phases of a severe accident, up to core temperatures of about 2800 K before the uranium dioxide and other ceramics like zirconium dioxide (~3100 K and ~2900 K respectively) and other ceramics melt. Such models were first developed by NRC contractors using data acquired from short-length and small-scale separate effects tests conducted for the NRC by PNL and other laboratories (Prater and Courtright 1987; and Knipe, Ploger and Osetek 1986). The NRC uses data from the FLHT tests to further develop, benchmark, and validate these models.

These models have also been incorporated into detailed mechanistic computer codes, such as MELPROG (Dosanjh 1989) and SCDAP/RELAP (Allison et al. 1989). These codes can be used to predict separate and integrated phenomena expected during severe accidents.

SCOPE

The scope of the CBDP Program provides for five tests in the FLHT test series and one proof-of-principal test to explore a range of conditions that could contribute to a severe LOCA. They are identified in the following table along with basic parameters of each.

Summary of Basic Test Parameters for FLHT Tests

<u>Test</u>	<u>Irradiated</u>		<u>Peak Temperatures, K</u>		<u>Time at Peak Temperature</u>	
	<u>Date</u>	<u>Rods</u>	<u>Goal</u>	<u>Actual</u>	<u>Goal</u>	<u>Actual</u>
MT-6B	6/84	0	1600	1400	0	0
FLHT-1	3/85	0	2150	2275	0	0
FLHT-2	12/85	0	2200	2500	4	4
FLHT-3	(Conceptualized only)					
FLHT-4	8/86	1	2500	2600	30	30
FLHT-5	5/87	1	2800	2600	60	60
FLHT-6	11/93	2	2600	--	5	NA

The conditions for each test in the series are designed to become progressively more severe, operating at increasing prototypic nuclear power densities, thermal gradients, and steam mass fluxes. They progress from a relatively short time at a relatively high temperature to longer times at even higher peak temperatures they become progressively more severe over the temperature range from 1400 K to about 2600 K. The program is limited to conducting tests that will provide data on and an analyses of the behavior of a reactor core from the time that the core begins to become uncovered until the temperature in the core reaches a peak of about 2800 K.

The scope of PNL activities in the CBDP program includes the following:

- conceptualizing suitable and productive in-reactor tests
- designing and developing the hardware and systems needed for the tests, with emphasis on the geometry of the fuel bundle
- fabricating, procuring and assembling the hardware and systems to be used during the tests, including the fuel bundle
- performing the required safety analyses required for approval to perform the tests in the NRU reactor
- performing pretest operational checks
- with the assistance of CRNL reactor staff, performing the tests
- conducting limited but immediate post-test examinations (PTEs) and analyses
- analyzing and reporting on the preliminary results of the tests to members of the agencies and countries participating in the program
- coordinating and guiding detailed post-irradiation examinations (PIEs) and analyses with the CRNL staff
- assessing the results of the PIEs
- analyzing and reporting on the final results and conclusions of the tests.

HISTORY

In 1978, PNL began tests under the large-break LOCA Program in the Canadian NRU reactor with studies on the behavior of full-length LWR fuel rods related to design-bases accidents. Studied were the heatup and reflood phases of simulated large-break LOCAs. A description and major results of experiments conducted from 1980 through 1982 are contained in NUREG-1230, Compendium of ECCS Research for Realistic LOCA Analysis.

Mainly as a result of the accident at the Three Mile Island, Unit 2 (TMI-2), in March 1979, the NRC redirected work at PNL on fuel behavior from

studies related to design-bases accidents, focusing on large-break LOCAs (LB-LOCAs), to severe accidents, focusing on severe small "break" accidents. Actual work with this new focus started in 1980.

During 1981 and 1982, PNL designed and developed SFD-type tests that could be accommodated in the short-length Power Burst Facility (PBF) reactor, Idaho Falls, Idaho. The SFD-type test train that was developed included a system compatible with the Canadian NRU reactor. Starting in 1983 test trains were designed to permit the safe conduct of severe damage tests on full-length LWR fuel bundles and the PNL severe fuel damage studies became part of the NRC SFD/ST international program.

Transition Test

In 1984, a proof of principal test (MT-6B) was performed to determine whether a test on a full-length fuel bundle could be safely performed to demonstrate the kind and extent of the damage that would result to fuel rods from a boilaway of reactor coolant. Emphasized were the severe damage conditions that would result in the core. In this proof of principal test, the LB-LOCA test geometry was used. Demonstrated during the test was that adequate thermal insulation can protect the reactor under severe conditions and that it is possible to control a boilaway transient; the conclusion was that it would be safe to conduct in-reactor tests that cause severe damage to reactor fuel rods from a loss of coolant.

High-Temperature Tests

In March 1985, FLHT-1, the first very high-temperature test on full-length fuel rods was performed. During the test, the coolant was allowed to boilaway causing severe damage to the fuel rods. Demonstrated by this test was that severe fuel damage tests on full-length fuel bundles can be safely conducted in the NRU reactor.

FLHT-1 was followed in about 9-month intervals by three similar tests, each becoming increasingly more severe:

- FLHT-2 in December 1985
- FLHT-4 in August 1986
- FLHT-5 in May 1987.

The following are the general findings from these tests:

- On-line data indicate that hydrogen generation continued after onset of Zircaloy melting and during and following relocation of the Zircaloy for as long as it remained hot and steam was present.

- Analyses of on-line data indicate continued hydrogen generation during and following metal relocation, which is corroborated by examination of metallographic specimens.
- Data from thermocouples indicate non-uniform bundle heatup and melt generation, which is corroborated by post-test metallographic observations of heterogeneous flow blockage areas.
- Post-test metallography for the FLHT tests by Chalk River engineers indicate extensive oxidation of Zr-bearing melt debris. The extent of bundle oxidation relates directly to the length of time at high temperatures.
- Metallographic data to date indicate only minor changes in coolant flow areas.

During 1988 and 1989, the data acquired from these FLHT tests were analyzed by PNL. Based on this analysis, a modified test train was designed and developed. It will be used in the last boilaway test to be conducted in late 1993. The NRU reactor was unavailable during 1991, 1992 and most of 1993 for performing the test. The purpose of this test, FLHT-6, is to study the unique aspects of the behavior of BWR fuel rods channel box, and control blade subjected to simulated conditions of a severe accident.

GENERAL DESCRIPTION OF CBDP TESTS

The tests are conducted to evaluate the behavior of fuel during simulated small-break LOCAs resulting in a partially uncovered reactor core. In a LOCA, as the coolant boils away and the fuel rods become uncovered, the temperature of the rods increases above design limits. As the temperature continues to increase, the cladding on the rods starts to melt and become damaged, and radioactive fission products are released. The boilaway phase of the test is started by reducing the flow of coolant to the bundle of full-length fuel rods while low fission power to simulate decay power is maintained; this boils the coolant away. The phase simulates system enthalpy and decay heat by supplying the energy that could cause an accident resulting from coolant boilaway.

In general, the boilaway of the coolant and the subsequent heatup of the exposed rods, and the exothermic autocatalytic oxidation reaction of Zircaloy with steam that result during a test lead to the following physical and chemical changes: 1) the generation of hydrogen caused by the oxidation of Zircaloy from steam; 2) a rapid increase in the fuel and cladding temperatures cause by the exothermic oxidation reactions; 3) the melting of the cladding on the fuel rods; 4) the liquefaction of the fuel by the molten cladding; 5) the relocation of molten fuel and cladding material; and 6) the release of fission products from the fuel, cause by the high temperatures.

Monitoring the Tests

Severe fuel damage test conditions are monitored by the following methods. Cladding temperatures and heat flow are monitored by thermocouples located along the length of the fuel rod assembly. Water flow and steam flow are monitored by flowmeters. The generation of hydrogen is measured by partial pressure and flow, and the release of fission products is monitored by gamma spectrometers. The test system consists of a test assembly, commonly referred to as "test train"; effluent control module (ECM) and steam cave closure (SCC); and NRU reactor and auxiliary flow loop.

A shielded ECM is used to 1) maintain the pressure in the test system by regulator valves and flow injection of pressurized nitrogen gas, 2) condense the steam from the effluent, and 3) route the gas/liquid fractions to waste storage tanks and the reactor stack.

The following are specific phenomena that are controlled and/or monitored during the tests:

- the NRU reactor power and the associated fission power in the test fuel bundle
- the water flow rate into the fuel bundle region
- the water level within the fuel bundle
- the temperature of the fuel bundle, within each fuel rod and various regions of the shroud
- the flow rate of the gaseous effluent from the fuel bundle
- the amount of hydrogen in the gaseous effluent
- the time the fuel bundle is at a temperature above 2477 K.

Safety Reports

Two safety analyses and an operations plan are prepared for each test under the CBDP program. First, PNL prepares a preliminary safety analysis report (PSAR). Several months later PNL then prepares a final safety report (FSAR). Both safety analysis reports (SARs) are reviewed and approved by safety engineers at CRNL and by the AECL Nuclear Safety Advisory Committee (NSAC). Therefore, these two reports are written mainly for AECL safety engineers at CRNL.

Finally, PNL prepares the experiment operations plan (EOP) for each test. This plan is also approved by test and operation engineers at the NRU reactor. Experimental operations plans are the detailed step by step procedures used to conduct the CBDP tests and are directly related to the test, reactor, equipment, measurements and controls.

Post-Test Examination

After each test, a post-test examination (PTE) is performed as follows. First, axial spectral gamma scans include the following: 1) along a special rod suspended in the plenum tube during a test, 2) along the outside of the plenum tube above the fuel rod assembly, and 3) along the outside of the shrouded fuel rod assembly. The axial extent of damage to fuel rods is then visually determined in the NRU reactor basin shortly after a test. Later, metallographic and gamma ray tomographic examinations are performed by CRNL staff on cross sections of the fuel rod assembly to quantitatively determine the amount of bundle damage.

DATA ON SOURCE TERMS

In addition to providing data on the behavior of fuel rods under severe conditions, the experiments also provide data on the behavior of radioactive fission products. These data, mainly on the release of radioactive xenon and krypton, can be used to help estimate "source terms."

SEPARATE EFFECTS TESTS

Another component of the NRC SFD/ST testing program on nuclear power reactors is a series of small-scale separate effects tests. The results of these tests are being confirmed and validated by the data that is being acquired during the CBDP tests. These small-scale tests, begun in the early 1980s, have concentrated on a single phenomena such as the rate of the oxidation of Zircaloy at very high temperatures, the rate of the reaction between Zircaloy and UO_2 , or the oxidation rates of stainless steel. The NRC has sponsored these tests at PNL and other laboratories. Much work in this area was also conducted at Kernforschungszentrum (KfK), Karlsruhe, Germany.

The small-scale tests are conducted in the Power Burst Facility (PBF) reactor at Idaho Falls, Idaho (Knipe, Ploger and Osetek 1986; and Martinson, Petti and Cook 1986); in the Annular Core Research reactor (ACRR), Albuquerque, New Mexico (Gauntt, Gasser and Ott 1989); and at several out-of-reactor facilities. In SFD experiments conducted at the PBF reactor, ~0.91-m (36-in.) -long 32-rod bundles were tested to 3000 K. Four tests were completed and provided data to support FLHT test operations.

NRU REACTOR

The CBDP tests are designed and conducted to take full advantage of the main capabilities of the NRU reactor. These include the following:

1. the capability for testing highly instrumented 12-ft long bundles of fuel rods, up to 14 rods in a bundle, under thermal-hydraulic conditions representative of contemporary LWRs
2. the capability for achieving prototypic power densities and axial power distributions with fuels of commercial enrichment

3. the capability for providing prototypic coolant mass fluxes at the fluid/vapor interface which are typical in a TMI-2 type boildown.

These unique capabilities reduce the uncertainties associated with scaling the length and power distributions and the interpretation of the experimental results from the small-scale separate effects tests.

UNITS OF MEASURE

Common practice is to use "familiar" units in all documents relating to the CBDP tests, units familiar to the actual test engineers and common to the equipment and measurement systems. The common test units and the units in the EOPs are inch-lb.-F-liter-second.

Because the on-line test data are evaluated during the test to direct test operations, the data are recorded in the same units as defined in the EOP; thus, data reports use the EOP's system of units as well as SI units.

DISSEMINATION OF RESULTS

At the conclusion of the FLHT tests, to ensure that the NRC and SFD partners received the basic results of the tests in a timely fashion, preliminary data reports are prepared based on preliminary reviews of the results; these reports are issued as draft PNL reports. The objective of issuing these preliminary drafts is to make the results of the test available to the research community as quickly as possible so that the results can be applied to reactor operations. After subsequent review and detailed analyses, final data reports on the tests are prepared and published.

Because the PSARs, FSARs and EOPs concern preparations for the CBDP tests and do not contain vital data on the results of the tests, they are published as PNL technical reports.

ACKNOWLEDGMENTS

The work on the FLHT-1, -2, -4, and -5 tests spanned the years 1984 to 1987, excluding post-irradiation examinations. During this period, many individuals at the Pacific Northwest Laboratory (PNL), EG&G Idaho, two consultants, Chalk River Nuclear Laboratories (CRNL), and the U.S. Nuclear Regulatory Commission (NRC) made significant contributions to these successful tests. Listed below are most of the key contributors to this program. However, as is always the case with a large-scale irradiation test program, there are more contributors than can be named. Those whose names have not been included know that their contributions have been invaluable to the success of the program.

PNL

Gordon Anderson designed and built the special electronic systems for the liquid level time domain reflectometer sensors that continuously measured the liquid level within the fuel bundle region, the molten metal penetration detectors that help protect the reactor during the tests, and the electrical heaters in the upper plenum of the test train. He also helped with pre-test instrument checks. He was assisted by Gene Wallace, Beve Taylor, and Tom Conroy.

Dale Fitzsimmons was the test director for the last two tests and significantly upgraded the experiment operations plan for those tests.

Max Freshley provided management oversight and was a key interface among PNL, CRNL, and the NRC.

Bob Goodman was responsible for test train instrumentation, he was supported by Bill Norton and Tom Fish.

Galen Hesson performed thermal hydraulic analyses and was the test director for the first two tests.

Walt Hensley was responsible for fission product measurements and analysis for the later tests.

Dave Hurley designed, fabricated, assembled, checked out, operated and modified the effluent control module. He was assisted by Duane Hobbs and Steve Burnum.

Urban Jenquin was responsible for the tests nuclear analysis including fission product production, radiation levels, and shielding analyses.

Loyd King was responsible for the test train design and post-test examinations; he also helped with test operations. He was assisted by designers Dave Hagerty, Ken Keene, Craig Sumner, and Andy Anthony. Also assisting were Bob Hagen, Don Smith, and Chuck Bigelow during post-test examinations.

Don Lanning, one of the authors of this report, performed many hand calculations to check computer predictions for both safety analysis and for actual test operations. He analyzed all test data but was the key analyst of the mass spectrometer results. Don was responsible for the safety analysis for the last two tests.

Nick Lombardo, also an author of this report, assisted in earlier thermal hydraulic analyses and was responsible for all the thermal mechanical analyses for the last two tests and was the assistant test director.

John Nageley provided editorial guidance and improved the clarity of many parts of the test report. Others who worked on earlier drafts of the report include Dave Hilliard, Nancy Waugh, S. Kesterson, R. Moreno, and J. Moore.

Frank Panisko was the project manager. He was assisted by Pam Cummings during the earlier tests and by Steve Cassidy during the last test.

LeeRoy Parchen was in charge of QA and the fabrication of the later test trains.

Chuck Bigelow assembled the test trains and helped put them into the reactor.

Ron Page was responsible for making flexible electrical connections for the test train instruments. He was assisted by Steve Faber, Tom Fish and Billy Norton.

Jim Pilger was responsible for the earlier test train fabrication development, actual fabrication and assembly. He was assisted by Stu Allison, Chuck Bigelow, Jim Vining, Ron Page, Billy Norton, Steve Faber, John Lester, and Norm Davis.

Bill Rausch did the development of the necessary software for the test data acquisition and control; he also operated the computer during the tests.

Glenn Russcher was responsible for the safety analyses of the earlier tests.

Nancy Wildung processed the millions of data points and converted them into visual representations. She was helped by Tracy Klem, Brian Reed, and Gary Springer.

Consultants

August Cronenberg performed key safety analyses for off-normal energetic events as well as for fission product transport and deposition.

Mark Wismer made major contributions to every aspect of the tests, design, assembly, setup and check out, especially in trouble-shooting instrument problems; he was the interface between the test director and the computer control systems during the tests and helped perform post-test fuel bundle and shroud examinations.

EG&G

Jack Hartwell performed fission product release measurements and analysis.

CRNL

Just as there are more PNL staff who contributed to the success of these tests than can be named here so also are there more CRNL staff than can be named here. However, the following CRNL staff provided major contributions to the success of most if not all the FLHT tests: Mike Atfield, Don Burton, Terry Chapman, Paul Fehrenbach, Nancy Howden, Neil Keller, Pat Kelly, Bob Lovoie, Ivan Martin, Bernie McCambridge, Bill McCrea, Danny Nishimura, Bill Shorter, Dave Thompson, and Dave Wilder.

NRC

This program was initially under the management of Bob van Houten and, after his retirement, temporarily under Bob Wright then under Dr. Tom J. Walker. This program was part of the U.S. NRC Severe Accident Research Program that was initially headed by Mel Silberberg and later by Farouk Eltawila. Ralph Meyer made timely, valuable contributions.

We also wish to thank the Electric Power Research Institute for permission to use the palladium hydrogen meter that they developed.

ACRONYMS

ADD	antideposition device
ALPM	actual liters per minute
CBDP	Coolant Boilaway and Damage Progression (Program)
CRNL	Chalk River Nuclear Laboratories
DACS	data acquisition and control system
DOE	U. S. Department of Energy
ECM	effluent control module
EFPH	effective full-power hour
EPRI	Electric Power Research Institute
EOP	Experiment Operations Plan
FLHT	full-length high-temperature (tests)
HBR	H. B. Robinson (reactor)
HEPA	high-efficiency particulate air (filter)
ID	inside diameter
INEL	Idaho National Engineering Laboratory
LCS	loop control system
LHGR	linear heat generation rate
LWR	light water reactor
MFM	mass flowmeter
MMPD	molten material penetration detector
MS	mass spectrometer
NCTFM	noncondensable turbine flowmeter
NRC	U. S. Nuclear Regulatory Commission
NRU	National Research Universal (reactor)
OD	outside diameter
PHM	palladium hydrogen meter ¹
PNL	Pacific Northwest Laboratory
PT	pressure transducer
PWR	pressurized water reactor
RCS	reactor control system (in other literature, RCS often means reactor coolant system)
SCC	steam closure cave
SFD/ST	Severe Fuel Damage/Source Term (Program)
SPND	self-powered neutron detector
TC	thermocouple
TCM	thermal conductivity meter
TD	theoretical density
TDR	time-domain reflectometer

¹ On loan from the Electric Power Research Institute.

DEFINITIONS

Significant words and phrases used in this report are defined below:

assembly	(also referred to as fuel rod assembly or fuel assembly) The 11 full-length fuel rods and 1 dummy gamma thermometer held in a square lattice array by grid spacers.
bundle	(also referred to as fuel rod bundle or fuel bundle) A term commonly used to designate the fuel rod assembly and its characteristics (e.g., bundle coolant flow and bundle oxidation power).
boilaway transient	The extended operation at 30-kW bundle nuclear power and low makeup flow (1.26 g/s) when the coolant boils away and the rods heated up. Cladding temperature then escalates; the cladding melts; and the oxidation accelerates--resulting in severe damage to the bundle.
carriers	Pieces of Zircaloy occupying the corners of the assembly that shield the thermocouple (TC) leads and route them, i.e., "carry" them to exit points from the assembly extremities.
caves	Lead-lined boxes containing and shielding various components within the steam closure cave/effluent control module (SCC/ECM).
cavity	The spaces in the shroud or plenum area that were pressurized with inert gas and monitored for pressure during the test; e.g., the shroud insulation cavity, the plenum insulation cavities, and the shroud molten material penetration detector (MMPD) cavity.
closure	(also referred to as closure seal or closure plug) The specially designed fixture at the top of the test train that provides a pressure seal for the reactor pressure tube and permits penetration of makeup flow lines, time-domain reflectometer (TDR) vent lines, plenum drain line, bundle effluent line, and test train instrument leads.
gamma thermometer	A device for measuring coolant level and axial power distribution which can be thought of as a solid stainless steel rod with internal thermocouples along its length, residing in a

Zircaloy guide tube in one of the cells of the square lattice of the fuel assembly. (In FLHT-5, a steel rod in a Zircaloy sleeve was used to simulate the gamma thermometer used in FLHT-4 to keep the two tests similar with respect to power distribution and material content.)

makeup flow	(also referred to as makeup) The inlet water flow to the bundle, especially after steaming and boilaway have begun. This small flow "makes up" for steaming losses.
level	The elevation in inches above the bottom of the FLHT fuel column.
plenum	The 4-m-long Zircaloy tube (effluent line) leading from the fuel bundle to the closure, including insulated heaters. The upper plenum is the approximately 3-m-long heated section beginning 1 m above the top of the bundle; the lower plenum is the unheated approximately 1-m section below the heated section. The two are separated by a metal diaphragm. Above the upper plenum is an evacuated double-walled plenum section leading through the closure.
SCC/ECM	A combined acronym denoting the steam closure cave (SCC) module and the effluent control module (ECM) which were contained in the same confinement system for the test.
shroud	The insulated multicomponent structure surrounding the fuel bundle. The components (listed in order from inside to outside) include the liner, insulation tiles, Zircaloy saddles, inner round, MMPD cavity with wire wrap, and the outer Zircaloy round. The oxidation of the liner and the carriers is included with that of the bundle in the measurement of bundle oxidation power and hydrogen generation because it is impossible to distinguish the separate contributions of these components during accelerated bundle oxidation.
liner	The Zircaloy inner lining of the fuel bundle shroud, which separates the bundle from the insulation tiles in the shroud.

test train The combined FLHT experimental apparatus inserted in the National Research Universal (NRU) reactor pressure tube, including fuel bundle, shroud, plenum, and closure fixture (approximately 9-m long).

time-domain
reflectometer A device for sensing coolant level in the test train. It consists of a tube running the full length along the outside of the test train in the bypass annulus and is held at constant pressure by means of a vent line. The tube interconnects the bottom of the fuel bundle and the top of the plenum and acts like a manometer with the fuel bundle as one leg; therefore, the tube acts like a stand pipe, indicating the collapsed coolant level in the fuel bundle.

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1.0 INTRODUCTION

Through the Severe Fuel Damage/Source Term (SFD/ST) Program,^(a) the U.S. Nuclear Regulatory Commission (NRC) is conducting or participating in numerous tests to study the behavior of reactor fuel rods and other core materials during severe accident conditions. As part of the SFD/ST Program, NRC sponsors the Coolant Boilaway and Damage Progression (CBDP) Program at Pacific Northwest Laboratory (PNL).^(b) In the CBDP Program, instrumented, insulated assemblies of full-length (4-m) light-water reactor (LWR) fuel rods are subjected to coolant flow reductions while operating at low heat ratings. This procedure simulates possible loss-of-coolant accident conditions in LWR cores. The consequent coolant boilaway, heatup of the exposed rods, and exothermic oxidation reaction of the hot Zircaloy cladding with steam then lead to cladding melting, fuel liquefaction, material relocation, hydrogen evolution, and fission product release.

A series of severe damage tests, designated full-length high-temperature (FLHT) tests, is being conducted by PNL under the CBDP Program in the National Research Universal (NRU) reactor at Chalk River Nuclear Laboratories (CRNL), Ontario, Canada. In the tests conducted so far, the severity of peak conditions has been increased from one FLHT test to the next as shown in Table 1.1. The peak conditions included maximum cladding temperature, maximum oxidation power and hydrogen flow, and the time period at high assembly temperature (>2477 K).

During the tests, cladding temperatures and heat flow were monitored along the length of the fuel rod assembly by thermocouples (TCs); water flow and steam flow were monitored by flowmeters; the evolution of hydrogen was measured by various means; and the release and transport of fission products were monitored by gamma spectrometers. A shielded effluent control module (ECM) maintained system pressure, condensed the steam in the effluent, and routed the liquid/gas effluent to waste storage tanks and to the reactor stack.

After each test, axial spectral gamma scans were performed along the plenum above the fuel rod assembly, as well as along a special rod suspended in the plenum during the test. The axial extent of fuel rod damage was visually inspected several months later in the NRU spent fuel storage basin; ultimately, metallographic examinations will be performed on cross sections of the fuel rod bundles.

-
- (a) Partners in this program with NRC include nuclear organizations from the following countries: Belgium, Canada, England, Federal Republic of Germany, Finland, Italy, Japan, the Netherlands, Republic of China (Taiwan), Republic of Korea, Spain, Sweden, and Switzerland.
 - (b) Operated for the U.S. Department of Energy (DOE) by Battelle Memorial Institute under Contract DE-AC06-76RLO 1830.

TABLE 1.1. Summary of Major Parameters for Current Coolant Boilaway and Damage Progression Severe Fuel Damage Tests in the National Research Universal Reactor^(a)

Test Designation	Test Date	Number of Irradiated Rods ^(c)	Nominal ^(d) Assembly Nuclear Power, kW	Peak Assembly Heat Generation Due to Oxidation, kW	Peak Cladding ^(e) Temperature K	Time at Peak Temperature min	Peak Hydrogen Generation Rate, mg/s
FLHT-1	3/85	0	23	21	2275	0	140
FLHT-2	12/85	0	23	31	2500	4.5	210
FLHT-3 ^(b)							
FLHT-4	8/86	1	23	26	2600	30	174
FLHT-5	5/87	1	30	28	+2600	45 ^(g)	190

- (a) All tests to date have been conducted with a nominal bundle inlet flow of 1.26 g/s during the boilaway transient.
- (b) A test designated FLHT-3 was canceled at the conceptual design stage.
- (c) The total rods per assembly is 12 (FLHT-1, -2) or 11 (FLHT-4, -5). The irradiated rods have a nominal burnup of 28 MWd/kgU and were discharged from the H. B. Robinson (HBR) reactor in May 1975.
- (d) This is the fission power confirmed by calorimetry of the water-filled bundle. The voided bundle is estimated to have 15% higher power, based on neutronic calculations.
- (e) These values are best estimates based on interpretation of thermocouple data.
- (f) The values are the period of time when cladding temperatures were greater than 2477 K.
- (g) 60-min total operation.

This report describes the fourth test in the series (designated FLHT-5) and presents test results. The FLHT-5 test differed from its predecessors by featuring 30-kW versus 23-kW bundle nuclear power during the boildown and heatup. The higher nuclear power was chosen to promote cladding melting and rundown. After the first measurement of high temperature (>2477 K), the hold period was extended to 60 min to probe the time required to reach cessation of oxidation/hydrogen generation in a full-length fuel assembly.

Section 2.0 presents an overview of the test objectives, design requirements, and operations plan. Section 3.0 contains a more detailed description of the test hardware, hardware arrangement, and effluent flow paths. The test operations are described in Section 4.0, and test results are shown in Section 5.0. Results from the post-test gamma scanning and visual examination are described in Section 6.0. Instrument layout and identification are given in the appendix.

2.0 TEST OBJECTIVES AND DESIGN REQUIREMENTS

The FLHT-5 test was nearly identical to the FLHT-4 test¹²³ in regard to test train design and hardware used. The test objectives differed, however, as described below.

2.1 TEST OBJECTIVES

The overall objective of the FLHT series was to produce severe damage behavior data for the validation of severe damage computer codes. Specific objectives for the FLHT-5 test included the following:

1. promote cladding melting and rundown by operating at 30-kW assembly nuclear power during the boilaway/severe damage transient (as opposed to the 23 kW used in preview tests)
2. determine the time at which oxidation and hydrogen generation ceased within the assembly and characterize the cause of the cessation
3. characterize the release, transport, and deposition of gaseous and volatile fission products (xenon, krypton, iodine, cesium, and tellurium).

To achieve the second objective, approval was obtained to operate the assembly for 60 min beyond >2477 K cladding temperature. For the third, the preconditioning period was extended to 5 effective full-power hours (EFPH) (versus 0 EFPH for FLHT-2 and ~1 EFPH for FLHT-4) to build in radioactive iodine and other fission products of interest.

2.2 DESIGN REQUIREMENTS

The design requirements and constraints imposed by the test objectives and by the use of existing NRU reactor equipment are described below. These serve as an introduction to the details of test train design and associated hardware design described in Section 3.0.

Overall, the test train and associated hardware had to 1) provide for coolant boilaway and rod heatup to 2800 K within the fuel bundle; 2) measure and record key parameters such as temperatures, flow rates, and pressures;

¹ Lanning, D. D., and N. J. Lombardo. 1986. Final Safety Analysis for FLHT-4 Test. PNL-5869, Pacific Northwest Laboratory, Richland, Washington (Draft).

² Fitzsimmons, D. E. 1986. Experiment Operations Plan for the FLHT-4 Experimentation in the NRU Reactor. PNL-6059, Pacific Northwest Laboratory, Richland, Washington (Draft).

³ Lanning, D. D., et al. 1988. Data Report: Full-Length High-Temperature Test 4. PNL-6368, Pacific Northwest Laboratory, Richland, Washington (Draft).

3) maintain test control; 4) sample the effluent; and 5) provide for operating safety and public safety throughout the test. The hot and radioactive effluent (at 1.38 MPa) was routed across the top of the reactor for gamma counting and sampling. That effluent and the associated sampling equipment had to be monitored, shielded, and contained by portable modules that could be installed and removed within a few days before and after the test, respectively.

Specific assembly and shroud requirements were as follows:

1. The insulating shroud surrounding the fuel assembly had to permit cladding melting and oxidation on the inside (in the bundle region), while maintaining acceptably low temperatures (<420 K) on the outside to maintain the reactor pressure tube integrity.
2. The fuel assembly and shroud had to fit within the 10.3-cm inside diameter (ID) L-24 pressure tube, with clearance for bypass coolant flow from the U-2 loop and for the bundle inlet flow tubes that brought metered inlet coolant to the bottom of the bundle. The fuel rods were to be full-length (3.63-m fuel column length).
3. Bundle and shroud instrumentation had to track the temperatures as a function of time and radial/axial position and also had to provide indications of any molten material penetration into the shroud.

The basic design chosen to meet the above requirements was an array of 12 pressurized-water reactor (PWR) rods surrounded by 1-cm-thick insulating tiles of low-density zirconium dioxide. Thermocouples were placed at 10- to 20-cm intervals along the length of the assembly, both within the fuel rods and in the shroud.

Specific requirements for the effluent piping and control system included the following:

1. Join the fuel assembly to the pressure tube closure seal (located 4 m above the top of the core) while avoiding steam condensation.
2. Allow for penetration of the effluent tube instrument lines and assembly coolant lines through the closure seal at the top of the pressure tube.
3. Route the effluent flow up through the closure and across the reactor top deck plate; provide flow and pressure control at 1.38 MPa; provide effluent sampling and gamma counting, plus shielding and confinement; and safely route the effluent into a 0.1-MPa existing discharge line.

The first requirement was met by an insulated and heated plenum, and the second was met by a specially designed closure plug. The ECM filled the third requirement by providing a steam condenser, a pressure control valve, nitrogen flow injection, sampling lines, hydrogen monitors, and penetrations in the surrounding lead shielding for gamma counting. The steam closure cave (SCC)

provided shielding for the effluent line S-bends from the vertical exit at the closure to the ECM and held a sample bomb manifold.

The specific requirements for the data recording and test control systems included the following:

1. Maintain a low, constant reactor power and loop (bypass/bundle) flow and pressure control throughout the test; in particular, maintain a high bypass flow (1 kg/s) while reducing the bundle inlet flow from a pretransient value of ~10 g/s to 1.26 g/s during the transient.
2. Provide for automatic rapid shutdown of the reactor (assembly) power and inlet flow should an off-normal condition occur and provide for monitoring all conditions throughout the experimental hardware.
3. Record data from ~250 test instruments more than once per second during the boilaway/severe damage transient.

These requirements were met by a computerized data acquisition and control system (DACS)⁴ working in conjunction with an existing reactor control system (RCS) and the U-2/U-1 loop control system (LCS).

⁴ Rausch, W. N. 1985. Full-Length High-Temperature Test Series Data Acquisition and Control System User's Guide. PNL-5639, Pacific Northwest Laboratory, Richland, Washington (Draft).

3.0 TEST HARDWARE

The hardware components for the FLHT-5 test were almost identical to those used for FLHT-4, which were described in the data report for that test.¹ In this section, the basic hardware arrangements are reviewed, and the changes from the FLHT-4 design are noted.

3.1 HARDWARE ARRANGEMENT AND EFFLUENT FLOW PATHS

Existing NRU building hardware included the reactor and the pressurized U-2 water loop that provided bypass coolant to the L-24 reactor position (the pressure tube containing the test train) and the control systems for the reactor and U-2 loop. The NRU waste stream and exhaust systems were also used, which included the reactor service space exhaust (attached to the ECM confinement space), the liquid waste line and catch tank, and the filter and stack release systems.

A sketch of the water coolant and steam effluent flow paths (during the boildown transient phase) is shown in Figure 3.1. The fuel assembly shroud and plenum were cooled externally by the bypass flow from the U-2 coolant loop. A separate system, called "makeup flow" (because it "made up" for steaming loss), supplied the assembly inlet flow. The makeup flow proceeded from pressurized accumulator tanks to the top of the test train. From there, it proceeded downward through the closure to oval tubes in the bypass annulus running alongside the plenum and the shroud to the bundle inlet. The makeup flow then went up through the bundle.

The steam-hydrogen fuel bundle effluent flowed up through the plenum and closure and through the SCC and ECM modules, where it was separated by a condenser into liquid and gaseous streams. These streams were later rejoined and swept by nitrogen dilution flow down a waste line to a catch tank in the basement of the NRU building. From there, the nitrogen, noble fission gases, and any carry-over iodine vapor were sent through activated charcoal filters to the reactor stack, 1 km away from the NRU building, for controlled release to the atmosphere. The SCC/ECM confinement reactor space was fed from service space air, and the air was exhausted through a high-efficiency particulate air (HEPA) filter and a charcoal filter back to the service space, which in turn exhausted the air through the reactor building filtration and stack release systems.

The arrangement of the major hardware components is shown in Figure 3.2, which shows both electrical and piping connections. The loop and makeup systems provided the bypass cooling and bundle makeup flows, respectively. The bundle effluent passed up through the plenum, then to the SCC and ECM on the deck plate and continued to the reactor waste system. Data from the test train and the ECM instrumentation were transmitted by electrical cables to the

¹ Lanning, D. D., et al. 1988. Data Report: Full-Length High-Temperature Test 4. PNL-6368, Pacific Northwest Laboratory, Richland, Washington (Draft).

FIGURE 3.1. FLHT-5 Flow Paths [flow control valve (FCV), pressure control valve (PCV), liquid control valve (LCV)]

FIGURE 3.2. Major FLHT-5 Test Hardware Components

DACS, located in the test control room (about 30 m from the reactor), for recording and comparator checks against control limits. Control signals went from the DACS back to the ECM for pressure/temperature/flow control, to the LCS for setpoints on bypass flow, and to the RCS in the event of an automated reactor trip or a manual trip by the test director. The LCS also had a trip-function communication with the RCS.

3.2 TEST TRAIN DESIGN

The test train hardware consisted of the following components:

- a modified 4 x 4 square lattice array of full-length, ~4-m-long PWR-type fuel rods, containing 10 unirradiated rods, 1 irradiated rod, and 1 dummy thermometer, held in place by a series of 8 grid spacers
- an insulating, instrumented shroud surrounding the fuel rod array along its full length
- a heated and insulated plenum 4 m in length, extending from the top of the assembly to the top of the reactor pressure tube
- a closure plug at the top of the reactor pressure tube, providing a pressure seal for the effluent line, the instrument leads, and for the bundle makeup line to pass through
- an inlet fixture channeling the bypass flow and leading the makeup flow into the bottom of the fuel rod bundle.

The arrangement of these components is shown in Figure 3.3. Note that the stainless steel deposition rod is suspended in the plenum from a blind flange above the closure in the SCC (not shown).

3.2.1 Fuel Rod Bundle

A cross section of the fuel rod bundle, the insulating shroud, and the surrounding pressure tube is shown in Figure 3.4a. Details of the shroud construction are shown in Figure 3.4b.

The fuel rod bundle was a modified 4 x 4 square array (a subsection of a PWR 17 x 17 rod assembly). The corner positions of the array were replaced with angled Zircaloy pieces ("carriers") that routed and protected the instrumentation hard lines leaving the bundle. Ten of the remaining 12 positions were occupied by 8 instrumented PWR-type unirradiated rods and 2 noninstrumented, unirradiated rods. These unirradiated rods were fabricated at PNL and contained 1.76% enriched, sintered, and dished uranium dioxide pellets with a 97% theoretical density (TD).

The eleventh position in the array was occupied by a three-cycle irradiated PWR rod from the HBR reactor. The initial uranium dioxide pellet enrichment was 2.55%, and the rod had a nominal (rod-averaged) burnup of 28 MWd/kgU and a cooling time of 13 years. The enrichment of the unirradiated rods (1.76%) was selected so that each unirradiated rod and the irradiated rod would produce the same nuclear power.

The twelfth position in the array was occupied by a stainless steel rod in a Zircaloy guide tube, and this rod simulated the gamma thermometer used in the FLHT-4 test. This rod position contained the dummy thermometer, rather

FIGURE 3.3. FLHT-5 Test Train Layout. Levels are elevations in inches from the bottom of the fuel column.

(a)

(b)

FIGURE 3.4. Cross Section of FLHT-5 Fuel Rod Bundle and Shroud: (a) Full Cross Section; (b) Detail of the FLHT-5 Shroud (dimensions in inches)

than a fuel rod, to maintain neutronic and thermal similarities between FLHT-4 and FLHT-5, which would later facilitate direct comparisons of test results.

The unirradiated fuel rods, the hardline carriers, and the grid spacers were instrumented with thermocouples at 10- to 20-cm axial intervals. The eight grid spacers, located every 53 cm (21 inches), held the fuel assembly together mechanically. The central four grid spacers were made of Zircaloy instead of the Inconel used in FLHT-4. The location of these grid spacers is shown in Figure 3.5.

3.2.2 Shroud

As shown in Figure 3.4, the multilayered shroud consisted of an octagonal Zircaloy liner. This liner was surrounded by interlocking tiles of sintered low-density zirconium dioxide insulation. Zircaloy transition pieces called saddles supported the tiles from the outside and provided the transition from the octagonal geometry to a round exterior cross section. Two concentric Zircaloy tubes were located outside the saddles. These were called the inner and outer "rounds"; they were separated by a sealed pressurized gas annulus. Eight independent wire wraps spiraled up the inner round inside this annulus. Four of the wraps formed continuous temperature sensors that monitored the heatup of the rounds. The wire continuity/resistivity and cavity pressure of this arrangement were monitored, and the whole annulus was designated as the molten material penetration detector (MMPD). The MMPD was capable of detecting the advance of molten material from the bundle toward the rounds; such an occurrence could jeopardize the pressure tube. Therefore, the loss of MMPD cavity pressure was a criterion for automatic reactor shutdown and test termination.

The zirconium dioxide insulating tiles for the shroud were the type used in previous FLHT tests. These tiles have proven to be thermally effective and mechanically resilient. The properties of the tiles (from vendor data) are as follows:

- physical density - 1440 kg/m^3 (30% of theoretical)
- fraction interconnected porosity - very high
- thermal expansion - $1.1 \times 10^{-5} \text{ m/m-K}$ (300 to 1700 K)
- melting temperature - 2866 K
- flexural strength - 8.3 MPa (length parallel to fiber orientation)
- estimated heat capacity - 700 J/kg-K
- thermal conductivity -
 - 523 K = 0.26 W/m-K
 - 873 K = 0.27 W/m-K
 - 1273 K = 0.37 W/m-K
 - 1673 K = 0.44 W/m-K
 - 1923 K = 0.52 W/m-K.

FIGURE 3.5. Axial Layout of the FLHT-5 Fuel Bundle. Levels are elevations in inches from the bottom of the fuel column.

Thermocouples were located in the shroud between the liner and the insulator at various azimuthal and axial locations and also on the outside of the Zircaloy saddles. A limited number were also placed on the outside of the shroud. Whereas the fuel rod and liner thermocouples failed very soon after accelerated oxidation and high temperatures were achieved within the assembly, the saddle thermocouples survived the test because they were exposed to only a small amount of steam and much lower temperatures. These saddle thermocouples recorded valuable information on transient radial heat flow and axial location of the oxidation reaction.

The high-temperature, Type C, thermocouples for the cladding and liner consisted of tungsten-rhenium wire couples, with beryllium oxide insulation and a zirconium-tantalum duplex sheath. Stainless-steel-clad compensating leads (hardlines) led through the closure plug to a potted junction with soft-line compensating leads proceeding to a 339 K reference temperature junction box. Copper lead wires proceeded from there to the DACS. The cladding thermocouples were strung down slots in the fuel pellets on the inside of the cladding. The liner thermocouples were strung along the outside of the saddles to the axial level of measurement where they passed through holes in the saddles and insulation and were spot-welded to the outside of the liner.

The low-temperature, Type K, thermocouples for the saddles and outer shroud were chromel-alumel couples with magnesium oxide insulators and stainless steel or Inconel sheaths. The route of their compensating leads was similar to that for the Type C thermocouples.

3.2.3 Plenum Section and Closure Plug

The plenum section extended from the top of the shroud (just above the fuel rods) to the closure plug at the top of the reactor pressure tube. It conducted the assembly effluent to the top of the reactor. An axial layout of the plenum is shown in Figure 3.6 and a cross section displaying radial dimensions is illustrated in Figure 3.7. The plenum basically consisted of a 2.5-cm (1.0-in.) ID Zircaloy tube (approximately 4-m long). A 1-cm (0.375-in.) outside diameter (OD) stainless steel rod was suspended inside the Zircaloy tube along its entire length; as mentioned previously, this suspended rod was the deposition rod.

The plenum tube was heated and insulated to prevent steam condensation on the tube wall. Beginning at level 171.4, four sequential sections of heater wire were spirally wrapped on the outside of the plenum tube, each with an approximate (tube) length of 0.83 m for a total heated length of 3.3 m. Each heater section had a main heater plus a backup; all were separately controlled by thermocouples through a master controller (located near the reactor on the building main floor).

The heated length was insulated with annular pieces of low density zirconium dioxide insulation (2.0-cm or 0.8-in. wall thickness), and the insulation was surrounded by an outer Zircaloy tube with a 7.6-cm (3.0-in.) OD. Above the heated length, an evacuated annulus replaced the zirconium dioxide insulation, and this continued through the closure plug. (Heater tape and

FIGURE 3.6. Axial Layout of the FLHT-5 Plenum and Closure Regions

FIGURE 3.7. Transverse Cross Section of the FLHT-5 Heated Plenum (units in inches)

wrapped fiber insulation were used on the effluent line from the closure plug through the SCC and on into the ECM.)

The plenum assembly was centered in the reactor pressure tube, which had an ID of 10.35 cm (4.07 in.). The 1.36-cm (0.54-in.) bypass flow annulus between the plenum, shroud and the pressure tube contained the four fuel assembly makeup flow lines: a plenum drain/blowdown line that came down through the closure from the reactor service space and entered the plenum near the top of the shroud; thermocouple and pressure transducer (PT) leads; and two TDR tubes, which functioned as manometer tubes and measured the "collapsed" coolant level in the assembly.

The plenum instrumentation included the control thermocouples for the heaters, eight thermocouples on the deposition rod (see Figure 3.8), and thermocouples inside and outside the insulation at various axial locations. Also pressure transducers measured the cavity pressures of both upper and lower plenums.

The closure plug was threaded onto a fixture on the top of the pressure tube (above the return-flow nozzle) called the "stump body." A stump body extension was welded onto this fixture to permit the FLHT-5 closure seal to be bolted in place. Flexible (carbon) packing seals maintained a pressure seal with the load of the expanding/contracting test train. The closure plug body

FIGURE 3.8. Schematic of the Deposition Rod for FLHT-5

had machined penetrations to permit passage of the main effluent line, three bundle makeup lines (one of which branched into two lines below the closure plug), four instrument hardline feed-through plugs, and the drain/blowdown line. Axial and transverse cross sections showing the basic design of the closure are shown in Figures 3.9 and 3.10, respectively.

3.2.4 Bundle Inlet Fixture

The axial layout of the bundle inlet region is shown in Figure 3.11. Below the bottom of the fuel column (Level 0), the fuel rod end caps had a special Zircaloy-to-stainless steel transition piece that facilitate bringing the cladding thermocouple leads out of the rods. These pieces locked into the tie plate. Belleville washers that bore the live load of the expanding/contracting shroud were located below the tie plate, inside the shroud. The flange for the four bundle coolant inlet locations was located below these washers.

3.3 STEAM CLOSURE CAVE AND EFFLUENT CONTROL MODULE HARDWARE DESIGNS

The SCC and the ECM were interconnected modules that were located on the deck plate. They provided flow and pressure control, shielding, and secondary confinement for the portion of the effluent line that proceeded across the top

FIGURE 3.9. Axial Schematic of the FLHT-5 Closure Plug

FIGURE 3.10. Transverse Cross Section of the FLHT-5 Closure Plug

of the reactor. The simplified effluent flow paths through the SCC and the ECM were shown in Figure 3.1. These flow paths are shown in more detail in Figure 3.12 to emphasize the main SCC/ECM components and their functions.

The effluent line outer/inner diameters were reduced in stages from 2.54/2.38 cm to 1.27/1.09 cm as the line emerged from the deck plate and entered the SCC. The smaller diameter line then proceeded past the antideposition device (ADD)² (a chemically polished high-speed nozzle with injected hot nitrogen) and the sample bomb manifold and entered the ECM confinement and shielding cave. The line proceeded inside the cave to a condenser and separator. The condenser split the effluent flow into a liquid (condensate) stream and a flow of noncondensable gases.

Nitrogen (50 standard L/min) was injected into the effluent in a split stream just upstream and just downstream from the condenser and combined with 20 L/min of nitrogen injected at the ADD and the two TDR locations. The non-condensable gaseous effluent proceeded through a shielded iodine filter and then past a gamma spectrometer on exiting the ECM; it rejoined the liquid stream after reentering the deck plate. The liquid steam proceeded past a gamma spectrometer before rejoining the gaseous stream.

² The purpose of the ADD was to keep a short section of the effluent line free of fission product deposition to facilitate gamma monitoring of the effluent without interference from deposits. In FLHT-4, the ADD design used hot nitrogen that was injected through a porous ring surrounding the effluent line. For FLHT-5, this design was changed to a nozzle injection across a polished section of the effluent line.

FIGURE 3.11. Axial Layout of the FLHT-5 Bundle Inlet Region

Two sample lines branched off the main effluent flow line inside the ECM cave. These lines had an ID of 0.3 cm (0.125 in.) and an OD of 0.61 cm (0.25 in.). The first, upstream from the condenser, conducted a sample through a series of pressure regulators to the vacuum chamber of a quadrupole mass spectrometer (MS). The mass spectrometer was installed to measure hydrogen/steam, hydrogen/nitrogen, and xenon/nitrogen ratios, from which to deduce the steam flow, hydrogen flow, and the noble fission gas release rate.

The second sample line (downstream from the condenser) conducted a sample (about 1% of the flow) to a heated palladium hydrogen meter (PHM), which provided a measure of hydrogen partial pressure in the sample stream. The sample went through a chiller and a drier, and then through a thermal conductivity meter (TCM), which measured the conductivity of the hydrogen plus nitrogen mixture, from which the hydrogen fraction could be deduced. The sample streams then rejoined the main gaseous effluent flows to the waste line.

3.3.1 Steam Closure Cave

The SCC was essentially a lead box with walls and lid that were 10-cm (4-in.) thick. The overall (outside) dimensions of the box were 1.2-m long x 0.6-m wide x 0.8-m high.

A layout of the SCC is shown in Figure 3.13a. The cave was a lead box that shielded the effluent steam line, the deposition rod blind flange, the sample bomb manifold, the makeup flow piping, and the manifold of rod vent lines and pressure transducer (the manifold is not shown in Figure 3.13a). The 1.1-cm ID steam line formed an S-bend after teeing off the deposition rod blind flange, thus permitting the deposition rod scanning port to be shielded from the remainder of the line, which was important during the gamma scan of the deposition rod.

The sample bombs were connected to the effluent line through air-actuated solenoid valves. These valves were opened and closed by the DACS, according to a preset program. A top view of the sample bomb manifold is shown in Figure 3.13b.

3.3.2 Effluent Control Module

The ECM provided equipment for controlling the pressure, the condensation of steam, and the flow of the steam, hydrogen, and fission gas effluent from the assembly. It also provided equipment for sampling and monitoring the effluent in various ways and provided shielding and secondary confinement of the effluent line as it proceeded along the top of the NRU reactor deck plate.

The basic components of the ECM were a steam condenser, a pressure control valve to maintain system pressure, and a nitrogen injection system to facilitate pressure control. These components were contained in a lead-lined "lower cave" with outside dimensions of 1.27-m long x 0.71-m wide x 0.7-m high. An "upper cave" of similar dimensions was added for FLHT-5 directly above the lower cave and contained the mass spectrometer components, i.e., roughing pumps, high-vacuum pump, vacuum chamber, and the quadrupole sensing

FIGURE 3.13. Steam Closure Cave: (a) Axial Schematic; (b) Detail of Sample Bomb Manifold (top view)

head. As in the SCC, the lead wall of the caves was 10-cm (4-in.) thick, and the lead was contained between 0.6-cm (0.25-in.) steel plates.

The lead caves also contained the two sample lines described previously. The first line, upstream from the condenser, routed a small sample past the mass spectrometer and then returned the sample to the waste line. The second sample line teed off the gas effluent line downstream from the separator and routed a sample past the PHM and the TCM.

Another shielded component was a multistage iodine filter, located on the main noncondensable gas line downstream from the separator. The filter contained various sequential filters for capturing iodine in organic and inorganic forms.

The caves, iodine filter, and the measurement instruments were surrounded by an airtight confinement box made of 0.32-cm (0.125-in.) steel plating with outside dimensions of 2.0-m long x 1.8-m wide x 1.9-m high. The box was maintained at a slightly negative pressure relative to the reactor building. The inlet air flow was from the reactor service space, and the outlet air flow returned to the reactor service space after passing through a HEPA filter and a charcoal (iodine) filter.

4.0 TEST OPERATIONS

The sequence of test operations for FLHT-5 is reviewed in this section.

4.1 TEST OPERATIONS PLAN

The FLHT-5 test was planned to be different from the previous FLHT-4 test in ways that would promote Zircaloy relocation, debris formation, and possible flow blockage. Throughout the boilaway transient, the bundle was to be operated at 30-kW assembly nuclear power versus the 23 kW used in FLHT-4 and FLHT-2. The higher power would expose more Zircaloy to steam and cause a more rapid boildown and faster heatup after dryout. The result would be that the cladding temperature would increase above the melting temperature (2100 to 2400 K) with relatively less premelting oxidation. This was predicted to enhance rundown and relocation of molten Zircaloy and liquefied uranium dioxide, possibly leading to complete flow blockage and cessation of the autocatalytic oxidation reaction. The plan was to continue the transient until the oxidation of assembly Zircaloy had ceased.

The basic phases of the test remained the same as for the FLHT-4 test as shown below:

- hardware installation and checkout
- commissioning of test train and ECM (in-situ hydrogen flow test)
- preconditioning of fuel assembly
- boilaway/severe damage transient
- post-test activities.

Within most phases of FLHT-5 there were some assembly and operational differences from the FLHT-4 experimental plan; the differences are noted in the following sections.

These phases are described below, together with major results.

4.2 HARDWARE INSTALLATION AND CHECKOUT

Pretest activities for FLHT-5 included installation of the irradiated HBR rod in the test train, installation of the test train in the NRU reactor, and installation of the SCC/ECM on the reactor top face. The mass spectrometer (MS) system in the ECM was replumbed to eliminate the possibility of sample line condensate falling back into the vacuum pumps, which occurred in FLHT-4.

Borated polyethylene slabs were installed around the gamma spectrometers to prevent the overloading of the detectors by the delayed neutron activity in the effluent, which occurred during FLHT-4. For FLHT-5, CRNL also installed a variety of neutron dose rate monitoring equipment around the ECM.

The test train arrived at CRNL in April 1987. The irradiated HBR rod was installed in a central bundle position. The test train was installed in the L-24 reactor position on May 5, 1987, and the closure seal was completed and found to be leak tight at 1.8 MPa (250 psig) on May 6, 1987. The instrument leads were then checked for continuity and were connected via cables to the DACS. The following instruments were found to be nonserviceable:

<u>Inoperative Sensor Type</u>	<u>Location and Description</u>
TC-Type C	Rod 1B, level 72
TC-Type C	Rod 4C, level 72
TC-Type C	Rod 3A, level 120
TC-Type K	Saddle, level 56, 0°
TC-Type K	Saddle, level 80, 270°

The SCC and ECM were installed on May 7. Water and electrical services were connected to them, and the hydrogen flow measurement systems were calibrated by May 9. The ECM piping passed a leak test at 2.4 MPa.

The gamma spectrometer installation was postponed until after preconditioning and was completed May 13, just before the boilaway transient. Seven spectrometers were installed, including two at the ADD, three at the iodine filter, one at the condensate, and one at the noncondensable waste lines.

4.3 COMMISSIONING AND IN-SITU HYDROGEN FLOW TEST

As part of the commissioning of the test train and SCC/ECM hardware, in-situ calibration and response checks were performed on the hydrogen measurement systems by injecting hydrogen gas through a specially calibrated mass flowmeter (MFM) into the ADD. In the ADD, the hydrogen gas was mixed with prototypic nitrogen injections and the mixture was sampled and analyzed for hydrogen, as would be done in the boilaway transient. Two levels of hydrogen injection rate were selected: 90 standard L/min (equivalent to expected peak flows in the boilaway transient) and 50 standard L/min. At the PHM/TCM sample line tee, these two injection rates corresponded to 56% and 42% (volume %) hydrogen in the hydrogen/nitrogen mix.

The PHM was first tested at varying pressures of static hydrogen and hydrogen nitrogen gas and found to be operational. Then the nitrogen injections were started (20 L/min at the ADD and 50 L/min at the condenser) and finally, the ADD hydrogen injection was started. The delay, rise time, and final levels of the noncondensable turbine flowmeter (NCTFM), TCM, PHM, and mass spectrometer, relative to the MFM are shown in Figures 4.1 and 4.2 and listed in Table 4.1. The mass spectrometer corroborated the MFM hydrogen injection rate within 5%. The PHM agreed closely with the MFM value; and the NCTFM and TCM were biased high and low by 10% to 20% and 10% to 15%, respectively. These biases and lag times are noted here but have not been used to adjust the instrument data presented in Section 5.0.

FIGURE 4.1. Performance of Thermal Conductivity Meter (TCM), Palladium Hydrogen Meter (PHM), Noncondensable Turbine Flowmeter (NCTFM), and Mass Spectrometer (MS) Relative to the Mass Flowmeter (MFM) During In Situ Hydrogen Flow Test at 90-L/min Hydrogen Injection

FIGURE 4.2. Performance of Thermal Conductivity Meter (TCM), Palladium Hydrogen Meter (PHM), Noncondensable Turbine Flowmeter (NCTFM), and Mass Spectrometer (MS) Relative to the Mass Flowmeter (MFM) During In Situ Hydrogen Flow Test at 50-L/min Hydrogen Injection

TABLE 4.1. Results of In-Situ Hydrogen Flow Tests

<u>Hydrogen Nominal Flow Rate, Standard L/min</u>	<u>Instrument</u>	<u>Bias in Hydrogen Flow Rate (relative to MFM), %</u>	<u>Approximate Lag Time for Instru- ment Response, s</u>
50	NCTFM	+20	<10
50	PHM	-1	230
50	TCM	+50	800
50	MS	-5	45
90	NCTFM	+25	<10
90	PHM	-5	230
90	TCM	+10	800
90	MS	+5	45

Other commissioning activities included the following:

- checking calorimetry flow (126-g/s water flowing through the test train and ECM)
- checking plenum drain and dryout
- checking TDR operation during assembly drain.

The piping, valves, and flowmeters were verified operational with 126-g/s water flow (the calibration flow rate). The plenum was drained in <2 min. The plenum heaters and controllers were checked at 690 K and 1.38-MPa steam flow and were found to dry the plenum and to maintain a preset temperature profile.

The assembly was drained and the TDRs were found to track each other successfully (Figure 4.3).

4.4 PRECONDITIONING OPERATION

Preconditioning (at full reactor power) for FLHT-5 was extended from 1 EFPH (for FLHT-4) to 5 EFPH. The planned maximum assembly power was maintained at 700 kW, and two power cycles (going from 100% to 1% and returning to 100%) were planned to ensure adequate fuel pellet cracking in the unirradiated rods.

The purposes of the preconditioning operation were to prototypically crack the unirradiated fuel pellets and to build up radioactive iodine isotopes in the fuel. The target parameters were an assembly power of 700 kW [27-kW/m peak rod linear heat generation rate (LHGR)] and a time at power of 5 h. The planned/actual operating conditions are summarized in Table 4.2.

The power was raised in steps to 100% full reactor power (equivalent to 700-kW assembly power) beginning at 11:16 a.m., May 12, 1987. Power was rapidly decreased twice to ~1% to promote cracking, was returned to 100%, and then was held until 6:30 p.m. for an integrated power of 3910 kW-h.

FIGURE 4.3. Performance of TDR Coolant Level Sensors During Fuel Bundle Drain

TABLE 4.2. FLHT-5 Preconditioning Operating Conditions

Test Section	Planned	Actual
Coolant	U-2 loop water	U-2 loop water
Inlet temperature	311 K	313 K
Peak outlet temperature	420 K	420 K
Flow rate	1.51 kg/s	1.55 kg/s
Outlet pressure	1.38 MPa	1.4 MPa
Total assembly power	700 kW	700 kW
Maximum linear rod power	27.8 kW/m	26.1 kW/m
Average linear rod power	17.4 kW/m	17.4 kW/m
Axial peaking factor	1.51	1.5
Radial peaking factor	1.06	1.0
<u>Bypass Annulus</u>		
Coolant	U-2 loop water	U-2 loop water
Inlet temperature	311 K	310 K
Outlet temperature	315 K	320 K
Flow rate	1.0 kg/s	1.0 kg/s
Outlet pressure	1.38 MPa	1.4 MPa

A plot of the assembly power during the preconditioning appears in Figure 4.4. Typical bundle and shroud temperatures at full assembly power are shown in Table 4.3.

FIGURE 4.4. Bundle Nuclear Power During Preconditioning Operation (zero time = 11:15:12 DACs time on 5/12/87)

TABLE 4.3. Axial Midplane Temperatures During Preconditioning

Level	Temperature, K				
	Cladding	Liner	Saddle	Carrier	Shroud
80	715	389	573	--	--
72	Failed	391	666	362	320
64	635	369	617	348	--

4.5 BOILAWAY TRANSIENT OPERATION

This operation was described at the beginning of this subsection. The plan for FLHT-5 was to maintain the 30-kW assembly power and 1.26-g/s makeup flow throughout the boilaway/severe damage transient and 60 min beyond the first cladding temperature escalation above ~2500 K, even if hydrogen generation ceased before the 60 min had passed.

The coolant pipe connection changes from preconditioning to transient operation were completed, and the reactor was brought to low neutron power at 10:50 p.m., May 13, 1987. Power calibrations and flow checks were carried out at 11:40 p.m. The assembly was calibrated at 30 kW using 126-g/s coolant. A

steady-state was established at 1.38-MPa system pressure, 11.3-g/s (90-lb/h) coolant flow, and a coolant level at the ~2.9-m elevation.

At 0:54:17 a.m. (DACS time), May 14, 1987, the inlet flow was reduced to 1.26 g/s (10 lb/h) (507 s on the plot), initiating the boilaway transient. The coolant level dropped rapidly, eventually achieving a steady-state level of about 0.76 m. First escalations to >2500 K were noted in the bundle at 01:02:30, ~100 s before the steady-state steaming level was achieved.

The 30% added nuclear power and higher steaming rate at escalation relative to previous FLHT tests had the following two effects: 1) the peak cladding temperatures were higher (>2600 K versus >2500 K) and 2) the axial extent of cladding involved in first escalation was much greater (~2 m versus 1 m). The measurable hydrogen generation increased sharply at the same time as the temperature escalations and continued at approximately full-conversion levels until 1:47:17, which corresponded well with the estimated arrival of the oxidation front at the rod plenum and upper end-cap regions. Nuclear power and makeup flow were continued for ~15 min after the reduction in hydrogen flow, but hydrogen generation never restarted and the bundle was cooling continuously. The reactor was shut down and the test terminated at 2:02:37 a.m., just over 60 min after the first escalations.

4.6 POST-TEST ACTIVITIES

The post-test examinations for FLHT-5 included activities that were similar to those planned for FLHT-4:

- deposition rod removal and gamma scanning
- ECM condenser flush and waste line gamma scanning
- SCC/ECM removal and sample bomb recovery
- test train removal and gamma scanning
- flux wire gamma scanning
- stack release analysis
- fuel assembly visual inspection and sectioning.

The deposition rod was built in pinned sections to facilitate hot cell entry and storage. The plan was to dispose of the contaminated internal components of the ECM (condenser, valves, piping, etc.) after the FLHT-5 test.

A bridge structure and multi-ton cask were installed for deposition rod removal and scanning. The rod bale was attached to the lifting winch inside the receiving canister and spectral gamma scanning began at 7:45 p.m., May 14, 1987, 18 h after the transient. One 1-min scan was performed every 5 cm of deposition rod length.

On May 15, 1987, the test train-to-SCC piping connections and test train instrument leads were severed, and the SCC and ECM were removed from the deck plate. The test train was also removed on May 15, 1987. The upper part of the plenum had a heavy deposition of radioactive material that made it impossible to gamma scan the fuel bundle, although a gamma scan of the plenum was performed on May 17, 1987.

The test train was stored in the rod bay, and a visual examination was conducted there in November 1987.

5.0 TRANSIENT TEST RESULTS

In this section, the major test results will be presented in relation to each other and to the test conditions. Appendix A contains listings of the operable instruments for the FLHT-5 boilaway transient.

The chronology of major events in the boilaway transient is summarized in Table 5.1. Note that the zero-time reference for this chronology is 0:45:50 a.m., May 14, 1987 (DACS time). This reference also applies to the data plots presented in this section. It was chosen to display the pre-transient state before the makeup flow was reduced to 1.26 g/s (10 lb/h) at 0:54:17 DACS time (507 s on the plots). A list of planned and actual operating conditions throughout the transient is shown in Table 5.2

The test results will be discussed under the following headings:

- makeup flow and coolant level
- temperature measurements^(a)
- hydrogen flow measurements
- cavity pressure measurements
- fission product release measurements.

5.1 MAKEUP FLOW AND COOLANT LEVEL

A pretransient quasi-steady-state condition was established with 30-kW bundle nuclear power, an inlet flow of 11.3 g/s (90 lb/h), a pressure of 1.4 MPa, and a coolant level at an elevation of 2.9 m. Reducing the flow to 1.26 g/s started an immediate boildown, which eventually reached a constant elevation of 0.76 m. The first temperature escalations occurred in the Level-120 region at 01:02:14 DACS time, 477 s after boildown was started. These escalations are shown in Figure 5.1, where the coolant level (from cladding dryout assessment), the TDR readings, and the Level-80 cladding temperatures are all plotted.

The TDR measured the collapsed (subcooled) coolant level and, therefore, the TDR measurements do not directly correlate to the coolant level in the assembly; but there is agreement on the boildown initiation time and the

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- (a) The cladding thermocouple data (above level 48) presented in Figures 18, 19, 24, 28, and 31 are valid only up to the point of escalation (~1000 s on the plots). Beyond that time, the thermocouples melted and rejunctioned; thus, any subsequent data represent temperatures at some unknown lower elevation or the data are spurious. Because these data are invalid, post-escalation cladding thermocouple data beyond ~1200 s have been removed from the figures listed above.

TABLE 5. FLHT-5 Test Sequence (May 14, 1987)

Date and Time (h:m:n:s)	Relative Time (s) (a)	Event	Remarks
<u>May 12, 1987</u>			
11:16		<u>Preconditioning</u>	
		Began preconditioning, reactor started	Slowed ramp to 125-MW reactor power
13:57		Reached reactor full power	Bundle power (698 kW) Integrated power (1024 kWh)
14:22		Began rapid decent to low power	Thermal cracking of pellets
14:26		Reactor at 1% power, began ascent to full power	
15:10		Reached reactor full power	Integrated power (1658 kWh)
15:14		Began second rapid decent to low power	
15:19		Reactor at 1% power, began ascent to full power	
16:34		Reached reactor full power	Integrated power (2352 kWh)
18:31		Began decent to zero power	
18:57		Reactor shutdown, preconditioning complete	Total Integrated power (3910 kWh)
<u>May 13, 1987</u>			
19:20		<u>Test Preparation and Startup</u>	
		Bollaway test preparations completed	
22:50		Began startup, reactor to low neutron level	Decay of xenon poisoning allowed startup
23:45		Adjusted reactor power for bundle calorimetry	Bundle power at 30 kW, flow at 126 g/s
<u>May 14, 1987</u>			
00:10:00	-1250	<u>Pretransient and Transient</u>	Began plenum drain
		Began bundle coolant flow reduction to pretransient state	
00:12:36	-1994	Bundle coolant flow at 12.6 g/s (100 lbm/h)	Closed plenum drain valve
00:16:16	-1774	Bundle coolant boiling swell began	Bundle pressure buildup began
00:18:45	-1625	Pressure surge peaked at 1.94 MPa (266 psig)	
00:22:30	-1410	Bundle coolant level stabilized	
00:24:00	-1300	Bundle coolant flow at 11.3 g/s (90 lbm/h)	
00:24:22	-1288	Began plenum heating	
00:45:50	0	Zero time for 5000-s data plots	

TABLE 5. (contd)

Date and Time (h:min:s)	Relative Time (s) (a)	Event	Remarks
May 14, 1987		Pretransient and Transient (contd)	
00:50:40	290	Plenum and deposition rod temperatures stabilized	
00:54:17	507	Initiated bundle coolant flow reduction to 1.26 g/s (10 lbm/h)	Began bollaway transient
00:54:30	520	Bundle coolant flow at 1.44 g/s (11.4 lbm/h)	
00:57:00	670	Bundle coolant flow at 1.26 g/s (10.0 lbm/h)	
00:59:03	793	Hydrogen detected	Noncondensable flowmeter
01:02:14	984	Peak cladding temperature to 1700K (2600°F)	Level 96
01:02:14	984	Autocatalytic oxidation began	Full conversion of inlet flow to H ₂
01:02:29	999	Rod cladding collapse/failure	50-psi Compression
01:02:34	1004	Cladding temperature 2475K (4000°F)	Started elapsed time clock
01:02:37	1007	Liner breached	Minimal pressure disturbance
01:07:06	1276	Heated plenum cavity breached	System pressure disturbance
01:08:56	1386	Level 48 oxidation excursion	Downward burn ended, upward burn started
01:33:27	2857	Level 136 oxidation excursion	Inferred from saddle TC
01:47:17	3687	Hydrogen generation ceased	48-min sustained H ₂ production
02:02:37	4607	Reactor shutdown. Inlet flow to zero	1-hr high temperature hold ended
19:45		Initiated deposition rod scanning	1-min counts every 5 cm
May 15, 1987		End deposition rod scanning	
01:45			
May 17, 1987		Started test train plenum scan	Scan only included plenum (not fuel assembly) because high upper plenum activity prevented further lifting of the test train
12:12		stopped test train plenum scan	
17:44			

(a) Time in seconds, relative to zero time for the 5000-s data plots..

TABLE 5.2. FLHT-5 Boilaway Operating Conditions

<u>Test Section</u>	<u>Planned</u>	<u>Actual</u>
<u>Coolant</u>	Makeup water	Makeup water
Inlet temperature	361 K	360 K
Makeup flow rate	1.26 g/s	1.26 0.13 g/s
Steam outlet temperature	<644 K	<644 K
Outlet pressure	1.44 MPa	1.46 MPa
Peak cladding temperature	2800 K	>2600 K
Total fission power	30 kW	30 kW
Average rod LHGR	0.746 kW/m	0.75 kW/m
Maximum rod LHGR	1.194 kW/m	1.12 kW/m
Axial peaking factor	1.51	1.5
Radial peaking factor	1.06	1.0
<u>Bypass Annulus</u>		
Coolant	U-2 loop water	U-2 loop water
Water inlet temperature	361 K	362 K
Water outlet temperature	374 K	374 K
Flow rate	1.0 kg/s	1.0 kg/s
Outlet pressure	1.38 MPa	1.4 MPa
Maximum pressure tube temperature	468 K	376 K
<u>ECM and Outlet Piping</u>		
ECM condenser water flow	0.315 kg/s	0.3 kg/s
ECM chiller water flow	0.063 kg/s	0.06 kg/s
ECM nitrogen carrier flow	70 L/min	73 L/min
ECM control pressure	1.38 MPa	1.41 MPa
Outlet piping nitrogen dilution flow	4000 L/min	4000 L/min

relative rate of decrease of the coolant level. At about 1000 s on the plots (the time of initial temperature escalations), the TDR trace changes direction (although boildown and coolant level decrease are not complete) and rises sharply. There is a more gradual rise through the remainder of the transient (from 2000 to 5000 s). These rises in TDR readings, which are not matched by rises in coolant level, probably result from increased back pressure in the bundle caused by progressively greater cumulative flow restrictions as severe damage progresses in the bundle.

5.2 TEMPERATURE MEASUREMENTS

In considering the measurements recorded by the test train thermocouples, it is important to realize that the leads for the thermocouples in the fuel rods and carriers proceed down, then out the inlet fixture and up along the bypass annulus, whereas the thermocouple leads from the shroud, including liner thermocouples and saddle thermocouples, rise and exit through the top of the assembly. Therefore, when the temperature escalation first occurred and the oxidation front moved rapidly downward from approximately Level-96, the cladding thermocouples for levels above 96 were rejunctioned by melting at undefined lower levels and were thereby invalidated. The rush

FIGURE 5.1. Coolant level (from dryout data and TDRs) and Level-80 Cladding Temperature During Boildown/Heatup Phase

of gas into the insulation following the rupture of the liner during the initial escalations may have burned out upper liner thermocouples. The cladding thermocouples from Level-96 downward demonstrate the rapid initial downward progression of the burn front as shown in Figure 5.2.

Most saddle thermocouples, however, survived the entire 60-min transient; these thermocouples are used to interpret the upward progression of the burn front. That progression is verified by sequential peaking in plots of saddle thermocouples versus time at progressively higher levels as shown in Figure 5.3. In this figure, peaks in the saddle temperatures occur at 1740, 2100, and 2660 s at Levels 96, 112, and 128, respectively. This slowly moving front is estimated to have burned out at the tops of the rods (Level-153) at 3570 s DACS time, which is just before the hydrogen generation ceased at ~3600 s.

The saddle temperatures also demonstrate that the assembly developed an azimuthal variation in temperature but only after escalation began. The development of a "hot" and "cold" side is shown in Figures 5.4 and 5.5.

In the plenum, thermocouple output values at eight axial levels along the deposition wall stayed very close to their counterparts at each axial level on the plenum liner. A history of the plenum temperature is shown in Figure 5.6. Up to ~1400 s, these temperatures are dominated by the action of the plenum heaters. At 1400 s, the rupture of the plenum insulation cavity drew hot effluent into the insulation at Level-178, heating it rapidly. This

FIGURE 5.2. Cladding Temperatures During Boilaway Transient (showing initial downward progression of the burn front)

FIGURE 5.3. Saddle Temperatures Throughout the Boilaway Transient (showing the upward progression of the burn front)

FIGURE 5.4. Development of the Azimuthal Temperature Gradient at Level-80

FIGURE 5.5. Development of the Azimuthal Temperature Gradient at Level-128

rapid heatup and slow cooldown account for the first temperature peak at Level 178. The second peak, at Level-178 around 3800 s, is related to the approach of the burn front to the top of the assembly.

5.3 HYDROGEN FLOW MEASUREMENTS

Four instruments were calibrated to measure hydrogen flow during the transient: the TCM, PHM, NCTFM, and mass spectrometer. At 8:30 p.m. on

FIGURE 5.6. Plenum Wall Temperature History Throughout the Boilaway Transient

May 13, 1987, 5 h before the boilaway transient, the high-vacuum pump for the mass spectrometer chamber ceased to operate, either because of a seized bearing or a rupture in the electrical power controller. A replacement controller could not be located on such short notice, and the pump itself was sealed within the confinement/shielding. Therefore, the decision was made to proceed with the mass spectrometer disabled rather than to delay an imminent test by 1 to 2 days for pump repair. Therefore, only measurements from the other three instruments are presented below.

The delay between the NCTFM response and hydrogen generation in the assembly was ~10 s, as deduced during the FLHT-4 test and confirmed during the in situ hydrogen flow test. Accordingly, the NCTFM output is plotted without time lag correction in comparison with cladding temperatures (Level-96) in Figure 5.7. A sharp increase in hydrogen generation was simultaneous with the onset of temperature escalation and autocatalytic oxidation in the fuel assembly. Greater than 95% of the hydrogen generation occurred after the cladding began to melt, and the hydrogen generation reduced at ~3600 s, over 900 s before the termination of the test. The end of the hydrogen generation corresponded with the time when the burn front reached the rod plenum/end cap region at Levels 144 to 151, where the lack of heating probably quenched the accelerated oxidation reaction, leaving only a cumulative "smoldering" production along the length of the burned-out fuel bundle.

The NCTFM output can be transformed into a direct estimate of hydrogen mass flow by accounting for the temperatures and pressures of the nitrogen injection and also those of the noncondensable gas effluent. Similarly, the output of the PHM and TCM can be translated to mass flows by reference to the measured nitrogen injection rate. The results of these calculations are shown in Figure 5.8, where hydrogen flow rates as deduced from all three sensors are plotted versus time. The same relative lag times and response characteristics that prevailed during the in-situ test are evident in this figure. No corrections have been made to these plotted data. On the basis of the in-situ

FIGURE 5.7. Noncondensable Turbine Flowmeter (NCTFM) Performance and Level-96 Cladding Temperature Throughout the Boilaway Transient

FIGURE 5.8. Hydrogen Flow Rates Deduced from the Noncondensable Turbine Flowmeter (NCTFM), Palladium Hydrogen Meter (PHM), and Thermal Conductivity Meter (TCM) and Output Throughout the Boilaway Transient

test, the NCTFM data were suspected to be 10% to 20% high with respect to hydrogen flow rate and total hydrogen generation; the TCM and PHM data were suspected to be within $\pm 5\%$ of true values for the average rate and quite low ($\sim 20\%$) with respect to the peak rate; the TCM and PHM were suspected to be low (by 10% to 15%) with respect to total hydrogen generated. Cumulative hydrogen values as measured by the three instruments are plotted in Figure 5.9. Total hydrogen generation values and peak and average flow rate data are listed in Table 5.3. Shown in the above discussion, the best estimate of total hydrogen generated in FLHT-5 is 300 ± 30 g.

This value is less than the equivalent of the conversion of all of the Zircaloy in the bundle (398 g) to zirconium dioxide but indicates $\sim 95\%$ oxidation of the exposed Zircaloy. The hydrogen generation does correspond closely to full conversion of the makeup flow throughout the period between oxidation escalation and cessation. This is emphasized in Figure 5.10, in which the hydrogen flow rate as measured by the NCTFM and the equivalent flow rate (corresponding to full conversion of the inlet flow) are plotted throughout the transient.

5.4 CAVITY PRESSURE MEASUREMENTS

The assembly cavity pressure measurements (fuel rod, shroud insulation cavity, plenum insulation cavity) indicated the time of pressure boundary rupture of various components. Level-96 cladding temperatures and the rod internal pressures are plotted in Figure 5.11 for Rod 2A. The initial increase in rod pressure from ~ 500 to 1000 s corresponds with the increase in cladding temperature following dryout. Then at ~ 1000 s the sudden increase in pressure (approaching the system pressure of 1.38 MPa) corresponds with the temperature escalation at Level 96 and signals cladding rupture.

Because the fuel rods fail well below the system pressure (from compressive loading), hot gasses within the bundle region are swept into the failed fuel rod gas volume (fuel-cladding gap, pellet slots, gas plenum and pressure transducer line), which diverts most of the effluent (steam hydrogen and fission product) from the nominal flow path through the plenum and ECM. Therefore, fuel rod rupture is accompanied by a temporary reduction in the NCTFM response.

The insulation cavity pressure follows a similar pattern, as shown in Figure 5.12. The pressure rose with post-dryout assembly temperatures from ~ 500 to 1000 s. At 1007 s, the rise in cladding temperature leads to a rise in the liner temperature and then to rupture as shown in the figure by the sudden increase in shroud pressure to system pressure.

When the upward burn started, the upper plenum cavities breached almost simultaneously at ~ 1400 s. The rupture of the unheated plenum slightly preceded that of the large heated plenum (Figure 5.6), and the failures correlated with a surge of hot gas past Level 178 as shown in Figure 5.13.

Note that the two initial peaks and valleys in the noncondensable gas flow corresponded with the shroud and plenum breaches (see Figure 5.8). It is

FIGURE 5.9. Cumulative Hydrogen Generation Deduced from the Noncondensable Turbine Flowmeter (NCTFM), Palladium Hydrogen Meter (PHM), and Thermal Conductivity Meter (TCM) and Output Throughout the Boilaway Transient

TABLE 5.3. FLHT-5 Hydrogen Generation Data

Parameter	Measured Values			
	NCTFM	PHM	TCM	Best Estimate ^(a)
Peak flow rate, mg/s ^(b)	182	114	109	152
Average flow rate, mg/s	90	70	67	75
Total hydrogen generated ^(c)	340	250	220	300 ± 30

- (a) Corrected according to the in situ hydrogen flow test results.
 (b) Multiply by 0.15 to obtain bundle oxidation power in kW.
 (c) Multiply by 0.25 to obtain % of total bundle Zircaloy (8972 g) oxidized.

FIGURE 5.10. Noncondensable Turbine Flowmeter (NCTFM) Measured Hydrogen Flow Rate Compared with Equivalent (fully converted) Makeup Inlet Flow Rate Throughout the Boilaway Transient

FIGURE 5.11. Rod Internal Pressure and Level 96 Cladding Temperature for Fuel Rod 2A During the Boilaway Transient

FIGURE 5.12. Shroud Insulation Cavity Pressure and Bundle Coolant Pressure During the Boilaway Transient

FIGURE 5.13. Plenum Shroud Cavity Pressure and Bundle Coolant Pressure During the Boilaway Transient

probable that these failures temporarily shunted steam and hydrogen into the cavities, resulting in an apparent or perhaps actual temporary reduction of hydrogen generation. It is also possible that the second valley is additionally influenced by cladding rundown following first escalation.

5.5 FISSION PRODUCT RELEASE MEASUREMENTS

Several real-time instruments were used to assess the rate and extent of fission product release during the FLHT-5 boilaway transient. These instruments included the following:

- a PNL gross gamma-dose monitor on the outside of the SCC shielding near the ADD
- the CRNL stack gas sampling and monitoring system, which detected noble gas activity in the reactor gas effluent
- an Idaho National Engineering Laboratory (INEL) gamma spectrometer, which viewed the ADD area through the lead shielding
- the PNL gamma spectrometers, which viewed the ADD and the iodine filters in the ECM through windows in the lead shielding
- the CRNL gamma spectrometer on the condensable and noncondensable effluent lines.

Data from the first three of these are available and will be discussed below. Although the PNL and CRNL spectrometers also functioned throughout the transient, the data from these spectrometers were not analyzed due to a lack of program funds.

5.5.1 Pacific Northwest Laboratory Dose Meter on the Steam Closure Cave

A Geiger counter was fixed to the outside of the SCC shielding and used as a dose meter. This dose meter measured gross gamma activity. It could not be used to identify the isotopes that contributed to the gamma dose, but it did indicate when high gamma activity first reached the ADD. Because the bundle effluent travel time from the fuel rods to the ADD was ~10 s, the first arrival of high activity at the ADD was essentially the moment of the first significant fission product release from the fuel. In Figure 5.14, the dose meter output is plotted (in mr/h) versus time along with the cladding temperature at Level 96. The effluent radioactivity was first detected at 1100 s, i.e., ~100 s after the cladding temperatures had risen to the ~2100 K cladding melting temperatures. The measured radioactivity rose to a plateau value and remained at that value even after reactor shutdown and test termination, indicating that the detected activity was dominated by deposited volatile fission products; therefore, this data can indicate only when fission product release began, not when it changed or ceased.

FIGURE 5.14. Output from the Geiger Counter Dose Meter (at the ADD ADD exterior) During the Boilaway Transient Compared with Level-96 Cladding Temperatures

5.5.2 Chalk River Nuclear Laboratories Stack Gas Monitoring System

The released noble gases (xenon and krypton) from FLHT-5 went to the reactor catch tank and from there to the gas waste line, which led through filters to the reactor stack building, about 1 km from the reactor building. A sample line led from the reactor stack gas duct through a flowmeter to a calibrated two-range Geiger counter system. This system recorded gamma activity in the sample stream on a strip chart. Noble gases were the only contributors to this activity, based on measured absence of any other species downstream from the reactor charcoal filter (see Figure 5.1). The count rate data was translated to the activity rate (Ci-MeV/s) of xenon plus krypton. The measured activity versus time is plotted in Figure 5.15 together with the hydrogen flow rate measured by the NCTFM.^(a)

- (a) The time of measurement for stack activity has been corrected for a 167-s discrepancy between the DACs time and the Geiger system chart time and has been further corrected for an estimated 406-s mean lag time from fuel bundle to reactor stack. The 406-s mean lag time adjusts for the extended rise time of the stack monitor (relative to the dose monitor) that is caused by the dispersion of the fission gas "front" within the nitrogen carrier gas as it travels 1 km from the reactor to the stack. The travel time, as judged by first detection at both monitors, is about 200 s.

FIGURE 5.15. Measured Noble Gas Activity at the National Research Universal (NRU) Reactor Stack During the FLHT-5 Boilaway Transient Compared with Bundle Hydrogen Flow. The timing of the gas release was corrected for the delay from fuel bundle to stack.

As shown in Figure 5.15, the measured stack gas activity increased at a (delay corrected) time close to that of the onset of hydrogen generation. This time corresponded with temperature escalation in the fuel bundle. The stack gas activity decreased as the burn front in the bundle progressed upward. This probably occurred because the speed of the front was decreasing and because the front was traversing a progressively less fraction of bundle inventory per unit time, therefore, releasing progressively less fission gas. The gas activity decreased more sharply at ~3700 s, when the fuel bundle essentially burned out as the burn front had traversed the entire length of the bundle. There was a further decrease at the time of reactor shutdown (4600 s) when both nuclear heating and fission gas production ceased, and the assembly began slowly cooling toward the ambient (bypass coolant) temperature.

Chalk River Nuclear Laboratories staff^(a) integrated the activity rate from Figure 5.15 with respect to time and obtained a total release activity of 4840 >780 Ci-MeV. This total can be divided by the decay corrected bundle curie inventory of noble gas isotopes to obtain an estimated fission gas release (FGR) fraction. If the decay correction is for 200 s, the bundle inventory is 5500 Ci-MeV, and the estimated FGR is 88%. If the decay correction is for 400 s, the bundle inventory is ~4200 Ci-MeV, and the FGR is

(a) Martin, I. C. 1987. "FLHT-5 Noble Gas Release." Memo to D. D. Lanning, Pacific Northwest Laboratory, Richland, Washington.

over 100%. Because at least 10% of the inventory was in fuel that was continuously water or steam cooled, this range for FGR (88% to >100%) is considered unreasonably large.

A more reasonable FGR range (based on FASTGRASS and CORSOR code predictions and Power Burst Facility (PBF) severe fuel damage test results) would be 50% to 80%. The high value of the integrated stack release activity (relative to the decay-corrected inventory) may be due to the fact that the integrated activity includes significant contributions from short-lived isotopes produced and released continuously during the 1-h transient; whereas, that integrated value is being compared with the one-time (instantaneous) bundle inventory at the end of the transient. The FGR rate and total fraction would be better estimated from release-to-birth ratios for the short-lived noble gas isotopes, derived from ECM gamma spectrometer data.

5.5.3 Idaho National Engineering Laboratory Gamma Spectrometer

One of the gamma spectrometers that viewed the ADD during the FLHT-5 test was supplied and operated by EG&G Idaho, Inc. This unit had been successfully used during the Severe Fuel Damage (SFD) 1-4 test in the PBF (Hartwell and Killian 1986). This spectrometer system included dual computers that 1) provided for remote control, 2) incorporated the technique of pulser injection with subsequent removal to provide spectrum specific energy calibration and count-rate-effects correction, and 3) implemented a combination of fast electronics and selectable collimation to yield the desired dynamic counting range.

The detector, collimator, and shield of the EG&G unit were combined on a single skid. This unit was placed on the reactor deck and aligned using a small alignment laser so that it viewed the center of the ADD. The heavy lead and polyethylene shielding of the detector unit was augmented with additional blocks of borated polyethylene surrounding the SCC in order to protect the germanium detector from fission product delayed neutrons. A control station was established ~15 m below the reactor deck on the main floor.

Detector efficiency calibration was accomplished using a semiempirical calculation method that combined the measurement of a point source in a simple geometry with a photon transport code analysis of the actual detector configuration used. Before positioning the detector skid on the reactor deck, a series of confirmatory measurements were taken on an ADD mockup source of known activity. The calculated curves checked within ~4% of the values measured from the mockup for the two larger collimator apertures used during the actual FLHT-5 monitoring. For the remaining smallest aperture, the calculated curves checked within ~20%.

The EG&G spectrometer unit acquired 132 spectra on-line before, during, and after the FLHT-5 shutdown transient. During the transient, each spectral count time was nominally 30-s long. Periodically (at about 400-s intervals) during the data acquisition period, the selectable collimator was momentarily rotated to the fully closed position in order to assess the background radiation of the detector; the measured background was never significant.

The acquired spectra were analyzed on INEL's CDC CYBER 176 computer using the analysis code GAUSS VIII.^(a) The spectral analysis results, expressed as concentrations in Ci/cc, were sorted by isotope and plotted versus the time at the midpoint of the counting interval.

Examinations of the isotopic concentration plots developed from the on-line data show strong evidence of a deposition of chemically active fission products on the viewed segment of the ADD walls. Concentration plots for iodine, tellurium, and cesium isotopes exhibit behavior characteristics indicative of domination by deposited activity. Figure 5.16 presents a comparison between the output of the PNL dose meter at the SCC with the measured concentration curve for ^{131}I . Both plots indicate that the primary contribution to the gamma ray emission rate at the SCC was from reactive fission product isotopes plated out on the effluent piping, as evidenced by a lack of response by the measured activity to the fluctuating bundle thermal conditions and to the test termination. The discrepancy between the monotonically increasing ^{131}I concentrations and the flat or slightly decreasing dose meter response after 3000 s occurred, because the dose meter was responding to a deposited activity including that of several isotopes with half-lives shorter than the 8-day half-life of ^{131}I . Little real-time information can be derived directly from the on-line iodine and cesium concentration data from the gamma spectrometer. However, significant fission product release information can be derived for a variety of noble gas (xenon and krypton) isotopes. The noble gas isotopes are not subject to plate out and deposition, and their release rates are similar to those of iodine, cesium, and bromine (Silberberg 1986). Consequently, the noble gases can be used to estimate the volatile fission product release profile for the test.

Calculations of noble gas release rates from the on-line measurements at the ADD require known effluent flow rates and transit times from the bundle exit to the monitoring location. The linear flow velocities maintained during the FLHT-5 transient resulted in short effluent transit times (~10 s) from the bundle exit to the ADD. Effluent flow rates at the bundle exit as a function of time were estimated from the system makeup flow, test conditions, and the output of the NCTFM. Using the assumption of "plug flow" in the effluent line, noble gas release rates in Ci/s were calculated from the on-line data^(b) by multiplying the measured noble gas concentrations by the bundle effluent

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- (a) Putnam, M. H., R. G. Helmer, and C. M. McCullagh. 1985. "GAUSS VIII A Computer Program for the Analysis of Gamma-Ray Spectra from Ge Semiconductor Spectrometer." EGG-PBS-6768, EG&G Idaho, Inc., Idaho Falls, Idaho.
- (b) That is, assuming negligible dispersion of the fission gas in the hydrogen/nitrogen effluent flow.

FIGURE 5.16. Comparison of Measured ^{131}I Concentrations with the Response of the Dose Meter (showing the similarity in response)

flow rate at the bundle release time and correcting for the density differences of the effluent between the measurement location and the bundle exit. During FLHT-5, the effluent transit time was short so that the differences between the estimated effluent volumetric flow rates at the bundle exit and the monitoring location were compensated by density differences. Consequently, noble gas release rates from the bundle are adequately estimated by multiplying the measured concentrations by the estimated flow rate at the ADD at the time of measurement.

Sample release rate profiles calculated in this manner are presented in Figure 5.17(a, b, and c). The release rate plots are presented with a linear ordinate, and positive and negative release rate values are plotted. The error bars represent one estimated standard deviation (a 20% uncertainty was assumed in the flow-rate values). The negative calculated values and the generally increasing statistical fluctuations (particularly large after about 2600 s) are the result of the high concentrations of deposited fission products late in the test. As the count rate from dominant (deposited) isotopes increases, detection of minor spectral components like the noble gas nuclides becomes more marginal and the results for these nuclides are subject to larger statistical fluctuations. Examination of Figure 5.17(b and c) for ^{135}Xe and ^{138}Xe , respectively, reveals several results between about 2800 s and 3500 s that are relatively high and have a moderate error. These results seem anomalous and are not supported by response of the NRU stack activity monitor. However, these elevated responses seem to appear consistently in the release rates for several noble gas isotopes, perhaps indicating a relation to a delayed release. Pending further investigation, these results will be labeled spurious.

FIGURE 5.17. Release Rates (Ci/s) for the Noble Gases: (a) ^{90}Kr , (b) ^{135}Xe , and (c) ^{138}Xe

The noble gas release rates agree with other fission product release timing information. The correlation between the bundle temperature escalation and the reactor stack gas activity was discussed in Section 5.5.2. Figure 5.18 presents a comparison of the measured stack activity with the spectrometer results for release rate of ^{135}Xe . There is qualitative agreement in the time behavior of the two plots up to ~2000 s when the spectrometer results become unreliable.

Fractional release rates and integrated fractional releases have been estimated from the isotopic release rate values, and a shutdown inventory has been calculated using the ORIGEN2 code.^(a) This type of computation requires careful selection of appropriate nuclides. Because the fission product inventories during the FLHT-5 transient were not modeled in detail using data for nuclides with short half-lives relative to the duration of the FLHT-5

(a) Jenquin, U. P. 1987. "Radioisotope Inventory and Activity Calculations for the FLHT-4 and FLHT-5 Experiments." Letter to D. D. Lanning, Pacific Northwest Laboratory, Richland, Washington.

FIGURE 5.18. Comparison of the Calculated Release Rate of ^{135}Xe with the Measured Stack Activity

transient to calculate release rates or release fractions would yield misleading results. Inventories for extremely short-lived isotopes (half-life < 100 s) would be continuously evolved following initial release, while the shutdown inventory would reflect the instantaneous saturation inventory. Thus the integrated release could be several times the shutdown inventory. On the other hand, extremely long-lived isotopes (half-life > 1 yr) were not detectable; isotopes whose inventories are strongly coupled to the behavior of precursors on a short time frame relative to the transient duration must be avoided. On-line release rates have, therefore, been calculated for ^{88}Kr (half-life = 2.8 h) and ^{135}Xe (half-life = 9.1 h). Both of the nuclides are moderately decoupled from their precursor's behavior and have half-lives that are acceptable. Because the bundle nuclear power was constant during the FLHT-5 transient, fractional releases and release rates calculated for these two nuclides using a shutdown inventory should be reasonably accurate.

Release fractions for noble gases during FLHT-5 were estimated from the measured release of ^{135}Xe and ^{88}Kr as

$$F_i = \left[\int_0^{t_{\text{end}}} R_i(t) dt \right] / [I_i(t_{\text{end}})]$$

where F_i = release fraction of isotope i

$R_i(t)$ = release rate of isotope i at time t (Ci/s)

$I_i(t_{\text{end}})$ = inventory of isotope i at shutdown (Ci).

To simplify the calculations a simple histogram integration of the release rate data was performed. The integration was terminated at 3000 s to avoid inclusion of the spurious results noted above. (Examination of the time-shifted stack activity curve indicates that 80% of the noble gas release occurred before 3000 s.)

The release fractions calculated for ^{135}Xe and ^{88}Kr were $0.22 (\pm 0.05)$ and $0.39 (\pm 0.26)$, respectively. The quoted uncertainties are estimated to be one sigma errors and are propagated assuming a 20% uncertainty in the inventory. The uncertainty calculations do not reflect the error associated with the use of a single shutdown inventory rather than a more detailed inventory model. Instead they include only estimated errors in the measured concentrations, the effluent flow rates ($\pm 20\%$), and the bundle shutdown inventory ($\pm 20\%$).

Conventional practice requires that fractional release rates be expressed as fractional release rate coefficients ("k" values) to be applied to the instantaneous bundle inventory (Silberberg 1986). Consequently, assuming the initial inventory to be equal to $I_i(t_{\text{end}})$, fractional release rate coefficients can be estimated as

$$k_i(t) = R_i(t) / \left[I_i(t_{\text{end}}) - \int_0^t R_i(t') dt' \right]$$

where $k_i(t)$ is the fractional release rate coefficient of isotope i at time t .

Fractional release rate coefficients for ^{135}Xe and ^{88}Kr are presented in Figure 5.19. Coefficients were not calculated for data beyond 2500 s because the release rate data are highly uncertain. Estimated uncertainties are not plotted but are generally reflected in the plot scatter.

During the period from between ~1200 to 2000 s, the estimated noble gas release rate coefficient was about $2.0 \text{ E-}04 \text{ s}^{-1}$. The CORSOR-M database (Silberberg 1986) predicts noble gas release rate coefficients of this magnitude for fuel samples at temperatures of about 2000 K. Large portions of the FLHT-5 test bundle were at or above 2000 K during this period, so these results appear reasonable. Perhaps better correlations between the FLHT-5 bundle noble gas release rates and the CORSOR-M database will be possible when more detailed bundle temperature-time profiles are developed.

FIGURE 5.19. Fractional Release Rate Coefficients Determined for ^{135}Xe and ^{88}Kr up to 2500 s

6.0 POST-TEST GAMMA SCANNING AND VISUAL EXAMINATION

The procedures used in and major results from post-test gamma scanning activities and visual inspection of the test train are described in this section. The gamma scanning activities were completed within a few days following the FLHT-5 test. These activities included the axial spectral gamma scanning of the deposition rod and the plenum and the axial gamma counting of the copper flux wire.

The visual examination of the assembly was conducted underwater in the NRU rod bay in November 1987. The examination included the progressive cutting of a 90 window through the shroud along the full length of the damaged fuel bundle to reveal in succession the outer round, the MMPD wire wrap, inner round, saddle pieces, insulation tiles, liner, and one pair of exterior fuel rods.

6.1 POST-TEST GAMMA SCANNING ACTIVITIES

Post-test gamma scanning activities included 1) an axial spectral gamma scan of the deposition rod 18 h after the test was terminated and 2) an axial spectral gamma scan of the plenum and top 60 cm of the fuel bundle 85 h after the test was terminated. The copper flux wire was removed from the test train and axially gamma scanned in June 1987.

6.1.1 Deposition Rod Gamma Scanning

The gamma scan of the deposition rod began at 7:45 p.m. on May 15, approximately 19 h after the boilaway transient. A shielded germanium gamma detector was placed in front of a beam port that penetrated the 10-cm lead wall of the SCC. The deposition rod was winched up beyond the port into a sealed canister that had been bolted to the steam-line extension. The canister was encased in a shielded cask supported by a portable bridge structure. After the rod was pulled into the canister during disassembly of the test, the steam line below the flange was crimped and cut, and the canister plus flange pulled into the cask. Therefore, the deposition rod was shielded and confined throughout the operation. Shielding and confinement of the deposition rod were maintained throughout the operation.

An 80-s scan was performed every 3 cm along the rod, from the top of the rod (Level 366) to the bottom of the rod (Level 173). Definitive spectra were obtained from every scan; an example is shown in Figure 6.1. Almost all the major peaks can be attributed to isotopes of cesium and iodine. Absolute concentrations of deposited fission products were deduced from the scans; however, these absolute values were subject to some uncertainty due to possible shifts in source geometry as the rod passed the detector. The ratio of measured concentrations of the two isotopes is not as subject to uncertainty as are the absolute values, because much of the uncertainty is divided out in the calculation of the ratio.

FIGURE 6.1. Gamma Spectrum from the Spectral Scan of the FLHT-5 Deposition Rod

6.1.2 Plenum Gamma Scanning

The test train was raised from temporary storage into a vertical multi-ton shielded cask at noon on May 18, 83 h after the test. Spectral data were taken in 1-min counts every 2.5 cm, from the top of the plenum at Level 340 to Level 120, 58 cm below the top of the fuel column. The test train gamma scan was to have included the full length of the plenum and the fuel bundle. However, there was a high amount of radioactivity deposited at the top of the plenum and a diminished amount of lead thickness in the tail-piece of the vertical cask (into which the test train was raised for scanning). These conditions combined to produce unacceptably high personnel exposure rates at the counting equipment when more than 60 cm of the upper fuel bundle was raised beyond the detector. Therefore, the gamma scan was limited to the plenum area and top 60 cm (24 in.) of the fuel bundle (i.e., from Level 120 to Level 360).

The general axial structure in the gross gamma scan is shown in Figure 6.2. The gross activity is much greater in the top of the fuel bundle than in the plenum above the fuel because of the activity contributed by nonvolatile (nonreleased) fission products in the fuel. The axial structure of the plenum gross gamma activity corresponds with certain structural features along the plenum length. The low broad peak in the plenum scan (from Levels 160 to approximately 200) corresponds with the lowest heated zone, where the heater was turned off during most of the severe damage portion of

FIGURE 6.2. Gross Gamma Activity Along the FLHT-5 Plenum Compared with Heater Positions

the transient. The next three heated zones are interspersed with plenum gamma activity peaks that correlate with the short unheated zones separating the heaters. The largest peak, at Level 310, corresponds to a bellows, where the effluent on its way up encountered a very large, cool, convoluted surface area.

Although absolute ratios of isotopic concentrations were derived from the axial plenum scan, these concentrations may have significant uncertainties associated with them, because the calibration models have not accounted for all of the changing geometries along the plenum. However, the relative ratio of the two isotopic concentrations has less associated uncertainty because it is independent of changes in source geometry.

The axial distribution of selected concentration ratios are compared with inventory ratios (i.e., the ratios calculated for the undisturbed fuel bundle inventory) in Figures 6.3 and 6.4. The ^{137}Cs to ^{133}I ratio is shown in Figure 6.3. The fact that the ^{137}Cs to ^{133}I ratio is fairly close to the inventory ratio indicates that cesium was deposited similarly to iodine along the plenum length. The situation is different when the ^{133}I to ^{132}Te ratio shown in Figure 6.4 is considered. The iodine is deposited to a much greater degree than the tellurium over most of the plenum length. The one exception is the valley at Level 300 in Figure 6.4, which corresponds to a peak in the absolute tellurium concentration.

The concentrations of some isotopes were converted to "fraction of fuel bundle inventory" (by decay correction to time-of-release) and summed over the plenum length from Levels 190 to 280 to yield an estimated fraction of total bundle inventory deposited on that portion of the plenum wall. This portion was chosen because it had uniform temperature during test (Figure 5.6) and

FIGURE 6.3. ^{137}Cs to ^{133}I Concentration Ratio in the Plenum Compared with the Inventory Ratio

FIGURE 6.4. ^{133}I to ^{132}Te Concentration Ratio in the Plenum Compared with the Inventory Ratio

relatively uniform deposited activity (Figure 6.2). These estimated fractions are shown in Table 6.1. Whereas the release fractions of cesium and iodine should theoretically be similar, a similar fraction of the cesium inventory is estimated to have been deposited in the plenum. This is consistent with the ratio plots just discussed.

TABLE 6.1. Estimated Fractions of Isotope Inventory Deposited on Plenum Walls

<u>Isotope</u>	<u>Estimated Fraction of Total Bundle Inventory Deposited on Plenum Walls (Levels 190 to 280)</u>
¹³¹ I	0.0068
¹³³ I	0.0065
¹³⁷ Cs	0.0087
¹³² Te	0.0002

6.1.3 Flux Wire Gamma Scanning

A 1-mm diameter copper flux wire was inserted in a hole along the outside of the FLHT-5 shroud in the zero degree orientation (see Figure A.1). The wire ran the full length of the assembly, with its lower tip at level -4.5 (i.e., 4.5 in. below the bottom of the fuel column). The purpose of this wire was to measure (by its neutron activation, determined from post-test gamma scanning) the relative axial distribution of thermal neutron flux and, hence, fission power during the transient. Accordingly, the lower 430 cm of the wire was gamma scanned in 1-cm increments several days after the test, when activity had stabilized. The activity is plotted versus elevation (above fuel bottom) in Figure 6.5. The expected (calculated) flux/power distribution for the fuel bundle during the boilaway transient is superimposed on this plot. The structure in the flux wire activity versus elevation near the bundle midplane (Level 72, 183-cm elevation) may reflect post-escalation fuel material redistribution in this area.

6.2 FUEL BUNDLE POST-TEST EXAMINATION

After the plenum had been gamma scanned, the fuel assembly was lowered in an elevator to the spent fuel storage pool located in the basement of the reactor building and was stored vertically. Six months after the test, the fuel assembly was visually examined. The assembly was rotated to a horizontal position and then placed on the disassembly-examination-reassembly machine (DERM)¹ underwater examination table.

¹ The DERM is a computer driven fixture that facilitates precision cutting and examination of the FLHT assemblies underwater in the NRU rod bay.

FIGURE 6.5. Gamma Activity Along the FLHT-5 Copper Flux Wire. Note correspondence to the expected cosine shape of axial power distribution.

The FLHT-5 shrouded fuel bundle was visually examined and photographed in the spent fuel storage pool using methods similar to the ones used in the MT-6A, MT-6B, FLHT-1, FLHT-2, and FLHT-4 tests. The initial examination and analysis concentrated on the shroud. The shroud main components (outer Zircaloy round, inner Zircaloy round, Zircaloy saddle pieces and zirconium dioxide thermal insulation) were visually examined, and selected regions were photographed. Greater emphasis was placed on the shroud examination (compared to previous tests), because the inner round ruptured about twenty minutes after the FLHT-5 test was terminated. This was the first occurrence of such a rupture. The examination included a close visual inspection of the complete outer surface of the inner round using a periscope to locate the rupture and determine its cause. The examination produced no unambiguous rupture location. The only obvious change in the round from the as-fabricated condition was surface discoloration (also observed in previous tests), which was hypothesized to be caused by Zircaloy surface reactions with impurity gases in the helium-filled MMPD cavity (the inner round forms the inner boundary of the MMPD cavity.) However, as the examination proceeded, cracks formed when the assembly was being clamped into position for the sawing operations. Visual examination of the inside surface of a section of the inner round revealed two likely rupture locations, both consisting of small transverse cracks. Portions of the inner round were highly embrittled. The metal embrittlement is presently thought to be caused by the absorption of oxygen or hydrogen. The oxygen would have come from the steam; the hydrogen would have come from Zircaloy-steam reactions that occurred within the bundle at lower elevations.

Following the examination of the inner round, an underwater saw was used to cut a 90° opening through all the remaining layers of the shroud, from Level 30 (0.76 m) up to Level 130 (3.03 m). After the sections had been cut, successive layers were visually examined and then removed. The layers consisted of the Zircaloy inner round, Zircaloy saddle pieces, zirconium dioxide thermal insulation and finally, the Zircaloy liner. Figure 6.6 a transverse cross section of the as-fabricated fuel bundle and shroud, showing the orientation of the 90° opening.

Because the insulation regions reacted (bonded or fused) in some regions with the liner, the two were inseparable at those points. Likewise, because the liner reacted (bonded or fused) with the fuel rods at most levels, these two were also inseparable. Thus only as much insulation was removed as was necessary so as not to cause changes in the liner. Then color photographs were taken every 2.54 cm along the liner between Levels 30 and 130. The five locations are at Levels 39, 65, 70, 94, and 120 (elevations of 0.96, 1.65, 1.77, 2.39, and 3.05 m).

The chemical reaction between the liner and the fuel bundle was more extensive than the chemical reaction between the liner and insulation. Manual methods (using long-handled tools) for removing the liner under ~2 m of water

FIGURE 6.6. Fuel Bundle Cross Section Showing Shroud Section Removed

were very tedious and not always successful, because there were to be minimal changes to the bundle geometry. After several regions of the liner were removed, photographs were again taken every 2.4 cm along the bundle between Levels 30 and 130.

Five pairs of photographs in black and white are presented in Figures 6.7 to 6.11. The photographs show the view at five different axial levels in the bundle. Each pair shows a specific axial location within the bundle, with and then without the liner in place. The five locations are at Levels 39, 65, 70, 94, and 120 (elevations of 0.96, 1.65, 1.77, 2.39, and 3.05 m).

Shown in Figure 6.7b (Level 39, liner removed) are two fuel rods (1b and 1c) near the bottom of the picture. The lighter colored cladding near the left-hand side represents heavily oxidized cladding that has been converted to zirconium dioxide. The darker colored cladding near the right-hand side of the picture is lightly oxidized cladding with a film of ZrO_{2-x} . For this test assembly, it is expected that the fuel rods below the "final" coolant level [Level 30 (0.76 m)] are in the as-fabricated condition; these fuel rods should have bright, shiny, silvery surfaces. Photographs show that between Levels 30 and 39 the cladding is dark colored with a protective film of ZrO_{2-x} . Above Level 39, as shown in the picture, the color of the cladding changes as the oxide thickness increases. The noticeable breakup of brittle fuel rods at the extreme left-hand side resulted from the chemical reactions between the Inconel grid and the cladding and resulted from the post-test examination that fractured the very brittle materials.

The set of photographs in Figure 6.8 (a and b) are of Level 65; the photo with the liner removed actually shows the bundle interior. It was taken after the fused liner and rod shards were removed and rotated 180° so that the top section became the bottom. The liner interior surface is not visible because of the extensive interactions of the bundle components with the liner.

Shown in Figure 6.9a (Level 70) is the normal "outside" view of the bundle. In Figure 6.9b (Level 94, liner removed) is shown a somewhat open (unblocked) region near the top of the photo. However, the views at Levels 70 and 65 appear to indicate that a complete flow blockage occurred, even though the data recorded during the test did not indicate a complete blockage.

Finally, shown in the photos in Figure 6.11 (Level 120) is the heavy damage that occurred near the top of the bundle, above approximately Level 108. The damage was caused more by heavy oxidation than by melting and liquid metal interactions.

The main conclusion from the visual examination is that between Levels 45 and 108 the bundle is extensively damaged, primarily by melting and liquefaction and above Level 108 by heavy oxidation. There is moderate damage below Level 39.

(a)

(b)

FIGURE 6.7. Visual Exam Results at Level 39: (a) Liner; (b) Fuel Bundle

(a)

(b)

FIGURE 6.8. Visual Exam Results at Level 65: (a) Liner; (b) Fuel Bundle

(a)

(b)

FIGURE 6.9. Visual Exam Results at Level 70: (a) Liner; (b) Fuel Bundle

(a)

(b)

FIGURE 6.10. Visual Exam Results at Level 94: (a) Liner; (b) Fuel Bundle

(a)

(b)

FIGURE 6.11. Visual Exam Results at Level 120: (a) Liner; (b) Fuel Bundle

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APPENDIX
INSTRUMENTATION

APPENDIX

INSTRUMENTATION

In this appendix, the location, nomenclature, output, units and DACS channel designations of the thermocouples, pressure transducers, flowmeters, and specialized sensors used in the full-length high-temperature (FLHT-5) test are presented. Digitized DACS-collected data are stored on tapes and disc files at PNL and are identified and retrieved by the DACS channel designation.

A.1 TEST TRAIN INSTRUMENTATION

The test train instrumentation included the following major groups:

- thermocouples [Type K (chromel-alumel) and Type C (tungsten-rhenium)] grouped into
 - plenum region above the fuel bundle
 - fuel rod assembly including the inlet region and inlet
 - shroud
- fuel rod and shroud cavity pressure transducers
- other instruments
 - self-powered neutron detectors (SPNDs) and flux wires
 - TDR coolant level sensors
 - MMPD wire.

A.1.1 Plenum Region Thermocouples

The thermocouples in the plenum region are listed in Table A.1 and were grouped as follows:

- deposition rod thermocouples - every 17 inches from Levels 178 to 297
- thermocouples (between liner and insulation) every 17 inches from Levels 178 to 297
- outer thermocouples - Levels 166, 226, 282, and 162
- steam-line thermocouple (above closure) - (Level 332).

TABLE A.1. Plenum Region Thermocouple Summary (output in °F)

<u>Thermocouple Nomenclature^(a)</u>	<u>Location</u>	<u>DACS Channel Number</u>
TCIK-332-SL-XXX	Outside steam line (above closure), 0°-(90°-180°-270°) LCS Trip Circuit	106 (0°)
TCIK(or s)-280-SI (or PI)-XXX	Outside of steam line, 0°, 180°	76, 75
TCIK-246-SI-XXX	Outside of steam line, 0°, 180°	129, 111
TCIK-212-SI-XXX	Outside of steam line, 0°, 180°	128, 47
TCIK-178-SI-XXX	Outside of steam line, 0°, 180°	110, 19
TCZC-166-IL-XXX	Outside of steam line, 0°, 180°	2, 32
TCIK-282-OP-XXX	Outside of outer liner, 0°, 180°	97, 48
TCIK-226-OP-XXX	Outside of outer liner, 0°, 180°	99, 113
TCIK-166-OP-XXX	Outside of outer liner, 0°-90°-180°-270°	119, 70, 63, 43
TCIK-297-DR-XXX	On deposition rod, 0°	11
TCIK-280-DR-XXX	On deposition rod, 0°	15
TCIK-263-DR-XXX	On deposition rod, 0°	18
TCIK-246-DR-XXX	On deposition rod, 0°	40
TCIK-229-DR-XXX	On deposition rod, 0°	77
TCIK-212-DR-XXX	On deposition rod, 0°	130
TCIK-195-DR-XXX	On deposition rod, 0°	22
TCIK-178-DR-XXX	On deposition rod, 0°	74

(a) Thermocouple designation includes type, axial location (level), component, and azimuthal location.

A.1.2 Bundle Thermocouples

The bundle thermocouples are shown in Table A.2 and included the following groups:

- Type C thermocouples - on the inside of the fuel rod cladding at 8-in. intervals, from Level 40 through Level 132
- Type K thermocouples - on the inside of the carriers, at 4-in. intervals, from Level 20 through Level 72

TABLE A.2. Fuel Bundle Thermocouple Summary (output in °F)

<u>Thermocouple Nomenclature</u>	<u>Location</u>	<u>DACS Channel Number</u>
TCZC-147.5-1B, 4C	Steam probe 1B, 4C	101, 137
TCZC-132-3D	Fuel rod 3D	3
TCZC-128-2A	Fuel rod 2A	35
TCZC-126.5-1B, 4C	Steam probe 1B, 4C	51, 103
TCZC-120-3A, 119.55-3C	Fuel rods 3A, 3C	5, 132
TCZC-112-1C	Fuel rods 1C	34
TCZC-104-1B, 4C	Fuel rods 1B, 4C	86, 140
TCZC-96-2A, 3D	Fuel rods 2A, 3D	104, 79
TCZC-88-3A, 86.73-3C	Fuel rods 3A, 3C	118, 100
TCZC-80-1C	Fuel rods 1C	59
TCZC-72-1B, 4C	Fuel rods 1B, 4C	108, 95
TCZC-64-2A, 3D	Fuel rods 2A, 3D	27, 44
TCZC-56-3A, 55.67-3C	Fuel rods 3A, 3C	123, 83
TCZC-48-1C	Fuel rod 1C	8
TCZC-40-1B, 4C	Fuel rods 1B, 4C	68, 69
TCIK-72-4A	Carrier 4A	38
TCIK-64-4A	Carrier 4A	136
TCIK-56-4A	Carrier 4A	135
TCIK-48-4A	Carrier 4A	94
TCIK-44-1A, 4D	Carrier 1A, 4D	134, 42
TCIK-40-1A, 4D	Carrier 1A, 4D	139, 93
TCIK-36-1A	Carrier 1A	226
TCIK-32-1A	Carrier 1A	227
TCIK-28-1A	Carrier 1A	228
TCIK-24-1A	Carrier 1A	189
TCIK-20-1A	Carrier 1A	191
TCIK-14-4A	Carrier 4A	96
TCIK-08-4A	Carrier 4A	109
TCIK-02-4A	Carrier 4A	91
TCIK--10-BI-XXX	Bundle inlet level--10, 57, 37, 62 0°, 120°, 240°	

- steam thermocouples - mounted on the grid spacers, at Levels 126.5 and 147.5 (Type C)

- Type K thermocouples - mounted in the inlet flow stream, below the fuel bottom (Level -10).

Rods and carriers are designated by grid location, shown in Figure A.1.

A.1.3 Shroud Thermocouples

The shroud thermocouples are listed in Table A.3 and included the following groups:

- pairs of diametrically opposed Type C thermocouples - on the liner outer surface, staggered 90° at 8-in. intervals, from Level 56 to Level 152 (fed down from the top of the assembly)
- companion pairs of Type K thermocouples - on the saddle outer surface, from Levels 8 to 136 (fed down along slots in the outside surface of the saddles)
- sets of Type K thermocouples - on the outside surface of the shroud, at specific axial locations.

FIGURE A.1. Instrumentation Nomenclature and Grid Locations for FLHT-5

TABLE A.3. Shroud Thermocouple Summary (output in °F)

<u>Thermocouple Nomenclature</u>	<u>Location</u>	<u>DACS Channel Number</u>
TCZC-IL-152-XX	Outside of liner, 0°-180°	117, 124
TCZC-IL-144-XXX	Outside of liner, 90°-270°	54, 105
TCZC-IL-120-XXX	Outside of liner, 0°-180°	6, 9
TCZC-IL-112-XXX	Outside of liner, 90°-270°	1, 7
TCZC-IL-104-XXX	Outside of liner, 0°-180°	13, 52
TCZC-IL-96-XXX	Outside of liner, 90°-170°	10, 25
TCZC-IL-88-XXX	Outside of liner, 0°-180°	67, 24
TCZC-IL-80-XXX	Outside of liner, 90°-270°	49, 23
TCZC-IL-72-XXX	Outside of liner, 0°-180°	29, 28
TCZC-IL-64-XXX	Outside of liner, 90°-270°	92, 56
TCZC-IL-56-XXX	Outside of liner, 0°-180°	26, 58
TCSK-136-IS-XXX	Outside surface of saddle, 0°-180°	115, 36
TCSK-128-IS-XXX	Outside surface of saddle, 90°-270°	225, 46
TCSK-120-IS-XXX	Outside surface of saddle, 0°-180°	90, 61
TCSK-112-IS-XXX	Outside surface of saddle, 90°-270°	81, 50
TCSK-104-IS-XXX	Outside surface of saddle, 0°-180°	121, 39
TCSK-96-IS-XXX	Outside surface of saddle, 90°	120, 66
TCSK-88-IS-XXX	Outside surface of saddle, 0°-180°	126, 60
TCSK-80-IS-XXX	Outside surface of saddle, 90°-270°	122, 71
TCSK-72-IS-XXX	Outside surface of saddle, 0°-180°	73, 30
TCSK-61-IS-XXX	Outside surface of saddle, 90°-270°	55, 89
TCSK-56-IS-XXX	Outside surface of saddle, 0°-180°	107, 17
TCSK-48-IS-XXX	Outside surface of saddle, 90°-270°	82, 131
TCSK-40-IS-XXX	Outside surface of saddle, 0°-180°	138, 41
TCSK-32-IS-XXX	Outside surface of saddle, 90°-270°	127, 20
TCSK-24-IS-XXX	Outside surface of saddle, 0°-180°	98, 31
TCSK-16-IS-XXX	Outside surface of saddle, 90°-270°	102, 21
TCSK-08-IS-XXX	Outside surface of saddle, 0°-180°	114, 78
TCIK-144-OS-XXX	Outside of shroud, 0°-90°-180°	33, 124, 14
TCIK-110-OS-XXX	Outside of shroud, 90°-270°	116, 4
TCIK-90-OS-XXX	Outside of shroud, 0°-180°	133, 16
TCIK-72-OS-XXX	Outside of shroud, 0°-180°	88, 12
TCIK--8.0-OS-XXX	Outside of shroud, 0°-90°-180° (bypass inlet), (level--8)	192, 190, 112

A.1.4 Test Train Pressure Sensors

The pressure sensors in the test train are listed in Table A.4 and included two groups:

- pressure transducers - on the fuel rods (10 total)
- pressure transducers - on the various shroud and plenum cavities.

TABLE A.4. Fuel Rod and Cavity Pressure Transducers (output in psig)

<u>Pressure Transducer Nomenclature</u>	<u>Location/Component</u>	<u>DACS Channel Number</u>
PTFR-05-1C	Rod 1C	182
PTFR-05-2A	Rod 2A	174
PTFR-05-2B	Rod 2B	184
PTFR-05-2D	Rod 2D	180
PTFR-05-3A	Rod 3A	178
PTFR-05-4B	Rod 4B	177
PTFR-05-3D	Rod 3D	179
PTFR-05-4C	Rod 4C	181
PTSG-PLENUM-UPPER	(Heated) upper plenum insulation cavity	175
PTSG-PLENUM-LOWER	(Unheated) plenum insulation cavity	185
PTSG-SHRD-INSUL	Insulation cavity in the shroud	173
PTSG-MMPD Cavity	MMPD cavity in the shroud (2 pressure transducers)	183, 176

A.1.5 Other Test Train Sensors

Other test train sensors included the four neutron sensors, two neutron flux wires, and the two TDR coolant level detectors. These sensors are listed in Table A.5.

A.2 EFFLUENT CONTROL MODULE INSTRUMENTATION

The instrumentation in the SCC and ECM included flowmeters, pressure transducers, thermocouples, and specialized instruments for measuring hydrogen flow. They were grouped as follows:

- flow measurements
 - steam
 - noncondensable gas
 - condensate
 - nitrogen injections at TDRs, ADD, and condenser
- pressure measurements
 - system
 - cross-train pressure differential
 - nitrogen injection
 - pressure at hydrogen measurements
- temperature measurements
 - steam line
 - hydrogen measurement instruments

TABLE A.5. Other Test Train Instrumentation

<u>Sensor Type</u>	<u>Nomenclature/ Location</u>	<u>Output Units</u>	<u>DACS Channel Number</u>
Self-powered neutron detector	SPND-86.75-OS-180	NV	156
Self-powered neutron detector	SPND-65.75-OS-180	NV	157
Self-powered neutron detector	SPND-44.75-OS-180	NV	158
Self-powered neutron detector	SPND-23.75-OS-180	NV	155
TDR coolant level detector	TDRF-162-OS-70	Level	144
TDR coolant level detector	TDRF-162-OS-250	Level	143

- hydrogen measurements
 - mass spectrometer
 - palladium partial pressure meter
 - noncondensable gas flowmeter
 - thermal conductivity meter.

A.2.1 Flow Measurements

The flow of gas and liquid through the ECM and the flows of injected nitrogen were measured. The various meters, units, and DACS designations are listed in Table A.6.

A.2.2 Pressure Measurements

The system pressures were measured at the ECM inlet, across the condenser, iodine filter, assembly, and at the nitrogen injection. The pressure transducers are listed with their designations in Table A.7.

A.2.3 Temperature Measurements

The effluent temperatures were measured at the ECM inlet and at the condenser inlet and outlet with Type K thermocouples. The gas sample stream temperatures were measured at the heated palladium meter and at the thermal conductivity meter. The temperatures of the nitrogen injection flows were also measured. The thermocouples are listed in Table A.8.

A.2.4 Hydrogen Measurements

Three instruments were installed in the ECM specifically to measure the hydrogen flow rate in the effluent from the test train during the accelerated oxidation/severe damage portion of the test. These instruments were the mass spectrometer (MS), the palladium partial pressure meter (PHM), and gas thermal conductivity meter (TCM). A fourth hydrogen flow measurement was deduced from the output of the turbine flowmeter connected to the noncondensable gas line downstream from the condenser. The last measurement proved to be the most reliable for reassuring both instantaneous hydrogen flow rate and total hydrogen generated.

TABLE A.6. Effluent Control Module Flowmeters

<u>Sensor Designation</u>	<u>Location and Function</u>	<u>Units</u>	<u>DACS Channel Number</u>
MSFM-ECM-TDRI-INJCT	Nitrogen injection for TDR	Std. liter/min	243
MSFM-ECM-TDR3-INJCT	Nitrogen injection for TDR	Std. liter/min	145
MSFM-ECM-ADDI-INJCT	Nitrogen injection for the ADD	Std. liter/min	147
TBDFM-ECM-N ₂ -IN	Nitrogen injection for ECM condenser	Absolute liter/min	222
TBFM-ECM-NON-COND	Turbine flowmeter, ECM noncondensable gas flow (downstream from the condenser)	Absolute liter/min	221
MSFM-ECM-BNDL-PURGE	Nitrogen bundle purge flow (post-test)	Std. liter/min	230
MSFM-ECM-BKMN	Sample flow past TCM hydrogen meter	Std. cm ³ /min	240
TBFM-ECM-BNDL-LO	Makeup flow to the fuel bundle	lbm/h	224
TBFM-ECM-CONDST-HI	Condensate flow (down- stream from the condenser)	U.S. gal/min	194
CLFM-ECM-STEAM-FLOW	Coriolis flowmeter on the steam line (at entry to ECM)	lbm/h	241

Mass Spectrometer (MS)

The mass spectrometer sampled a small fraction of the effluent flow and output relative partial pressures of steam, nitrogen, hydrogen, and helium. The interpretation of the data required a knowledge of the mass flow of nitrogen (from metered injection) in the effluent and the assumption of complete gas mixing. The data was output to a video recorder (not to the DACS). The mass spectrometer was successfully calibrated with a gas mixture and performed adequately during the ECM commissioning. However, it was shut down because of inoperative hardware just before the boildown transient and did not collect useful data on hydrogen.

TABLE A.7. Effluent Control Module Pressure Sensors (output in psig)

<u>Sensor Designation</u>	<u>Location and Function</u>	<u>DACS Channel Number</u>
PTSG-ECM-BNDL-IN	Bundle inlet pressure	238
PTSG-ECM-Inlet	ECM inlet pressure	234
PTSG-ECM-CONDIN	Condenser inlet pressure	237
PTSG-ECM-NONCOND-OT	Condenser outlet pressure	236
PTSG-ECM-N ₂ -IN	Pressure at the condenser nitrogen injection	233
DELP-ECM-TRAIN	Differential pressure across the test train	232
DELP-ECM-I2-SAMPLER	Differential pressure across the iodine filter	231
PTSG-ECM-BKMN-SAMP	Pressure on the gas sample line at the Beckman meter	235

TABLE A.8. Effluent Control Module Thermocouples (output in °F)

<u>Sensor Designation</u>	<u>Location</u>	<u>DACS Channel Number</u>
TCIK-ECM-Inlet	Inlet temperature at the ECM	85
TCIK-ECM-COND-IN	Inlet temperature for the condenser	80
TCIK-ECM-NONCON-OUT	Outlet temperature of condenser (gas steam)	84
TCIK-ECM-CONDST-OUT	Outlet temperature of condenser (liquid steam)	87
TEMP-ECM-PD-HEATER	Temperature of inlet gas to the palladium hydrogen partial pressure meter	141
TCIK-ECM-BKMN-SAMP	Temperature of inlet gas to the thermal conductivity meter	72
TCIK-ECM-N ₂ -IN	Temperature of nitrogen gas injected into the condenser	105
TEMP-ECM-SL-HEATR	Control TC on heater tape around the effluent line inside the SCC (just upstream from the ADD)	169

Palladium Partial Pressure Meter (PHM)

This meter was a passive device consisting of a coiled palladium alloy tube inside a large, 6-inch-long chamber. The chamber was heated to $\sim 588^{\circ}\text{K}$ thereby heating the coiled tube and the incoming gas. The hydrogen rapidly and selectively diffused through the heated tube walls from the sample gas passing through the chamber. The ratio of the hydrogen pressure inside the tube to the total gas pressure in the chamber is the hydrogen fraction in the sampled effluent gas. This ratio must be combined with the total measured nitrogen injection gas flow to calculate the hydrogen flow. The DACS sensor number for the hydrogen partial pressure output (in psig) is 218.

Thermal Conductivity Meter (TCM)

Downstream from the PHM the gas was chilled and dried and then a sample of the effluent gas was led through a chamber, where its thermal conductivity was measured by a hot-wire technique. The hydrogen fraction was then deduced from a calibration function, assuming the effluent composition of hydrogen plus nitrogen. This instrument proved to have such a long sampling delay time and inherent inertia that it was not used to estimate the instantaneous hydrogen flow rate but only total hydrogen. Its output (in % hydrogen) is recorded in DACS channel 217.

The Noncondensable Gas Flowmeter (NCTFM)

For the majority of the hydrogen generation period, the noncondensable gas consists only of hydrogen plus injected nitrogen. Because the flow of injected nitrogen is measured, the nitrogen flow can be subtracted from the total noncondensable gas flow to obtain the hydrogen flow. The accuracy of this measurement of hydrogen flow is limited only by the inherent accuracy of the turbine flowmeter and the accuracy of the instantaneous measurement of nitrogen flow. (The nitrogen injection flows are relatively constant, so the few seconds of lag time from injection measurements to the noncondensable gas flow measurement do not significantly impair the accuracy of the deduced hydrogen flow.) Overall, the hydrogen flows deduced from this flowmeter have proven to be the most reliable real-time determinations of instantaneous flow rates because the sampling lag time is the smallest and the instrument response is the best. As a consequence, this flowmeter also provided the most reliable measure of total hydrogen produced during the FLHT-5 test. Its output [in actual liters per minute, (ALPM)] is recorded in DACS channel 221.

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