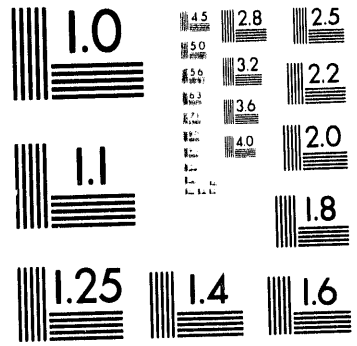
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FULL-LENGTH HIGH-TEMPERATURE SEVERE FUEL DAMAGE TEST #1

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MASTER

ABSTRACT

This report describes the first full-length high-temperature test (FLHT-1) performed by Pacific Northwest Laboratory (PNL) in the National Research Universal (NRU) reactor at Chalk River, Ontario, Canada. The test is part of a series of experiments being performed for the NRC as a part of their Severe Fuel Damage Program and is one of several planned for PNL's Coolant Boilaway and Damage Progression Program. The report summarizes the test design and test plan. It also provides a summary and discussion of the data collected during the test and of the photos taken during the post-test examination. All objectives for the test were met. The key objective was to demonstrate that severe fuel damage tests on full-length fuel bundles can be safely conducted in the NRU reactor.

SUMMARY

Full-length high-temperature (FLHT) in-reactor experiments are being performed by Pacific Northwest Laboratory (PNL) to evaluate degraded core behavior and the progression of light water reactor (LWR) fuel damage resulting from loss-of-coolant accidents. These experiments, which are part of the U.S. Nuclear Regulatory Commission Severe Fuel Damage Program, are being conducted in the National Research Universal (NRU) reactor, Chalk River, Ontario, as part of the PNL Coolant Boilaway and Damage Progression (CBDP) Program.

This report presents a summary of the FLHT-1 test operations. The test was performed on March 2, 1985. In the report, the actual test operations and data are compared to the planned operations and predicted test behavior. The report also provides a brief description of the test assembly design and instrumentation, the effluent control module, and the data acquisition and control system. The report concludes with a summary and brief discussion of the data collected during the test, and photographs from the post-test examination.

The test plan called for a gradual temperature increase to approximately 2150 K (3400°F). However, during the test, the fuel cladding began to rapidly oxidize, causing local bundle temperatures to rapidly increase from about 1700K (2600°F) to 2275K (3635°F), at which time the test was terminated. Much of the Zircaloy cladding in the central region (axially) of the 3.7-m-long (12-ft) fuel bundle was heavily oxidized, and some Zircaloy cladding melted.

All objectives for the test were met. The key objective was to demonstrate that severe fuel damage tests on full-length fuel bundles can be safely conducted in the NRU reactor.

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

	Irradiated		Peak Temperatures, K		Time at Peak Temperature	
Test	Date	Rods	Goal	Actual	Goal	Actual
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 2800 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
- with the CRNL staff, 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 (Cronenberg, et al.)

- 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 Zry-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 2000 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.

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- B. J. Webb performed thermal-hydraulic analyses.
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- D. E. Hurley was responsible for the design, fabrication, assemble, checkout and operation of the effluent control module, ECM.
- L. J. Parchen managed the quality control function within this project and arranged shipping.
- G. A. Anderson provided all the engineering, design, fabrication, calibration, and operation associated with processing the electrical signals from the MMPDs and TDRs.
- R. R. Lewis provided support in getting the ECM, test train, and other equipment prepared for the test.
- U. P. Jenquin provided the calculations of fission product inventory, radiation levels, and shielding requirements.
- N. J. Wildung provided computer generated figures.
- J. W. Nageley and D. K. Hilliard edited this document and provided publication assistance.
- G. E. Russcher prepared the Safety Analysis Report for the FLHT-1 test.

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DEFINITIONS

ACRR	Annular Core Research Reactor (at Sandia National Laboratory)
A/D	analog to digital
AECL	Atomic Energy of Canada Limited
BWR	boiling-water reactor
CBDP	coolant boilaway and damage progression (program at PNL)
CRNL	Chalk River Nuclear Laboratories (Ontario, Canada)
DACS	data acquisition and control system
DERM	disassembly-examination-reassembly machine
ECM	effluent control module
EOP	experiment operations plan
FLHT	full-length high-temperature
FSAR	final safety analysis report
LCS	loop control system
LOCA	loss-of-coolant accident
LWR	light-water reactor
MMPD	molten material penetration detector
NCTFM	noncondensable turbine flowmeter
NRC	U.S. Nuclear Regulatory Commission
NRU	National Research Universal (reactor at CRNL)
PBF	Power Burst Facility (Idaho National Engineering Laboratory)
PIE	post-irradiation examination
PNL	Pacific Northwest Laboratory
PSAR	preliminary safety analysis report
PTE	post-test examination
PWR	pressurized-water reactor
SFD/ST	severe fuel damage/source term
SLPM	standard liters per minute
SPND	self-powered neutron detector
TC	thermocouple
TCM	thermal conductivity meter
TDR	time-domain reflectometer
TMI	Three Mile Island
Type K	thermocouple made of chromel/alumel with MgO insulation
Type C	thermocouple made of W-5Re/W-26Re with BeO insulation

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1.0 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) Severe Fuel Damage (SFD) program[a] consists of well-controlled, highly instrumented separate effects and integral experiments that provide data on fuel rod and core behavior during severe accident degraded core conditions. Dissolution of fuel in molten Zircaloy, formation and coolability of core debris beds, hydrogen generation, fission product release, and aerosol transport are among the phenomena being investigated.

The Coolant Boilaway and Damage Progression (CBDP) program is part of the NRC SFD program. As part of CBDP program a series of integrated full-length high-temperature (FLHT) tests is being performed by Pacific Northwest Laboratory (PNL)[b] to study the behavior of full-length light-water reactor (LWR) fuel rods during CBDP conditions. This is a report on the first of these tests. They are being conducted in the National Research Universal (NRU) reactor, Chalk River, Ontario; equipment and staff are being provided by Chalk River Nuclear Laboratories (CRNL).

The results from both separate effects tests and integral experiments will provide the database for developing and verifying computer codes such as SCDAP (Berna et al. 1982), MELPROG (Rivard and Camp 1983), and TRAPMELT (Jordan et al. 1979), which are designed to predict the individual and combined phenomena expected during severe accidents. Separate effects tests are being conducted in the Annular Core Research Reactor (ACRR), Albuquerque, New Mexico, and at several out-of-reactor facilities to obtain detailed data on the phenomena for analysis. In-reactor tests were completed at the Power Burst Facility (PBF), Idaho Falls, Idaho. In the PBF SFD tests, ~0.91-m (36-in.)-long 32-rod bundles were tested to ~2200°C (4000°F). Four tests were completed and provide proof test data to support FLHT operations.

This report presents the data collected during the FLHT-1 test and post-test visual examination of the fuel bundle. The planned test is described and compared to the actual test operations. Descriptions of the test train design, supporting instrumentation, and control systems are also included.

(a) Partners with the NRC in this program include nuclear organizations from the following countries: Belgium, Canada, England, Federal Republic of Germany, Italy, Japan, The Netherlands, Republic of China (Taiwan), Republic of Korea, Spain, Sweden, and the United States.

(b) Operated for the U.S. Department of Energy (DOE) by Battelle Memorial Institute under Contract DE-AC06-76RLO 1830.

2.0 TEST OBJECTIVES

The FLHT-1 test was designed and performed to provide full-length integral core effects data for LWRs by simulating and evaluating SFD accidents. Test objectives included the following:

- Provide data that are prototypic of LWRs during coolant boilaway and core damage progression near typical decay heat levels.
- Correlate fuel temperatures with the hydrogen generation history, fuel bundle coolant level (elevation), steaming rate, and fuel relocation.
- Provide data to compare full-length fuel temperature profiles, fuel rod rupture effects, and damage progression phenomena with short core data from the PBF and with scaled-up separate effects data from other sources to determine the validity of applied scaling factors and separate effects correlation techniques.
- Compare measured hydrogen generation with analytic code predictions, and provide data for validation of scaling factors applied to short core data from the PBF.
- Demonstrate that full-length LWR fuel bundle SFD tests to temperatures as high as 2200 K (4000°F) can be safely conducted in the NRU reactor.

3.0 EQUIPMENT CONFIGURATION

The equipment for the FLHT-1 test consisted of three major components in addition to the components of the NRU reactor loop systems: the test train and instrumentation, an effluent control module (ECM), and the data acquisition and control system (DACS). These components are described in the following.

3.1 TEST TRAIN AND INSTRUMENTATION

The FLHT-1 test train assembly consisted of a closure region, a plenum region, a core region, and an inlet region. Longitudinal schematics of the test train assembly plenum, and core regions are shown in Figures 3.1 and 3.2, and a horizontal core region schematic is shown in Figure 3.3. The test train assembly was ~10-m (31-ft) long and was suspended in the L-24 position of the NRU reactor. The closure region contained the hardware components that support the test train and maintained the primary system pressure boundary. Pressure boundary penetrations for instrument lines, fuel rod pressurization lines, bundle coolant lines, the effluent (or outlet steam) line, and the flush water line were included in the closure region. The plenum region, ~4.3-m (14-ft) long, occupied the portion of the test train between the closure region and the core region. The fueled core region, ~2.6-m (12-ft) long, contained a 12-rod fuel bundle assembly enclosed within an insulated shroud. The inlet region contained the shroud- to-bundle interface and the bundle coolant inlet system. The bundle coolant was a once-through system that was isolated from and independent of the bypass loop coolant system. The bundle coolant system was designed for zero leakage to or from the bypass coolant.

3.1.1 Closure Region

A cross section of the closure region is illustrated in Figure 3.4. These components formed the primary system pressure boundary between the bypass loop coolant used by the FLHT-1 test and the reactor hall environs. The closure plug was sealed to the L-24 position stump body by two commercial metallic gaskets.

The closure plug contained the pressure boundary penetrations for all test train instrument lines, fuel rod pressurization lines, and shroud and plenum cavity vent lines. The instrument lines exited the plug through three lead feedthroughs. The system pressure boundary at the feedthrough location was maintained by packing gland seals. In addition to the above penetrations, an effluent line, a flush water line, and two bundle coolant lines passed through the closure plug. The effluent line was welded to the closure plug. The pressure boundary at the flush water line and the bundle coolant lines was provided by commercial autoclave fittings.

Four Type K thermocouples (TCs) were attached to the effluent line above the closure plug. These thermocouples provided a safety group to measure the temperature of the effluent.

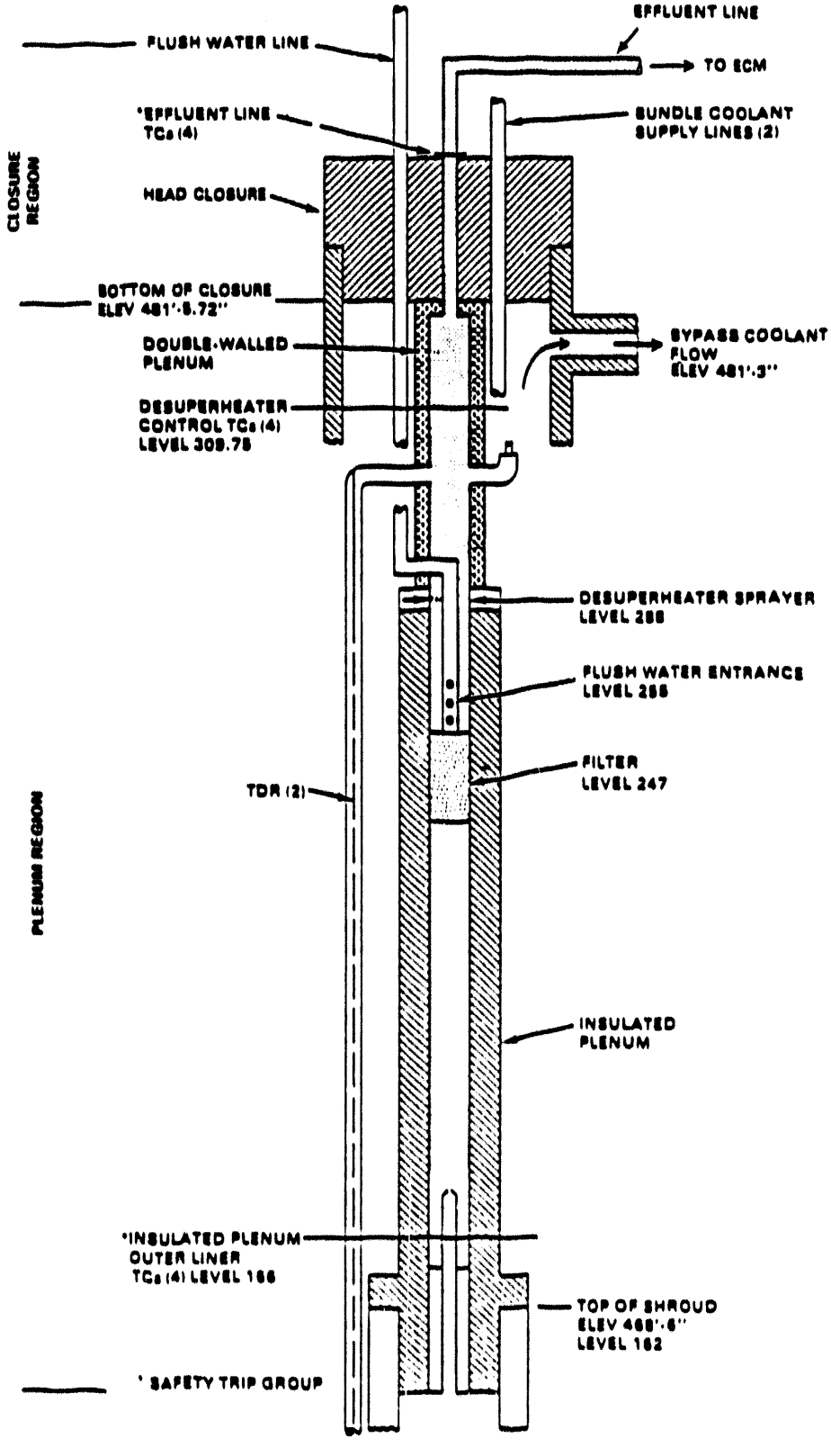

FIGURE 3.1. Longitudinal Cross Section of FLHT-1 Upper Test Train Assembly

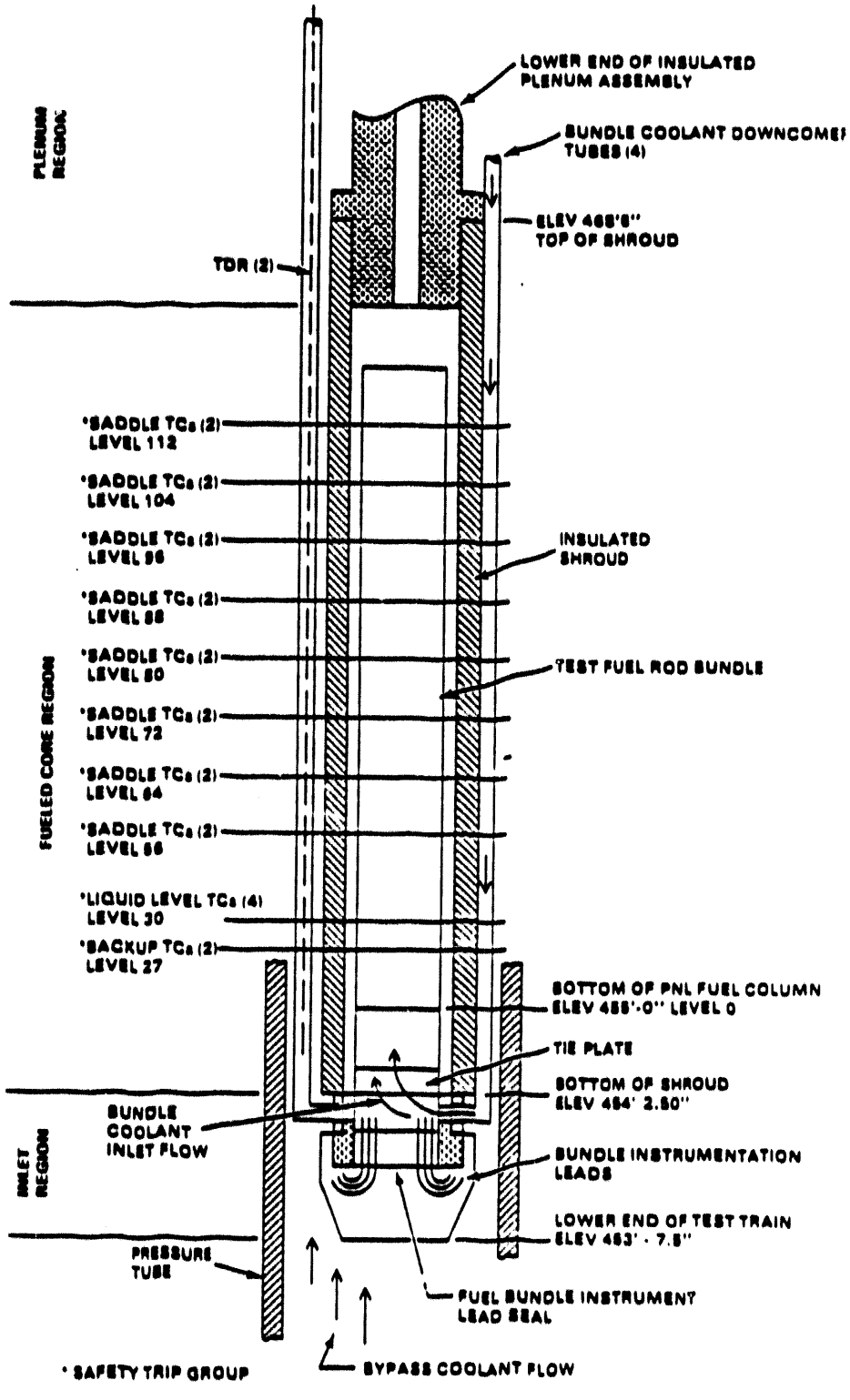

FIGURE 3.2. Longitudinal Cross Section of FLHT-1 Lower Test Train Assembly

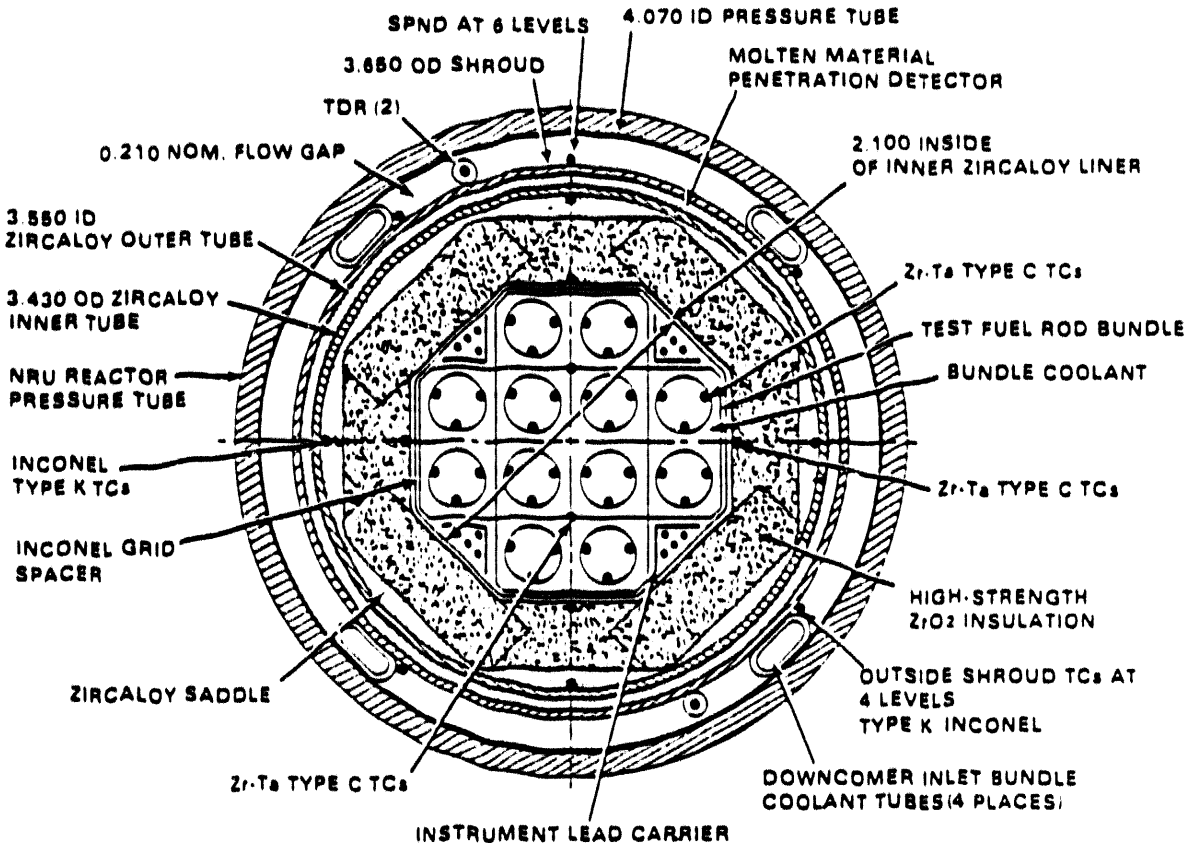

FIGURE 3.3. Core Region Cross Section of FLHT-1 Test Train Assembly (dimensions in inches)

3.1.2 Plenum Region

The plenum region (Figure 3.1) contained two plenum assemblies. The upper plenum was an evacuated double-walled assembly (Figure 3.5), and the lower plenum was a zirconium-dioxide-insulated assembly (Figure 3.6). The plenum region contained the mechanical structure that positioned and suspended the fuel region of the test train within the reactor core. The plenum region also provided the effluent flow path, filtration, temperature measurement, and sensor instrumentation.

Double-Walled Upper Plenum

The double-walled upper plenum assembly was attached to the bottom of the closure plug (Figure 3.5) by hexagonal nuts threaded into the bundle coolant and flush-line extensions. Four Type K thermocouples were attached to the effluent flow boundary along the support tube to measure the effluent temperature at various levels. The region below the upper flange was a double-walled structure with an evacuated annular space. The evacuation tube was located at the upper flange. The annular space ended at the lower flange housing. The lower flange provided connections for the flush line and two

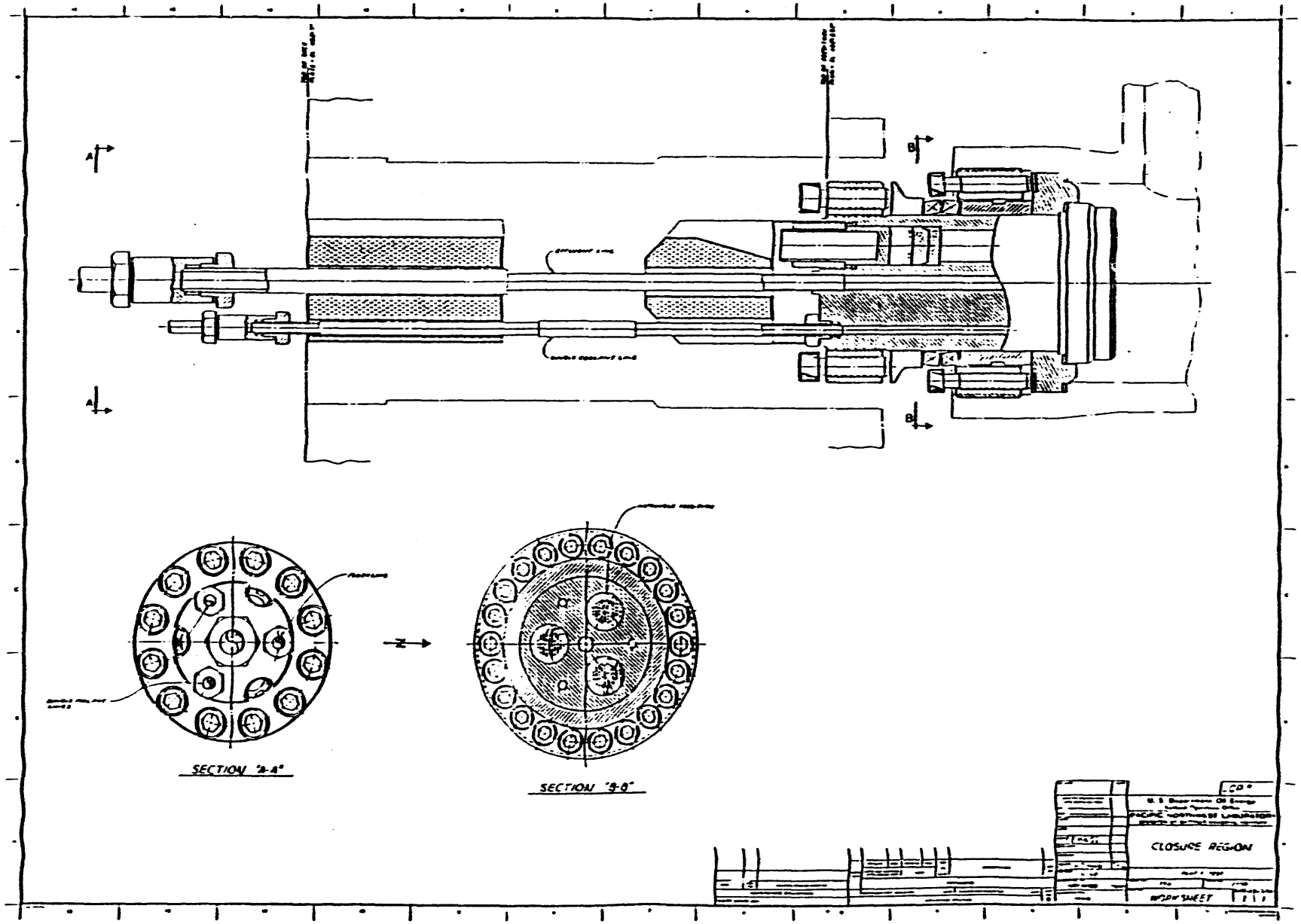

FIGURE 3.4. Test Train Closure Region

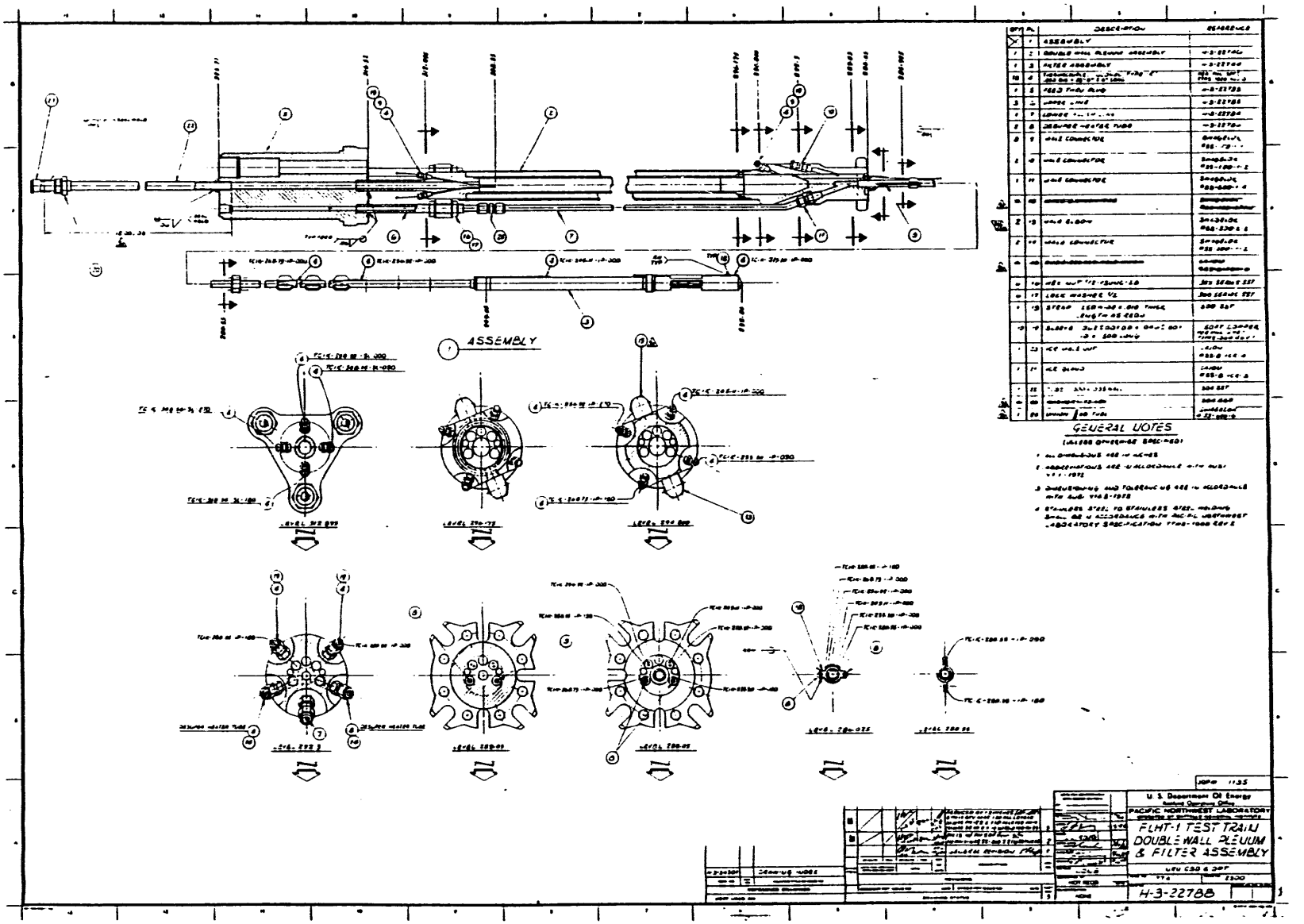

FIGURE 3.5. Double-Walled Plenum and Filter Assembly

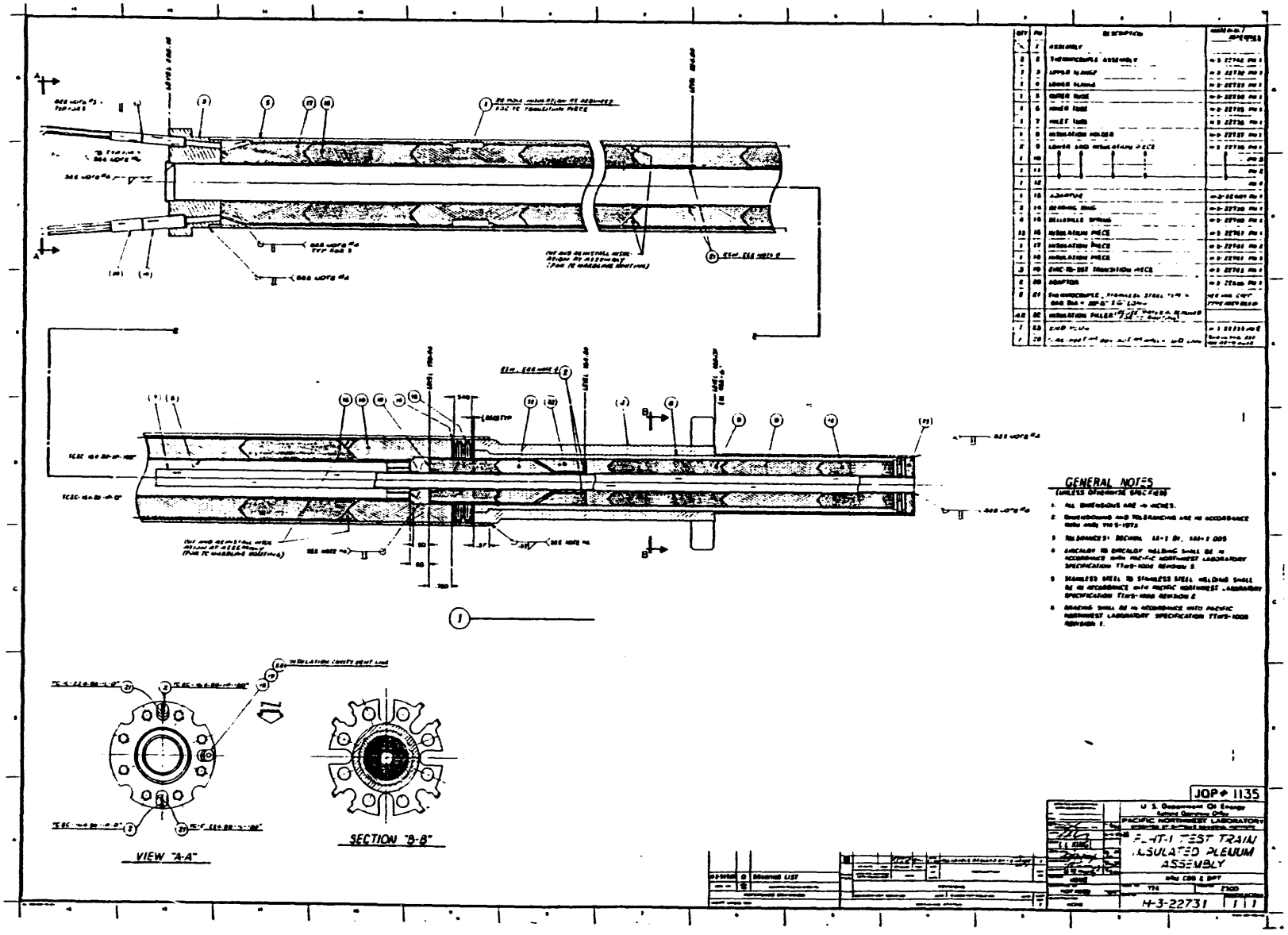

FIGURE 3.6. Insulated Plenum Assembly

time domain reflectometer (TDR) lines. Six thermocouples and two desuperheater lines also entered the effluent flow path through the lower flange. Two thermocouples were located below the desuperheater at level 281.75; the desuperheaters were at Level 288; and the desuperheater control data were provided by four thermocouples at Level 309.75. In addition, four thermocouples were mounted on the flush-line extension and filter holder at Levels 269.9, 258.1, 246.3, and 234.5, respectively. Flush water entered the effluent flow path above the filter at about Level 255.

Insulated Lower Plenum

The insulated lower plenum (Figure 3.6) attached to the lower flange of the double-walled plenum at Level 289.25. The insulated plenum was a coaxial tubular structure with low-density zirconium-dioxide insulation [1440-kg/m^3 (90-lb/ft^3)] in the annular space between the Zircaloy tubes. The insulation cavity was hermetically sealed and backfilled with argon at 0.1-MPa (1-atm) pressure; the cavity was monitored during the test by a pressure transducer.

Effluent entered the insulated plenum through the inlet tube at the lower end. The inlet tube was surrounded by low-density zirconium-dioxide insulation to minimize heat loss from the effluent to the bypass coolant. The low-density zirconium dioxide in this zone was contained within a high-density zirconium-dioxide holder, which acted as a support and container for the low-density material.

The inlet tube temperature was monitored by two Zircaloy/tantalum-sheathed Type C thermocouples at Level 166. The inner tube temperature was monitored at Level 226 by two stainless steel-sheathed Type K thermocouples. All thermocouple sheaths exited the plenum insulation cavity through two transition piece adapters at the top flange.

The exterior temperature of the insulated plenum was also monitored at the highest temperature region, Level 166. Four Type K thermocouples were attached to the exterior Zircaloy tube and constituted a safety trip group of sensors.

3.1.3 Core Region

The core region of the test train contained two coaxial assemblies: the shroud and the fuel bundle. Several views of the shroud assembly are shown in Figures 3.7, 3.8, and 3.9; and the bundle assembly is shown in Figures 3.10, 3.11, and 3.12. The shroud assembly provided the mechanical support and physical and thermal containment for the fuel bundle. The bundle assembly contained the fuel rods and the instrumentation that monitored thermal conditions in the fuel rods and in the bundle coolant channels during the test.

Shroud

The shroud assembly attached to the lower flange of the insulated plenum at Level 162. The shroud was a built-up tubular structure containing an equilateral octagonal Zircaloy tube as the inner liner. Surrounding the octagonal

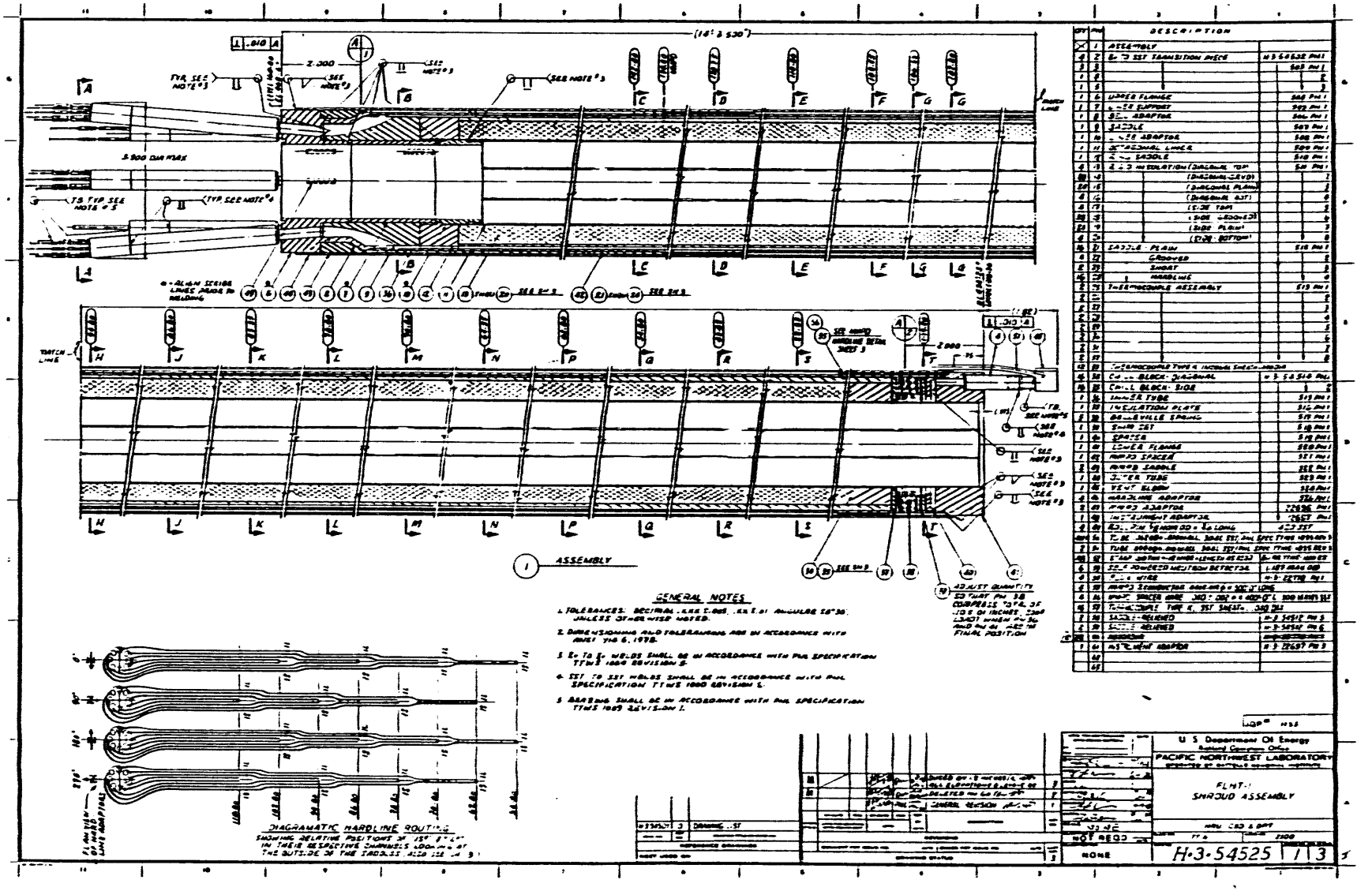

FIGURE 3.7. Longitudinal Cross Section of FLHT-1 Shroud Assembly

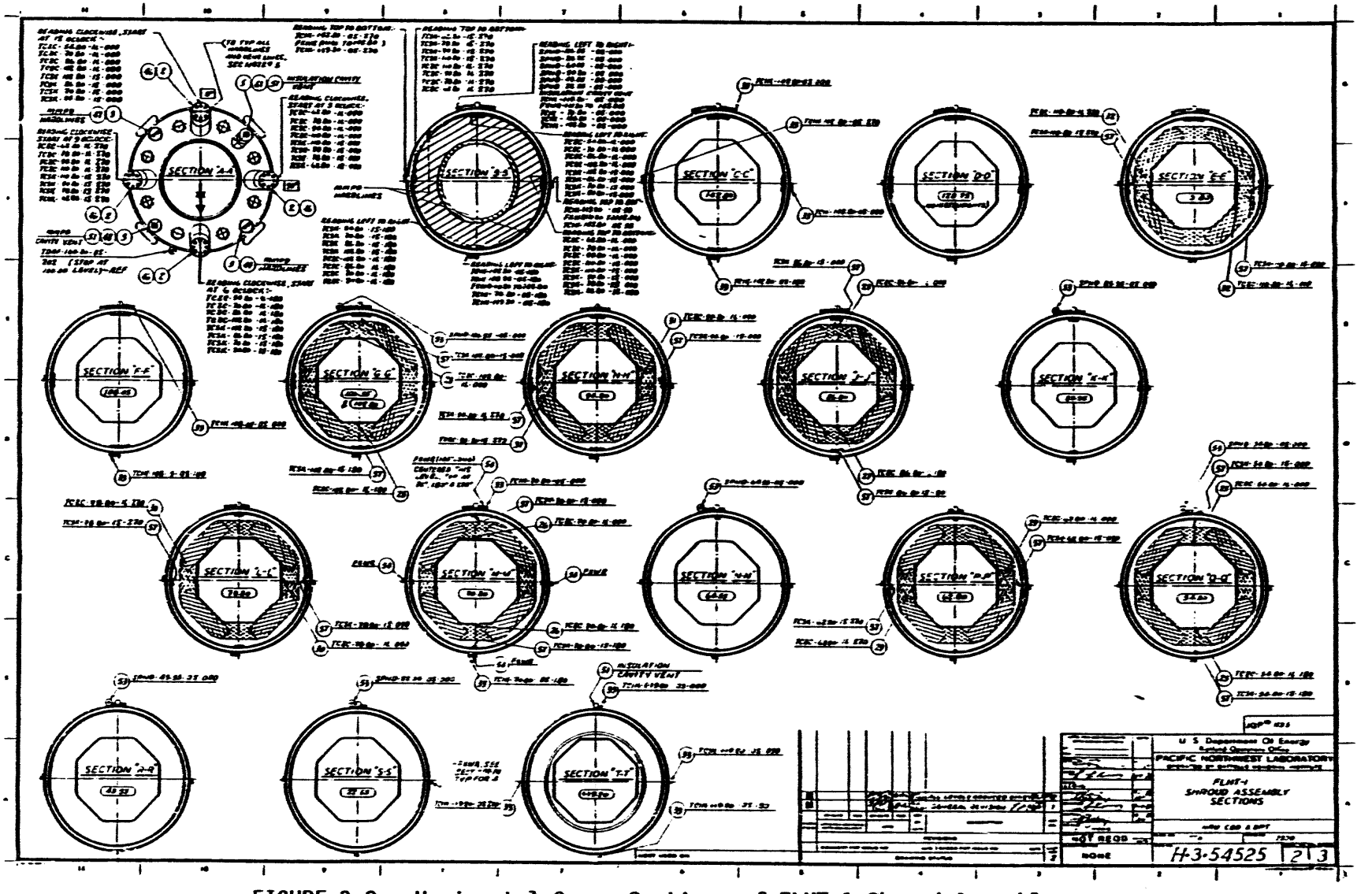

FIGURE 3.8. Horizontal Cross Sections of FLHT-1 Shroud Assembly

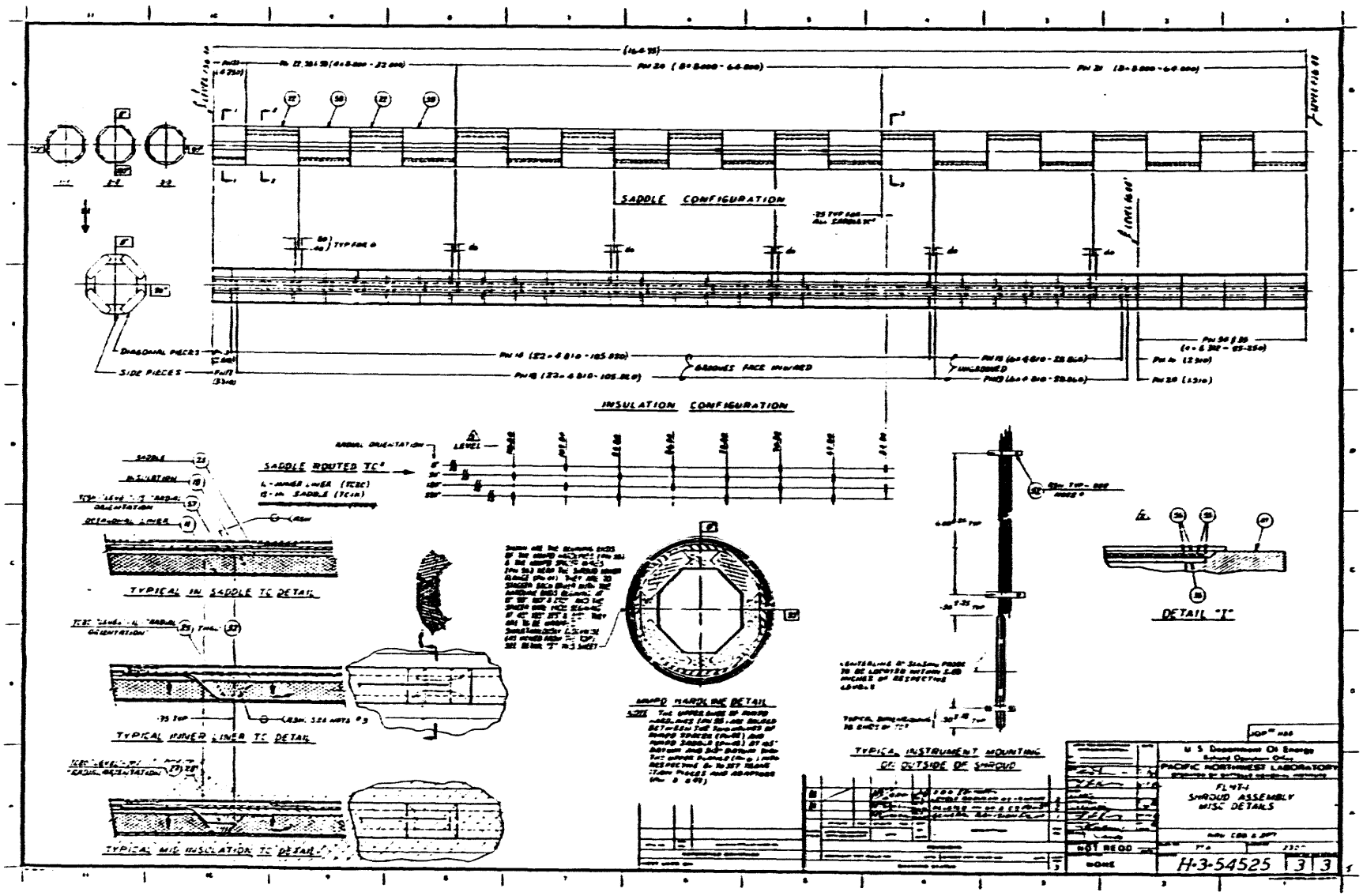

FIGURE 3.9. Details of FLHT-1 Shroud Assembly

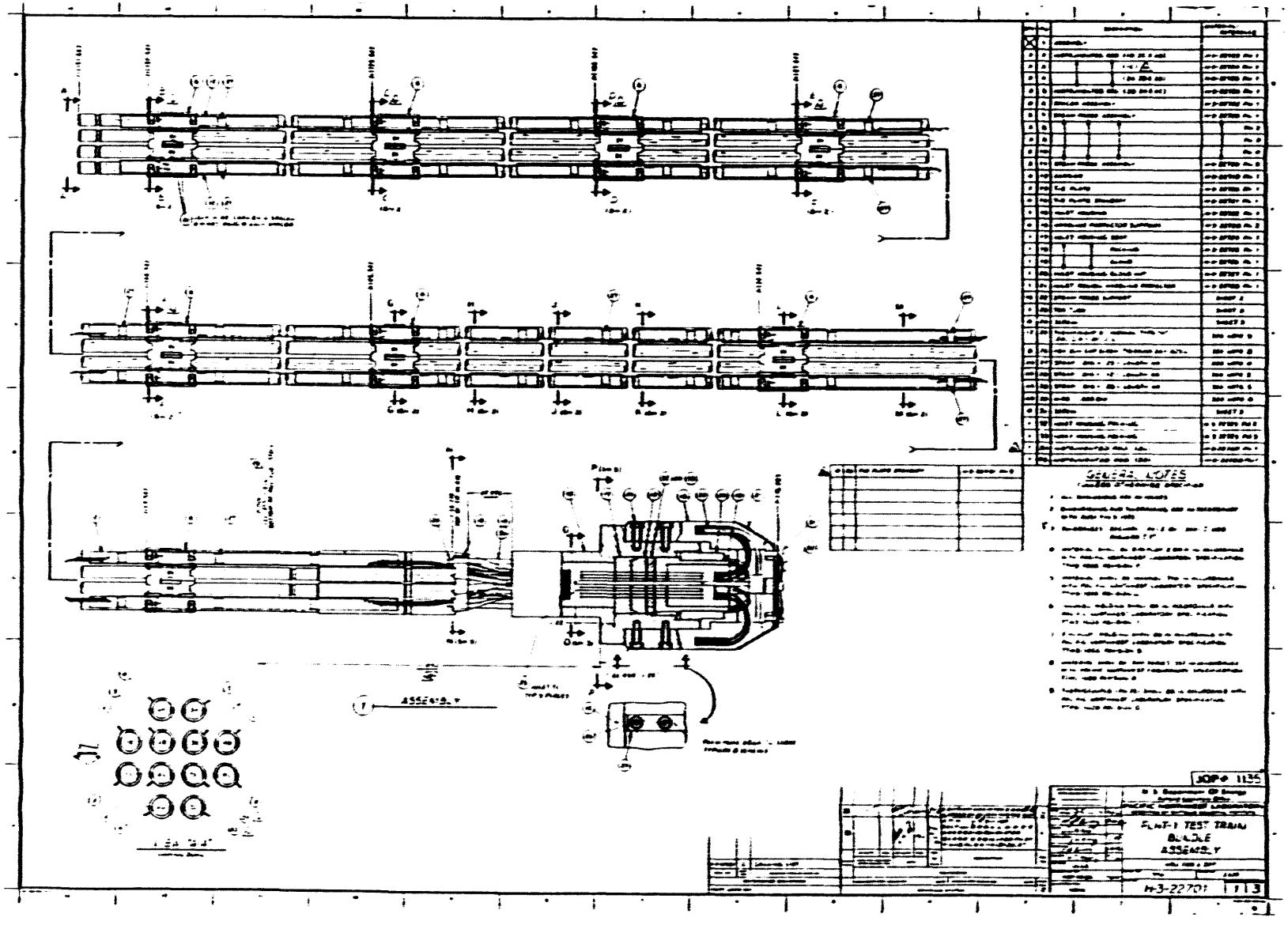

FIGURE 3.10. FLHT-1 Fuel Bundle Assembly

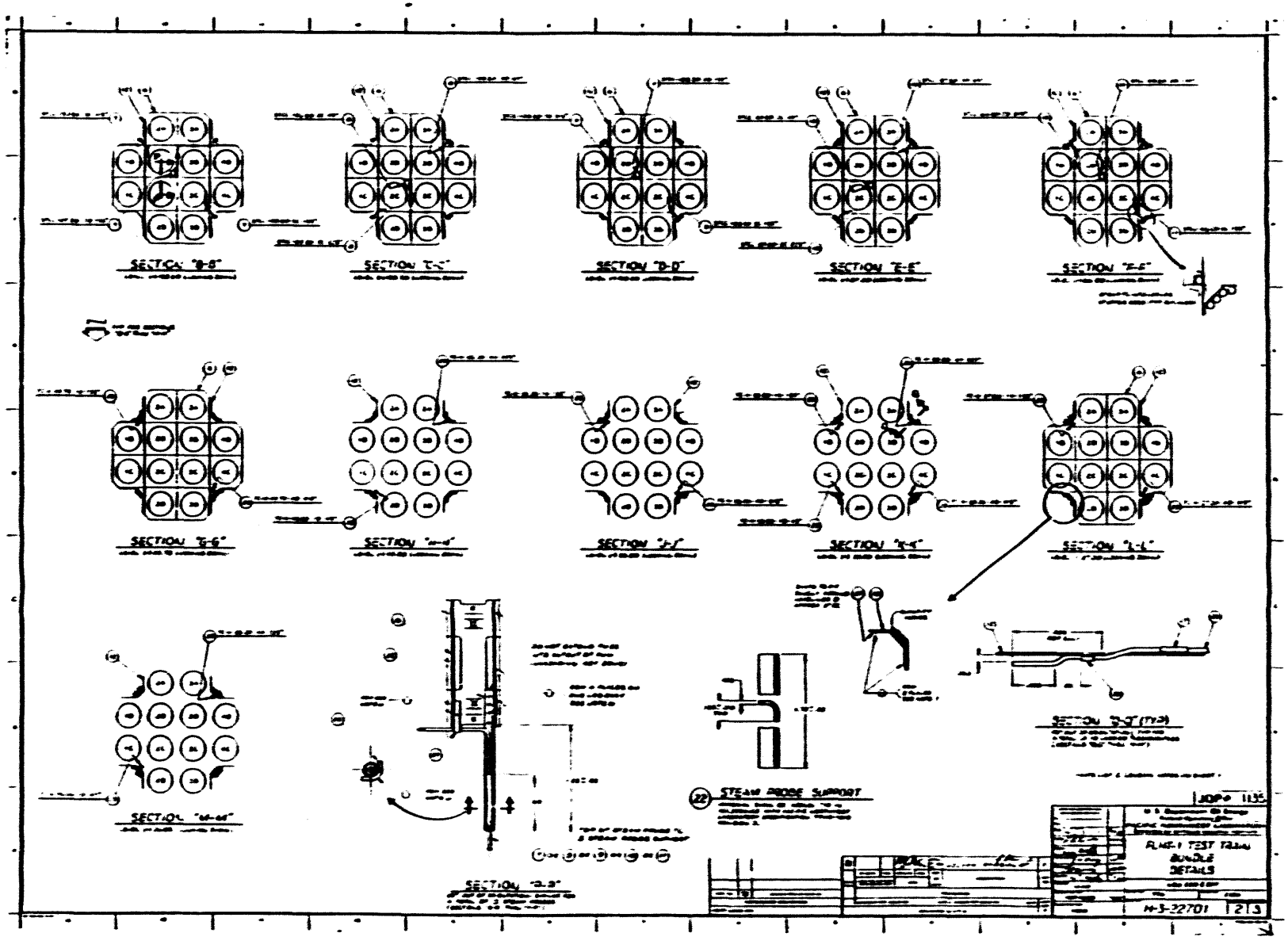

FIGURE 3.11. Fuel Bundle Details, Sections B-B Through Q-Q

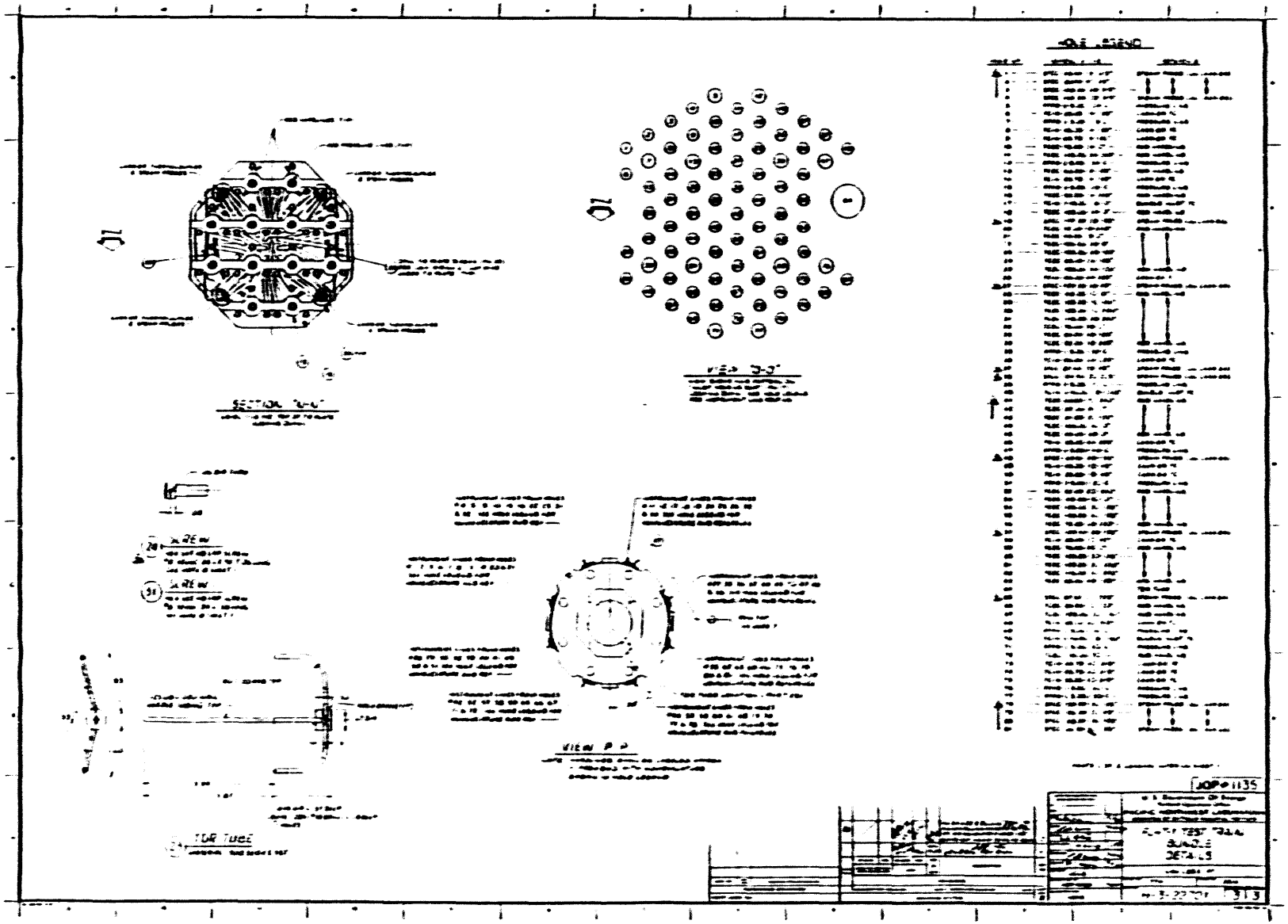

FIGURE 3.12. Fuel Bundle Details, Selected Views

liner tube was 10.4 mm (0.41 in.) of insulation. The insulation was high-strength low-density 1522-kg/m^3 (95-lb/ft^3) zirconium-dioxide tiles arranged in an interlocking configuration. The configuration was self-supporting and would not displace inwardly if the inner liner tube were removed. Zircaloy saddles radially retained the insulating configuration and made the transition from an octagonal shape to a round cross section. This built-up substructure was surrounded by a Zircaloy tube and welded end fittings, which created a hermetically sealed insulation cavity.

The shroud insulation cavity contained 32 thermocouples: 16 Type C thermocouples were attached to the inner liner to measure high temperatures at Levels 56, 64, 72, 80, 88, 96, 104, and 112; and 16 Type K thermocouples were placed adjacent to them on the exterior of the shroud saddles to measure temperatures at the same levels and radial orientations. The 16 shroud saddle thermocouples constituted four safety trip groups of sensors to monitor the exterior shroud temperature in the core region. These thermocouple locations are shown in Figure 3.8. All thermocouple leads exited the sealed insulation cavity through Zircaloy/stainless steel transition pieces. The leads were brazed at the stainless/stainless interface.

The shroud insulation cavity also contained upper and lower vent lines. The vent lines penetrated the sealed insulation cavity through upper and lower shroud end fittings and monitored the insulation cavity pressure. When the shroud liner breached, a pressure transducer on the vent lines indicated the associated increase in pressure and the time of the event.

Four independent molten material penetration detector (MMPD) circuits were wrapped around the insulation cavity outer tube from Level 8.5 to Level 132. Each MMPD was a 1-mm-dia (0.040-in.) instrument and was separated from its adjacent neighbor by a 1-mm-dia (0.040-in.) spacer wire. The inner and outer tubes were welded to the shroud end fittings, creating a hermetically sealed MMPD cavity.

The MMPD leads exited the sealed cavity through Zircaloy/stainless steel transition pieces that were welded into the shroud end fittings. The leads were brazed at the stainless/stainless interface.

The MMPD cavity pressure was measured with a pressure transducer on the MMPD cavity vent line. Increasing shroud temperatures were indicated by the decreasing electrical resistivity of the MMPD circuits. A breach of the MMPD cavity, which did not occur, would have been indicated by a rise in cavity pressure up to the system pressure.

Exterior shroud instrumentation consisted of thermocouples and self-powered neutron detectors (SPNDs). A total of 12 Type K thermocouples were located at Levels -8, 72, 109.25, and 114; SPNDs were located at Levels 23.75, 44.75, 56, 65.75, 86.75, and 107.75. The locations and orientations of all shroud instrumentation are detailed in Figure 3.8.

Fuel Rod Bundle

The fuel rod bundle assembly contained 12 instrumented fuel rods, a spacing framework, support structure, instrumentation, and an instrument lead seal. A schematic horizontal cross section of the assembly is shown in Figure 3.3.

Each instrumented fuel rod in the bundle contained a pressurization tube and three Type C thermocouples. The thermocouples were resistance welded to the interior of the fuel cladding at various elevations, and the sheaths exited the bottom of the fuel rods through brazed closure. Fuel rod details are shown in Figure 3.13.

The rod-to-rod spacing was established by prototypic PWR Inconel 718 grid spacers. Eight grid spacers were located on 53-cm (21-in.) centers along the length of the bundle beginning at Level 2.75. Each corner of the bundle contained a Zircaloy instrument carrier for routing thermocouple leads. The carriers were anchored to the tie plate and provided the longitudinal structural anchors for the grid spacers. The thermocouple lead carrier/grid spacer/tie plate subassembly provided the overall spacing framework for the fuel rod bundle assembly.

Three Type C thermocouples were located at each of five grid spacer locations from Level 67.76 through Level 151.75. One thermocouple was routed across the bottom of the grid to the center of the bundle and extended 3 cm (1.25 in.) below the grid. The other two thermocouples were routed up through opposing corners of the grid and were located at the upper face of the grid. Seventeen Type K thermocouples were also located on the bundle assembly: 14 were located on the inside of carriers (two each at Levels 20, 27, 36, 40, and 44.75, and four at Level 30); three Type K thermocouples were located at Level -10 to indicate bundle coolant inlet temperature. Level 30 thermocouples constituted a safety sensor group. However, those at Level 27 were also available for use as an alternate safety group. The locations of all bundle instrumentation, except internal fuel rod thermocouples, are detailed in Figures 3.14, 3.15, and 3.16.

The tie plate at the lower end of the bundle (Level -6.118) was attached to the inlet housing by standoffs. The inlet housing contained a packing gland seal to isolate the bundle coolant channel from the bypass coolant channel where the instrument leads exited the bundle. Structural support for the bundle was provided by the inlet housing.

3.1.4 Inlet Region

The inlet housing provided protection for the instrument leads at the lower end of the test train. The inlet region contained the bundle coolant lines and the inlet flange. The design layout for the region is shown in Figure 3.17. The inlet flange provided the interface between the bundle and the lower shroud flanges. The bundle and shroud flanges were sealed to the inlet flange by crushable copper seals. The bundle coolant lines were welded into the inlet flange.

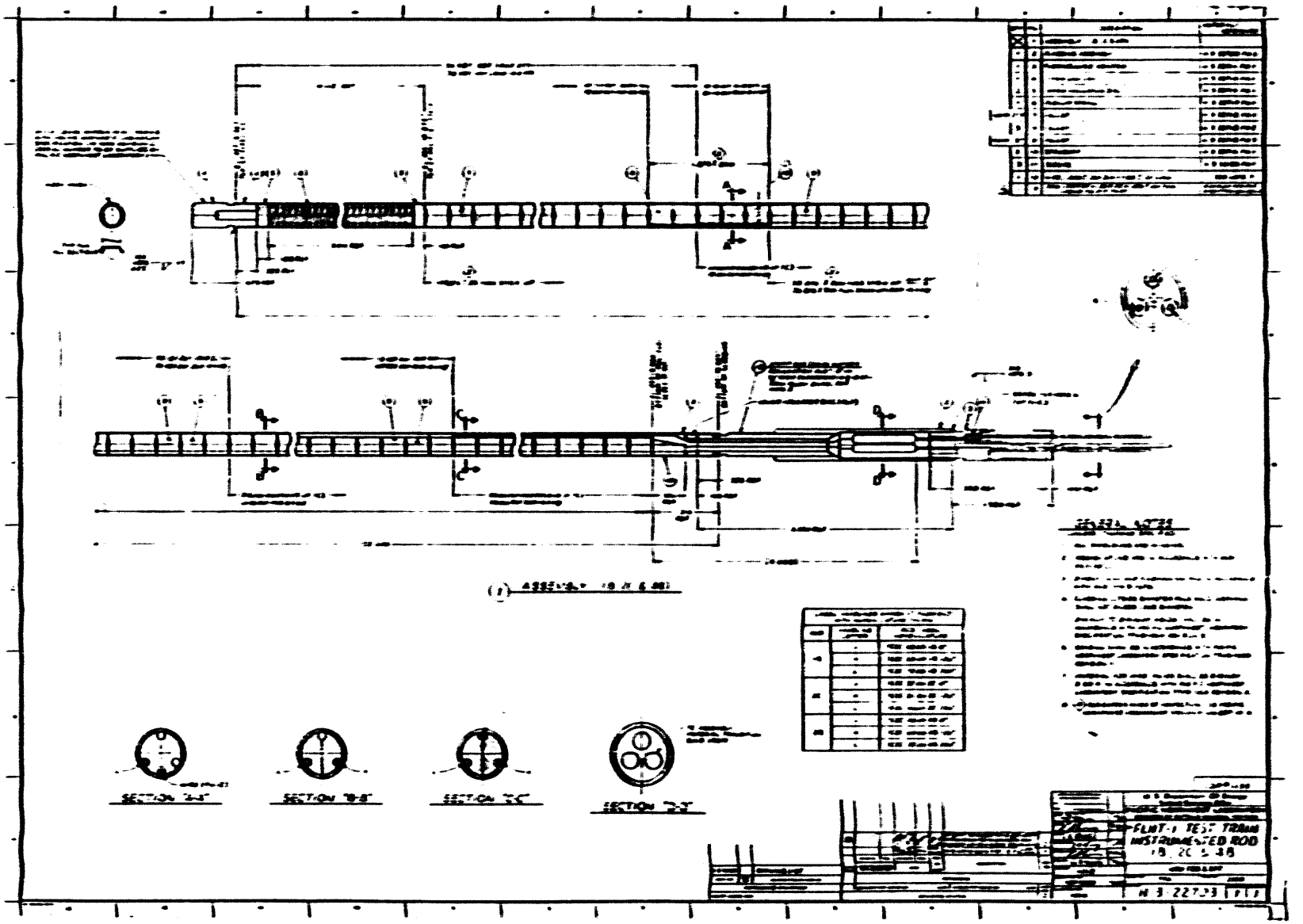

FIGURE 3.13. Fuel Rod Assembly

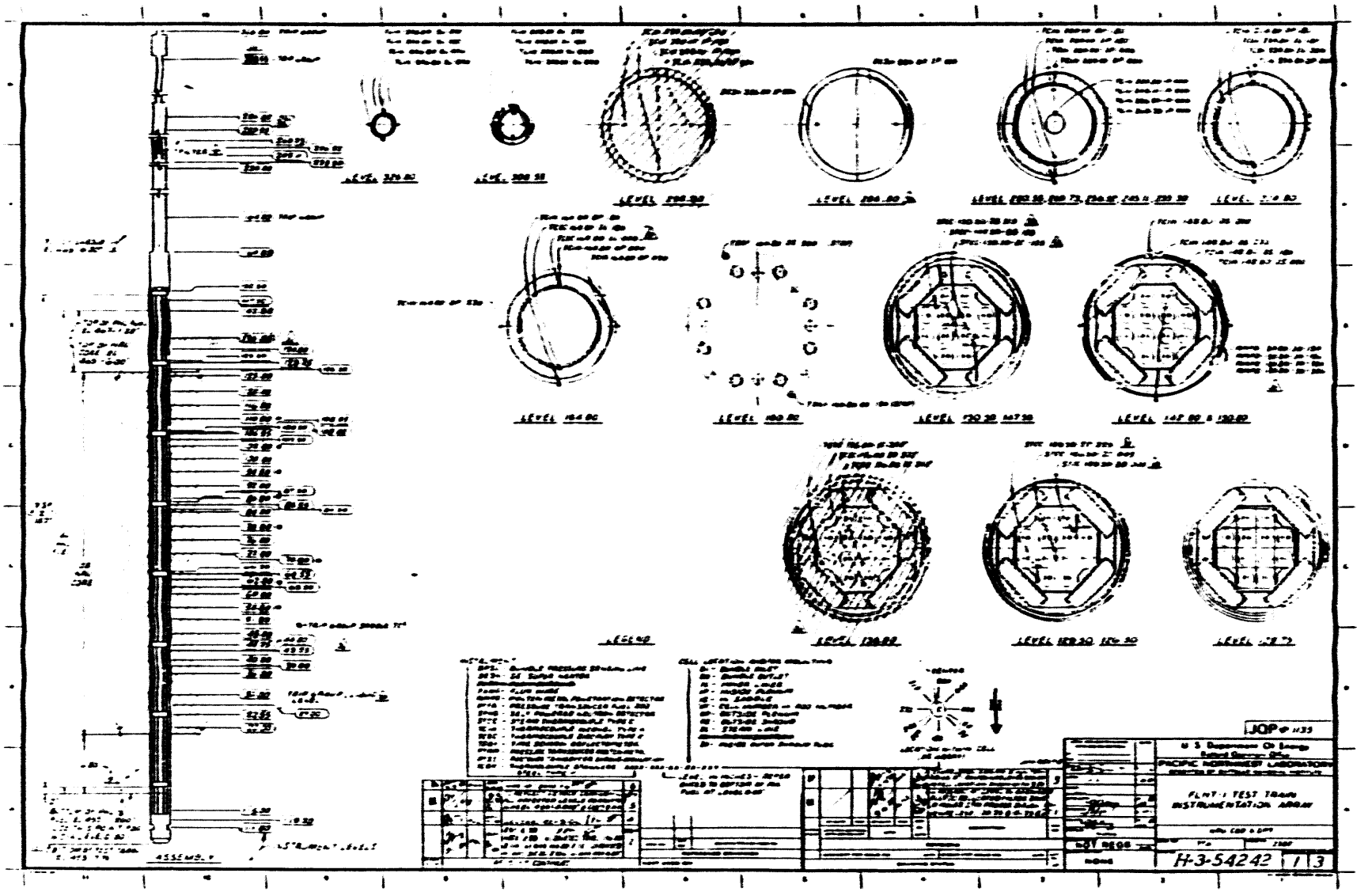

FIGURE 3.14. Instrumentation at Levels 128.75 to 328

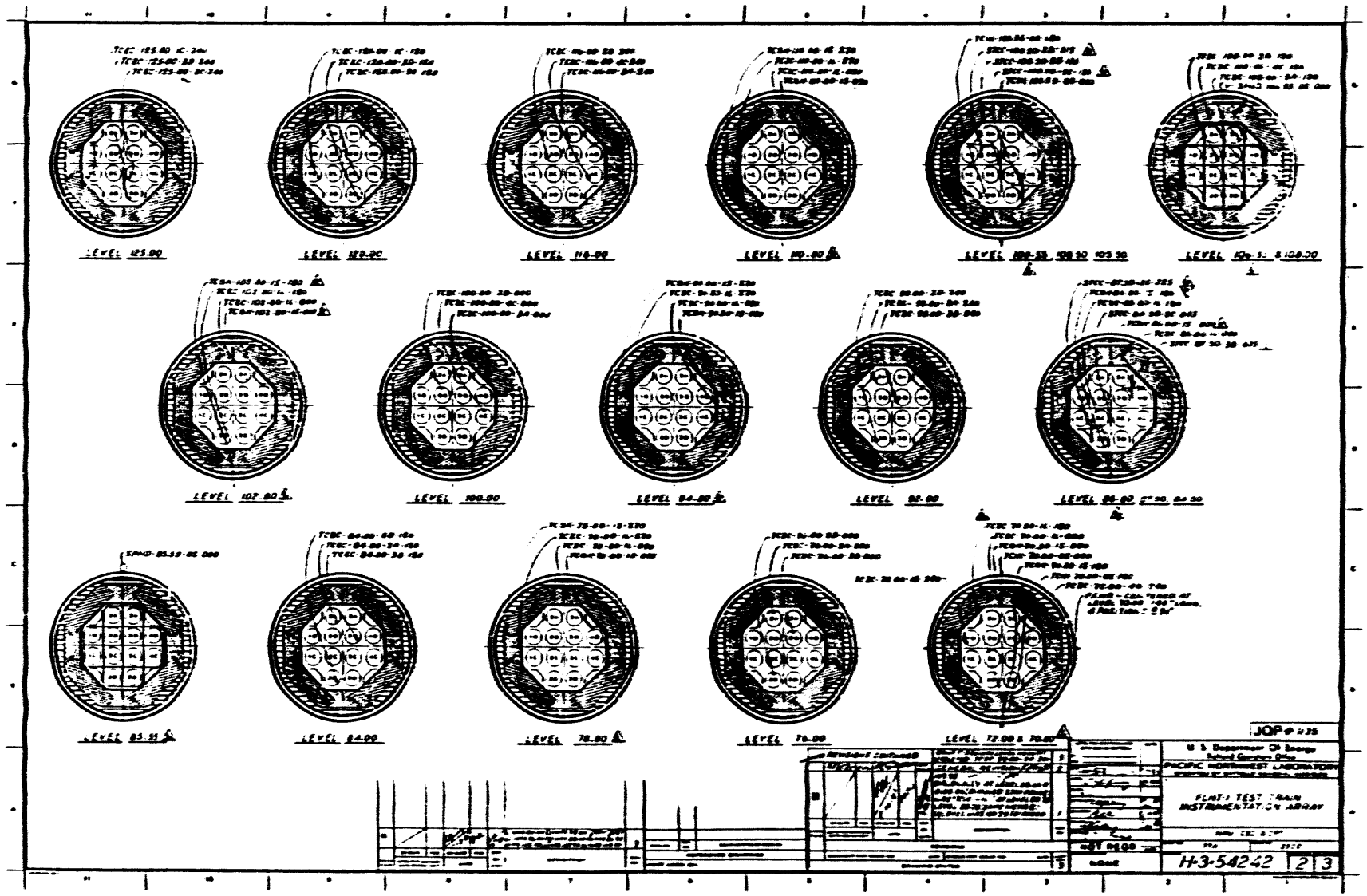

FIGURE 3.15. Instrumentation at Levels 72 to 125

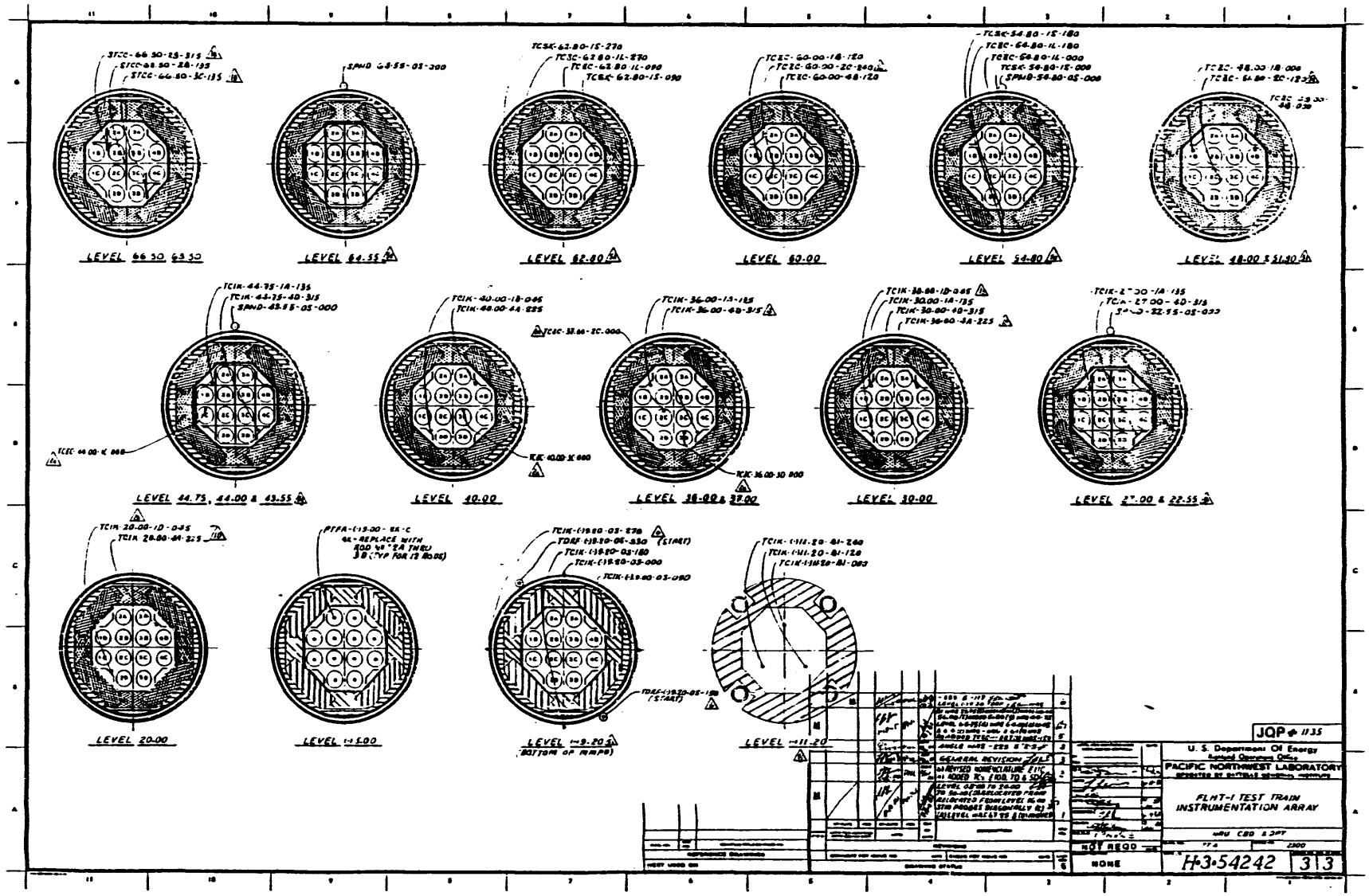

FIGURE 3.16. Instrumentation at Levels -10 to 67.25

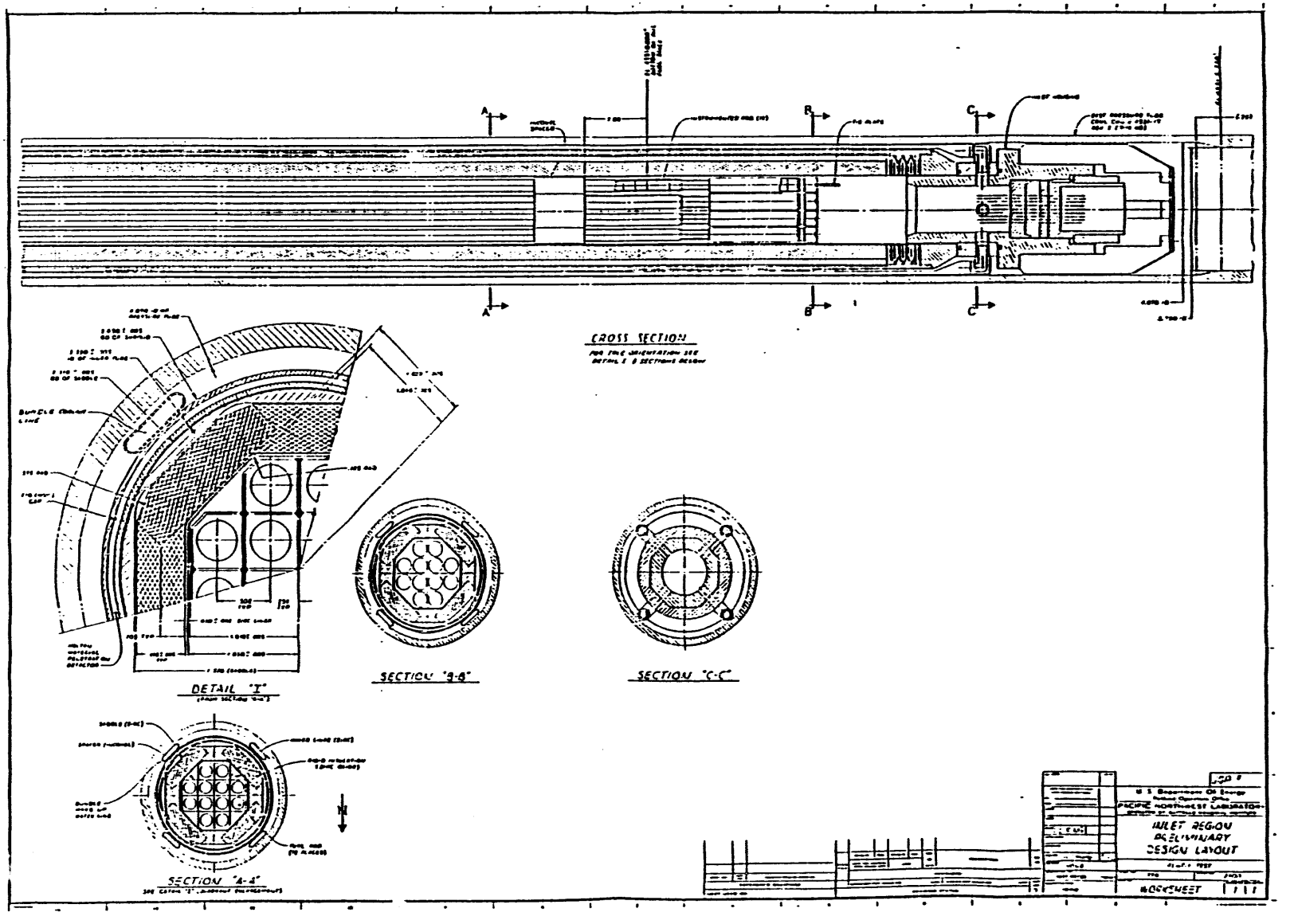

FIGURE 3.17. Inlet Region Design

3.2 EFFLUENT CONTROL MODULE

The ECM condensed steam from the test train effluent and provided pressure control for the fuel rod bundle coolant. It also measured hydrogen generation and fuel fission products that migrated to the ECM. The primary components of the ECM were a condenser; porous metal filters; a thermal conductivity cell; various blocking, relief, and control valves; various instruments; and lead shielding of fission-product-contaminated components for radiation protection. A plan view shows the location of the ECM on the NRU reactor deck (Figure 3.18), and a schematic flow diagram of the ECM within its secondary confinement is shown in Figure 3.19. The hydraulic relationships between the ECM, the FLHT-1 test train, and NRU reactor loop facilities are illustrated in Figure 3.19. A more detailed flow diagram of the ECM is shown in Figure 3.20.

3.2.1 Pressure Control

Nitrogen was injected into the ECM condenser to control the fuel bundle coolant pressure and to provide a carrier gas to sweep hydrogen through the thermal conductivity cell. The nitrogen and the test train effluent (both gas and liquid) exited the ECM through one outlet pipe and passed to the loop radioactive waste catch tanks.

3.2.2 Confinement and Shielding of Fission Products

This equipment was located within a sheet metal confinement that was anchored to the NRU reactor deck. A jumper pipe connected the test train to the ECM and another jumper pipe connected the ECM to a radioactive waste outlet pipe and the loop catch tanks that are below the NRU reactor deck. These jumper pipes were shielded and within the ECM confinement system. The condenser, filter, and pipes of the ECM that could be potentially contaminated with radioactive fission products were all shielded with lead.

The ECM used a special ventilation system provided by CRNL to exhaust any accidental releases of radioactive gases to the ECM confinement. These releases were processed through the existing reactor filtration systems and discharged through the reactor stack.

3.2.3 Fission Product Measurement

A 2-μm (79-min.) filter in the main effluent line trapped aerosols and fission products, and a condenser provided a plateout site for other selected fission products. Soluble and gaseous fission products passed through the ECM to the loop catch tank. However, gamma detectors monitored fission product species as they exited the test train through the effluent pipe to the ECM and as they passed through the filter and condenser.

3.2.4 Commissioning

The ECM was commissioned at the NRU facility before the FLHT-1 test. The ECM was pneumatically and helium leak tested before it was attached to the

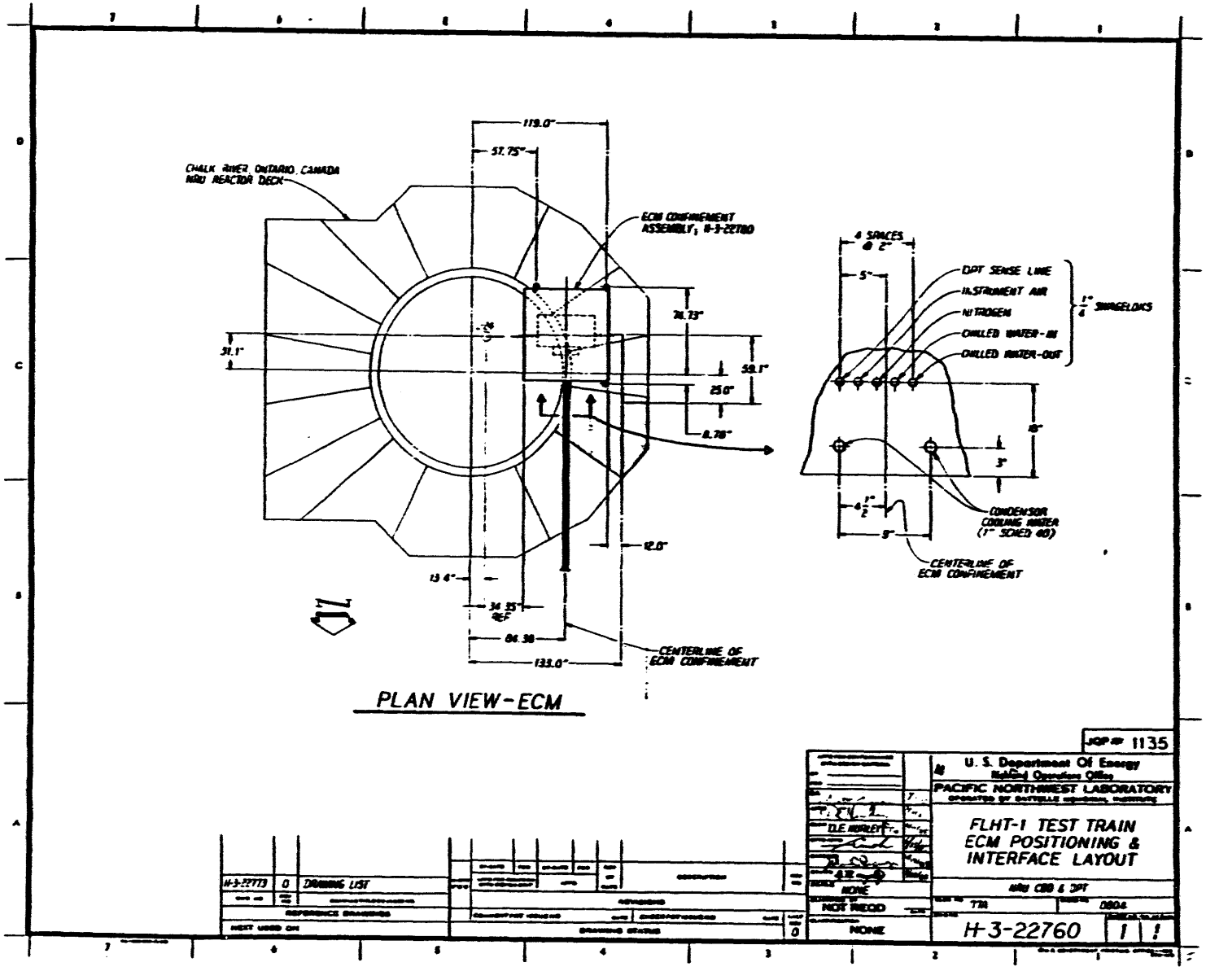

FIGURE 3.18. Plan View of Effluent Control Module

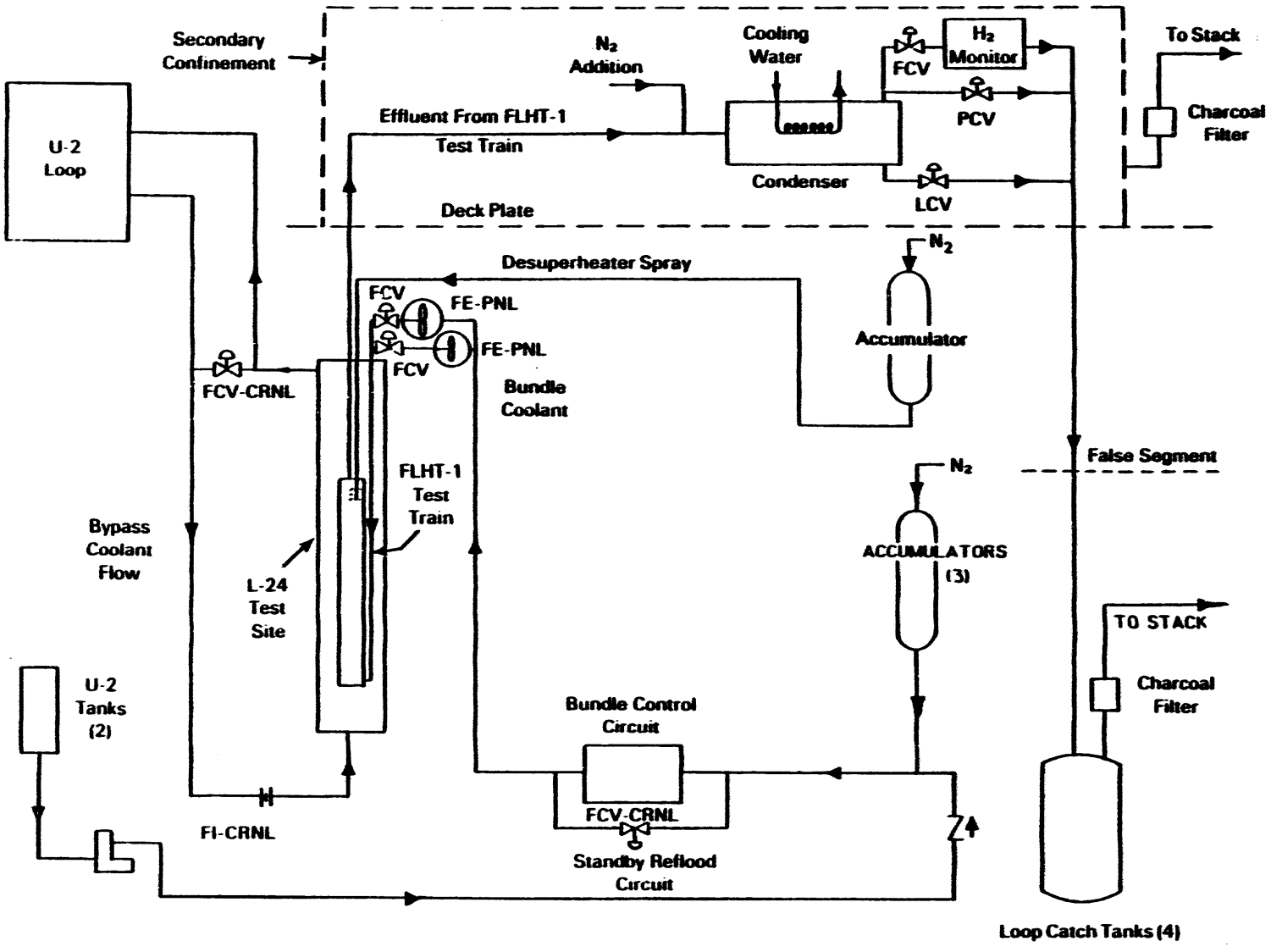

FIGURE 3.19. FLHT-1 Coolant Loop Flow Diagram

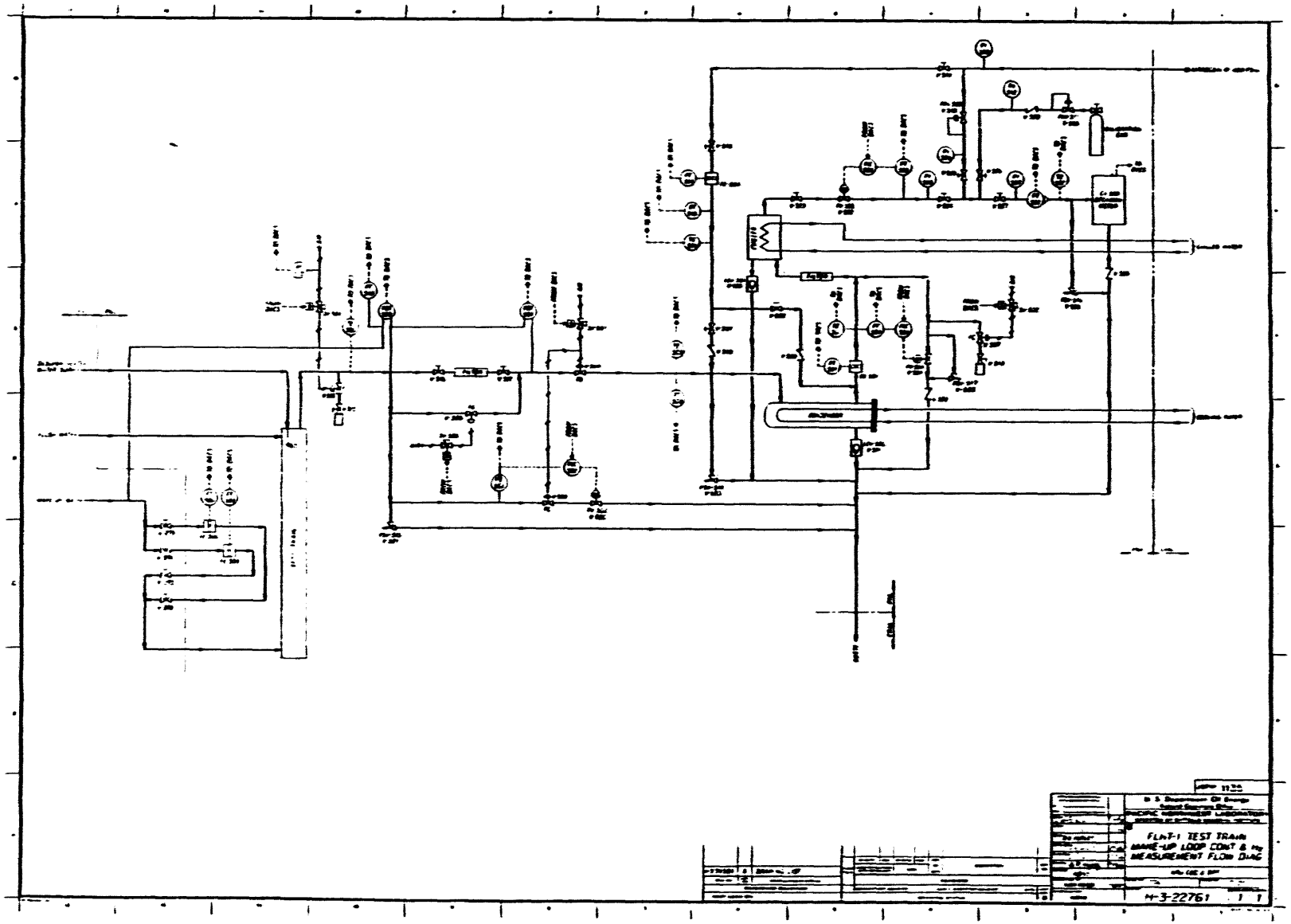

FIGURE 3.20. Effluent Control Module and Hydrogen Measurement Flow Diagram

test train. After it was connected to the test train and loop catch tank piping and before the test began, it was pressure tested along with the test train. After the test, the jumper pipes were removed and the ECM was lifted clear of the NRU deck. The ECM and the pipes were cleaned, placed in storage, and refurbished as necessary for future use.

3.3 DATA ACQUISITION AND CONTROL SYSTEM

The DACS used for FLHT-1 comprised several major components: a super-minicomputer, an analog-to-digital (A/D) subsystem, two color graphics terminals, and several character terminals. The super-minicomputer used the AOS/VS virtual memory operating system and was equipped with two megabytes of semiconductor memory, two 1600-bpi tape drives, two 190-megabyte disk drives, and a line printer. A small dot matrix printer was attached to one of the terminals; two other terminals were connected to a hard copy unit so that either of them could print a screen copy.

3.3.1 Operational Objectives

The DACS hardware and software were designed to accomplish the following objectives:

- data handling and scanning
- tape and disk input and output
- on-line graphics and terminal input/output (I/O)
- test control (calibration, startup, and controlling the bundle coolant flow)
- test termination (automatically or manually shutdown the NRU reactor)
- post-test data examination and output.

3.3.2 System Configuration

The terminals were set up in the following configuration. One character terminal was used as the console to control the DACS; one character terminal and one graphics terminal were used by the senior test engineer for data monitoring and evaluation; and one character terminal and one graphics terminal were provided in a separate room for the use of test observers not involved in actually running the test. These two terminals were equipped with a variety of monitoring functions, but no control functions. The major components and the personnel stations for operating and observing the test are shown schematically in Figure 3.21.

The DACS software was designed to use the function keys of the terminals to initiate desired routines. Certain functions available to the console

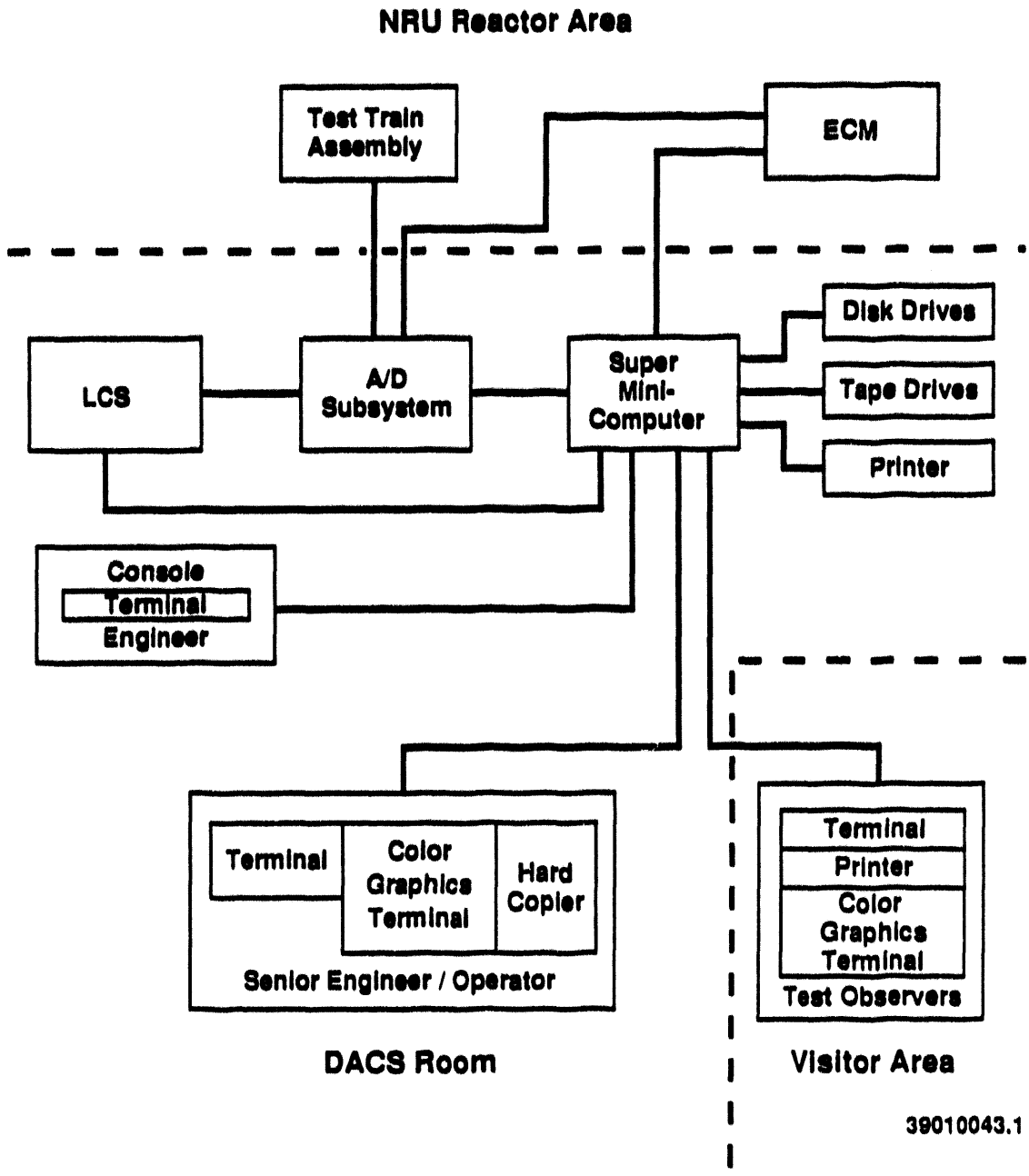

FIGURE 3.21. Connections for the DACS, LCS, ECM, Test Train Assembly, and Data Stations

operator were disabled in the other terminals. These special functions were necessary to operate the computer system, but they had no data reporting capability.

3.3.3 Reactor Safety Trips

Had it been necessary to terminate the test before completing the test, two means of tripping the reactor from the DACS were available--one automatic and one manual. The automatic trip, and its logic, is described first in the following. A brief note on the manual trip by the DACS console operator then follows.

Automatic Reactor Safety Trip

The hardware for the test was designed so that the reactor would be automatically tripped if set point or preset safety limits were exceeded during the test. These set points are listed in Table 3.1a. As a result of an automatic trip, a signal from the DACS would have been sent to the loop control system.

The automatic trip system worked in the following way during the test. The DACS scanned the instruments between one and five times per second recording data on the performance of the reactor. After each scan, the

TABLE 3.1a. Test Safety Trip Functions, Automatic

Automatic Trips	Setpoint	Responsibility
Temperature		
Shroud saddle exterior - high	640 K (700°F)	PNL
Bypass coolant outlet - high	450 K (347°F)	CRNL
Upper plenum outside - high	465 K (370°F)	PNL
Plenum steam outlet - high	640 K (700°F)	CRNL
Bundle coolant low level Level-30 TC Group	510 K (455°F)	PNL
Pressure		
ECM test train effluent - low (85 psig)	0.59 MPa	PNL
Flow Rate		
Bypass coolant - low (300 lb/h)	163 kg/h	CRNL
Bypass coolant - high (2000 lb/h)	909 kg/h	
Power Change		
Mean power log rate - high	15%	CRNL
Bundle Coolant		
Accumulator weight (25 lb)	11.3 kg	

readings of the instruments listed in Table 3.1a were checked against the preset limits. If after two successive scans of data they indicated that the reactor was operating beyond the established set points, a signal to trip the reactor would be sent from the DACS.

Thermocouples were the principal instruments used to record the data needed to determine whether the reactor was operating within established set points and preset safety limits. In most cases, the thermocouples were combined into arrays of four (M=4). Then, during the test, if two (N=2) of the four thermocouples in the array sent data indicating that the reactor, at the place where the array was located, was operating in excess of the set point, a software flag was set.

If the next scan of the instruments had then also shown N of the M thermocouples in that array reading beyond the set point, the DACS would have automatically tripped the reactor.

If, on the other hand, the second scan of the thermocouples had shown at least N of M thermocouples in that array reading beyond the set point, the software flag activated by the first scan would have been cleared and the reactor would not have been tripped.

This "two scans in a row" technique was used so that spurious electronic noise would not cause premature test termination. Because no dangerous or limiting conditions were encountered during the FLHT-1 test, the reactor did not have to be tripped, either automatically from the DACS or manually.

Other instruments were also part of the safety logic. For example, if two (N=2) of three (M=3) pressure transducers in the ECM piping measured low pressure, the reactor would have been shutdown after the second scan.

Whenever one thermocouple from a group of four safety sensors failed--read full-scale positive or below zero--during the test, it was removed from the safety sensor group definition without interrupting the test. The safety sensor group using that thermocouple was redefined as a two of three safety sensor group.

Manual Reactor Safety Trips

Conditions identified in Table 3.1b might have arisen during the test that would have caused the senior engineer to call for a shutdown of the reactor. The DACS enables the console operator to manually shut it down. During the FLHT-1 test, the senior engineer requested the NRU reactor operator to shut the reactor down, even though none of the conditions listed in Table 3.1b were abnormal.

TABLE 3.1b. Test Safety Trip Functions Manual

Manual Trips[a]	Condition	Responsibility
Sensor		
Bundle coolant differential pressure	high	PNL
Bundle coolant flow	large difference between measured versus demand	
MMPD continuity	change	PNL
MMPD resistivity	low	PNL
MMPD cavity pressure	high	PNL

(a) These conditions were continuously monitored and assessed to evaluate the course of the test. An approach to an unsafe condition would have been cause for a manual trip as outlined in the Final Safety Analysis Report (Russcher et al. 1985).

4.0 TEST OPERATIONS

This section provides an overview of the operations that were planned for the FLHT-1 test and of the operations that actually occurred. Information is included on the test equipment installation, planned operations, and actual operations.

The basic FLHT-1 test plan involved establishing a bypass flow through the annulus between the test train shroud and the pressure tube to cool the tube and maintaining the cold boundary necessary for the shroud to contain the high-temperature test. Once the bypass flow was set, a test assembly power equivalent to 0.524 kW/m (0.160 kW/ft) of fuel rod was generated by raising the reactor power and calculating the bundle power using coolant calorimetry. The water flow was then reduced to a value at which steam was exiting the assembly. A series of flow reductions were to follow until conditions were reached for which a predicted peak cladding temperature of 2150 K (3400°F) was calculated. The test was to then be terminated with no reflood flow; i.e., the test train was to be maintained dry, if possible.

The proposed operating conditions are summarized in Table 4.1. The time required to progress through the flow steps was calculated to be between 6 and 12 h. The actual test events varied somewhat from the planned operations. The actual operations are described in Section 4.3.

4.1 INSTALLATION AND CHECKOUT

Test installation involved inserting the test train into the L-24 position in the NRU reactor, mounting the ECM, and making the necessary service connections.

There were two piping systems for the FLHT-1 test. The first system supplied the bundle coolant to the test train. This coolant passed through the top closure, through the two bundle coolant lines, through four bundle coolant downcomer tubes, and then entered the bottom of the test train below the fuel rods. The coolant exited the test train through the effluent line to the ECM.

The bundle coolant entered the test train at the closure region. During pretransient operation, it subcooled the test train so that the test train power could be determined by means of a heat balance. The coolant was routed to the loop catch tanks.

The second piping system--the bypass coolant system--received coolant from the U-2 loop, measured it, and directed it up the annulus between the L-24 pressure tube and the shroud. It then returned the coolant to the U-2 loop.

Piping installation and checkout involved connecting the inlet and outlet cooling lines, ensuring proper flows and flowmeter calibrations, and ensuring leak tightness. The connections between the test train closure and the ECM were leak tested. These connections included the pressurization and pressure

TABLE 4.1. FLHT-1 Test Nominal Operating Conditions

Flow Rate	Value
Bundle coolant - calibration	0.151 kg/s (1200 lb_m/h)
Bundle coolant - operation	0.013 to 0.002 kg/s (100 to 16 lb_m/h)
Bypass coolant	0.063 to 0.252 kg/s (500 to 2000 lb_m/h)
Desuperheater water	≤8 g/s (60 lb_m/h)
ECM condenser water	315 g/s (2500 lb_m/h)
ECM chilled water	≤0.038 L/min (0.01 gpm)
ECM nitrogen	90 L/min (3.18 ft^3/min) STP
Power	
NRU reactor	~4% neutron full scale
Fuel rod - linear	0.524 kW/m (0.160 kW/ft)
Bundle	23 kW (22 Btu/s)
Temperature	
Peak fuel cladding	2150 K (3400°F)
Peak shroud saddle interior	425 K (300°F)
Bundle coolant inlet	340 K (150°F)
Bundle coolant saturation	467 K (382°F) at 185 psig
Bypass coolant inlet	340 K (150°F)
Bypass coolant outlet	365 K (200°F)
Bypass coolant saturation	470 K (382°F) at 185 psig
Peak plenum	1645 K (2500°F)
Peak plenum outlet	645 K (700°F)
Pressure	
Bundle coolant	1.28 MPa (185 psig)
Fuel rod	1.34 MPa (195 psig)
Bypass coolant	1.28 MPa (185 psig)
MMPD cavity	0 MPa (0 psig)
Shroud insulation cavity	≤0 MPa (≤0 psig)
Plenum Insulation cavity	≤0 MPa (≤0 psig)
Total Coolant Required	
Bundle coolant	540 kg (1200 lb_m)
Desuperheater	54 kg (120 lb_m)

measuring lines for each of the fuel rods, the shroud insulation cavity, the MMPD cavity, and the pressure transducer manifold (located inside the ECM). Finally, the ventilation system for the ECM was connected. The required flow conditions are shown in Table 4.1.

Once the test train was installed, the instrumentation cables were connected. Instrumentation was provided to collect data, control the test, and provide appropriate safety trips. Each safety trip listed in Table 3.1 and each controlled parameter listed in Table 4.1 was checked to ensure that it functioned properly.

4.2 PLANNED OPERATIONS

This section of text describing the planned test operations has been extracted from the experiment operations plan for FLHT-1. It is written in future tense.

> The FLHT-1 test will begin with pretransient operation to set the reactor power to give the desired test assembly power of 23 kW (22 Btu/s). Once the power is set, the test will be started through its transient operation. The term transient is somewhat of a misnomer; operation will consist of a series of preplanned, discrete flowreduction steps. The size and duration of each reduction is selected to control the steam-Zircaloy reaction--and hence the temperature ramps and hydrogen generation rate.
>
> Pretransient Operation
>
> Establishing the 23-kW (22 Btu/h) test train power will be accomplished by setting the bundle coolant flow rate to about 0.126 kg/s (1000 lb_m/h) and an inlet temperature of about 310 K (100°F). The reactor power will then be adjusted to the 23-kW (22 Btu/h) level as determined from the bundle flow rate and inlet-to-outlet temperature difference. This difference is about 45 K (80°F), which gives an outlet temperature of about 355 K (180°F), well below the saturation temperature of 467 K (382°F). The exact values of the bundle coolant parameters are not critical; once they are set, however, their values should be maintained constant. The reactor power necessary to give a test train power of 23 kW (22 Btu/s) will depend on the U-235 enrichment of the fuel; it is expected to be 4% to 6% of full power.
>
> Bypass cooling will not be needed during this time but will be established at its initial value before the pretransient operation is started. Thus, it will be at the proper value when the transient is started. It will always be on whenever the reactor is operating.
>
> Transient Test
>
> The transient test is started after the test train power is established. The start is accomplished by using DACS input to the LCS to

cause the bundle coolant flow rate reduction necessary to give peak cladding temperature of 925 K (1200°F)--well below the expected clad rupture temperature of ~1275 K (1800°F). The nitrogen flow into the ECM will be at a pressure of about 1.28 MPa (185 psig). The test train will then be drained until the liquid level is in the core region. After the drain has been stopped, the crossover connection and drain line will be capped. The bundle coolant flow will be adjusted to the rate of the first planned step, and boildown of the coolant will be allowed until a steady-state temperature and liquid level are reached. The expected temperature and liquid level are 1300 K (1900°F) and 183 cm (72 in.). Fuel rod rupture will probably occur during this step.

The bundle flow rate will then be decreased in a series of precalculated flow steps (Table 4.2). The duration of the time between steps is dictated by the time needed to reach near steady state and also by the requirement that the Zircaloy-steam reaction be limited. About 14 steps, each of about 1/2-h duration, are expected. The last flow reduction step will be calculated to give a peak cladding temperature of about 2150 K (3400°F).

Bypass coolant will be maintained during the flow reductions. The bypass system pressure will be maintained at 1.28 MPa (185 psig), the same as the bundle coolant pressure, to minimize the possibility of any leaks between the two systems. The bypass coolant inlet temperature will be ≤340 K (≤150°F) and the flow rate will be adjustable between 0.063 kg/s (500 lb_m/h) to 0.252 kg/s (2000 lb_m/h). The

TABLE 4.2. Expected Bundle Coolant Flow Steps for FLHT-1

Time, h	Bundle Coolant Flow Rate kg/h	Bundle Coolant Flow Rate lb/h
0.0	31.8	60.0
0.5	15.7	34.6
1.0	14.0	30.7
1.5	13.1	28.8
2.0	12.2	26.8
2.5	11.3	24.9
3.0	10.5	23.0
3.5	9.9	21.8
4.25	9.4	20.7
5.0	8.9	19.7
5.5	8.5	18.7
6.0	7.9	17.4
6.5	7.4	16.3
7.0	6.9	15.2
7.5	6.3	14.0
8.0	5.8	12.8

flow will be adjusted to give temperature increases from inlet to outlet large enough to allow reasonably accurate measurements of heat losses. A maximum outlet temperature of less than 365 K (200°F) is expected. This is well below the saturation temperature of 467 K (382°F).

Test Termination

The prime criterion for determining the success and termination point of the FLHT-1 test is achievement of a peak fuel cladding temperature of approximately 2150 K (3400°F). Once conditions giving this temperature have been reached, the NRU reactor will be manually shut down. The bundle coolant flow will be shut off, but the bypass coolant flow will be continued. This shut-down method will provide the least thermal shock and therefore minimize post-test fuel bundle damage. Nitrogen flow to the ECM for pressure control will be maintained, but dilution nitrogen may be stopped after hydrogen release rates fall to negligible values.

4.3 TEST OPERATIONS

The data recording for the FLHT-1 test began at 8:10 a.m. on March 2, 1985. The NRU reactor was brought to low neutron power at 8:15 a.m. At 9:21 a.m., the DACS trips (safety circuits) were enabled and reactor power was increased until the test fuel bundle power was 23 kW.

By 10:00 a.m., after the bundle power had stabilized, the type C thermocouples were showing temperatures below ambient; it became obvious that the temperatures signaled by those thermocouples were in error. The test was maintained at power while the test engineers discussed the unexpected readings. They concluded that the thermocouples had been wired incorrectly during assembly of the test bundle. The DACS software was modified to accommodate the wiring arrangement, and the test was resumed about 1:00 p.m.

4.4 UNPLANNED EVENTS

Unplanned events were experienced in establishing the initial steady-state condition with the liquid level in the top 0.6 m (2 ft) of the test assembly. This step would have been followed by the first coolant rate reduction to about 16 kg/h (35 lb/h), which would have shown a peak clad temperature of about 1310 K (1900°F). Attempts to achieve the initial steady-state condition were made by adjusting the bundle coolant flow.

When the first attempt to drain coolant from the plenum region above the fuel rods was made, the coolant in the fuel region was essentially stagnant. This coolant heated up rapidly, causing a "steam bubble" to form which then rose to the top of the test train. To avoid premature damage to the fuel from excessive temperatures, the drain was stopped and additional coolant re-introduced to the fuel. Unfortunately, this action left the test train in the same subcooled state it was in before the drain began.

An additional problem was that the TDRs did not work as planned. The steam condensed along the sides of the plenum and ran down into the TDR tubes, causing faulty TDR readings. The drain was continued more cautiously, and eventually the liquid level was near the top of the fuel. However, problems with the control of the liquid level persisted, mainly because of steam condensation on the plenum walls.

The deviation from the expected FLHT-1 test progression is attributed to refluxing. When the test was started, the approximately 4.6-m-long (15-ft.) plenum above the fuel was cool--below saturation temperature. The steam coming from the test section condensed as it entered this cool region and fell back into the test section. In effect, this refluxing provided a larger bundle coolant supply.

The excessive radial heat losses to the bypass coolant were attributed to the condensation and refluxing. However, the measured heat losses were not large; moreover, calculated radial heat flow when the inside plenum was at saturation temperature was very small. Evidently, the plenum, closure head, and vertical outlet piping acted as a heat sink. Eventually, they would have heated until they no longer caused condensation and refluxing.

4.5 STEADY-STATE CONDITION

A steady-state condition, with the liquid level around the top of the fuel, was never achieved. Instead, one of the many fluctuations in liquid level caused by the refluxing and operator adjustment of drain and reflood valves allowed the liquid level to drop well below the top of the fuel. This low coolant level allowed the fuel rods to heat rapidly, until a rapid metal-water reaction occurred, resulting in high temperatures in the fuel region.

Typical cladding temperature behavior at one position in the assembly during the test is shown in Figure 4.1. At about 60 to 70 min along the abscissa, a temperature increase was started when the flow rate was about 9 kg/h (20 lb/h). The temperature increased until about 95 min and 1450 K (2150°F), at which time the bundle coolant rate was increased to 18 kg/h (40 lb/h) to stabilize the temperature. However, the temperature rapidly dropped to about 1060 K (1450°F). The bundle coolant flow rate was then decreased through a series of steps to a minimum of 9 kg/h (20 lb/h). This action stopped the temperature decrease and started another temperature rise. When the temperature reached about 1475 K (2200°F), the bundle coolant flow was again increased to stop the temperature ramp. This led to a stabilized condition. The flow was increased in steps and reached a maximum of about 15 kg/h (34 lb/h). These flow rates did not stop the temperature rise, and a rapid metal-water reaction raised the temperatures rapidly until the test director requested that the reactor power be reduced to zero power.

4.6 TERMINATION OF EXCURSION

The reactor power was decreased at approximately 17:11:07, 85 s after the start of the excursion (approximately 131 minutes in Figure 4.1). The reactor

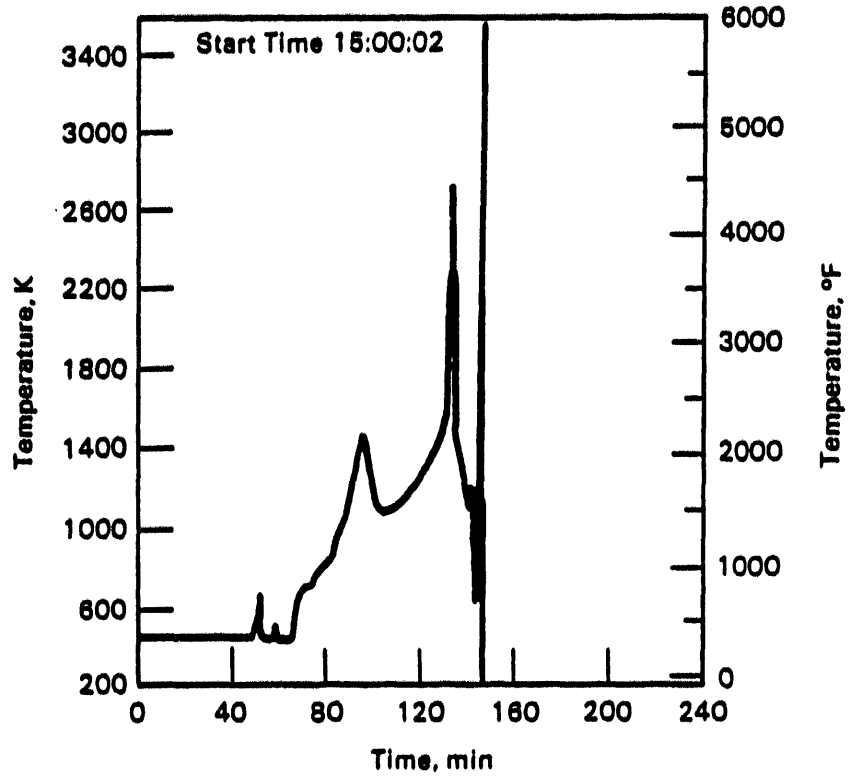

FIGURE 4.1. Typical Cladding Temperature Behavior

reached 10% of the initial power approximately 35 s later and reached low neutron level in another 30 s.

There were two indications at the time of the test that raised doubt that the shutdown of the reactor had effectively terminated the temperature excursions. The first indication was rising temperatures from bundle and liner thermocouples that gave no positive indication of failure. The second indication was a rising hydrogen level shown on the thermal conductivity hydrogen monitor.

A review of the thermocouple data led to the conclusion that the temperatures were not rising after the reactor shutdown. Typical cladding, coolant, and liner temperatures immediately after the reactor shutdown are shown in Figures 4.2, 4.3, and 4.4, starting at 17:12:00. The temperatures shown are somewhat erratic and show noise (probably associated with some thermocouple damage), but the general trend is downward, indicating an effective shutdown.

Additional indications of an effective test shutdown are shown by the saddle temperature, MMPD response, and bypass coolant power (radial heat loss) after the reactor power shutdown. Typical data from these sources are shown in Figures 4.5 through 4.7. All three of these indicators show steadily decreasing temperatures. Table 4.3 is a summary of the events of the FLHT-1 test.

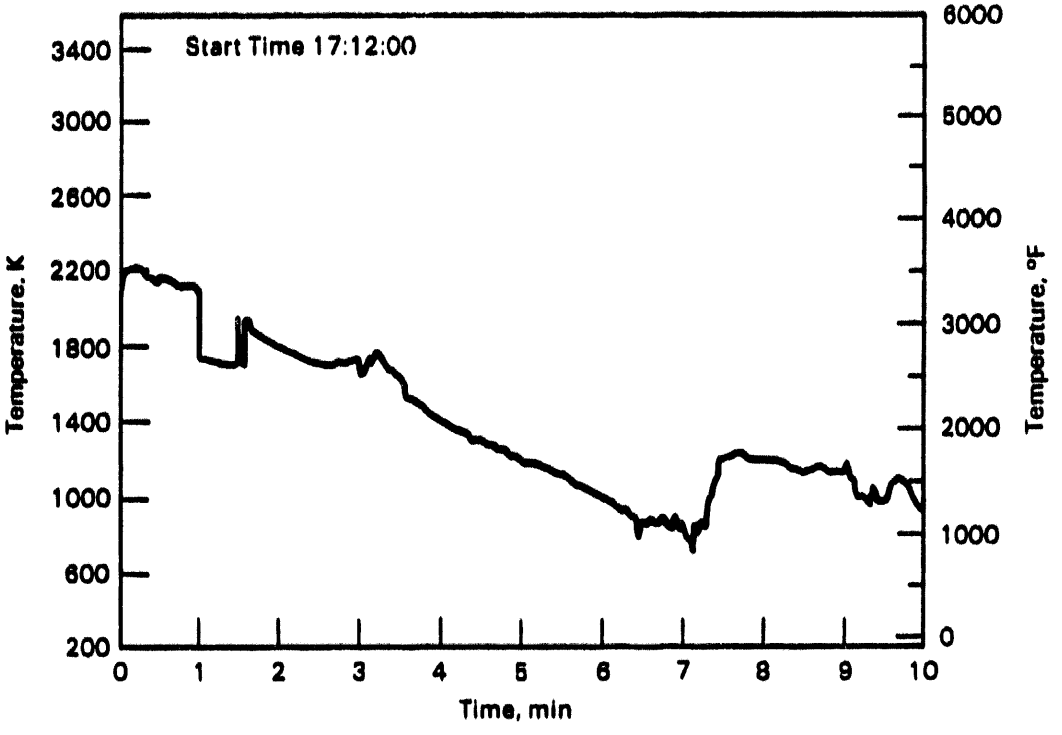

FIGURE 4.2. Typical Post-Shutdown Cladding Temperature Behavior

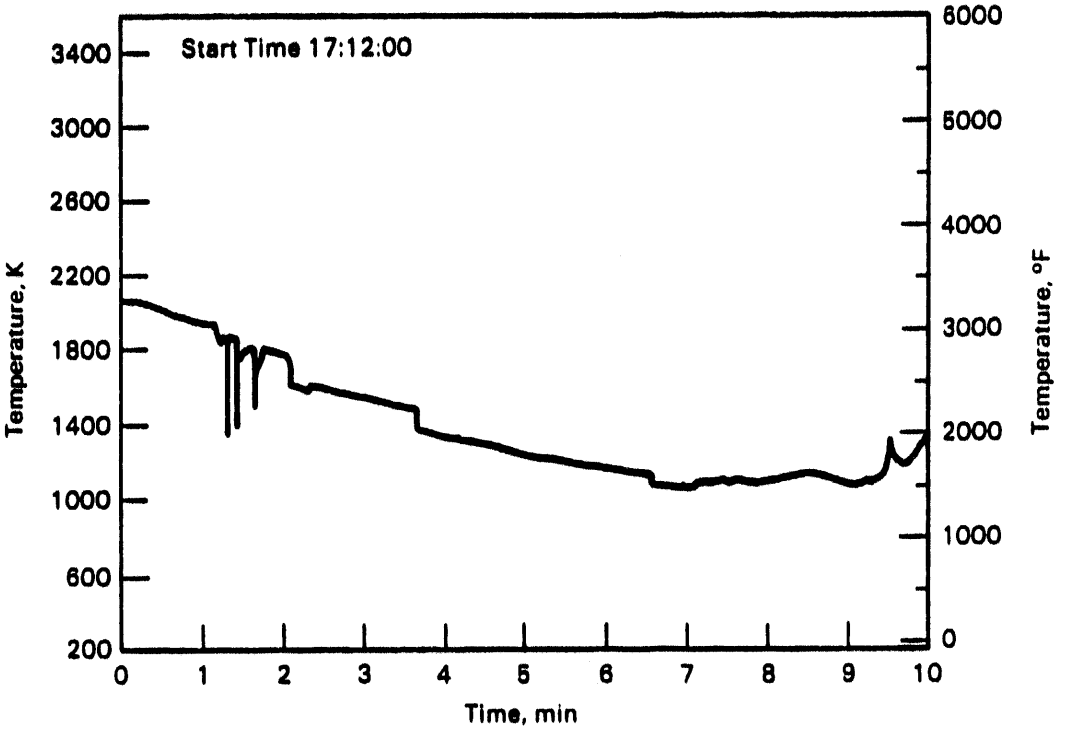

FIGURE 4.3. Typical Post-Shutdown Coolant Temperature Behavior

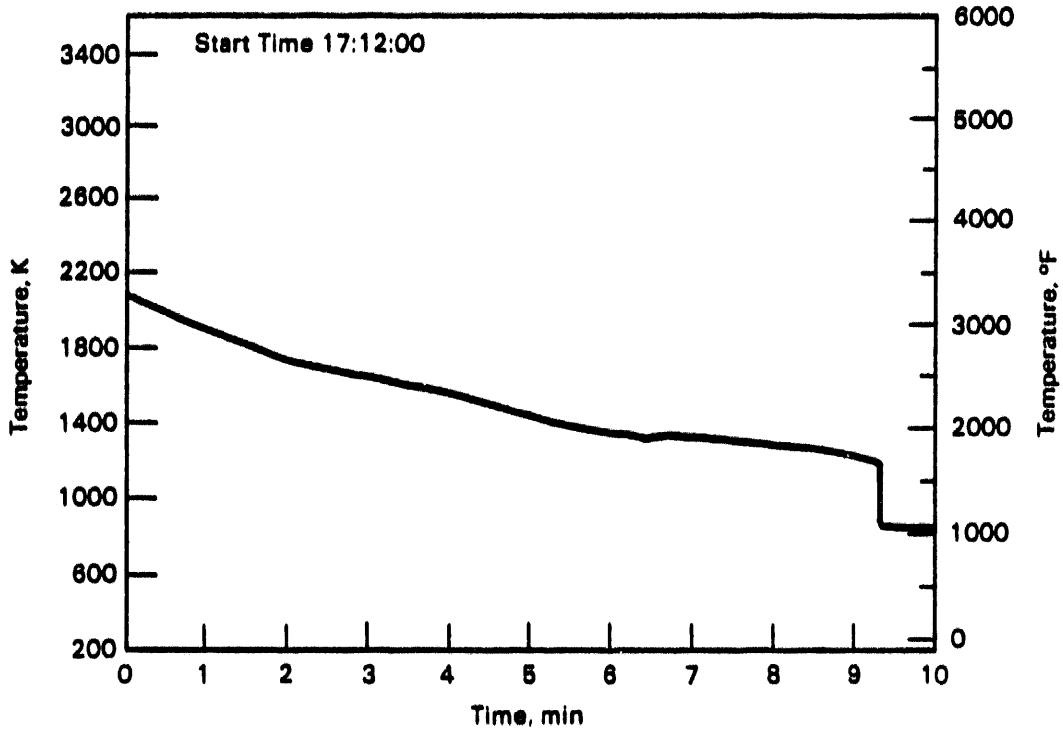

FIGURE 4.4. Typical Post-Shutdown Liner Temperature Behavior

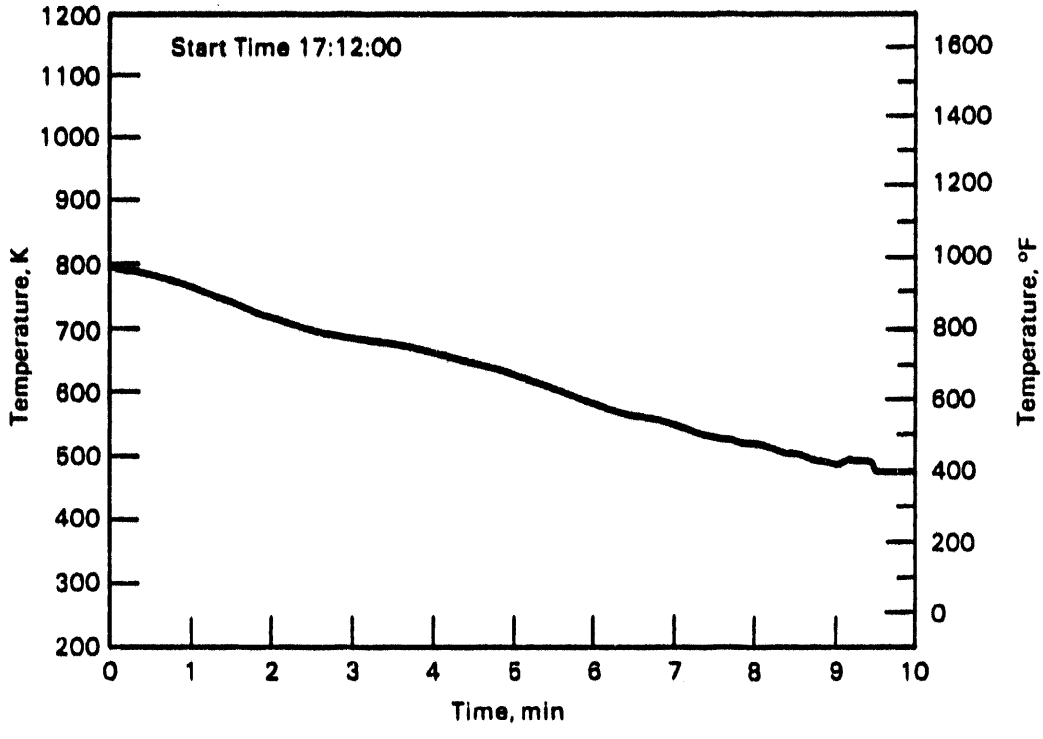

FIGURE 4.5. Typical Post-Shutdown Saddle Temperature Behavior

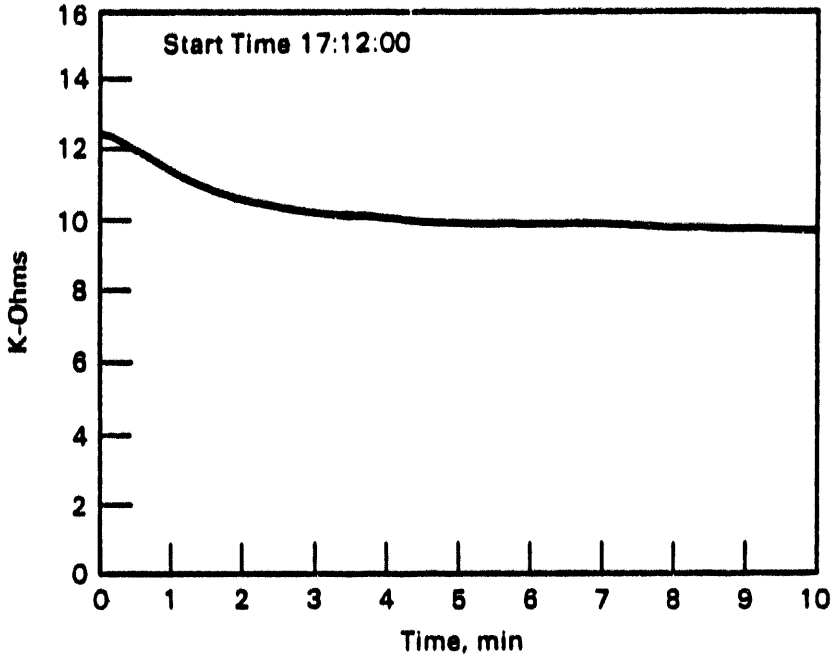

FIGURE 4.6. Typical Post-Shutdown MMPD Response

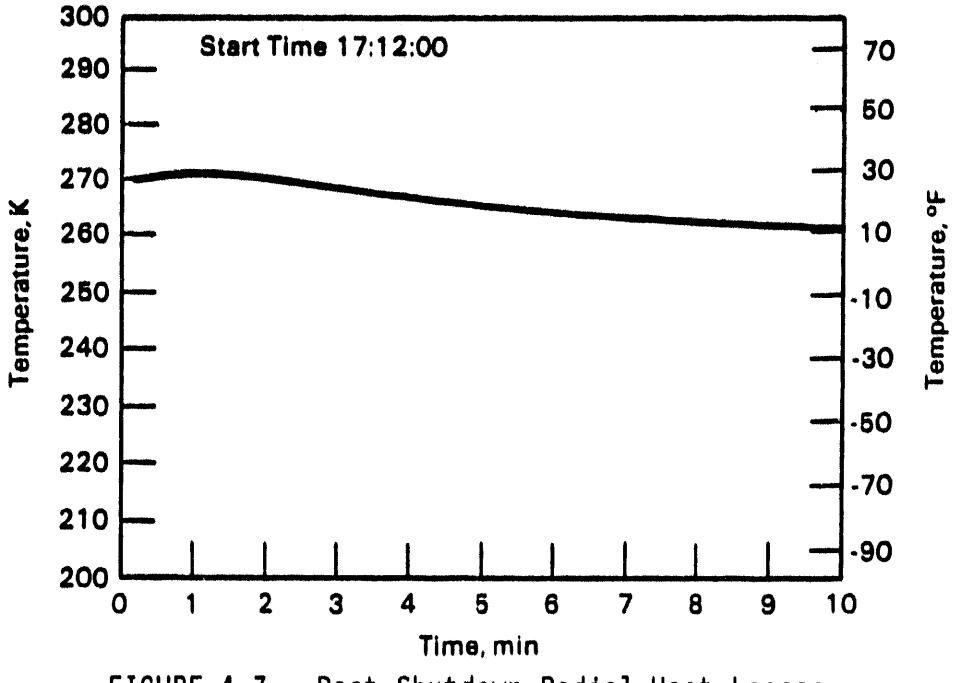

FIGURE 4.7. Post-Shutdown Radial Heat Losses

TABLE 4.3. Sequence of FLHT-1 Test Events (March 2, 1985)

Time	Event	Remarks
~10:00:00	Reactor to power, ~7%	Test assembly power = 23 kW (22 Btu/s)
16:20:00	Temperature rise started	Flow = 0.0, small heat losses < 1 kW (1 Btu/s)
16:36:00	Rods ruptured	
16:36:00	Temperature 1410 K (2080°F)	Flow = 40 lb/h (5 g/s)
16:56:00	Temperature 1120 K (1558°F)	Flow = 20 lb/h (2.5 g/s); heat loss ~8 kW (8 Btu/s)
17:05:00	Temperature ~1420 K (2100°F)	Flow = 20 lb/h (2.5 g/s)
17:09:50	Insulation cavity breached	
17:09:90	Shroud inner liner fails	
17:10:00	Excursion started ~1700 K (2600°F)	Flow = 30 lb/h (3.8 g/s)
17:11:07	Peak temperature reached	2275 K (3635°F)
17:11:07	Reactor power reduction started	Flow = 35 lb/h (4.4 g/s)
17:11:44	Reactor power = 10% of initial by SPND	
17:12:14	Reactor at low neutron level	Saddle TC peaked at 750 K (900°F) and 2.8 m (110 in.) Flow = 0
17:18:01	Peak hydrogen generation rate	Coolant turned on intermittently at 35 lb/h (4.4 g/s)
17:19:08	Minimum hydrogen generation rate	
17:20:08	Peak hydrogen generation rate	
17:24:00	Low hydrogen generation	

5.0 TEST RESULTS

During the hours the FLHT-1 test was underway, the DACS recorded over 17 million data points. These data were primarily temperature and pressure measurements from instruments in the test assembly but also included control information such as flow rates and valve positions from the LCS and the ECM. This data is summarized in Section 5.1. It is also compared with pretest predictions. Graphs of the data are included.

Section 5.2 presents results from the post-test examination of the test assembly, including photographs of the shroud, the shroud insulation, and the fuel bundle.

5.1 RECORDED DATA

The independent variables in this test were the bundle coolant flow rate and fission power. The following are the dependent variables for which data were collected:

- cladding temperature
- coolant (steam) temperature
- hydrogen concentration
- shroud liner temperature
- fuel rod rupture temperature and pressure
- radial heat losses through the shroud.

In addition, other data were collected for safety reasons; for example, bypass flow data, MMPD information, and shroud saddle temperature data. Still other data were taken to ensure controlled conditions. These data included system pressure, reactor power, and nitrogen carrier gas flow rate.

5.1.1 Temperature Measurements

The enormous amount of temperature data collected during the test will be abstracted in this report to show important trends and to present specific information such as the peak fuel temperature as a function of time, radial temperature gradients, and axial temperature profiles.

The predicted temperatures for FLHT-1 are summarized in Figures 5.1, 5.2, and 5.3. Figure 5.1 shows the predicted peak cladding temperature as a function of time into the test. Figures 5.2 and 5.3 show the predicted axial and radial temperature profiles.

Figure 5.1 shows the "steps" mentioned in Section 4.2 (planned test operation). Figure 4.1 (referred to in Section 4.3) shows a "typical" thermocouples response during the test. Figures 5.4, 5.5, and 5.6 show "pseudo sensor" readings for the peak temperature region of the fuel, shroud inner liner,

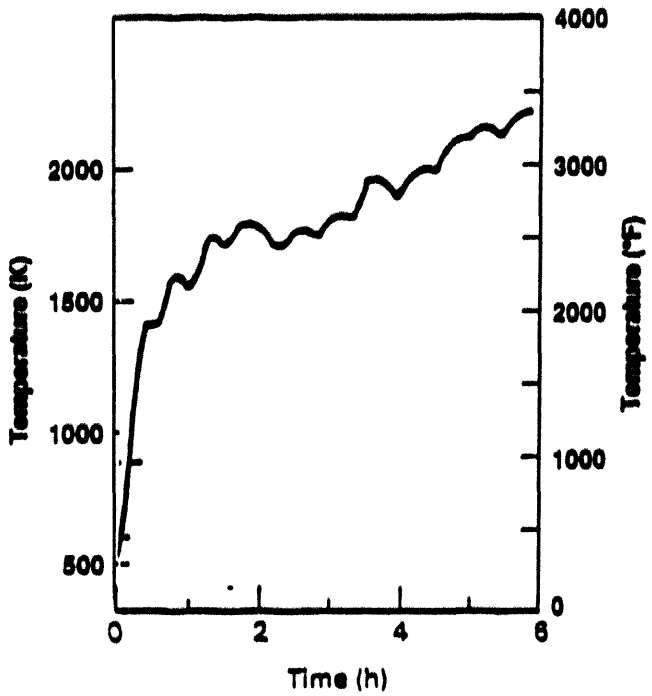

FIGURE 5.1. Predicted FLHT-1 Peak Cladding Temperatures with Zircaloy + Water Reaction But No Fuel Redistribution

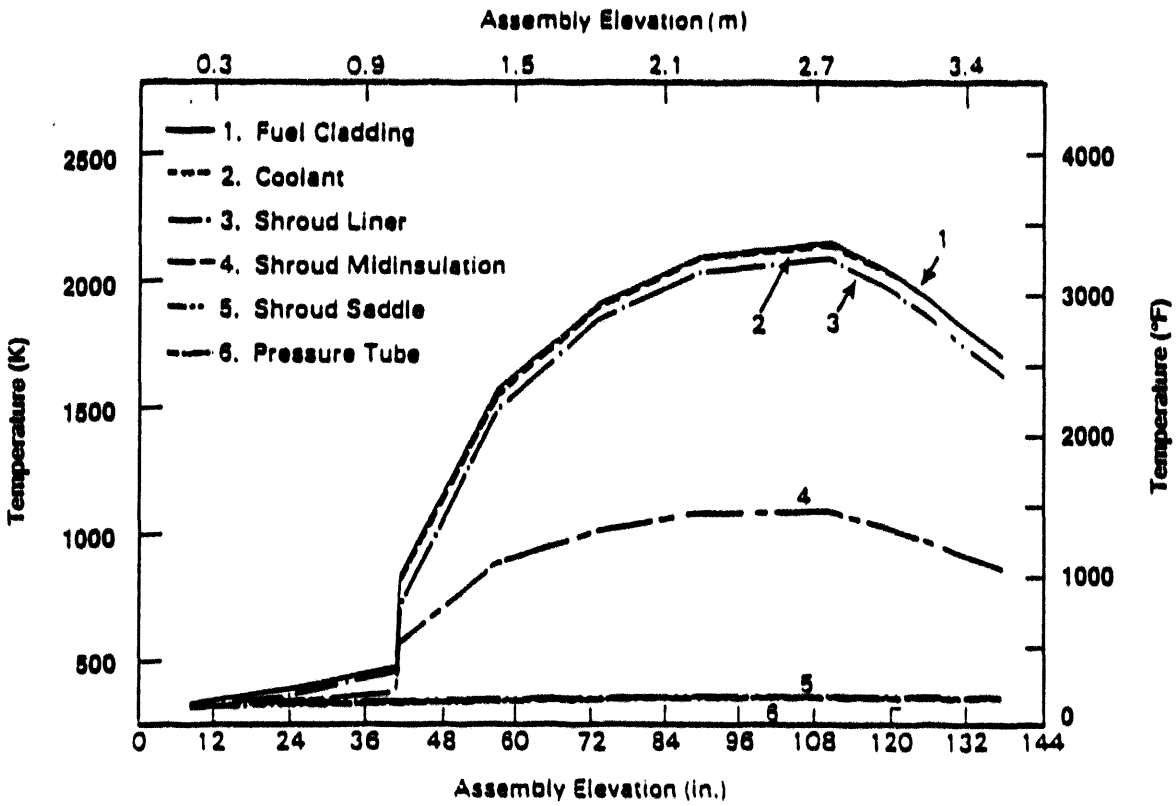

FIGURE 5.2. Predicted Axial Temperature Profiles for FLHT-1 with Zircaloy + Water Reaction and an Average Rod Power of 0.188 kW/ft

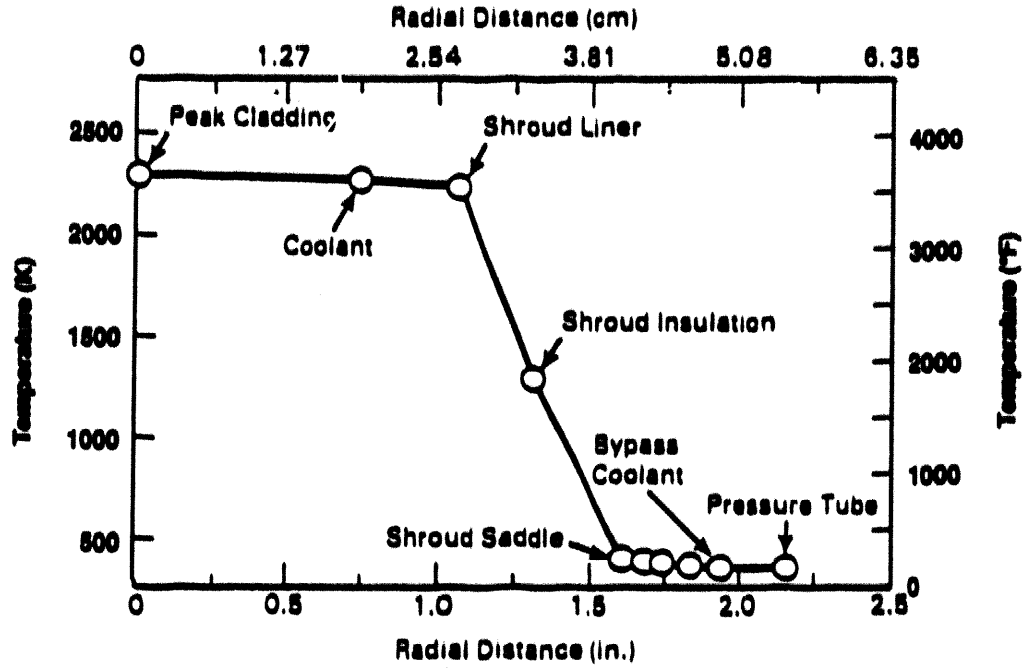

FIGURE 5.3. Predicted Radial Temperature Profile for FLHT-1 with Zircaloy + Water Reaction and an Average Rod Power of 0.188 kW/ft

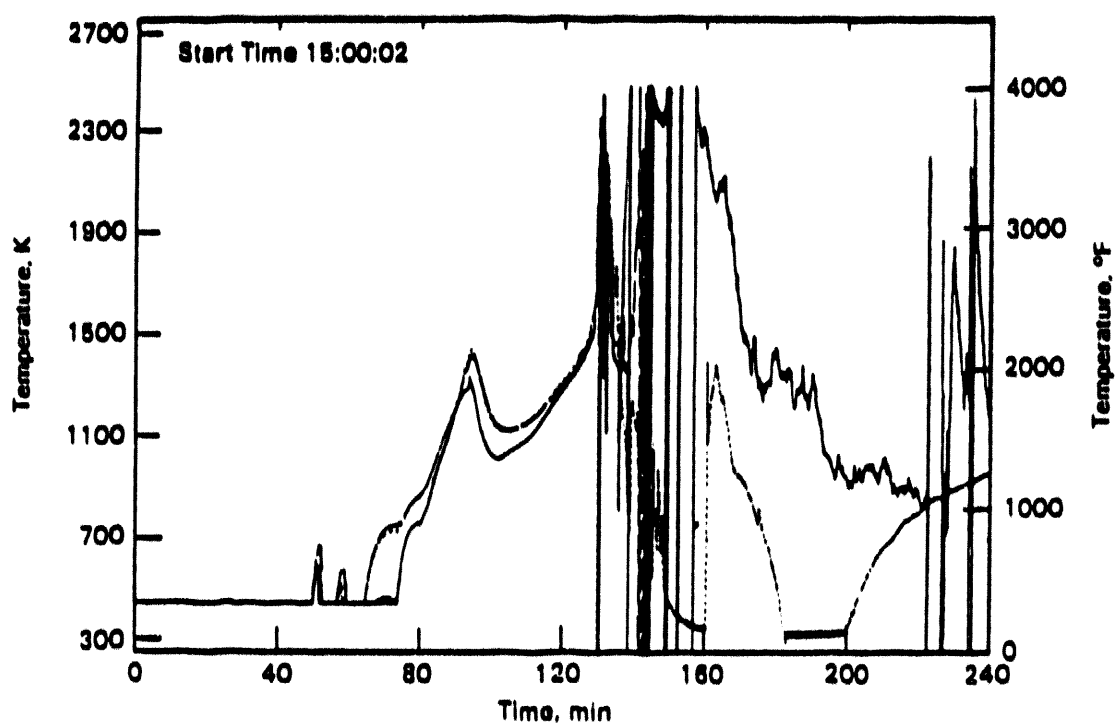

FIGURE 5.4. Pseudo Sensor Readings for Fuel Peak Temperature Region

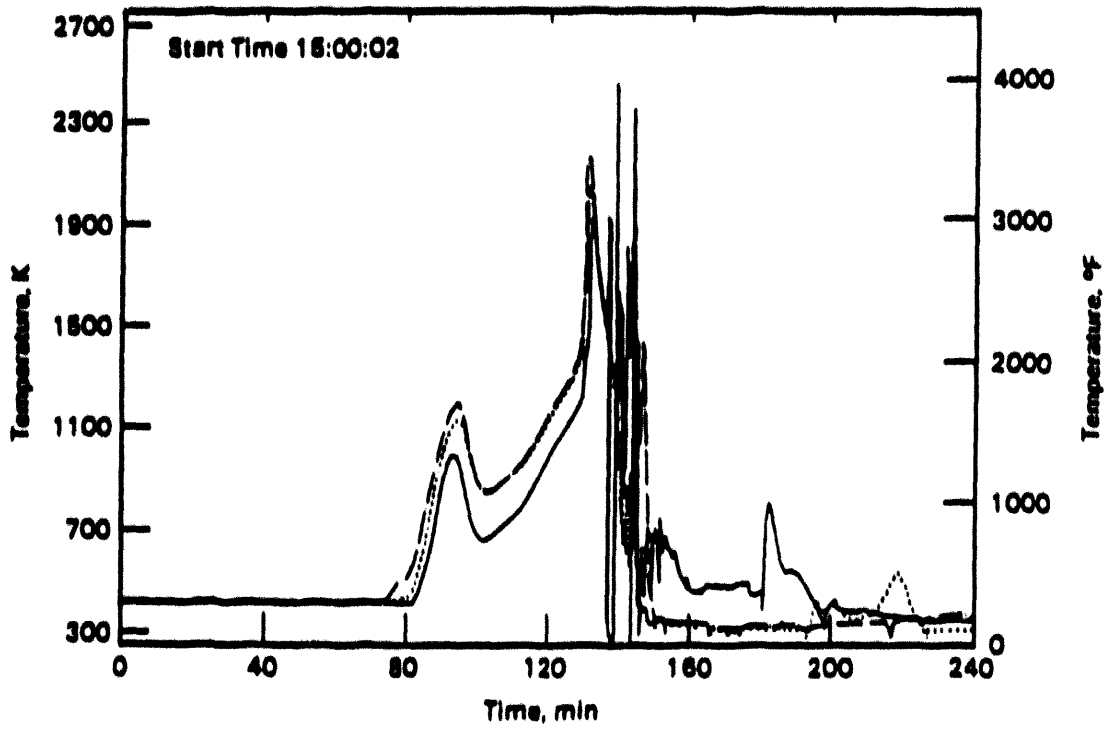

FIGURE 5.5. Pseudo Sensor Readings for Shroud Inner Liner Peak Temperature Region

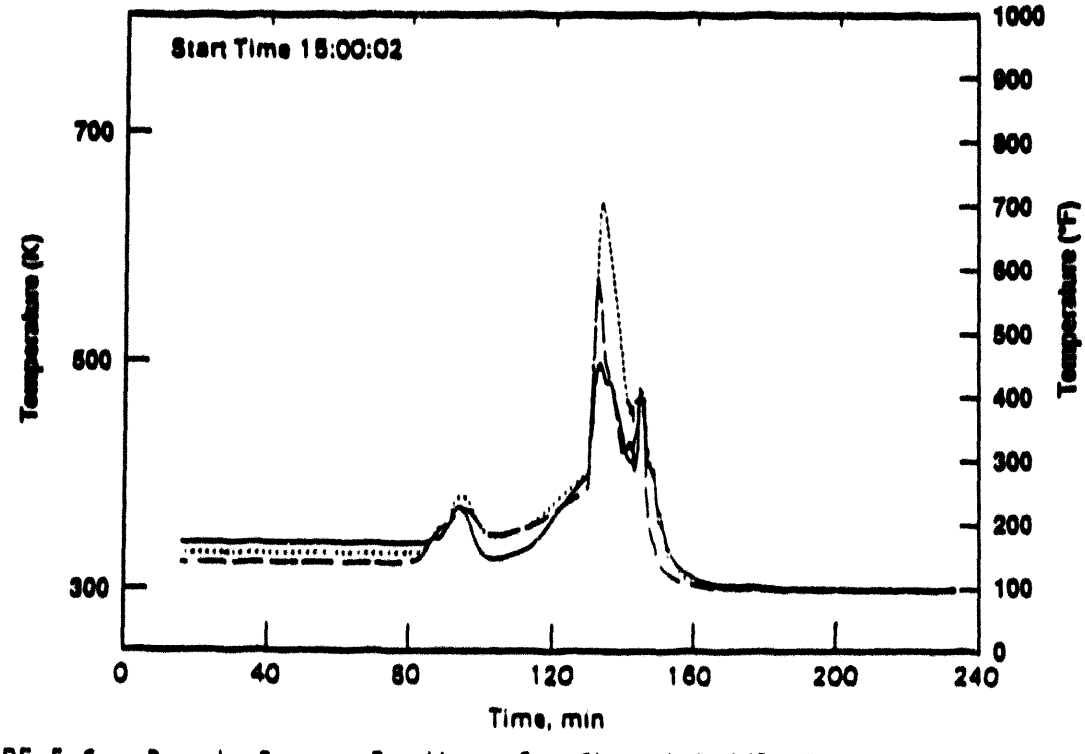

FIGURE 5.6. Pseudo Sensor Readings for Shroud Saddle Peak Temperature Region

and shroud saddles. (Pseudo sensors are averages of two or more real thermocouples). Figure 5.7 shows the axial temperature profile of the cladding, liner, and the saddle during the time of the high-temperature excursion.

The abrupt changes in temperature indicated by the almost vertical lines appearing in the thermocouple plots are not considered to indicate real conditions. Rather, the changes indicate some sort of failure mode in the thermocouple. Final judgement should be reserved on the apparently normal post-failure readings from some thermocouples. The data from them may very well be valid, but the post-failure thermocouples have not been tested in any controlled manner to recalibrate their outputs.

The measurement of actual peak cladding temperatures was not possible; the thermocouples for measuring those temperatures failed at about 2200 K. However, careful analysis of the entire data set measured by the thermocouples in the test train indicates a peak temperature on the order of 2275 K (3625°F). This conclusion is supported by the observations made during the post-test examination (Section 5.2), when some cladding melting was found.

5.1.2 Hydrogen Measurements

Pretest calculations showed that significant amounts of hydrogen would be generated when the Zircaloy-steam interface temperature exceeded 1200 K

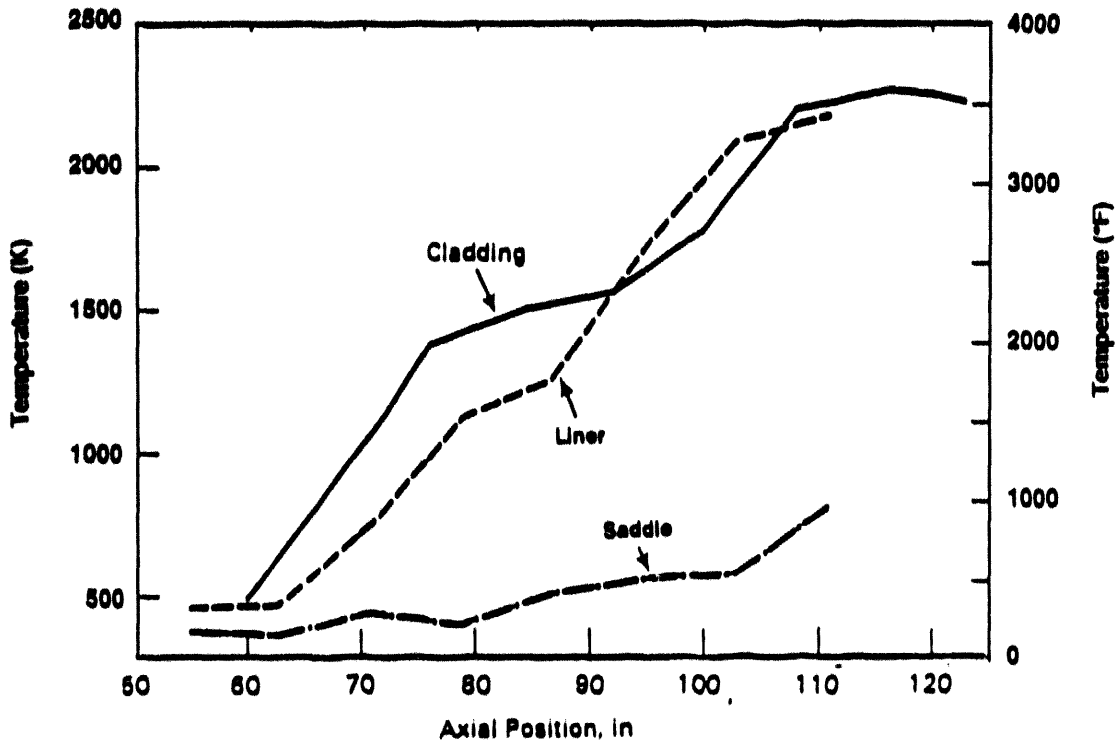

FIGURE 5.7. Axial Temperature Profile During High Temperature Time Period

(1700°F) (see Figure 5.1). A peak hydrogen generation rate of approximately 0.023 g/s (0.18 lb_m/h) was predicted, with as much as 220 g (0.5 lb) of hydrogen generated during the entire test (Figures 5.8 and 5.9).

The amount of hydrogen measured was only about 35 g (0.08 lb_m) for the entire test, but the hydrogen generation rate peaked at approximately 0.07 g/s (0.56 lb_m/h), which was about three times the predicted peak rate. This much hydrogen would be generated by oxidizing about 17% of the Zircaloy in the top half of the fuel bundle.

The effluent gas thermal conductivity meter measurement of the hydrogen shows two separate peaks (Figure 5.10). The first peak is that of hydrogen evolved from the metal-water reaction during the high-temperature excursion. The second coincides with the introduction of cooling water into the fuel bundle after the high-temperature excursion. Some of this additional water boiled and the resulting steam probably pushed hydrogen (formed during the high-temperature time period but not flushed out of the system) out the exit line and past the Beckman.

5.1.3 Pressure Measurements

Shown in Figure 5.11 are typical pressures recorded from the fuel rod pressure transducers, and shown in Figure 5.12 are the pressures of the shroud insulation cavity and the system pressure. From these traces it is very easy to identify the time of rupture of the fuel rods and of the shroud inner liner.

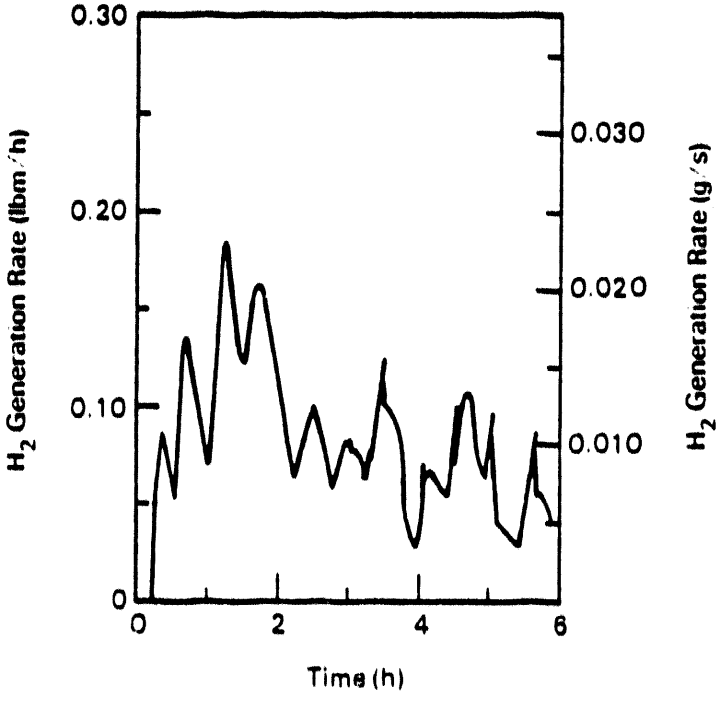

FIGURE 5.8. Predicted FLHT-1 Hydrogen Generation Rate with Zircaloy + Water Reaction But No Fuel Redistribution

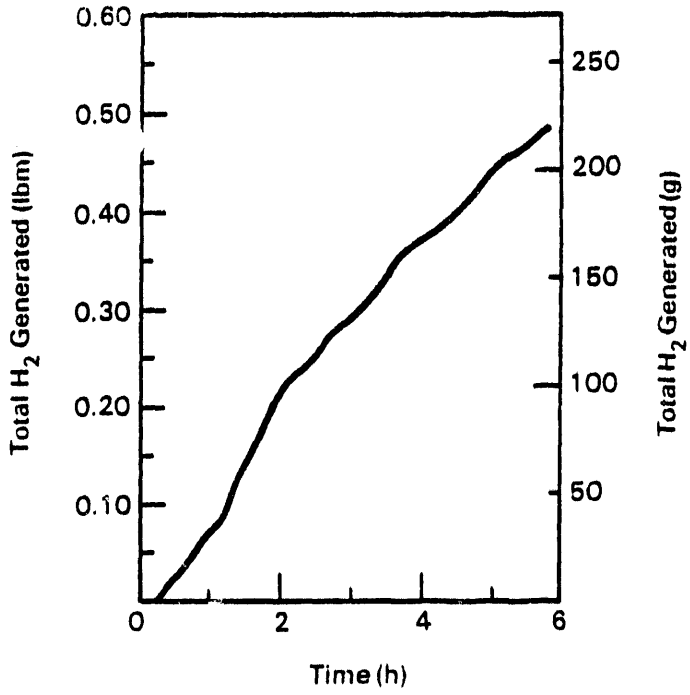

FIGURE 5.9. Predicted FLHT-1 Hydrogen Generation with Zircaloy + Water Reaction

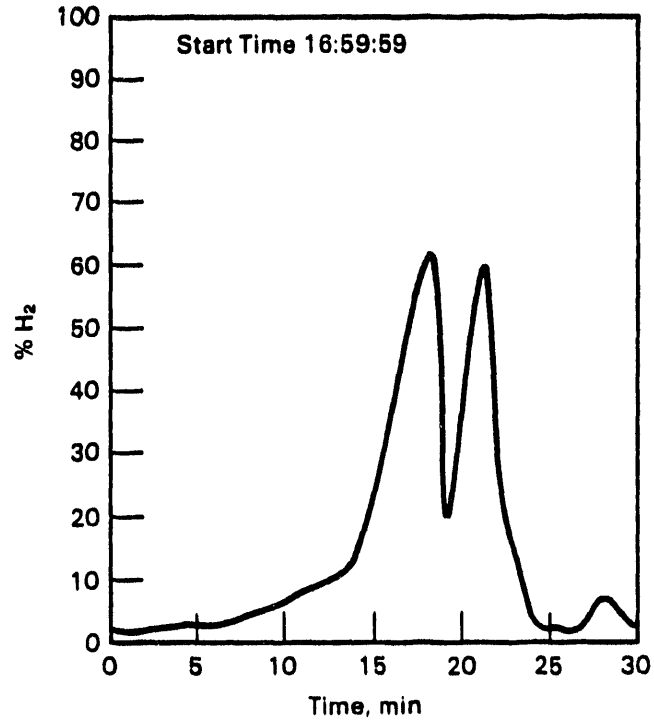

FIGURE 5.10. Hydrogen Generation

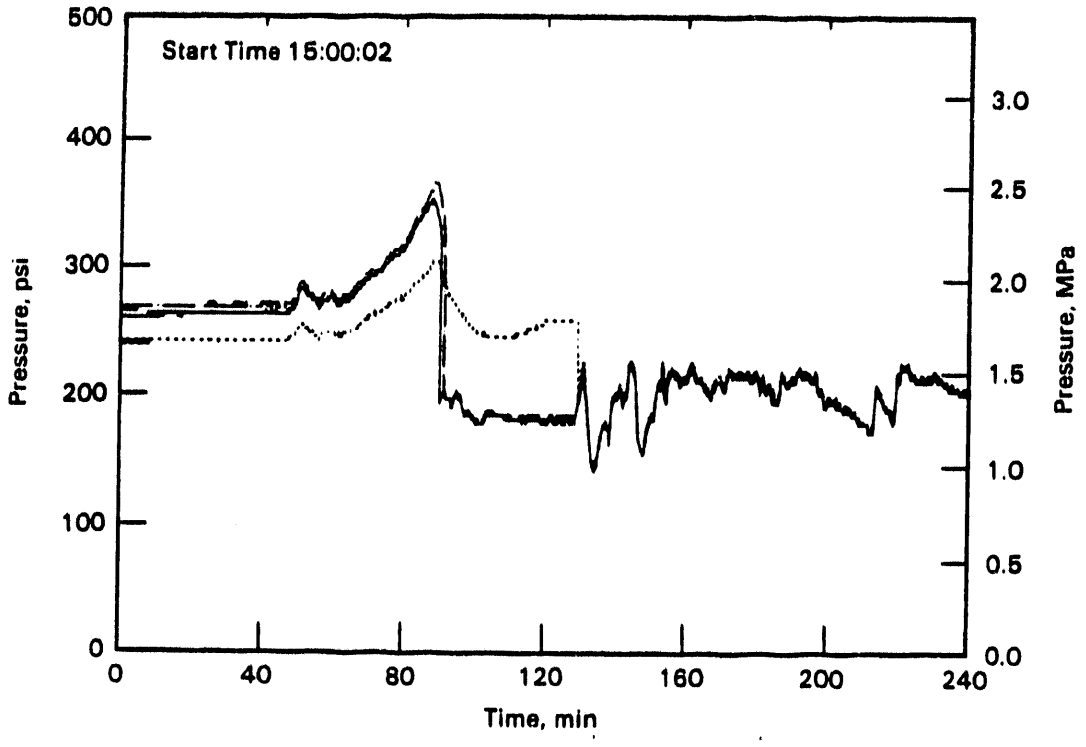

FIGURE 5.11. Typical Fuel Rod Pressures

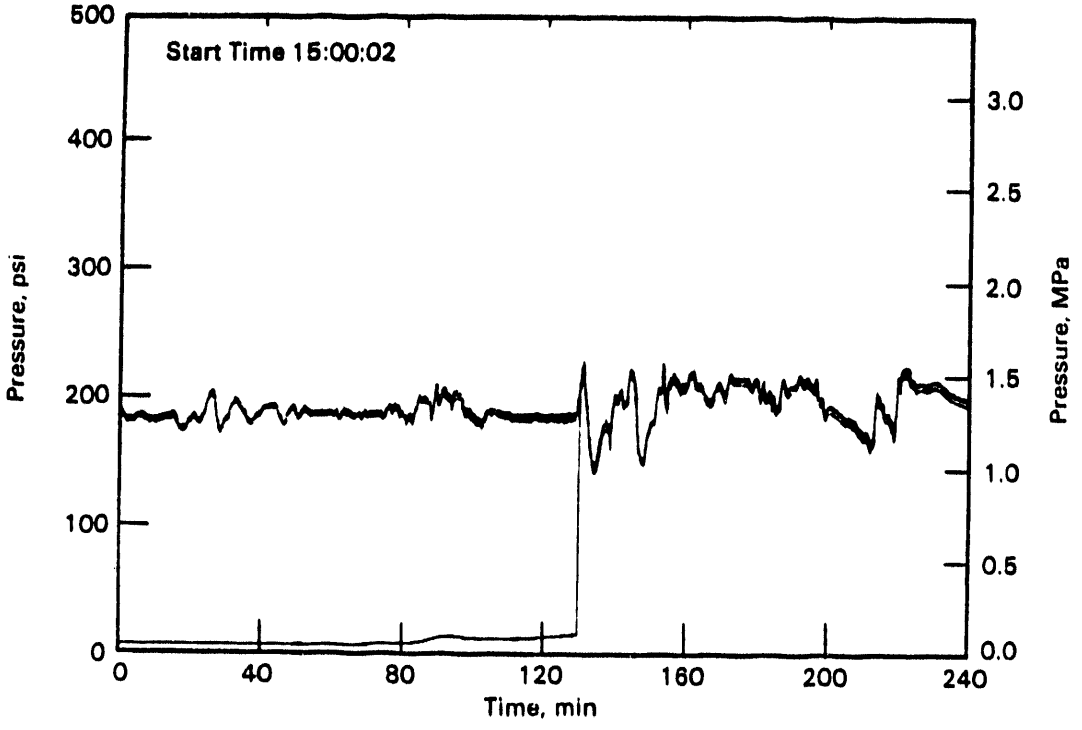

FIGURE 5.12. Shroud Insulation Cavity Pressure and System Pressure

The time of rupture of the shroud inner liner corresponds with an abrupt temperature increase measured by the saddle thermocouples (Figure 5.13). Figure 5.14 shows the coolant temperature response at that time. The conclusion drawn from this data is that when the liner ruptured, a large quantity of hot steam [at around 13 atm (190 psi) pressure] was injected into the shroud insulation cavity [which was at about 1 atm (15 psi) pressure]. This steam "jet" carried a thermal energy into the insulation cavity and caused the step change in the saddle temperatures. However, once the pressures equilibrated between the insulation cavity and the rest of the system, the temperature response of the saddle thermocouples soon returned to normal.

5.1.4 Bundle Coolant Level Measurements

Two TDRs were mounted in manometers connected to the fuel bundle cavity. They were attached in this fashion to protect them from the high temperatures that would be experienced in the fuel region of the test assembly. The readings from these instruments turned out to be of little value during the test, because they more or less continuously indicated that the test train was full of water. The experiences of the test operators during calibration and then during the early portions of the test indicate that the problem was probably caused by steam condensing in the TDR opening at the top of the test train. The condensed liquid then ran down inside the TDR tube and caused the erroneous readings.

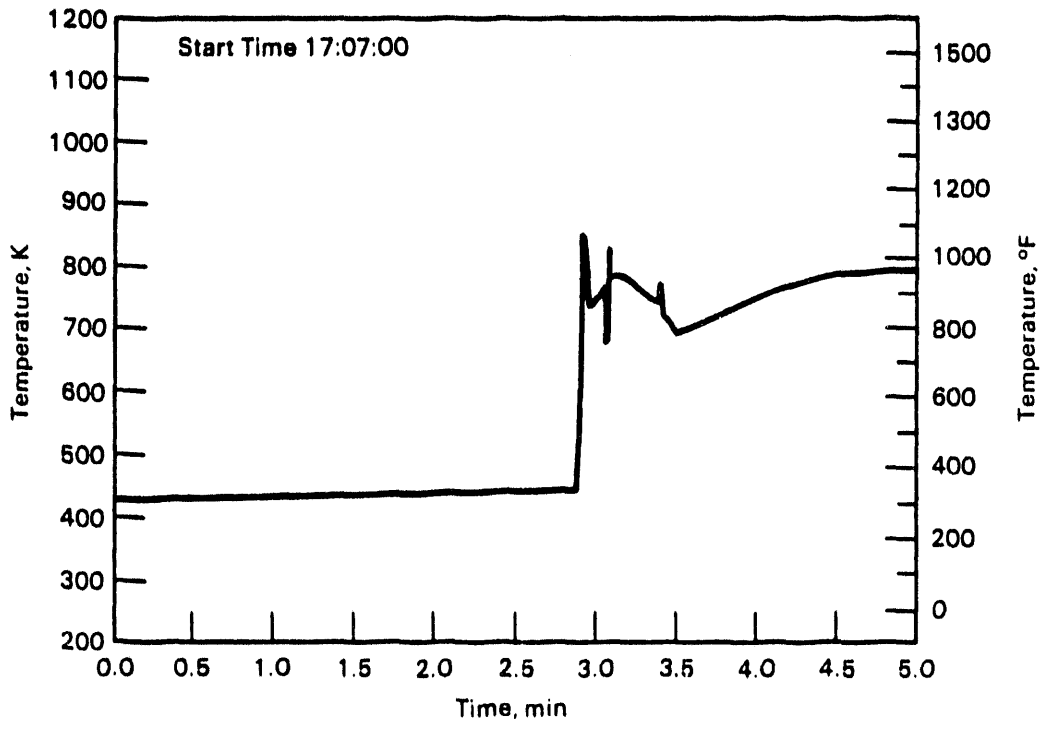

FIGURE 5.13. Top Instrumented Saddle Average Temperature

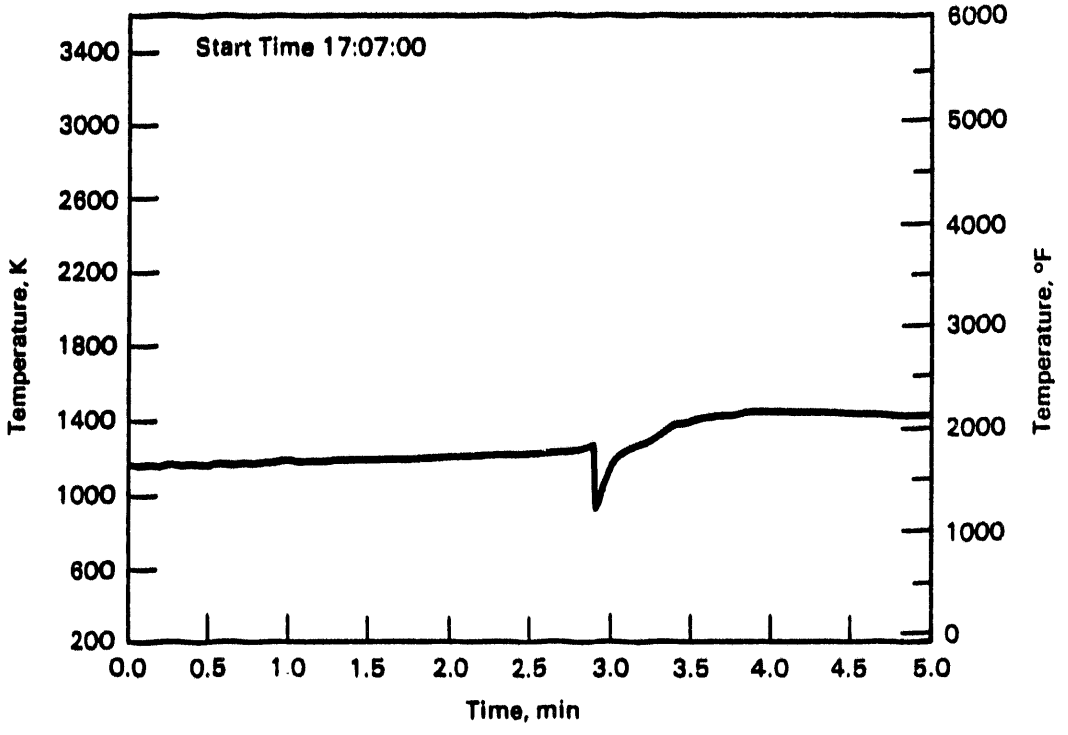

FIGURE 5.14. Coolant Response to Liner Rupture

5.2 POST-TEST EXAMINATION

After the test, the fuel assembly was transported to the spent fuel pool in the NRU reactor building and stored there. After several weeks, the fueled region of the test train was moved to the horizontal disassembly-examination-reassembly machine (DERM). (The computer-controlled DERM was developed by PNL during an earlier phase of NRC sponsored research at the NRU reactor). The DERM allows operators to mount various cutting tools, measuring tools, or cameras on a fixture, along or around a fuel rod or bundle, and move them with great precision.

The post-test examination of the FLHT-1 test train involved cutting into and removing sections of the shroud and then removing the insulation and liner layers of material to exposure the fuel bundle. Photographs--still and video, color and black-and-white--were taken as each new feature came into view. No samples were taken from the test train. After the examination was completed, the test train was returned to the vertical position and stored in the spent fuel pool.

5.2.1 Preparation

The post-test work began by cutting all the instrument leads at the shroud/plenum interface. Next, the plenum region was unbolted from the shroud. Close visual examination of the plenum, particularly the hottest

portion just above the top of the fuel bundle, revealed no damage. A thin, uniform layer of oxidation had formed on the Zircaloy surfaces exposed to steam during the test.

5.2.2 Description of Examination Procedure

Using an underwater saw and various other tools, layers of the shroud were visually examined, photographed and removed in succession until the fuel bundle was visible. Figures 5.15 through 5.18 show each photographed region, starting near the top of the shroud (above Level 153) and extending downwardly below Level 81. (The level value is the distance in inches from the bottom of the fuel column; for example, the top of the fuel is at Level 144.)

Each figure has as many as six views (starting at the top of each figure) showing surfaces of the 1) outer shroud tube (Figure 5.15 only); 2) inner shroud tube; 3) saddles; 4) insulation; 5) liner; 6) first view of fuel region; and 7) second view of fuel region. There is some overlap on the edge of each figure to permit the reader to construct (assemble) a mosaic of the entire length of the examination region (called a window). Each "window" view is a photographic mosaic of the exposed surface of the examination area. The brief view in Figure 43 of the exterior surface of outer tube is shown just for completeness and to help orient the reader. A description of the remaining views on each figure is shown below.

The window covered one-fourth of the shroud circumference and was 213-cm (7-ft) long. The window was cut from Level 153 down to Level 69. The hottest elevation during the test was at about Level 114 (290 cm) (9.5 ft). In the following, the results are presented in the same order as the examination proceeded, starting at the shroud exterior and moving into the fuel bundle region. The shroud layers examined were the outer tube, MMPD wires, inner tube, saddles, insulation, liner, and fuel bundle.

5.2.3 Outer Tube and MMPD Wires

The exterior surface of the shroud outer tube was visually examined under approximately 180 cm (6 ft) of water and found to be in the as-fabricated condition--no discoloration (oxidation) and no distortion (mechanical deformation). See the top view of Figure 5.15. The outer tube was removed and the inside surface was essentially as-fabricated. Likewise, the MMPD wires appeared to be in the as-fabricated condition.

5.2.4 Inner Tube and Saddles

After the MMPD wires were cut and removed, the outer surface of the inner Zircaloy tube was exposed (see the top views of Figures 5.16 through 5.18). This surface has one discolored area that begins at Level 93. The discoloration intensity increases to a maximum between Levels 115 and 121 and then fades away at Level 131. The discolored area is identified in the top view of Figure 5.17, but the actual discoloration is difficult to see. The Zircaloy surface reacted with impurity gases that remained in the MMPD cavity. The cavity was evacuated and backfilled with helium to 1-atm (15-psi) pressure before the test. No temperature sensors were located exactly at the peak

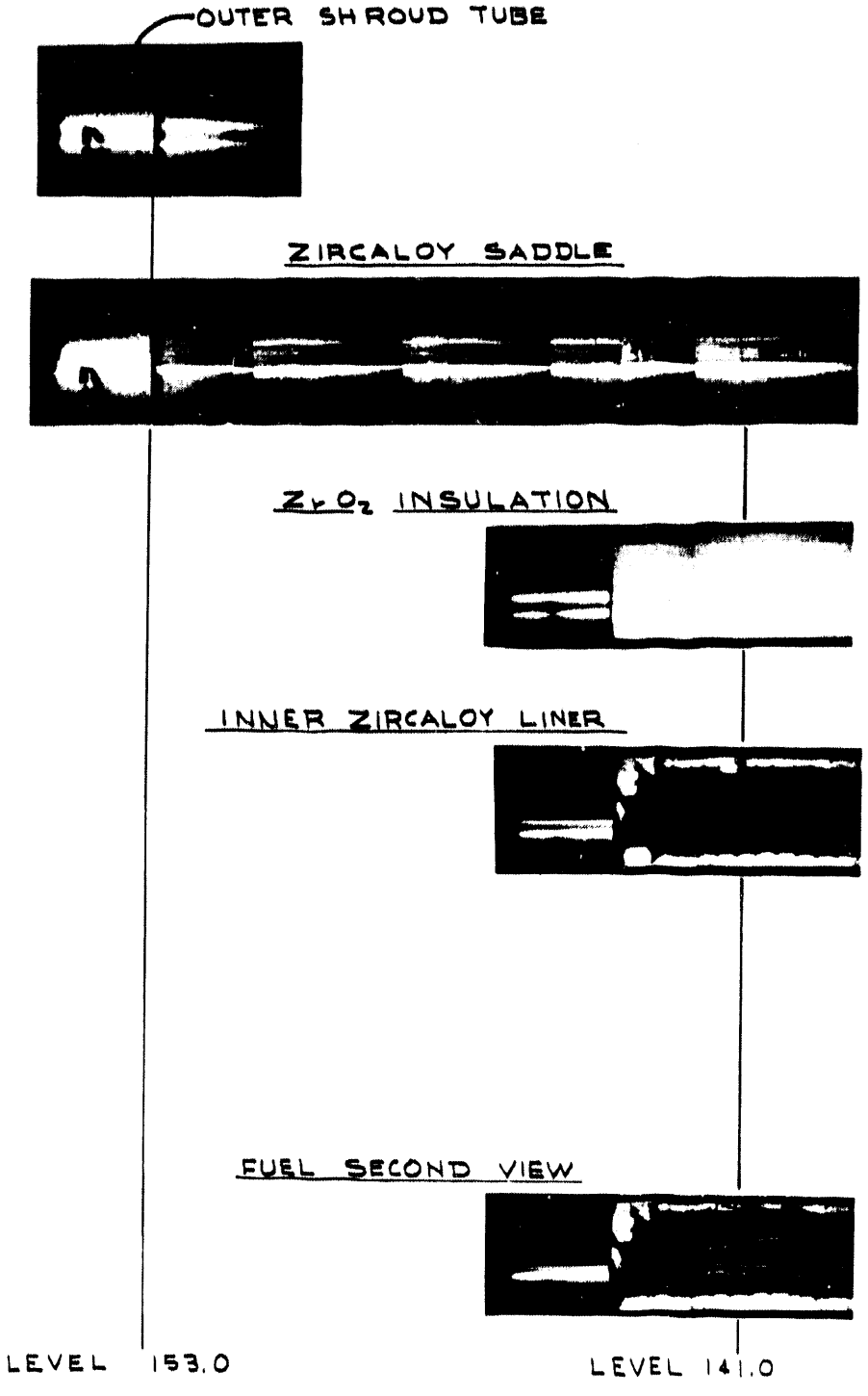

FIGURE 5.15. FLHT-1 Post-Test Examination, Level 141 to Level 153

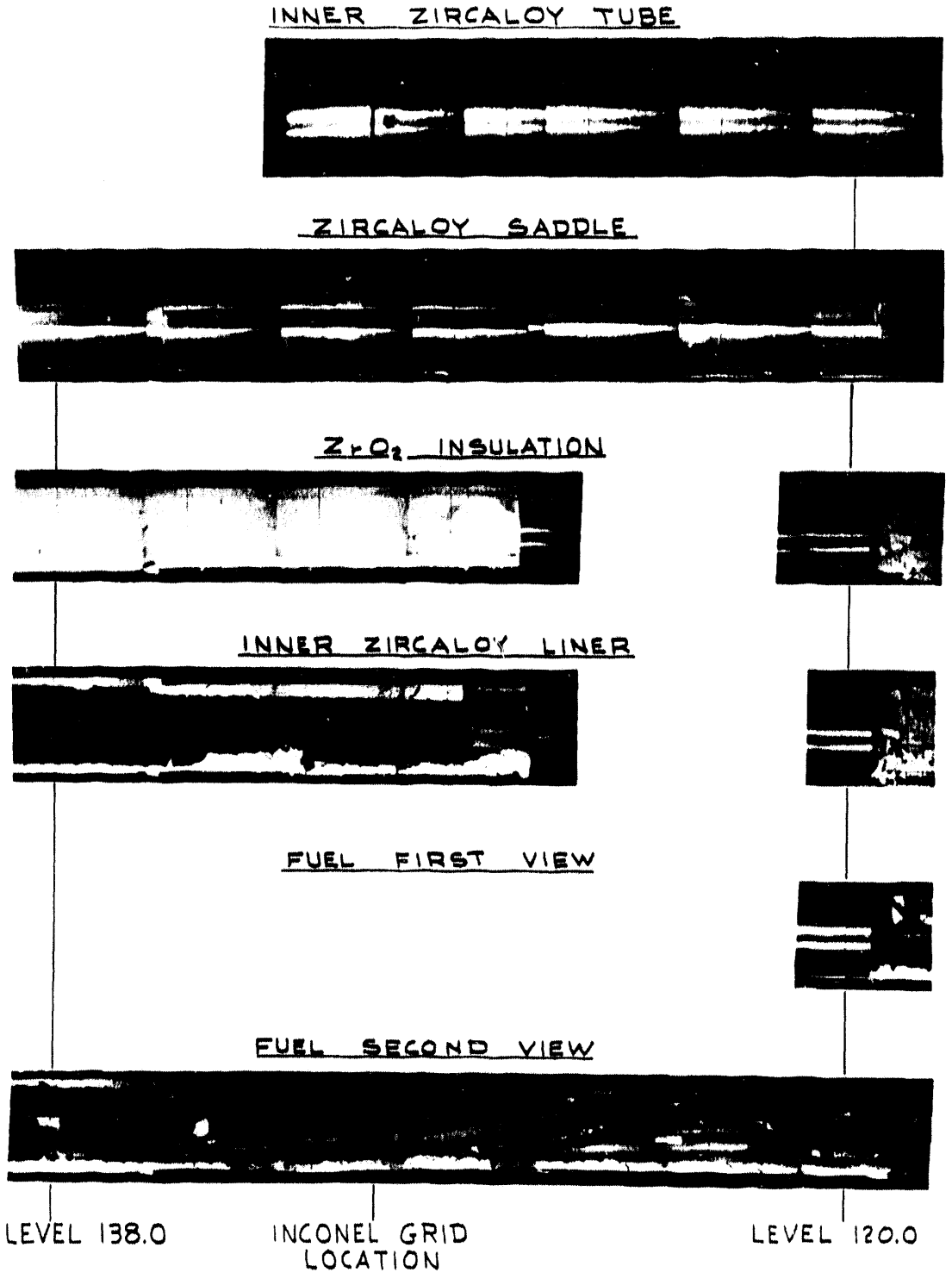

FIGURE 5.16. FLHT-1 Post-Test Examination, Level 120 to Level 138

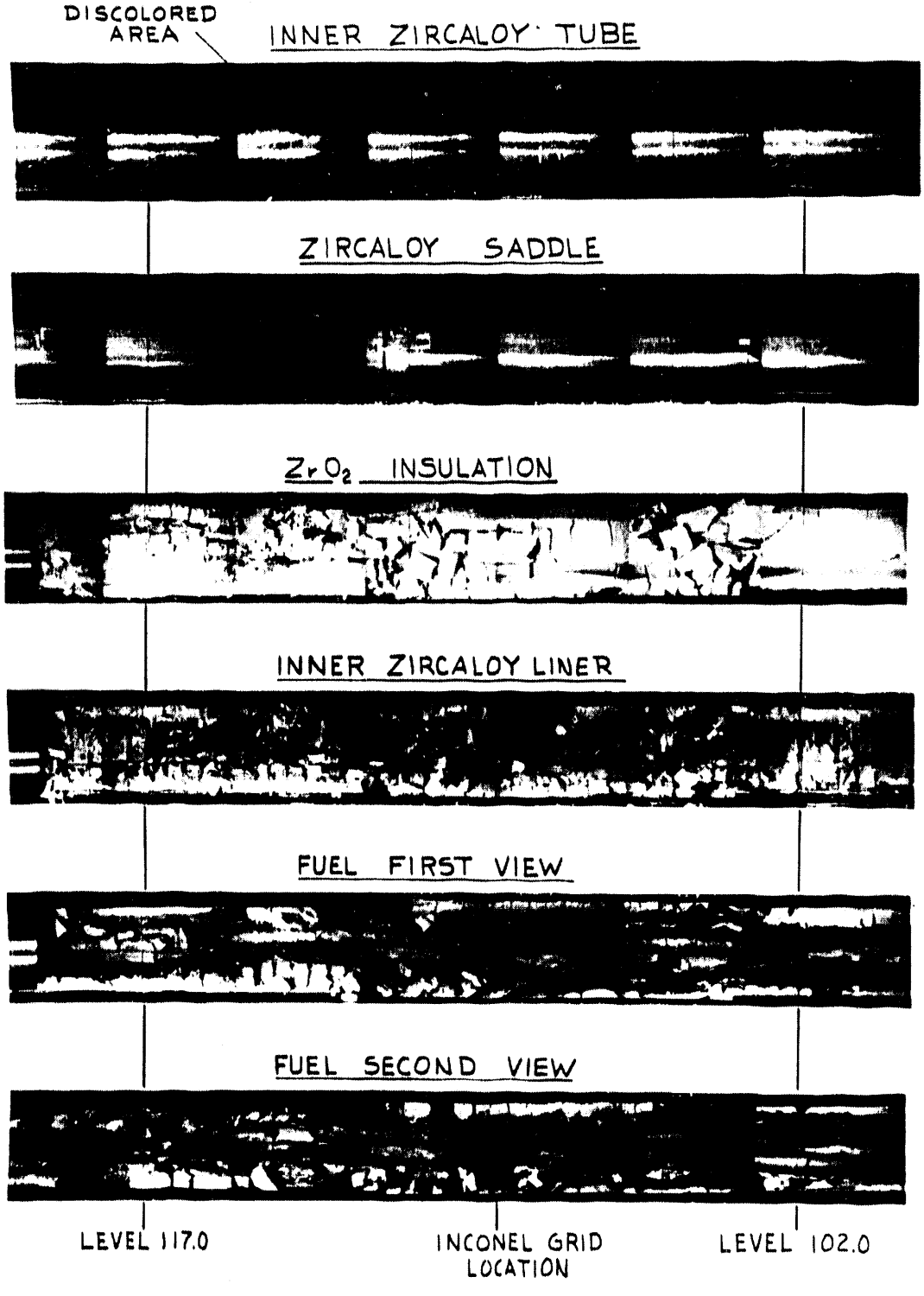

FIGURE 5.17. FLHT-1 Post-Test Examination, Level 102 to Level 117

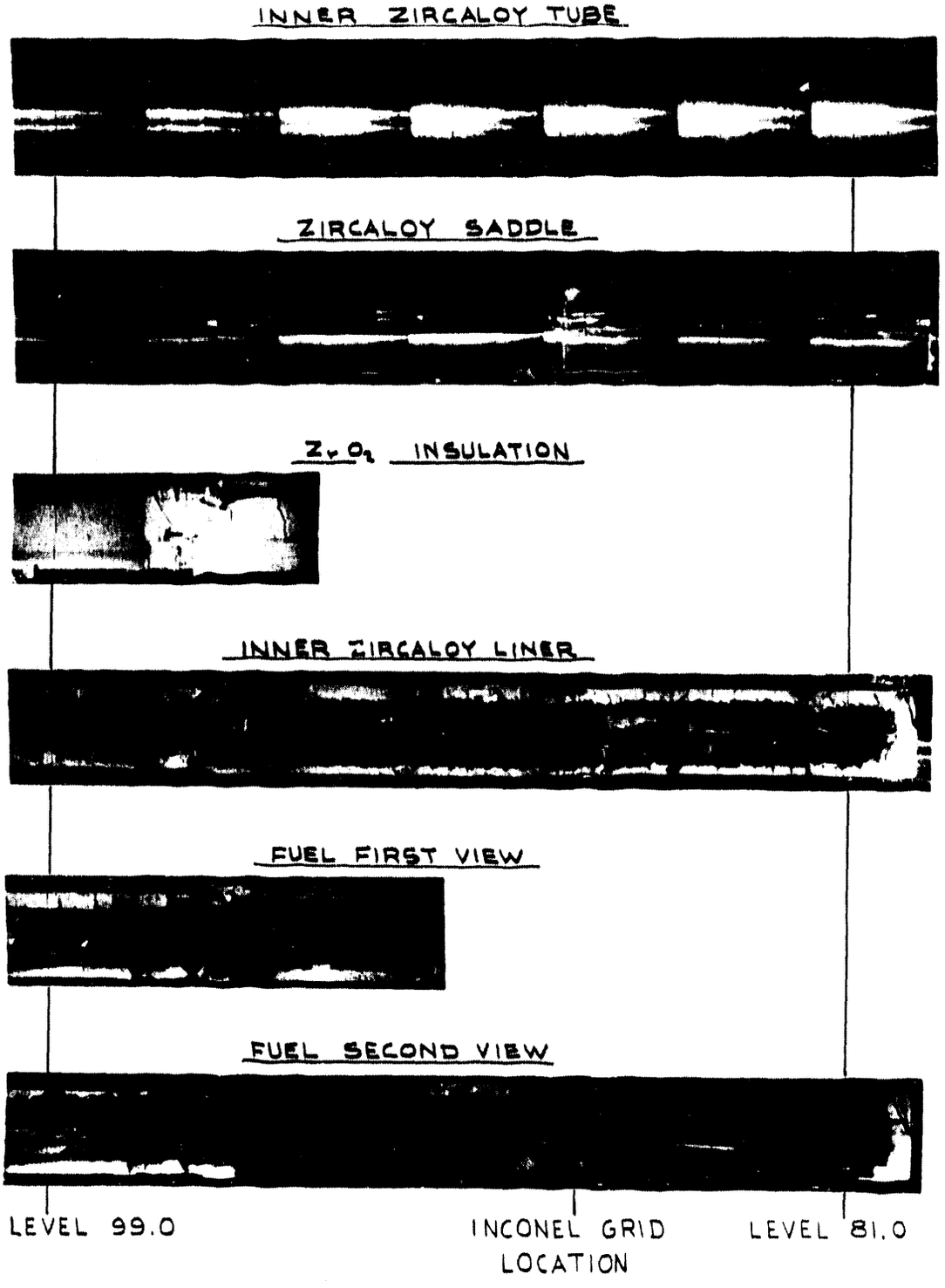

FIGURE 5.18. FLHT-1 Post-Test Examination, Level 81 to Level 99

discolored region, but an estimated peak temperature during the test based on saddle thermocouples at Level 110.8 was about 825 K (1000°F) for as long as 1 min.

Next a window was cut in the inner Zircaloy tube to expose the Zircaloy saddles. The inside surface of the inner tube was discolored like the outside surface. The discolored region spanned Levels 125 to 100, with the peak intensity at about Level 115. The color intensity faded away to a point at the extremities (Level 125 and 100), indicating that the area was a localized hot spot. A matching discolored region appeared on the outer surfaces of the saddles. This region on the saddles is shown on Figure 5.16, but again the discoloration is difficult to see.

The discoloration is probably a film of zirconium dioxide that formed during the few minutes after the liner ruptured and high-temperature steam rushed into the region. The peak steam and liner temperatures were ~2000°C (3625°F) near Level 115. Steam flowed into the insulation/saddle cavity until the pressure in the cavity became equal with the bundle coolant region pressure. Just before the liner ruptured, the insulation cavity pressure was about 1 atm (15 psi) and the bundle region pressure was about 13 atm. Other than the discolored surfaces, the inner tube and saddles exhibited no change from the as-fabricated condition.

Then the saddles were cut by a saw and lifted out of place. However, three saddles were later found attached. Because some saddles had been removed, exposing insulation, these attached saddles were temporarily left in place so the examination could continue without undue movement of the insulation. The most obvious "attached" saddle is seen from Level 127.6 to Level 119.4 in the insulation photographs. The other two partly severed saddles are at either extremity of the window.

5.2.5 Insulation

As the saddles were removed, the zirconium-dioxide tile insulation was visible. The tiles at each end of the window above Level 127 and below Level 95 were essentially intact, with some minor cracking. The enhanced cracking at Levels 105 and 95 was caused by the forced removal of the overlaying saddles.

The fragmented insulation between Levels 120 and 108 is probably the result of the rupturing liner and the associated localized stress imposed on the tiles. There was no evidence that the fragmented insulation moved. The thermal resistance of the fragmented insulation was similar to the intact tiles as shown by the similar cooldown behavior of thermocouples at Level 110.8 (located in a fragmented region) and Level 102.8 (located in an intact region).

5.2.6 Liner

After the insulation was inspected and photographed, it was removed piece by piece, exposing the inner liner. Over most of the exposed length, the liner had ridges that resulted from straining into the fabricated grooves on

the mating zirconium dioxide tile surfaces. This design was used to accommodate liner axial differential expansion. The ridges occurred over the entire exposed region of the liner above Level 86. The short length of exposed liner below the Level 86 shroud showed no ridging. The ridging occurred at high temperature because of the approximate 12-atm (175 psi) pressure gradient across the liner wall. The liner was broken into pieces from Level 117 to Level 102, probably due to the rupturing.

The liner surface was "washed" to remove the zirconium dioxide dust before the photographs of the liner were taken (Figures 5.15 and 5.16). This dust is shown on the liner in Figure 5.18.

After examination of the exposed liner, it was removed piece by piece, but where it was still intact above Level 138 and below Level 90, pieces were broken by repeated bending.

5.2.7 Fuel Bundle

Two examinations were performed on the fuel bundles. The first occurred with the saddle in place between Levels 127 and 119. The second occurred after the saddle was cut and removed. Photographs taken during these two examinations are shown in the bottom two pictures of Figures 5.15 through 5.18.

The fuel bundle is described as follows: from Level 143.5 to about Level 117, the fuel rods are highly oxidized, embrittled, and expanded, and broken into 8- to 15-cm-long (3- to 6-in.) rodlets. Fuel pellets are not bound to the cladding; some fell out during the examination.

The fuel pellets appear in the as-fabricated condition. Few pellets are cracked because of the very low power rating during the test [0.06 kW/m (~0.2 kW/ft) of rod].

The cladding near Level 117 is more highly oxidized as evidenced by a lighter color. Some cladding melted in very localized areas such as at Levels 135 and 104.

The outer portion of the Inconel grid at Level 128 was stuck to, and removed with, the liner. The grid was assembled at Level 128; however, as seen in Figure 5.10, it is now located about 8 cm (3 in.) higher. The increase in level is due to the growth of the Zircaloy cladding during oxidation caused by the volume increase of the oxidized metal.

A broken piece of Zircaloy instrument carrier [a long, bent (90° angle), narrow piece of Zircaloy sheet] is shown at the bottom of Figure 5.17 just below the indicated grid location. A carrier, located in each of the four corners of the bundle, provided a pathway for instrument leads and support for the grid spacers.

The most changed bundle region was from Level 117 to Level 102, including an Inconel grid at Level 108. The major changes include complete chemical reaction of the Inconel grid, tight bonding of the rods to each other, and a

large area of once-molten Zircaloy. The grid reacted with the cladding and/or melted and flowed down into the bundle. On inspection, the former location of the grid appeared greenish, probably due to the formation of oxides from the Inconel elements.

The tight bonding of the rods by once-molten material suggests that this was the hottest region during the test. Data collected during the test support this observation. The large area of once-molten Zircaloy may be partially due to chemical reactions with Inconel, forming lower melting alloys.

Below Level 102 the fuel bundle is similar to the top region, containing oxidized, embrittled and fractured rodlets. Below Level 90 the fuel rods appear to be intact.

6.0 CONCLUSIONS

The FLHT-1 test was planned as a slow boiiaway transient to limit the rate of rapid Zircaloy oxidation and limit the resulting peak temperatures to 2275 K (3635°F). The test was intended to demonstrate that such high temperatures could be achieved safely and to obtain severe fuel damage data on full-length assemblies. The slow boiiaway was not achieved because of excessive plenum heat loss and the resulting steam condensation and "rain-back" on the fuel bundle. However, rapid Zircaloy oxidation did occur and peak bundle temperatures over 2200 K (3500°F) were safely reached, providing data on severe fuel damage. The test train discharge operation went as planned, and the results of a post-test visual examination showed that the shroud performed as expected and thus demonstrated that similar tests could be safely performed in the future. Thus the primary objectives of the test were achieved.

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