PNL-5547
UC-520

Full-Length High-Temperature Severe Fuel Damage Test #2 Final Safety Analysis

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September 1985 – Completion Date
September 1993 – Publication Date

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

Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
by Battelle Memorial Institute

Battelle

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PACIFIC NORTHWEST LABORATORY
operated by
BATTELLE MEMORIAL INSTITUTE
for the
UNITED STATES DEPARTMENT OF ENERGY
under Contract DE-AC06-76RLO 1830

Printed in the United States of America

Available to DOE and DOE contractors from the Office of Scientific and Technical Information, P.O. Box 62, Oak Ridge, TN 37831; prices available from (615) 576-8401. FTS 626-8401.

Available to the public from the National Technical Information Service, U.S. Department of Commerce, 5285 Port Royal Rd., Springfield, VA 22161.

NOTE

The analysis in this report was completed several months prior to the FLHT-2 test which was conducted in December, 1985. A report was not made available to the public then because the FLHT-2 test was part of the NRC Cooperative Severe Accident Research program and program participants received timely reports in draft form. This report is now being published in order to assist any future studies of fuel behavior during severe accidents as well as to formally document the extensive analysis effort that was made to prepare for this severe fuel damage test.

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ACKNOWLEDGMENTS

The authors and project staff gratefully acknowledge the staff of Chalk River Nuclear Laboratories (CRNL) for their assistance. A special acknowledgment is due to I. C. Martin for his dedication and interest and for his efforts in preparing the safety technical notes for the FLHT-2 tests for the Canadian Nuclear Safety Advisory Committee. The support and direction of R. Van Houten, the NRC program manager, are also appreciated.

ABSTRACT

Hazardous conditions associated with performing the FLHT-2 experiment have been analyzed. Major hazards that could cause harm or damage are 1) radioactive fission products, 2) radiation fields, 3) reactivity changes, 4) hydrogen generation, 5) materials at high temperature, 6) steam explosion, and 7) steam pressure pulse. As a result of this analysis, it is concluded that with proper precautions the FLHT-2 test can be safely conducted.

SUMMARY

The second full-length high-temperature experiment (FLHT-2) is scheduled to be performed in the National Research Universal (NRU) Reactor at Chalk River Nuclear Laboratories (CRNL), Ontario, Canada. The hazards associated with the test have been analyzed, and test conditions have been identified so that the hazards can be properly and adequately handled. The major hazards that could cause harm or damage are radioactive fission products, radiation fields, reactivity changes, hydrogen generation, materials at high temperature, steam explosion, and steam pressure pulses. Each of these hazards is summarized below:

* Radioactive fission products will be contained within the low-pressure primary system. If a leak or break develops during the critical few minutes of the test, the secondary confinement will contain any radioactive materials.

* No high radiation fields are expected near reactor personnel during this test. However, as a precaution, effluent lines are shielded with 10 cm of lead.

* Reactivity calculations for fuel relocation and light-water voiding indicate that no uncontrollable reactor conditions can be produced.

* Hydrogen will be generated in the fuel bundle during the test. Downstream from the effluent control module (ECM), hydrogen will be diluted to less than 4% by adding nitrogen gas.

* High-temperature Zircaloy, zirconia, urania, and mixtures containing them will reach temperatures near 2200°C during the test. Materials at this high temperature must be isolated from the reactor piping. This test, like its predecessor FLHT-1, will utilize a chill block, double-cold-walled shroud design to cool any high-temperature materials before they can contact the nearby reactor piping. The main requirement for

safe conditions during the test is maintaining an adequate supply of water for cooling the shroud external surface.

* The possibility of a steam explosion from molten fuel/coolant interaction was analyzed. Only one of five necessary conditions for steam explosion is satisfied; therefore, a steam explosion is not possible.

* High-pressure steam pulses due to either molten fuel/coolant interaction or Zircaloy/water chemical interaction were also analyzed. Calculated maximum pressure pulses can be contained by the test primary containment system. Again, secondary containment will provide additional safety.

As a result of these analyses, it is concluded that with proper precautions the FLHT-2 test can be safely conducted. This report also provides a description of the experiment objectives, hardware, conditions, and the expected test results.

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INTRODUCTION

The Coolant Boilaway and Damage Progression (CBDP) Program is conducted by Pacific Northwest Laboratory (PNL)[a] as part of the U.S. Nuclear Regulatory Commission (NRC) severe fuel damage/source term (SFD/ST) program.[b] The CBDP Program consists of in-reactor experiments using full-length light-water reactor (LWR) fuel rods to determine fuel bundle behavior and fission product release during severe accidents similar to the one that occurred at Three Mile Island Unit-2 (TMI-2). The CBDP experiments are being performed to evaluate fuel behavior during a simulated small-break loss-of-coolant accident (LOCA) that results in a partially uncovered reactor core. As the coolant boils away and the fuel rods become uncovered, the temperature of the rods increases above design limits. As the temperature increases, the rods become damaged and potentially dangerous radioactive fission products are released from the fuel.

The CBDP Program consists of six tests designed to investigate fuel bundle damage behavior from 930 to 2600°C in a series of progressively more severe tests at prototypic power densities, thermal gradients, and steam mass fluxes. Fission heating is used to simulate decay heat generation to boil the coolant. Three tests have been completed (Table 1). Two of these tests studied fuel bundle behavior during coolant boilaway conditions that resulted in peak temperatures as high as 2000°C. The experiments use full-length LWR fuel rod bundle test assemblies and are being performed in the National Research Universal (NRU) Reactor at Chalk River Nuclear Laboratories (CRNL) at Chalk River, Ontario. Highlights of the test conditions are given in Table 1.

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

(b) Partners in this program with NRC 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, and Sweden.

TABLE 1. CBDP Program Test Matrix Highlights

Test	Peak Test Temperature, °C	Preirradiated Fuel Rods	Hydrogen Measurement	Test Date
MT-6A	930	No	No	Completed (5/84)
MT-6B	1280	No	No	Completed (6/84)
FLHT-1	2000	No	Yes	Completed (3/85)
FLHT-2	2200	No	Yes[a]	11/85
FLHT-4	>2600	Yes	Yes[a]	6/86
FLHT-5	>2600	Yes	Yes[a]	12/86

(a) Using enhanced measurement instruments.

Well-characterized data for evaluating the effects of coolant boilaway and core damage progression in an LWR are being developed in the program. Coolant boilaway is achieved using low level fission heat to simulate system enthalpy and decay heat that supply the energy that causes a postulated coolant boilaway accident. These data provide a basis for developing accident mitigation strategies, for evaluating postulated coolant boilaway accidents, for developing concepts for accident prevention and quantifying safety margins, and for developing, benchmarking, and validating computer codes such as SCDAP and MELPROG.

The following data will be obtained from the CBDP tests and will be used to confirm the validity of results obtained from separate effects tests that are being sponsored by the NRC at PNL and other laboratories:

* axial temperature distribution for full-length fuel bundles as a function of liquid level

* fuel bundle damage progression (core degradation) behavior

* cladding melt progression (dissolution and resolidification of UO_2)

* core debris and grid spacer interaction

* coolant boilaway behavior

* debris bed formation and coolability

* flow channel blockage behavior

* hydrogen evolution

* fission product release and transport

* inner and outer diameter cladding oxidation and embrittlement

* test train design verification for possible subsequent tests.

The CBDP experiments utilize the following advantages of the NRU Reactor: 1) the capability to test highly instrumented, multirod 12-ft-long fuel bundles under thermal-hydraulic conditions representative of contemporary LWRs; 2) the ability to achieve power densities and axial power distributions typical of TMI-2 accident conditions using preirradiated fuel rods with commercial enrichment; and 3) the ability to provide prototypic coolant mass fluxes at the fluid/vapor interface typical of a TMI boildown condition. These unique capabilities will reduce uncertainties associated with length and power distribution scaling factors and the interpretation of the experimental results from small-scale separate effects tests.

The CBDP tests are the only known full-length in-reactor pressurized water reactor (PWR) and boiling water reactor (BWR) multirod boilaway tests being performed. The deformation, rupture, fission product release, and debris bed data can be used to evaluate LWR accident codes like SCDAP and MELPROG and to help quantify the safety limits used in the nuclear industry.

In order to obtain approval to conduct the FLHT-2 experiment, PNL must perform preliminary and final safety analyses of the proposed test and submit the results to the safety engineers at CRNL. CRNL safety engineers review the PNL results and then prepare safety technical notes that are submitted to the Canadian Nuclear Safety Advisory Committee (NSAC) for review and approval.

CRNL test engineers review and approve not only the safety analysis but also the detailed experiment operating plan and expected test conditions. This report then documents the safety analysis performed for the FLHT-2 experiment. Since most of the PNL analyses and the actual reactor hardware utilize the English pound-Fahrenheit-hour units and NSAC requires the metric gram-Centigrade-second units, both types of units are used in this report.

Before presenting the safety analyses, a description of the FLHT-2 experiment objectives, hardware, conditions, and expected results is provided.

EXPERIMENT OBJECTIVES

The FLHT-2 experiment will provide full-length fuel bundle damage behavior data for LWRs by simulating a natural boildown accident. Experiment objectives include:

* Provide data that are prototypic of LWRs on full-length fuel during coolant boilaway and core damage progression near typical decay heat levels.

* Correlate fuel temperatures up to 2200°C 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 failure effects, and damage progression phenomena with short core data from the Power Burst Facility (PBF), Annular Core Research Reactor (ACRR) 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.

* Test the gamma spectrometry systems that will be used on subsequent fission product source term tests.

EXPERIMENT HARDWARE

The experiment hardware consists of the test train assembly, the effluent control module (ECM), the NRU reactor coolant system, instrumentation, and the data acquisition and control system (DACS). The hardware arrangement is depicted in Figure 1. The figure illustrates the test train hanging inside one of the reactor pressure tubes. The ECM is located on the top of the reactor near the test train. Individual (separate and independent) coolant supply systems are connected to the test train and the reactor pressure tube. A superheater located on top of the reactor near the test train preheats the upper portions of the test train. This superheater will help prevent premature steam condensation. The superheater is supplied with steam from a reactor coolant system and delivers superheated steam to the test train. The DACS is located in a room about 30 m from the test train. Electrical cables connect the test train assembly and the ECM to the DACS. The experiment components are highly instrumented, especially the test train assembly. Instruments measure mainly local pressure, temperature, and flow.

TEST TRAIN ASSEMBLY

The approximately 8-m-long test train assembly that hangs inside the reactor pressure tube consists of four sections that occupy different regions of the reactor: the closure, plenum, reactor core, and inlet regions (Figure 2).

Closure Region

The closure hardware is located at the top of the test assembly and contains the components that support the rest of the assembly and seal it to the reactor pressure tube. The closure hardware consists of the closure plug, seal ring, two gaskets, two bolting rings, two Belleville washers, and three feed-through plugs. The closure plug is sealed to the reactor pressure tube using two metallic gaskets, the seal ring, the Belleville washers, and the two bolting rings arranged as shown in Figure 3.

The smaller diameter gasket seals the plug to the seal ring; the larger gasket seals the seal ring to the reactor pressure tube. The lower bolting ring is threaded in the reactor pressure tube, and the bolts threaded through the ring compress the larger diameter gasket. The upper bolting ring maintains a live load on the smaller diameter gasket by deforming the Belleville spring washers. The live load maintains sealing pressure and compensates for temperature differences between the closure plug and the pressure tube.

Pressure boundary penetrations are made through the closure plug for instrument lines, pressurization tubes, two coolant supply lines, a flush line, and the bundle effluent line. The superheated steam supply line penetrates the closure inside the bundle effluent line. The closure region is illustrated in Figure 4.

The effluent line penetration through the FLHT-2 closure plug is thermally isolated from the plug to help prevent premature steam condensation (such as occurred during the FLHT-1 test). The steam flows through the inside tube of two concentric tubes. The region between the tubes is evacuated, thus forming a "thermos" bottle. A metal bellows is welded to the inner tube below the closure plug to accommodate axial differential thermal expansion between the two tubes. The outer tube is seal-welded to the closure plug.

The test train instrument lines and pressurization tubes penetrate the closure plug through three feed-through plugs. As many as 55 leads can penetrate one feed-through plug. The pressure boundary for the feed-through plug is provided by grayfoil packing gland seals. The pressure boundary for the flush line and the two bundle coolant lines is provided by standard autoclave fittings. If necessary, the flush line is used after the test to reduce radiation levels near the closure region and thus provide easier personnel access during the test train assembly discharge operation.

All pressure boundary seals are tested after final test train assembly. The seal of the closure plug to the reactor pressure tube is tested after the test train is loaded into the reactor but before other experiment hardware is put into place.

Plenum Region

The approximately 4-m-long plenum section connects the closure section to the core section (Figure 5). In addition to providing the appropriate mechanical features to support and position the lower sections of the test train, the plenum also provides the effluent flow path and supports the desuperheater. The plenum contains two sections to permit hardlines and tubes to penetrate the region. These penetrations are located in the flange between the two sections. The upper plenum is an evacuated double-walled insulated assembly; the lower plenum is a ceramic-filled insulated assembly. The lower flange of the upper plenum provides penetrations into the steam region for the flush line, a time domain reflectrometer (TDR) line, and the desuperheater.

The desuperheater is a vertical small-diameter tube with small horizontal holes near the top of the ceramic-insulated plenum. If the effluent temperature near the closure plug exceeds a safe value during the test, the desuperheater is automatically activated and high-pressure low-temperature water is sprayed into the plenum until the temperature is reduced to an acceptable value.

The upper plenum is bolted to the bottom of the closure plug using tube extensions to the two bundle coolant lines and the flush line; the extension tubes are welded to the bottom of the closure plug. A seal weld connects the steam line in the plug to the upper plenum.

The lower plenum is a coaxial tubular structure with low-density fiberboard zirconia insulation in the annular space between two Zircaloy tubes. The insulation region is hermetically sealed. Just before the test, the region is evacuated and backfilled with argon at 1-atm pressure. During the test, the pressure is monitored using a remote pressure transducer that is connected to the region with a capillary line. A breach in either Zircaloy tube during the test is indicated by an abrupt pressure increase. Although such a breach would not create an unsafe condition, it could alter some of the test results (for example, hydrogen measurement, if hydrogen should flow into the insulation after the breach).

A set of Belleville springs near the bottom of the plenum compensates for thermally driven changes in the lengths of the inner and outer Zircaloy tubes.

Effluent enters through an inlet tube at the lower end of the plenum. The tube is surrounded by low-density ZrO_2 fiberboard to minimize radial heat loss. The fiberboard is contained within a high-density (approximately 100% TD) zirconia holder that supports the fiberboard if the inner Zircaloy tube loses strength. The small diameter of the inlet tube reduces radiant heat transfer to the plenum interior wall. An annular space between the plenum interior wall and the inlet tube serves as a condensation trap during the steam heatup phase of the test.

The bottom end of the lower plenum contains the superheater outlet and the plenum drain inlet. The bottom flange connects to the top flange of the core region of the test train assembly.

Reactor Core Region

The reactor core region that hangs below the plenum contains two major components: a thermally insulated full-length 12-rod LWR fuel bundle and a double-walled Zircaloy capsule that surrounds the fuel bundle (see Figure 6). For ease in fabrication, assembly, and post-test examination, the bundle insulation is assembled with the double-walled capsule: this assembly is called a shroud. Because the test train core region hardware is fabricated as bundle and shroud, they will be described as such.

The fuel bundle contains 12 full-length instrumented fuel rods fixed into a square array with a 1.3 cm pitch by eight Inconel grid spacers that are evenly spaced along the length of the rods. Each fuel rod contains a 3.63-m column of unirradiated UO_2 pellets enriched to 2.0%. The pellets are slotted to make room for 1-mm-diameter thermocouples (TCs) that extend along the inside of the cladding. Each fuel column is clad with a Zircaloy tube with end caps welded at both ends. TCs are resistance-welded to the cladding interior surface at various elevations. The TC leads exit through the bottom of the rods. An Inconel spring is located at the top of the fuel column to provide a compressive force on the column and thus prevent formation of axial gaps during handling and shipping. The lower end of each rod has a special duplex Zircaloy/stainless steel end cap that permits the end cap to be welded to Zircaloy tube and TC leads to be brazed to the stainless steel section of the end cap. The bundle is supported from the bottom by a flange at the bottom of the test assembly; the rods are free to expand at the top end of the bundle.

The shroud consists of two concentric Zircaloy tubes, Zircaloy saddles, zirconia thermal insulation, and a Zircaloy liner. The two tubes form a double-walled capsule to isolate and protect the reactor pressure tube from the high-temperature fuel bundle components. Each tube has a minimum yield strength (0.2% offset) of 400 MPa (58,000 psi), which equates to an internal yield pressure of 11.7 MPa (1700 psi) for the inner tube and 11.0 MPa (1600 psi) for the outer tube.

Bypass coolant flows up the annulus between the outer tube and the reactor pressure tube. As long as bypass water is present to keep the two concentric tubes cool, the tubes will contain the hot test bundle components and protect the reactor pressure tube. The two tubes are maintained concentric by eight 1-mm-diameter wires wrapped side-by-side around the outside of the inner tube. Each of the eight wires is about 180 m (560 ft) long.

Four of the eight wires act as continuous TCs to indicate changing temperatures along the inner tube. These four sensors are called molten metal penetration detectors (MMPDs) because their function is to indicate the presence of hot (molten) material near the inner tube. Such hot material would have penetrated the fiberboard insulation, possibly impacting and freezing on the saddles.

Zircaloy saddles located inside the inner tube provide a smooth transition from the circular inner concentric tube to the octagonal-shaped blanket of insulation that surrounds the fuel bundle. The thermal insulation is a low-density (approximately 30% TD) rigid high-strength zirconia fiberboard in the shape of interlocking tiles. The low density helps provide high thermal resistance. The rigid high-strength fiberboard provides easy handling and machining plus resistance to compressive loadings that exist during the test. The interlocking design is used to keep the tiles from moving into the bundle region if and when the liner moves. The insulation is required so that high bundle temperatures can be reached; it is not required to protect the double-walled capsule or the reactor pressure tube. The double-walled capsule provides the mechanical support for the insulated fuel bundle.

The shroud also includes a Zircaloy liner that protects the insulation during shipping and bundle insertion. During the early portions of the test, the liner prevents water from permeating the insulation. Such water ingress would reduce the thermal resistance of the insulation. During the high-temperature part of the test, the Zircaloy liner simulates additional fuel rod cladding surrounding the 12 fuel rods.

A water annulus surrounds the outer tube. This water annulus is called the bypass coolant and is the key safety component for the FLHT-2 test. It cools the outer tube and indirectly cools the inner tube. The reactor pressure tube is the secondary containment for the test assembly and forms the outer boundary for the bypass water annulus.

Inlet Region

The inlet region contains the fixture that supports the bottom of the bundle, seals the bundle region to the bottom of the shroud, and provides sealed passageways for the bundle coolant, instrument, and pressurization lines and two TDR tubes. The bundle leads that exit through the inlet housing are sealed to the housing using grayfoil gaskets like those used in the closure region.

EFFLUENT CONTROL MODULE

The ECM, located on the top of the reactor, condenses steam from the fuel bundle and uses nitrogen gas to control the bundle coolant pressure.

ECM instruments measure the hydrogen concentration from a sample of the noncondensable effluent. In addition, the ECM provides access for two gamma spectrometers that monitor the filter and condensed steam for gamma-emitting fission products. The ECM is bolted to the top of the reactor to reduce the risk of breaking any pipes (tubes) connected to the test train and the catch tank.

The primary ECM components are a condenser, porous metal filter, thermal conductivity cell, float, control, relief and block valves, and the necessary tubing to properly connect the various components. The ECM also provides about 10 cm of lead shielding in the form of a "cave" that envelops the piping system for the main effluent stream.

The ECM is a 2-m-cubical box with sheet metal walls, floor, and ceiling that form a secondary containment. During the test, the interior of the box is maintained at a small negative pressure by exhaust fans. If any gaseous radioactive fission products leak from the piping inside the box, they would be swept by air through a charcoal filter and a high-efficiency particulate air (HEPA) filter before being safely released through the reactor stack.

The flow paths through the ECM are shown in Figure 7. At the left-hand side of the figure, effluent exits from the top of the test train into a heated steam line. The effluent flows past an evacuated sample bomb that is activated by a test engineer at the DACS. The effluent then takes one or more different routes. During a test, the effluent (steam-H_2-fission products) would normally flow through the porous metal filter and into the condenser. If the filter should plug, as indicated by a large differential pressure, the effluent would automatically be routed around the filter and into the condenser. The third route contains a safety relief valve that opens when the effluent pressure exceeds 2.4 MPa (350 psi).

A safety relief valve that connects the effluent with the nitrogen back-pressure line opens when the system pressure exceeds 1.6 MPa (240 psi). When the effluent reaches the condenser, steam is turned to water and flows out of the condenser into either of two parallel float valves. One valve provides the capability to handle large calorimetry water flows, and the other small valve handles the small condensate during the boildown. The liquid flows from the float valves into a waste line that connects with catch tanks located below the reactor.

All noncondensable gases exit through the top of the condenser into a separator (to remove any water) and through a pressure control valve. Once past the control valve, the gases flow with the condensate through a waste line to catch tanks.

Another line, located just before the control valve, routes some of the effluent stream through a filter, hydrogen meter, chiller, and then to a second hydrogen meter. The first meter continuously measures the local hydrogen partial pressure; the second provides the mass fraction of hydrogen in nitrogen carrier gas.

REACTOR/LOOPS

The NRU Reactor is operated by the Atomic Energy of Canada Limited (AECL). Full-length fuel bundles with commercial enrichment levels can operate in the reactor at nominal LWR power levels. The 130-MW heavy-water-moderated and cooled reactor has two loops (U-1 and U-2) that can be connected to various pressure tubes for experiments. The Zr-2 pressure tube used for the FLHT tests has an inside diameter of 10.4 cm (4.07 in.) and spans the length of the 9-m (30-ft) reactor vessel, including the active core length of 2.8 m (9 ft).

The U-1 loop provides steam and reflood water for steady-state and transient thermal-hydraulic conditions at low flows and at pressures up to 1.3 MPa (200 psi). The U-1 loop will supply steam to the superheater during the FLHT-2 test.

The U-2 loop can provide cooling water for steady-state and transient thermal-hydraulic conditions that simulate BWR and PWR coolant systems. The loop can supply water at 10.3 MPa (1500 psi) at inlet temperatures up to 315°C

(600°F). The U-2 loop will supply bypass coolant at 38°C, 1.3 MPa, and 1 kg/s during the FLHT-2 test.

The bypass coolant supplied by the U-2 loop enters the bottom of the reactor pressure tube and flows up the annulus between the pressure tube and the test train assembly. The water exits through the top of the pressure tube and flows back to the U-2 loop pumps.

Separate loop hardware supplies the bundle coolant for the test from several pressurized tanks. The water flows from the tanks to the top of the reactor through two bundle coolant lines at the top of the test train. Past the closure region, each of the two streams splits and flows down the outside of the plenum and core regions. The flow streams enter the bundle region through four penetrations in the inlet fixture.

Once injected into the inlet region, the bundle coolant flows up along the fuel rods, is heated by fission power, converted to steam, and reacts with the high-temperature Zircaloy cladding and liner to form hydrogen. The bundle steam/hydrogen effluent then flows through the plenum, the closure, the steam line, and through the ECM to the catch tank.

Loop systems also provide a source of water (from another pressurized tank) for the spray desuperheater, steam for the superheater, and water for the ECM condenser and chiller. Loop hardware also supplies nitrogen gas for the ECM and hydrogen dilutions.

A 15-kW superheater located near the top of the test train assembly is capable of supplying 13 g/s (100 lb/h) steam at 650°C (1200°F) and 2 MPa (300 psi). Superheated steam is used just before the boildown phase of the test to preheat the plenum and ECM effluent lines to prevent premature steam condensation during the boildown phase.

INSTRUMENTATION

More than 250 instruments will be used in the FLHT-2 experiment to measure local pressure, temperature, flow, neutron flux, liquid level, and hydrogen generation. A detailed listing of all the test instruments is provided in the FLHT-2 Experiment Operations Plan (EOP). In this section, the safety function instruments are presented. A layout of the test train instruments is shown in Figure 8, and ECM instruments are shown in Figure 9. During an actual test, almost all the test instruments are monitored by the test engineers. If it appears that test conditions (indicated by one or more instruments) are either becoming unsafe or are such that the main test objectives cannot be attained, then the test will be terminated. If corrections can be made in a reasonable time (one or two days), they will be made and the test will be restarted.

Most instruments used in the test are safety related because they indicate at least a potentially unsafe condition. However, some instruments monitor key test conditions that automatically or manually cause termination of the test if preset limits are exceeded. A test will be terminated when temperature, pressure, power, or weight values exceed preset high or low limits.

All of the FLHT-2 safety instruments were either used in FLHT-1 or are the same type of instrument used in FLHT-1. All the instruments used in the FLHT-1 test to monitor for unsafe conditions performed satisfactorily. Just prior to the FLHT-2 test, each safety instrument circuit will be checked to be sure the limits are correct and the electrical circuits are operational.

All of the instruments for the FLHT-2 test train assembly are new but are the same type as used for FLHT-1. The ECM and loop instruments are the same as used for FLHT-1 except for four new instruments that were added to the ECM. Two flow meters were added to measure the condensate flow out the parallel float valves. A hydrogen partial pressure gage and a mass spectrometer were also added to the noncondensable sample line in the ECM.

Most of the instruments provide data on temperatures, pressures, and flows needed to perform the test and subsequently needed to analyze the results. Some instruments provide only information for the safe conduct of the test; for example, MMPD signals, reactor power log rate meters, and bundle coolant supply weight. Some instruments provide both test data and information to assure a safe test.

DATA ACQUISITION AND CONTROL SYSTEM

The DACS is composed of the following major components: a Data General (DG) MV/6000 super-minicomputer, a NEFF A/D (analog-to-digital) subsystem, two Tektronix 4027 color graphics terminals, and several DG character terminals. The MV/6000 uses the AOS/VS virtual memory operating system and is 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 is attached to one of the terminals; two other terminals are connected to a Tektronix hard copy unit and either of them can initiate a data copy.

The DACS hardware and software are designed to accomplish the following operations:

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

The DACS is arranged in the configuration shown in Figure 10. One character terminal is used as the console to control the DACS; one character terminal and one graphics terminal are used by the test director for data monitoring and evaluation; and one character terminal and one graphics terminal are provided in a separate room for the use of test observers not involved in actually running the experiment. These terminals are equipped with a variety of monitoring functions, but no control functions. The major components and the personnel stations for operating and observing the experiment are shown schematically in Figure 11.

The DACS software is designed to use the function keys of the terminals to initiate desired routines. Certain functions available to the console operator are disabled in the other terminals. These special functions are necessary to operate the computer system, but they do not have any data reporting capability.

EXPERIMENT CONDITIONS

This section provides an overview of the operations planned for the FLHT-2 experiment. Detailed experiment procedures will be provided in the EOP. Information on the test equipment installation, preliminary operations, test performance, trip logic, and expected data is included in this section.

The test performance involves establishing a bypass flow through the annulus between the test train shroud and the reactor pressure tube to cool the tube and maintain the cold boundary necessary for the shroud to contain the high-temperature test. Once the bypass flow is established, a test assembly power (23 kW) equivalent to 0.524 kW/m (0.160 kW/ft) of fuel rod will be established by means of bundle water flow calorimetry. The bundle coolant flow will then be reduced to a value that will maintain the liquid level in the top 2 ft of the test bundle. The water in the plenum will then be drained to a level just above the top of the fuel, and superheated steam will be introduced at about the same elevation into the plenum. Operation under these conditions will continue until the upper plenum and ECM piping are dry and heated to a temperature that will preclude refluxing, approximately 315°C (600°F). The transient will then be started by stopping the bundle coolant flow. The test will continue until peak cladding temperatures of 2200°C (4000°F) are reached. At that time, the test will be terminated by reducing the reactor power to zero. The duration of the transient has been calculated to be about 15 min. The test will be terminated with no reflood flow, i.e., the test train will be maintained dry, if possible, until the bundle region has cooled below about 400°C; water will then be added. The test train will be discharged with water in the bundle and plenum regions.

The gases exiting the test assembly, mainly steam and hydrogen, will pass into the ECM. The ECM will control the effluent pressure, using nitrogen gas as backpressure, condense the steam, and measure the noncondensable gas stream to determine the hydrogen concentration.

FLHT-2 operating conditions are summarized in Table 2. The test, including the calorimetry phase, will last about 1 to 2 h.

INSTALLATION AND CHECKOUT

Experiment installation will involve inserting the test train into the L-24 position in the NRU Reactor, mounting the ECM, and connecting the necessary services, including the superheater.

Test Train Piping

There are two major piping systems for the FLHT-2 test (Figure 1). The first system supplies the bundle coolant to the test train. This coolant passes down through the closure, through two bundle coolant lines, through four bundle coolant downcomer tubes, and enters the bottom of the test train below the fuel rods at the inlet region.

During pretransient operation, the flow is sufficiently large that the test train power can be determined by means of a heat balance (calorimetry phase of the test). The coolant exits the test train through the effluent line to the ECM and from there to the loop catch (waste) tanks.

The second piping system--the bypass coolant system--receives coolant from the U-2 loop, measures it, directs it up the annulus between the L-24 pressure tube and the shroud, and then returns it to the U-2 loop from the top of the pressure tube.

Two minor piping systems were added for the FLHT-2 test. One provides a means of draining the plenum to a level slightly above the top of the fuel before the transient. The other piping system injects hot (superheated) steam at the bottom of the plenum. The hot steam is used to dry and heat the plenum before the start of the transient.

Piping installation and checkout involve connecting the inlet and outlet cooling lines, ensuring proper flows and flow meter calibrations, and ensuring leak tightness. The connections between the test train closure and the ECM

TABLE 2. FLHT-2 Operating Conditions

Flow Rate	Value
Bundle coolant - calibration	0.126 kg/s (1000 lbm/h)
Bundle coolant - operation	≤0.0076 kg/s (≤60 lbm/h)
Bypass coolant	1.3 kg/s (10,000 lbm/h)
Desuperheater water	≤8 g/s (60 lbm/h)
ECM condenser water	315 g/s (2500 lbm/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	approximately 4% neutron full scale
Fuel rod - linear	0.524 kW/m (0.160 kW/ft)
Bundle	23 to 27 kW
Temperature	
Peak fuel cladding	2200°C (4000°F)
Peak shroud saddle interior	540°C (1000°F)
Bundle coolant inlet	38°C (100°F)
Bundle coolant saturation	194°C (382°F) at 185 psig
Bypass coolant inlet	38°C (100°F)
Bypass coolant outlet	42°C (112°F)
Bypass coolant saturation	194°C (382°F) at 185 psig
Peak plenum	1370°C (2500°F)
Peak plenum outlet	370°C (700°F)

Pressure	Value
Bundle coolant	1.28 MPa (185 psig)
Fuel rod cold fill	(a)
Bypass coolant	1.28 MPa (185 psig)
MMPD cavity (helium filled)	0 MPa (0 psig)
Shroud insulation cavity (inert gas filled)	0.3 to 0.7 MPa (50 to 100 psig)
Plenum insulation cavity (argon)	$\leq$0 MPa ($\leq$0 psig)
Total Coolant Required	
Desuperheater	54 kg (120 lbm)

(a) To be presented in the FLHT-2 Experiment Operations Plan.

will be made and leak tested. These connections include the pressurization and pressure-measuring lines for each of the fuel rods, the shroud insulation region, the MMPD region, and the pressure transducer manifold (located inside the ECM). Finally, the safety ventilation system for the ECM will be connected.

Instrument Interfaces

Once the test train is installed, the instrumentation cables will be connected. Instrumentation will be provided to collect data, control the experiment, and provide appropriate safety trips. The FLHT-2 safety trip set points are listed in Table 3, and the controlled parameters are listed in Table 2. Each of these safety trip circuits will be checked to assure that they function properly.

Mechanical Interfaces

The mechanical interfaces have been proven in previous tests (see FLHT-1 Final Safety Analysis Report). In particular, the remote disassembly tools needed to separate the test train, the ECM, and the support systems were used after the FLHT-1 test and performed satisfactorily. Minor modifications to the shielding and confinement between the test train and the ECM will be made to facilitate disassembly.

PRELIMINARY OPERATIONS

The following preliminary operations will be necessary to show that there are no leaks:

* Instrument lead seals and mechanical seals between the bypass coolant system and the bundle coolant system will be tested before the test train is installed in position L-24.

* The test train head closure seals will be pressure tested to assure leak tightness.

TABLE 3. FLHT-2 Experiment Safety Trip Functions

Automatic Trips	Responsibility
Temperature	
Shroud saddle exterior - high	PNL
Bypass coolant outlet - high	CRNL
Plenum steam outlet - high	CRNL
ECM - high	PNL
Pressure	
ECM test train effluent - low	PNL
Flow Rate	
Bypass coolant - low	CRNL
Bypass coolant - high	CRNL
Power Change	
Mean power log rate - high	CRNL
Manual Trips	
Sensor	
Bundle coolant differential pressure - high	PNL
MMPD continuity - change	PNL
MMPD resistivity - low	PNL
MMPD cavity pressure - high	PNL
Others	
Bundle coolant - low accumulator weight	CRNL
Bypass coolant - low surge tank level	CRNL

* The compression gland seals around the instrument leads will be pressure tested at the same time as the head closure seals to assure that there are no bypass coolant leaks.

* Connections between the FLHT-2 test train, the ECM, and the loop catch tank will be tested to ensure leak tightness so that fission product disposal will be controlled as planned.

* An exhaust ventilation system will be attached to the ECM confinement to assure that any radioactive steam/off-gas leakage will discharge through charcoal and HEPA filters to the NRU reactor stack.

* FLHT-2 fuel rods will be pressurized, leak tested, and monitored to verify their integrity and to indicate their operating conditions during the experiment.

The following preliminary operations will be required to show the necessary experiment control:

* Safety trip operations will be verified, trip parameters will be programmed, and trip set points will be activated (see Table 3).

* The integrated DACS/loop control system (LCS) system will be tested to confirm control operability.

* The operating capabilities of the bundle coolant flow, bundle coolant pressurization, and desuperheater control systems will be tested and calibrated.

TEST PERFORMANCE

The FLHT-2 experiment will begin with a pretransient operation that will set the reactor power to give the desired fuel bundle power of 23 kW. Once the reactor and bundle powers are set, transient operations will begin by stopping the bundle coolant flow. The experiment will be terminated when conditions producing a predicted peak cladding temperature of 2200°C (4000°F) are reached.

Pretransient Operation

The 23-kW fuel bundle power will be established by setting the bundle coolant flow rate at about 0.126 kg/s (1000 lb/h) and the supply temperature at about 38°C (100°F). The reactor power will then be adjusted to a level where the 23 kW is reached, as determined from the bundle flow rate and inlet-to-outlet temperature difference. This difference will be 44°C (78°F), which will result in an outlet temperature of about 82°C (178°F)—well below the saturation temperature of 194°C (382°F). The exact values of the bundle coolant parameters are not critical; once they are set, however, their values should remain constant. The reactor power necessary to give a test train power of 23 kW is expected to be about 4% of full power. Bypass cooling is not required during preconditioning, but it will be established before preconditioning begins, so it will be at the proper value when the transient begins.

Transient Test

The transient test will begin after the bundle power is established during the pretransient test. DACS input to the LCS will cause the bundle coolant flow to decrease to a value that is calculated to give dry outlet steam at a temperature of 425 to 540°C (800 to 1000°F). The liquid level in the test assembly will be 3.0 to 3.4 m (10 to 11 ft) above the bottom of the fuel. The test section drain valve will then be opened, which will drain coolant from the plenum region to about 28 cm (11 in.) above the top of the fuel. The drain discharge flow rate will be higher than the bundle coolant flow rate so that the coolant in the ECM and plenum will be drained. The drain will continue until the plenum is empty and the liquid level is down to the drain elevation. Next, a hot gas (dry steam or nitrogen) will be introduced into the bottom of the plenum until the plenum interior and ECM piping to the condenser are above the saturation temperature and steam condensation and refluxing are no longer likely. The drain valve will be closed when two-phase flow through the drain line no longer exists, i.e., dry steam flow only. The hot gas preheating flow will be stopped when the plenum and ECM piping are dry and heated.

When steady-state conditions are reached as indicated by a constant steam outlet temperature and constant liquid level, the transient will be started by stopping the bundle coolant flow. The transient will be allowed to continue until conditions producing a temperature of 2200°C (4000°F) are reached. At that time, the test will be terminated by automatically decreasing the reactor power to a low neutron level and then tripping the reactor.

Because the test assembly TCs are not expected to remain reliable at the peak temperatures planned for this test, a method was defined to determine when the desired termination temperature (2200°C) is reached. The elapsed time between two temperatures will be measured during an early test period when the TCs are still reliable. This information, together with the results of parametric TRUMP-BD calculations, will be used to define the elapsed time that must be allowed to reach the 2200°C (4000°F) desired temperature.

In addition to the "timed" definition of the desired peak cladding temperature, liner TC temperatures will be used to assess the approach to desired temperatures and the termination time will be adjusted if necessary. The liner temperatures will be used for this purpose because data from the FLHT-1 test showed that several liner TCs not only survived the test but also indicated that the temperature increase had stopped before the reactor was shut down.

The calculated peak cladding temperatures, the bundle coolant liquid level, and the core region outlet coolant temperatures are presented in Figures 12, 13, and 14, respectively. The calculations assumed no relocation of the fuel or the cladding.

Bypass coolant conditions will be maintained constant 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 38°C (100°F), and the flow rate will be adjusted to 1.26 kg/s (10,000 lb/h). A maximum outlet temperature of about 45°C (112°F) is expected, which is well below the bypass coolant saturation temperature (194°C).

Test Termination

The prime criterion for determining the success and termination of the FLHT-2 experiment is achieving a peak fuel cladding temperature of 2200°C (4000°F). Once conditions that result in 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 shutdown method will provide the least thermal shock and, therefore, will minimize post-test fuel damage.

Test Restart

If a reactor trip should occur during FLHT-2, a restart will be considered. The major factors in considering a restart are the number of instruments—primarily bundle TCs—that are still functioning and the cause of the reactor trip. If a restart is deemed feasible, the bundle coolant rate will be set to a value that would stabilize the liquid level when the reactor power is again established. The reactor power will be set to the level at which the trip occurred. The bundle coolant will then again be stopped.

A restart is also being considered to about one-third of the test power after the termination of the test. This restart would test the TCs and supply confirmatory data concerning TCs that had failed.

Steam Generation Rate

Steam will be generated by boiling of the bundle coolant below the steam/water interface. The rate of steam formation will be directly proportional to the total bundle power below that interface and will, therefore, decrease as the boiloff continues and the interface lowers. The steam generation rate as a function of time after the start of the transient is illustrated in Figure 15.

The steam generated will react to some extent with the hot Zircaloy cladding and liner. The steam exiting the test section will therefore be reduced from that quantity generated. The calculated outlet steam flow rate is shown in Figure 16. The outlet steam flow drops off starting at 8 or 9 min, when the test assembly temperature becomes high enough to cause significant Zircaloy oxidation.

TRIP LOGIC

The DACS has the capability to send a trip signal to the LCS. This signal will be sent automatically when certain safety sensor data or safety sensor group data exceed set points or preset safety limits. This trip signal may also be sent by the DACS console operator as a manual trip. The logic used by the DACS for automatic trips or by the console operator for manual trips is described below.

Automatic NRU Reactor Trip

The DACS will scan the instrument data at least once per second during the experiment. After each scan, the readings of certain instruments will be checked against the preset safety limits. If one or more instrument is beyond a limit, appropriate action will be taken by the computer. This action will vary depending on the instrument.

The TCs that are to be used as safety sensor instruments are combined in most cases into safety sensor groups of four. If two of the four sensors read in excess of the set point, a software flag is set for that safety sensor group. If the next scan of the instruments also shows two of the four sensors in that group beyond the limit, then the automatic trip is initiated by the DACS. If a two-out-of-four condition is not found in the second scan, the software flag for that safety sensor group is cleared. This technique is used so that spurious electronic noise will not cause premature test termination.

Prior to or during the beginning of the FLHT-2 experiment, it is conceivable that one or more of the sensors in the safety sensor groups may become inoperable. In this case, the safety sensor groups that have failed may be redefined, either by replacing the failed sensors with good ones or by redefining that group to a two-out-of-three safety group (the appropriate CRNL personnel will be notified).

Manual NRU Reactor Trip

Certain conditions might arise during the test that would cause the test director to shut down the NRU Reactor. The DACS provides the capability to manually initiate a trip. For a nonemergency condition, the test director would request that the NRU control operator shut down the reactor. One condition that could lead to a shutdown is: TDRs indicating full bundle coolant level and fuel rod bundle temperatures also reading high, indicating that the fuel bundle coolant channel may be blocked. Safety trip criteria proposed for the FLHT-2 test will be listed in the EOP.

EXPECTED TEST DATA

Data are desired for the following variables:

* temperature
* hydrogen concentration
* pressure
* bundle coolant level
* fission products.

In addition, other data will be collected for safety reasons; for example, bypass flow data, MMPD information, and saddle temperature data. Still other data will be taken to assure controlled conditions; for example, system pressure, reactor power, and nitrogen carrier gas flow rates.

Temperature Measurements

Fuel cladding and structural component TCs will provide radial and axial indications of test assembly temperatures as long as the TCs remain operational. High-temperature Type C (W-5 Re/W-26 Re, BeO-insulated, and Zr/Ta-sheathed) TCs will be used to measure interior fuel cladding temperatures. However, TC failures are anticipated to occur between approximately 1540°C and 2040°C (2800°F and 3700°F). High-temperature Type C TCs will also be used in the bundle coolant region and on the liner at various elevations for fuel cladding temperature comparisons. Predicted temperature gradients between the fuel and the steam and between the fuel and the liner are approximately 30°C and 110°C (50°F and 200°F), respectively. Consequently, liner temperature histories will be extrapolated to deduce peak fuel temperatures after the fuel cladding TCs have failed.

Type K TCs will be used extensively throughout the FLHT-2 lower temperature region—below 1100°C (2000°F)—where their continued operation throughout the experiment is expected. These TCs will provide temperature histories of the inlet and outlet of the bypass coolant channel. They will also be used in the inlet region, the lower bundle coolant channel, the saddle, downstream of the desuperheater spray heads in the double-walled plenum, the bundle coolant channel, and on the test train effluent and ECM piping.

Radial and axial fuel/coolant/shroud temperatures were calculated for FLHT-2. Radial temperature gradients are quite flat, particularly at the high temperatures that occur late in the test. Radiation is the predominant mode of heat transfer, which is a factor in the uniformity of the temperature gradients. Consequently, at the hottest elevation, the liner is only about 110°C (200°F) cooler than the hottest rod. Axial temperature profiles for the hottest rod, liner saddle, and inner round temperature are shown in Figure 17. The similarity of the cladding and liner plots illustrates the flatness of the radial temperature profiles in the fuel rod bundle.

Hydrogen Measurements

Significant amounts of hydrogen will be generated when the Zircaloy/steam interface temperature exceeds 1100°C (2000°F). The hydrogen generation rate will depend on the cladding temperature, the presence of steam, and the thickness of the ZrO_2. The hydrogen concentration will be measured with partial pressure, mass, and fraction thermal conductivity meters in the ECM outlet piping above the reactor deck. Predicted hydrogen generation rates are shown in Figure 18.

The amount of hydrogen generated in the upper steam plenum is insignificant because the temperatures are too low.

Pressure Measurements

The bundle coolant region will be monitored with differential pressure sensing lines that tap the test assembly inlet and outlet pipes (Figure 9). These lines are connected to transducers located in the ECM to provide data on the bundle coolant operating pressure.

Bundle Coolant Level Measurements

TDRs monitor the liquid level in the test assembly even though they will be mounted in manometer tubes in the bypass annulus. The liquid level data are expected to provide a coolant level history for the controlled boilaway experiment that will be correlated with fuel bundle TC data.

Fission Product Monitoring

The fuel rods are expected to rupture near a cladding temperature of 980°C (1800°F). Small quantities of Xe and Kr fission products will then be released through the ruptures. Three gamma spectrometers will monitor for fission products in the effluent pipe, ECM filter, and condenser.

POST-TEST OPERATIONS

Post-test operations include ECM and test train removal and postirradiation examination (PIE).

ECM and Test Train Removal

The test train will be cooled with bypass coolant until the bundle and bypass coolant system can be depressurized. This procedure is expected to require several hours after the test has been completed. The driving force for heat removal late in the cooldown may be so small that a complete cooldown would be inordinately long. In this case, small amounts of bundle coolant will be added to decrease the cooling time.

Radiation monitors will be used to ascertain the radiation fields near the deck plate and around the ECM. If the fields are higher than acceptable, the test train may be flooded with bundle coolant water. The flush line would then be used to flush clear water through the plenum and out the steam line until acceptable radiation fields are achieved.

When radiation fields are low enough to permit ECM and test train removal, temporary (jumper) pipes and all other instrument and power connections to the ECM will be removed to facilitate access to the FLHT-2 closure region. The test train instrument leads will be severed, and all tubes between the ECM and test train will be crimped/sealed and severed with long-handled tools to minimize radiation exposure. The closure plug, seal, and hold-down components will also be removed with long-handled tools; a grappling attachment will be installed on the FLHT-2 closure plug. The FLHT-2 test train will be withdrawn from the reactor in the shielded cavity in the J-Rod flask and transported to the fuel elevator for transfer to the rod bay. If necessary, the test train may remain water filled during these transfer operations. The ECM filter may be removed for gamma spectroscopy and a minimal hot cell examination.

Postirradiation Examination

As part of the transfer operation to the fuel elevator, the core region of the test assembly will be gamma scanned to detect relocated fuel. The test train assembly and shroud will be sectioned in the rod bay with a saw mounted on the disassembly, examination, reassembly machine (DERM). Photography will be the primary means to document the effects of the CBDP experiment on the fuel bundle. Fuel and core debris may be characterized in the CRNL hot cells.

PROJECT QUALITY ASSURANCE

The FLHT-2 quality control plan (QCP) is based on the elements of PNL's quality assurance (QA) manual (PNL-MA-65). All activities that affect quality will be monitored, starting with design, analysis, test predictions, and instrument development and continuing through materials and procurement handling, fabrication, testing, inspection, storage, and shipment. Detailed specifications and procedures for these activities are contained in material, product, welding/brazing, fabrication, and assembly documents. Approved drawings of all assemblies and components will provide easily understood information to aid in machining, fabrication, or assembly operations.

A central project file with individual task files has been established for material specifications, welding/brazing specifications, drawings, procurement documents, inspection logs, analytical data, and other information deemed necessary for traceability of materials, fabrication, and analysis. At the completion of the experiment series, these files will be combined into a main project file. The project manager has appointed a project quality control representative (PQCR); and both have the organizational freedom to make independent assessments of quality and direct the attention of other project members to any quality problem, its cause, and recommended corrections. An engineer from the QA organization provides assistance in implementing the QC elements described in the QCP and may conduct independent audits of the project. A materials review board processes all nonconformance reports on discrepancies of materials or components.

The objective of the QCP is to assure that the quality of the design, fabrication, assembly, and supporting analytical work meets as high a standard as practicable. To accomplish this, the QCP will provide the basis for selecting elements to assure that a materials, component, or instrument system will perform satisfactorily during the FLHT-2 experiment.

ELEMENTS OF QUALITY ASSURANCE

Quality Assurance Plan. The purpose of a Quality Assurance Plan (QAP) is to assist in planning QC activities for the project and to identify special QA requirements specified by PNL's QA organization or sponsor.

Design Control and Method Review. The purpose of this element is to assure that the design requirements are formally documented in design drawings, specifications, and procedures.

Procurement. The purpose of this element is to provide documented records of all procurement activities and to establish a central file of all purchased materials and components.

Instructions, Procedures, and Drawings. The purpose of this element is to assure that activities affecting quality are accomplished in accordance with documented instructions, procedures, or drawings.

Document Control. The purpose of this element is to assure that documents affecting quality are properly prepared, identified, reviewed, approved, distributed to the affected work location, and maintained.

Material Identification and Controls. The purpose of this element is to assure that only acceptable materials or components are used and that these meet special requirements for identification, storage, and use.

Key Fabrication and Special Processes. The purpose of this element is to provide documented verification of the control of key fabrication processes that affect the quality.

Inspection and Testing. The purpose of this element is to assure that items requiring inspection and testing conform to specifications and design requirements.

Calibration. The purpose of this element is to assure that all measurement and test equipment are suitable for the intended purpose and are maintained in accordance with specified calibration and service procedures.

Handling, Storage, and Shipping. The purpose of this element is to assure that all materials or components requiring special instructions are handled, stored, or shipped to reduce unnecessary damage, deterioration, or loss.

Nonconformance and Corrective Action. Regulations concerning nonconformance to all requirements. Copies of all pertinent documents are maintained in the project file.

PROJECT RECORDS

Project records include but are not limited to:

* purchase orders/certifications
* design drawings
* material specifications
* product specifications
* welding/brazing specifications
* inspection reports
* special process procedures
* nonconformance reports
* calibration reports and records
* audits

* testing reports

* design analysis

* assembly procedures

* shipping documentation

* project correspondence and reports.

SPECIFICATIONS AND PROCEDURES

Different types of specifications and procedures are being developed, implemented, and maintained for the CBDP Program:

* Specifications
  — material
  — product
  — welding/brazing
  — heat treatment

* Procedures
  — assembly
  — welding/brazing
  — special processes or operations
  — instrument calibration
  — shipping
  — procedure for test train insertion into the L-24 position of the NRU Reactor
  — testing or checkout
  — experiment operations.

SAFETY CONCERNS DURING NORMAL OPERATION

Safety concerns that could cause harm or damage during the FLHT-2 experiment are radioactive fission products, radiation fields, reactivity changes, hydrogen generation, materials at high temperature, steam explosion, and steam pressure pulses. The radioactive fission products are produced in the 12 UO_2 fuel rods during the short (approximately 1 h) calorimetry and during the subsequent boilaway phases. The radiation fields are created by the decaying radioactive fission products and the decaying neutron activation products. Once the test train assembly is loaded into the NRU Reactor, reactivity changes occur due to the removal of the bundle water from the core region and the axial movement of UO_2 in the test bundle. Hydrogen is produced from the chemical reaction of Zircaloy with steam in the high-temperature bundle region. A steam explosion could occur if molten (U, Zr, O) reacted with water under certain conditions. A steam pressure pulse would occur in the bundle region if hot material, including Zircaloy, fell into the water pool at the bottom of the fuel bundle region.

All of these hazards were analyzed for the FLHT-1 test and were reanalyzed for the FLHT-2 PSAR. Previous analysis that is still applicable to the FLHT-2 test is summarized in this section; recent analyses performed after the FLHT-2 PSAR submittal are presented in detail. Previous analysis on the following hazards still apply to FLHT-2:

* radioactive fission products (FLHT-1 FSAR and FLHT-2 PSAR)
* radiation fields (FLHT-1 FSAR)
* reactivity changes (FLHT-1 FSAR)
* steam explosions (FLHT-2 PSAR)
* steam pressure pulses (FLHT-2 PSAR).[a]

(a) Slightly modified as a result of a recent test train assembly design change.

Because of some recent changes in the analytical tools (in the TRUMP computer code) used to predict temperatures within the core region of the test train assembly and a shift in the emphasis of the test safety features, new analyses are summarized below concerning the hazards associated with hydrogen and materials at high temperature.

RADIOACTIVE FISSION PRODUCTS

The presence of radioactive fission products (contamination) is an inherent hazard associated with nuclear energy. Proper hardware design and handling procedures based on years of experience and scientific study provide for safe working conditions. The fission product inventory in the FLHT-2 test will be less than that generated in the FLHT-1 test. The radioactive species will be contained within the same or identical hardware used for the FLHT-1 test.

The primary containment system (bundle steam/hydrogen flow path) was designed for pressures above 2.4 MPa (350 psi); the test pressure will be 1.3 MPa (185 psi). The primary system will be tested at the test pressure to assure leak tightness. Should a leak develop during the test, the hazardous location will be downstream from the test train closure plug. The steam/hydrogen flow path from the top of the closure has a secondary confinement (maintained at a slight negative pressure) to prevent the release of gases into the reactor hall. If the FLHT-1 fission products behavior is duplicated in FLHT-2, essentially none will be released from the UO_2 pellets.

All of the released fission products are expected to be contained with the low-pressure 1.3-MPa (185-psig) bundle coolant system including the ECM piping. The system will be cold pressure checked to 1.7 MPa (250 psia) to insure good integrity and leak tightness. Should any leaks develop during the tests, the escaping component will be contained with a low pressure exhaust system.

Components in the ECM have different pressure and temperature design limits, all of which are comfortably above the operating conditions. Lower limit components are the condenser and float valves. The condenser is designed for 6.9 MPa (1000 psig) at 450°C (850°F); the float inside the float valve will collapse above 3.1 MPa (450 psig) at 260°C (500°F). The condensate from the condenser that flows into the float valves has a maximum temperature of 194°C (382°F).

Since the FLHT-2 peak fuel temperature will be about 200°C higher than temperatures in FLHT-1, it is possible that some volatile fission products will be released from the fuel. These products will be contained within the bundle coolant system.

RADIATION FIELDS RESULTING FROM FISSION PRODUCT RELEASE

Radiation fields associated with FLHT-2 are due to the presence of both fission and activation products; few, if any, activation products are expected to contribute to the existing working background radiation levels. Actual radiation levels during the FLHT-2 test are expected to be about the same as those for FLHT-1. The radiation levels during FLHT-1 were close to background.

Argon gas (approximately 1-atm pressure) was used to improve the thermal resistance of the ZrO_2 fiberboard during FLHT-1. However, when the liner breached, neutron-activated argon was released from the insulation into the bundle steam and hydrogen effluent. The presence of activated argon temporarily increased local radiation levels while the argon flowed through the ECM en route to the catch tank.

Several gases were evaluated as a possible replacement for the argon fill gas. Currently, two gases appear to be acceptable: nitrogen and neon. Nitrogen is the first choice because it has lower neutron activation and lower thermal conductivity than neon. The only concern with nitrogen is the reaction with the Zircaloy liner. Such a reaction appears acceptable because it proceeds more slowly and releases less heat than the Zircaloy/steam reaction.

The FLHT-2 experiment will be very brief, using 12 nonirradiated LWR fuel rods. These fuel rods will probably be exposed for less than 3 h—0.615 kW/m (0.188 kW/ft)—at about 5% of full power in the NRU Reactor. During that period, a minimal fission product inventory will be produced (less than that for FLHT-1). Inventories were calculated with the ORIGEN-2 code and are compared in Table 4 for the FLHT-1 and FLHT-2 fuel rod bundles. The percentages of these fission products to be released from the fuel rod bundles (that retain their structure) were calculated using temperature- and time-dependent release coefficients from NUREG-0956.

In every case, the released fission product activity is at least a factor of four less for FLHT-2 than predicted for FLHT-1. In some cases, it is lower by at least a factor of 10. It could be concluded that the FLHT-2 experiment will be less severe and the fuel will not release any radioactivity.

However, the possibly molten state of fuel or core debris must be taken into account. In the molten state, fission product mobility is considerably enhanced. Although some safety analyses have assumed that molten fuel will release all of its volatile fission products, recent SFD experiments have measured fission product release for both trace exposure and high-burnup LWR fuel. It was concluded that considerably smaller release fractions are released from severely degraded trace-irradiated fuel. Release data from SFD 1-1 were chosen to represent the FLHT-2 experiment because both use trace exposure fuel heated to about 2200°C (4000°F) but without reflood cooling.

SFD 1-1 empirical fission product release fractions are also shown in Table 4. They are comparable with analytical predictions. Evidently, the enhanced fission product release from molten fuel is counteracted by transport to other in-core locations where plateout or chemical reactions preclude or retard fission product release from the test train assembly. These empirical release fractions are used to calculate "expected FLHT-2 fission product activity" that may be released in the FLHT-2 effluent. This activity is tabulated in Table 4 for 13 elements.

The expected fission product release activities for FLHT-2 are less than half of the release activities predicted for all radioisotopes that could be

TABLE 4. Fission Product Inventory and Release Estimates

Element	FLHT-1 Calculations				FLHT-2 Calculations				SFD 1-1 Data	Expected FLHT-2 Release[a]	
	Inventory, Ci	Release			Inventory, Ci	Release				Activity, Ci	
		%	Activity[b] 0 h	24 h		%	Activity 0 h	24 h	Release, %	0 h	24 h
Kr	4,873	35.51	1730	2.5	4,318	14.09	608	0.52	15	647	0.55
Xe	5,524	35.51	1961	95.6	5,226	14.09	736	14.8	15	783	15.8
I	5,607	35.51	1991	85.8	4,468	14.09	630	12.2	12	536	10.4
Rb	6,759	35.67	2411	0.8	6,415	14.11	905	0.19	9.8	629	0.13
Cs	6,245	35.67	2228	0.02	6,310	14.11	890	—[c]	9.8	618	—
Te	5,166	1.99	103	1.46	4,845	0.45	22	0.12	0.4	20	0.11
Ag	106	12.81	13.5	0.58	96	3.72	3.6	0.07	NA[d]	3.6	0.07
Sb	2,544	3.19	81.2	0.18	2,438	1.11	27	0.03	NA	27	0.03
Ba	6,793	1.27	86.3	0.32	6,483	0.35	23	0.03	0.5	33	0.04
Mo	3,899	1.58	61.6	1.42	3,816	0.31	12	0.10	NA	12	0.10
Sr	7,262	0.19	13.8	0.21	6,449	0.10	6.5	0.05	NA	6.5	0.05
Zr	5,620	0	—[c]	—	5,381	0	—	—	NA	—	—
Ru	375	0.02	0.08	—	269	0.01	0.03	—	NA	0.03	—
Total[e]	99,940				92,530						

(a) Based on SFD 1-1 data.
(b) Time after reactor shutdown.
(c) Less than 0.1% of the total activity at the time.
(d) Not available.

released from FLHT-1. Because essentially no radioactive fission products were released during FLHT-1, fission product release from FLHT-2 does not present a significant hazard.

REACTIVITY EFFECTS

The reactivity effects of voiding the (poison) light-water from the bundle and bypass coolant regions and also the reactivity effects of relocated fuel were calculated for the FLHT-1 experiment. These calculations were reviewed for FLHT-2 and found to be appropriate, conservative, and acceptable.

Voiding water from the bundle region increases the reactivity 0.48 mk and also increases the bundle power from 23 to 27 kW. This voiding of the bundle coolant is a planned part of the FLHT-2. Voiding of the bypass water increases the reactivity to 0.57 mk. The reactivity effect of the combination voiding of both bundle and bypass water is +1.07 mk.

The calculated reactivity effect of relocated fuel (described later) is +0.94 mk and the total bundle power would increase slightly. These calculated reactivity changes are within the acceptable limit of +6.0 mk for the NRU Reactor; however, as mentioned earlier, the loss of bypass coolant is not an acceptable condition because of the resulting damage due to the thermal loading on the shroud and reactor pressure tube.

The reactivity increase associated with the assumed movement of fuel from the top of the fuel bundle down near the core centerline was calculated by CRNL for the FLHT-2 test; the reactivity increase is +0.94 mk. This value is for the 2% enriched FLHT-2 fuel. Allowance was made for the use of two aluminum nitride rods in reactor loading. The aluminum nitride rods were used to reduce the L-24 local flux and allow a higher (more controllable during restart) reactor power for the required 23-kW bundle power. The increase in power for the relocated fuel was also calculated and is discussed later with debris bed thermal analyses.

HYDROGEN GENERATION

Hydrogen is one of the products of the FLHT-2 experiment. The measured generation rates during the course of the experiment are important data. The presence of hydrogen requires careful handling/processing because hydrogen can burn and explode. To prevent any hydrogen reactions (mainly with air), the hydrogen is diluted with nitrogen so that the hydrogen concentration is less than 4%. The constant dilution rate with nitrogen is based on the predicted peak hydrogen generation rate of 0.18 g/s (1.4 lb/h) 130 L/m (STP). The predicted hydrogen rates during the course of the experiment are shown in Figure 18.

Hydrogen is produced by the reaction of hot Zircaloy with steam. Zirconium, the main constituent in Zircaloy,[a] reacts with steam as follows:

$$Zr + 2H_2O \longrightarrow ZrO_2 + 2H_2 + Q$$

where Q = the exothermic heat of reaction
= 143 kcal (598 kJ) per mole Zr
= 1555 cal (6506 J) per gram Zr
= 3972 cal (16.6 kJ) per gram H_2O
= 35.7 kcal (149 kJ) per gram H_2
= 3192 cal (13.4 kJ) per liter H_2.[b]

The rate of hydrogen production depends on the surface area, temperature, and amount of prior oxidation of the Zircaloy. The TRUMP computer code was used to calculate the hydrogen production rate. The core region test train assembly geometry and material characteristic plus the fission heat and starting bundle coolant conditions were input to the code. Using this input, the code calculated as a function of time 1) the steaming rate (from the fuel rod fission heat), 2) the water liquid level, 3) the Zircaloy reaction with steam, associated heat, and hydrogen production, and 4) new temperatures due to the fission

(a) Zircaloy-4 contains 98.2% Zr, 1.5% Sn, 0.2% Fe, and 0.1% Cr.

(b) At 1 atm and room temperature.

and chemical heats. The Zircaloy oxidation rate as a function of temperature was calculated within the TRUMP code using the experimentally developed correlation of Urbanic and Heidrick. Parabolic kinetic rates were assumed as a function of time.

Until very recently, the TRUMP Zircaloy oxidation model included an oxidation rate inhibitor that increased with hydrogen concentration in the steam. Based on new data,[a] no such inhibition occurs at least until the hydrogen concentration is above 95 mol%. Therefore, the model was dropped from the TRUMP code and the TRUMP calculations presented in this report exclude the so-called hydrogen blanketing model. The calculated hydrogen generation rate, exit coolant fraction, and cumulative production are shown in Figures 18, 19, and 20, respectively.

The removal of the hydrogen blanketing model accelerates the test temperature ramp rates, reducing the overall length of the experiment. Even though the hydrogen generation rates are now calculated to be larger than with the hydrogen blanketing model, the predicted total hydrogen production is reduced. The chemical energy associated with the Zircaloy oxidation and hydrogen generation is addressed in the section on high-temperature materials.

MATERIALS AT HIGH TEMPERATURE

The peak fuel bundle temperature planned for the FLHT-2 experiment is 2200°C (4000°F), which is about 200°C (360°F) hotter than the peak temperature attained in FLHT-1. Bundle region materials at such high temperatures can cause damage to reactor components. No damage was seen to the outer tubes after the FLHT-1 test. This section analyzes the conditions associated with the containment of bundle materials at least as hot as 2200°C. First, an analysis is presented that shows that the heat flux through the Zircaloy inner and outer tubes can be much greater than the experimental heat fluxes provided bypass coolant is present. Next, effects of chemical power from the Zircaloy

(a) J. T. Prater and E. L. Courtright. 1985. High-Temperature Oxidation of Zircaloy-4 in Steam and Steam-hydrogen Environments. PNL-5558, Pacific Northwest Laboratory, Richland, Washington.

oxidation are presented to help quantify the experimental heat fluxes. Next, an analysis is presented to show how the test termination time will be determined so that peak bundle temperatures reach 2200°C.

This section of the report examines the thermal conditions that might breach the outer shroud and cause possible damage to the pressure tube. It is concluded that these consequences are not possible. The primary pressure boundary for these tests is the two shroud tubes; and as long as they remain cool, no problems will exist regardless of what happens inside the test train. The thrust of the analysis is directed toward determining that the inner and outer tubes are indeed maintained cooled. The shroud insulation is not intended or needed as a safety feature. Rather, it is necessary to achieve the high temperatures desired for the fuel rods.

The inner and outer Zircaloy tubes are cooled by water flowing through the annulus between the shroud and the reactor pressure tube. A flow of at least 1 kg/s (8000 lb/h) of water with an inlet temperature of about 38°C is planned. The mass flux, linear velocity, and heat transfer coefficients corresponding to this flow condition are 615 kg/s-m^2 (0.452 x 10^6 lb/h-ft^2), 0.7 m/s (2.3 ft/s) and 0.5 W/cm^2-°C (875 Btu/h-ft^2-°F).

The energy source to cause heating of the shroud tubes and the bypass coolant is the radial heat flow out through the shroud, driven by the temperature difference between the test assembly interior and bypass coolant. The magnitude of this heat flow rate has been calculated to be 17 kW for FLHT-2. The measured heat flow rate into the bypass heat sink for FLHT-1 peaked at about 28 kW. The difference between the two heat flows is attributed to two factors. First, the duration of the FLHT-2 test, as calculated, is much shorter than the duration of the FLHT-1 test. The thermal time constant of the shroud is fairly long and the thermal inertia of the shroud delays its heating during faster transients.

The second factor contributing to the 28-kW heat flow was the injection of hot steam into the insulation cavity when the shroud liner breached. The quantity of steam injected was not measured but was calculated to be as much as 100 L (3.7 ft^3). It nearly instantaneously carried approximately 1040 kW-s (1000 Btu) of heat energy into the shroud cavity. This injection of heat into the shroud cavity increased the heat flow into the bypass coolant. The FLHT-2 test is intended to be operated with the shroud cavity somewhat pressurized to reduce the pressure difference across the liner. This will reduce the magnitude of the steam injection when the liner breaches.

To be conservative, however, a peak heat flow of 28 kW was used to assess the cooling of the Zircaloy tubes for FLHT-2. The 28-kW heat flow is distributed axially nearly directly proportional to the liner to bypass coolant temperature difference. The average temperature of the liner above the liquid level was determined and used to calculate the ratio of the liner to coolant temperature difference at the peak axial elevation to the average difference for the test assembly. This ratio is 1.74. Below the liquid level, the temperature difference is so small that the heat flow would be negligible. The maximum specific heat flow is therefore calculated by dividing the 28-kW heat flow into the bypass by the 2.9 m (9.5 ft) of test assembly above the liquid level and multiplying by the 1.74 peak-to-average axial heat flow. The maximum specific heat flow was calculated to be 16.8 kW/m (5.13 kW/ft). When used with the 0.5 W/cm^2-^oC heat transfer coefficient, this heat flow gives a maximum difference between the outside surface of the shroud and the bypass coolant of 12^oC (21^oF).

The 28-kW radial heat flow into the 1.0-kg/s bypass coolant will increase the coolant temperature by about 3^oC (6^oF) at the elevation of peak radial heat flow and by about 7^oC (12^oF) at the outlet. Assuming an inlet temperature of 38^oC (100^oF), the coolant outlet temperature and the peak outside shroud surface temperature are calculated to be about 45^oC (112^oF) and 53^oC (127^oF), respectively. Structural integrity of the Zircaloy tubes is expected at these low temperatures.

To provide an estimate of the safety margin, conditions to cause bulk boiling at the bypass outlet and conditions to cause subcooled local boiling at the peak radial heat flow elevation were determined as shown in Table 5.

TABLE 5. Bypass Coolant Margins

Parameter	Change in Parameter to Cause: Bulk Outlet Boiling	Subcooled Local Boiling
Coolant flow	-2400%	-4400%
Radial heat flow	+2400%	+1300%
Inlet temperature	+150°C (270°F)	+142°C (255°F)

Neither the condition of bulk coolant outlet boiling nor subcooled boiling should be construed as safety limits. They are used to provide a convenient method of illustrating the huge safety margins in the radial heat sink. It can be seen that the margins are large, about two orders of magnitude.

The inner Zircaloy tube and saddle temperatures are also of interest. The inner tube temperature was calculated for the given radial heat flows by using a lumped-parameter approach. The thermal conductivity of the MMPD was calculated to be an average of the resistance of the helium gas and the detector wires. The major uncertainty in this calculation was the gap between the saddle and the inner tube. The gap has a cold design value of 0.51 mm (0.02 in.); but since the saddles are not rigidly fixed, the gap can vary significantly from the design. Data were therefore taken during FLHT-1 at a time prior to the excursion to evaluate a conservative saddle-to-outside surface thermal resistance. The evaluation was done by first determining the ratio of the temperature difference between the liner and saddle to the temperature difference between the saddle and the outside surface. The temperature difference ratio is directly proportional to the thermal resistance of the two heat paths in series. The thermal resistance from the liner to saddle is primarily that of the insulating tiles, which is reasonably well known. The calculation of the saddle to outside surface thermal resistance was found to be 13.3°C/kW-m (0.023°F/Btu-ft). When this value is used with the peak radial heat flow of 16.8 kW/m, the saddle temperature is 222°C (400°F) above the outside surface temperature or 268°C (514°F). If the radial heat flow that causes local subcooled boiling, as given above, is used, the saddle temperature is 2886°C above the outside surface temperature. This requires a saddle temperature far above the saddle trip temperature. Finally, for a saddle trip temperature of 982°C (1800°F), a radial heat flow of 70.6 kW/m (73,600 Btu/h-ft), 4.2 times the probable value, would cause a reactor trip.

The inner Zircaloy tube temperature will be 109°C (277°F) under normal conditions and 325°C (615°F) at the time an increased heat flow would cause the saddle temperatures of 982°C (1800°F) to trip the reactor.

The radial heat flow through the shroud as a function of insulation thickness is presented in Figure 21. The following assumptions were used to develop the curve:

* Operation is at steady state.

* The inner surface temperature of the insulation is 2760°C (5000°F).

* Bypass coolant temperature is 65°C (150°F).

* The thermal resistance from the inside surface of the liner to saddle is proportional to the insulation thickness.

* The thermal resistance from the saddle to the outside of the outer tube is derived from FLHT-1 data.

The heat flow can then be expressed as

$$Q = C\,(T_1 - T_0)$$

where $C = \frac{1}{R_T}$

R_T = sum of resistance values in series

or $$Q = \frac{1}{R_I + R_S}\,(T_1 - T_0)$$

where Q = radial heat flow

R_I = thermal resistance of the insulation

R_S = resistance from the saddle to the coolant

T_I = inner insulation surface temperature.

The heat flow was then calculated assuming that through any means whatsoever the inner surface of the insulation was maintained at 2760°C (5000°F). The calculation was repeated for various insulation thicknesses. The results are plotted in Figure 21, which shows a heat flow of approximately 17.1 kW/m (5.2 kW/ft) with the full thickness of insulation and an increasing heat flow as the insulation is removed until a maximum of 204 kW/m (62 kW/ft) is calculated for no insulation remaining. The heat flux from the outer shroud surface, even for the case of no insulation, is not large—only 697 kW/m^2 (0.221×10^6 $Btu/h\text{-}ft^2$).

There is significant conservatism in the calculation leading to Figure 21. The calculations assume that the inner insulation surface remains at 2760°C (5000°F) and does so long enough to reach steady state; in any real case, it would not. For significant insulation losses, the large heat loss would cause the temperatures of the material giving the 2760°C to decrease. Furthermore, the time frame of the FLHT-2 test is short and the shroud acts as a transient heat sink. That is, for the short term, heat entering the shroud through the inner insulation surface will raise the average temperature of the shroud rather than passing through the shroud to the bypass coolant as was assumed in these calculations.

The saddle and inner Zircaloy tube temperatures were also calculated for the heat fluxes arising from loss of insulation. If 85% of the insulation is gone, the saddle temperature would just reach the proposed saddle trip temperature of 982°C (1800°F) (Figure 22). The inner tube temperature would reach only 325°C (617°F), a temperature at which the inner tube would retain most of its room temperature strength.

In addition to the conservatisms pointed out previously, there is yet another one in these calculations. The thermal resistance from the saddle out to the outer surface was derived from FLHT-1 data. This resistance includes a contact resistance between the saddle and inner tube. Its value can vary, which was evident in the FLHT-1 test. In that test, the value of the resistance when the test assembly to insulation cavity pressure difference was 0.24 MPa (185 psi) was about half its value late in the test, after the liner had breached and the pressure difference had equalized. Early in the test, the pressure load was transmitted to the saddle/inner tube interface, providing a reasonable contact resistance. After the loading was relieved, the contact resistance increased. The early lower contact resistance value was used in these calculations. If the larger resistance had been used, the calculated heat flows would have been slightly less. More important, however, the temperature difference between the saddle and the inner tube would have been much greater (perhaps twice as large), and the sensitivity of the saddle temperature trips would greatly increase. A saddle trip would occur at a 70% insulation loss and at an inner tube temperature of about 220^{o}C (428^{o}F).

These discussions indicate that there are large margins to unsafe shroud temperatures, even based on the inner tube. The outside Zircaloy tube will be even cooler and will provide another stronger barrier.

It must be emphasized that safety of the test requires that the tubes of the shroud MMPD, particularly the outer one, be maintained cooled. The cooling is controlled by the flow rate and inlet temperature of the bypass coolant and by the radial heat flow. It is independent of what is inside the test assembly, whether it be nominal operation, debris bed, loss of bundle coolant, loss of insulation, or anything else, except as how these phenomena affect the radial heat flow. Adequate cooling can be easily maintained by providing sufficiently subcooled inlet temperatures and sufficiently high flow rates for the bypass. This ability to provide these is easily within the capabilities of the NRU reactor loop coolant systems.

Preventing damage to the two shroud tubes and the reactor pressure tube caused by high-temperature materials requires bypass coolant. Loss of bypass coolant flow could create a hazardous condition. An analysis of the loss of bypass coolant and other safety concerns is presented in the last section of this report.

The chemical power produced by the Zircaloy/steam reaction can be significant. The calculated power is given in Figure 23, which shows a peak of 26.4 kW. The total power released by the metal/water reaction was calculated to be 826 kcal (. 295 Btu).

The removal of the hydrogen blanketing model from the TRUMP computer code significantly changes the calculated course of the test, particularly late in the test when the temperatures and reaction rates are high. The consequences are illustrated in Figure 12, which shows the peak cladding temperature history which increases at an ever increasing rate. At about 1540°C (2800°F), the rate sharply increases to an excess of 55°C (100°F) per second due to the step change in the Urbanic-Heidrick rate correlation at that temperature.

However, at about 13 min, the peak cladding temperature is 2290°C (4150°F) and starts to turn around. At 13.7 min, the fission power was turned off in the calculation and the balance of the curve of Figure 12 shows the cooling of the fuel as heat flows out to the bypass.

The course of the peak cladding temperature if the termination of the fission power is delayed is illustrated in Figure 24. The temperature goes through a series of oscillations, with successive increasing peaks. This calculational behavior is an artifact of the model used in TRUMP-BD. Each new peak occurs as the liquid level drops into a new node. A better description of the peak cladding temperature would be described by the dotted line in Figure 24 through the peaks.

The test will be terminated when it is determined (by the test director) that bundle temperatures have reached 2200°C. Because no reliable temperature measurements can be made at such high temperatures under the test conditions, the time to stop the test is based on a parametric series of pretest predictions as well as real-time data produced during early portions of the FLHT-2 experiment (when temperature measurements are still reliable).

The results of the pretest predictions will be plotted to show the relationship between the following variables:

* the predicted time for peak temperatures to increase from 1093 to 1371°C (2000 to 2500°F)

* the predicted time for peak temperatures to increase from 1371 to 2200°C (2500 to 4000°F).

Such a relationship is plotted as the upper curve in Figure 25; however, the predictions include the effect of hydrogen blanketing.

The input parameters that were varied were fiberboard thermal resistance, fission power, and power change across the liquid/steam interface. The lower curve is currently based on only three sets of calculations that exclude the hydrogen blanketing model. The lower curve is the result of the accelerated course of the oxidation reaction without any inhibition from hydrogen.

Figure 25 was created to provide operational guidance to determine test termination time. The elapsed time from 1093 to 1371°C would be measured during the test. The appropriate curve on the figure would then be used to determine the further elapsed time that would give a temperature of 2200°F, the desired peak cladding temperature.

Based on the distance between the two curves, a 300°C (540°F) temperature difference is due to the presence or absence of hydrogen blanketing. The change in the relationships shown in Figure 25 from hydrogen blanketing does not arise from changes in peak temperatures, but rather in the timing. This is expected since both blanketing and nonblanketing calculations stop the chemical reaction when steam starvation occurs (when all steam in a node is reacted). The blanketing model slows the reaction prior to steam starvation so that cessation of the reaction due to steam starvation is somewhat delayed. However, the steaming rate at the time of steam starvation is within approximately 5%; hence, the power and temperatures will not be very different.

The same method of determining a time to terminate is still planned for FLHT-2. Parametric calculations will be done to provide a curve such as shown in Figure 25 to define the necessary test time to reach the desired test temperature. The correlation given in the figure eliminates or greatly reduces the influence of operational uncertainties on the time sequence of the test. Such things as uncertainties in test assembly power or insulation resistance will affect both the earlier and later elapsed times and the information of Figure 25 will accommodate these.

The following paragraphs summarize work performed for PNL by Dr. Cronenberg of ESA. Three possible sources of overpressurization resulting from the interaction of hot material with water were analyzed:

* Energetic and extremely rapid thermal interactions between molten fuel rod debris and water, leading to shock pressurization above the reactor pressure tube dynamic pressure limit (called a steam explosion).

* Milder debris/water thermal interactions where slow overpressurization may result as a consequence of an overly constricted off-gassing system, with pressurization beyond the pressure tube static pressure limit (called a steam spike).

* Energetic oxidation of molten Zircaloy by water, which can lead to a shockwave or pressure spiking conditions depending on the configuration of the Zr-melt/water reaction mixture (called an energetic chemical reaction).

Pressure increases from any of these three sources if of sufficient magnitude would result in failure of the shroud outer tubes and the reactor pressure tube. The results of these analyses show that no such damaging pressures are expected.

STEAM EXPLOSION

For each class of interaction, the "necessary" conditions for inducement of debris/coolant interactions are evaluated in the context of the FLHT test conditions and conclusions are drawn.

To quantify debris/coolant interaction energetics, the physical, thermal, and chemical characteristics of the melt debris must be defined. The estimate of the physical quantity should conservatively bound the upper limit of the melt debris inventory that might be expected from consideration of the test sequence. The melt debris characteristics are estimated based on the predicted cladding temperatures for intact geometry. All cladding that exceeds the melting point of alpha-Zr (2000°C) is assumed to be completely molten. Based on solubility and phase diagram considerations, a 25% dissolution (liquefaction) of UO_2 is also assumed to accompany cladding melting. To this melt inventory is added an equivalent length of molten Zircaloy liner (same as axial length of cladding above 2000°C), resulting in a total "corium" melt debris mass of 2850 g with an effective melt temperature of 2127°C and a mass weighted density of 8.1 g/cm^3.

To assess the steam explosion hazard, five conditions considered necessary for inducement of such explosions were evaluated:

* a period of stable film boiling and coarse intermixing of melt debris and coolant
* destabilization of film boiling by thermal- and/or pressure-induced means
* extensive fuel fragmentation and intermixing with liquid coolant, resulting in a large effective heat transfer area for rapid coherent coolant vaporization
* intimate liquid-liquid contact between molten debris and coolant
* sufficient system constraint resulting in shock pressurization.

Evaluation of each of these criteria indicated the following trends relative to the FLHT test conditions. Since the contact interface temperature between corium melt and coolant is estimated to be several times the thermodynamic critical point of water (374°C), the initial film boiling requirement would be satisfied.

Consideration of the energy requirements for vapor film collapse indicates that the kinetic energy associated with debris free-fall impact at the FLHT fuel bundle height conditions would not be sufficient to destabilize the film boiling mode. In the absence of an external trigger, the second condition would not be satisfied for the FLHT test conditions. A similar conclusion is reached concerning the fine-scale fragmentation/intermixing requirement, where estimates of the available thermal energy of the system are less than that required for rapid fine-scale intermixing. Thus, an external trigger would be required to satisfy the requirement for fine-scale intermixing. No such plausible trigger is visualized for the FLHT test conditions.

Thermal and hydrodynamic considerations concerning the liquid-liquid contact requirement indicate that for water pool conditions near saturation, extensive steam formation upon initial corium contact with coolant can be expected. Such steam formation tends to vapor blanket additional corium fuel entry from direct contact with water, thus destroying the potentially explosive configuration of large-scale molten corium/water contact (i.e., liquid-liquid contact). Likewise, system constraint considerations, based on the residual water depth at the bottom of the test assembly, indicate an acoustic relief time of approximately 1 ms, which is about an order of magnitude less than the dwell time for significant thermal energy transfer from fuel to coolant.

Lack of attainment of any one of these five conditions would be sufficient to preclude the possibility for explosive steam formation. The fact that four of five criteria are not satisfied for the FLHT test conditions is indicative of a nil probability for inducement of an energetic steam explosion.

STEAM PRESSURE PULSES

Thermal Reaction

A steam spike involves vapor generation on a time scale that is considerably longer than that for explosive vaporization. Consequently, no shock wave is developed. However, depending on steam venting characteristics, slow overpressurization can occur. If steam production should overwhelm the venting capacity of the off-gassing system, a choked flow situation could result in escalating pressurization of the test loop, a condition that should be avoided.

When considering the steaming process due to quenching of corium melt in water, the principal features of interest are the mass and temperature of the melt material, the mass of water, the available condensation heat sink, and the steam relief paths within the test loop. This analysis conservatively neglects potential steam condensation effects. All steam and hydrogen (due to steam/Zircaloy reaction) effluent is vented through the ECM filter line or its bypass, in conjunction with the pressure relief line actuated at a differential set pressure of 0.17 MPa (25 psi). For the filter and bypass lines the off-gassing capacity is dictated by similar isolation valves with an orifice opening of 7.9 mm (0.312 in.), while the equivalent flow diameter of the pressure relief valve is 9.5 mm (0.374 in.).

Calculational results indicate that the gas venting capacity of the FLHT effluent control system, at a differential driving pressure of 0.17 MPa (25 psi), exceeds the estimated steaming rate for debris/coolant interaction by a factor of about 50. It is also noted that to reach the estimated steam choked-flow condition of the FLHT effluent control system [i.e., P = 2.34 MPa (340 psi), m = 422 g/s], the steam production rate would have to reach a level almost 135 times the maximum steaming rate predicted for debris/coolant interaction, a situation that is difficult to envision. It is therefore concluded that steam spike pressures in excess of 0.17 MPa (25 psi) above system pressure (i.e., the set pressure of the pressure relief valve) are realistically impossible.

Chemical Reaction

Experiments were conducted in the mid-1950s and early-1960s demonstrating that finely divided molten zirconium can react explosively with water. In all cases, explosive reactions occurred only when the melt was forcibly fragmented into a fine particulate state, with mean particle diameters less than 500 microns. Based on such experimental evidence, the following criteria are considered necessary for initialing violent Zr(melt)/H_2O chemical reactions:

* The Zr-melt temperature (2000°C) must be exceeded.

* The Zr-melt must be in a relatively unoxidized state at the time of quenching.

* Fine-scale Zr-melt fragmentation and intermixing with coolant must occur during quenching such that a large effective surface area is available for rapid oxidation.

* The oxidation of the Zr-melt inventory must be rapid and occur in a constrained condition such that significant overpressurization results.

The first two criteria would be satisfied for FLHT test conditions. However, with respect to the criterion of rapid fine-scale fragmentation and intermixing, calculations indicate that the energy for rapid intermixing is so large that it precludes a self-triggered interaction process. No plausible external trigger could be visualized that would satisfy the energy requirements for fine-scale intermixing.

With respect to the necessity for rapid/simultaneous reaction in a constrained condition, calculational results indicate only limited penetration of molten Zr into the water before system expansion begins. The maximum reaction pressure was calculated to be 11.7 MPa (1700 psi), which is below the failure limits of both the outer shroud wall and the reactor pressure tube. Thus, energetic reactions leading to system failure are not indicated.

The Zircaloy tubes have a 0.2% offset elastic yield strength of 400 MPa (58,000 psi) or yield at an internal pressure of 12.0 (1747) and 11.3 MPa (1633 psi) for the inner and outer shroud tubes, respectively. The combination of inner tube, MMPD wire wrap, and outer tube internal yield pressure limit is about 25.5 MPa (3700 psi). The expected ultimate tube strength is about 20% greater than the yield strength.

It should be noted that several conservative assumptions were made, which lead to higher-than-expected calculated pressures. These assumptions include 100% thermal-to-mechanical energy conversion, completely unoxidized Zr-melt available for reaction, and the absence of both hydrogen and steam blanketing effects. In reality, less than 100% of the reaction energy is available for slug explosion. Likewise, metallic Zircaloy, free from both dissolved UO_2 and oxygen, would not be expected. Rather, a molten corium mixture where the (U, Zr)-oxide phase is much less reactive than the pure Zr-melt would be expected. Molten corium quenching in near-saturated water can also be expected to result in a film boiling condition, which tends to separate the melt from direct contact with coolant. The resultant protective steam/hydrogen layer can be expected to inhibit ready access of the Zr-melt to water. For these reasons, rapid/coherent Zr-melt oxidation in a constrained liquid-liquid contact model is not predicted, but rather limited oxidation in a separated state.

Analysis of known governing phenomena indicate that the necessary conditions for inducement of energetic thermal and chemical interactions are not satisfied based on credible estimates of the FLHT test conditions. It is therefore concluded that the energetic melt debris/water interaction does not pose a threat to system integrity.

UNCERTAINTIES ASSOCIATED WITH NORMAL OPERATION

The actual fission product inventory as a function of time depends on the specific FLHT-2 bundle power history. The uncertainty is much less than

±100%; but, if the actual inventory were twice the inventory used in the hazards analysis, no serious unsafe conditions would result. The actual inventory could be greater (even greater than the 6- to 7-h FLHT-1 test) and the hazards associated with radioactive fission products would not increase significantly.

The predictions of radiation fields near reactor/test personnel are uncertain for FLHT-2 as they were for FLHT-1. Actual fields during and after the FLHT-1 test were very low (near background); however, depending on assumptions, calculated fields were as high as 2500 R/h.

An example of the calculated radiation field as a result of fission product deposition on the bottom of the closure plug as a function of time after the test is shown in Figure 26. The following assumptions were made:

* 8-1/2 h of operation at 27-kW bundle power

* the following fission product release fractions from the fuel

Fission Product Group	Release Fraction
I	0.3551
Cs, Rb	0.3567
Te	0.0199
Ag	0.1281
Sb	0.0319
Ba	0.0127
Mo	0.0158
Sr	0.0019
Zr	0.0000
Ru	0.0002
Fuel	0.0000
Cladding (Zr)	0.0000
Cladding (Sn)	0.0319
Structural	0.0011

* 50% of the released fission products deposited uniformly on the bottom of the closure plug.

The accuracy of reactivity calculations for tests in the NRU Reactor for a fixed reactor loading is probably within ±10%. Some uncertainty in the calculations occurs because the loading is not known until very near the test date. Greater uncertainty is associated with the configuration of the fuel bundle once fuel movement is possible. Minor movement is possible during the cladding ballooning and rupture portion of the test; but based on the results of past tests (MT series and FLHT-1) essentially no axial fuel movement occurs during this phase of the test.

The potential for fuel movement is greatest once bundle temperatures are high enough to melt the cladding. If the molten cladding should flow away from the pellets, the pellet columns could "collapse" into a pile of rubble. If the molten Zircaloy flows down the pellet surfaces, UO_2 is dissolved, thus relocating into coherent debris. The amount of dissolution, the rate of downward flow, and the location, size, and shape of the solidified debris are also uncertain, thus introducing uncertainties into the reactivity calculations. For this safety analysis, conservative values were assumed to maximize the possible reactivity effects. The results are quite acceptable from a reactor safety viewpoint.

The predictions of the hydrogen generation rate, the hydrogen flow rate, and the total hydrogen production are based on the following assumptions:

* the Zircaloy temperatures are known

* the Zircaloy oxidation rates as a function of time and temperature are known

* the local hydrogen concentration is a function only of the hydrogen generation rate (no hydrogen concentration occurs in the effluent system).

To assure that combustible mixtures of hydrogen do not occur during and right after the test, the nitrogen dilution rate was set based on the predicted peak hydrogen generation rate.

During the first several minutes of the test, the nitrogen dilution gas will be sweeping air out of the gas flow path. Near the end of the test (last few minutes), the gas mixture will essentially consist of nitrogen. During the test period when the peak hydrogen generation occurs, the gas fixture downstream the ECM will be diluted to less than 4% hydrogen. This mixture is not flammable regardless of the amount of air in the system. No such air leakage is expected.

Many factors contribute to the uncertainty associated with predicting FLHT-2 temperatures. In determining the temperature distribution within the bundle and shroud regions, the greatest uncertainty lies in predicting the local peak cladding temperatures of the test fuel rods, due primarily to the uncertainty in predicting the chemical power and the combined convective and radiative heat transfer from the oxidizing cladding. Much less uncertainty is associated with predicting temperatures in the exterior of the shroud, specifically those of the inner and outer Zircaloy tubes due to the low temperatures, the absence of the metal-water reaction, and the conduction-only heat transfer. Estimates of the uncertainty in the predicted inner and outer tube temperatures are $\pm 50^{\circ}C$ for FLHT-2. The uncertainty in the temperatures is due primarily to the uncertainty in the MMPD thermal resistance; as-measured values of this resistance were determined from the FLHT-1 tests and is therefore well defined. The safety of the test depends on maintaining the tubes at low temperature, and these temperatures can be accurately predicted. Thus, the uncertainty associated with the bundle temperature predictions is not a safety issue, provided adequate margins exist for containing hydrogen and fission products.

Thermal analysis based on the TRUMP code has uncertainties due to code limitations like node size, material properties, no material movement, and modeling local steam starvation.

There is some uncertainty in determining the time when the peak bundle temperature has reached $2200^{\circ}C$. The approach presented in an earlier section that uses the time for temperatures to increase from 1093 to $1371^{\circ}C$ as a basis to estimate the time for temperature to increase from 1371 to $2200^{\circ}C$ was chosen to minimize the uncertainties of test assembly power, hydrogen blanketing, and fiberboard insulation resistance.

It is not possible to quantify the effect of these uncertainties on temperature ramp rates; however, several observations can be made. First, complete elimination of hydrogen blanketing provides the fastest ramp rates. If some sort of a partial or localized hydrogen blanketing should occur, the temperature ramps late in the test will be slower and the method of defining time of test termination will result in failing to achieve the goal temperature. Candling or draining of melted Zircaloy will start at the peak temperature elevation. As the Zircaloy is removed, the reaction obviously stops at that elevation with the result that the reaction is moved to a lower cooler elevation and the peak cladding temperature at any given time will not change greatly, only its location.

There are several important observations concerning the consequences of removal of the hydrogen blanketing model from the calculations. First, and most important, the FLHT-1 results only partially support the decision to remove the hydrogen blanketing model. The removal explains the higher-than-expected temperature ramp rates noted early in the test before the TC behavior became erratic. However, there were two TCs whose behavior indicated they did not fail during the course of the transient. These were two of the liner TCs very near the peak axial temperature elevation. These both showed rapid temperature ramps just after the runaway excursion started. However, both leveled off shortly thereafter, one at a temperature about 1820^{o}C (3300^{o}F) and the other at 1980^{o}C (3600^{o}F). Both remained fairly constant at these values for about 50 s until the reactor power reduction, after which they both decreased. The fairly constant temperature level for the 50-s period is not consistent with the expected metal/water reaction. Visual observation of the test assembly supports the peak temperature levels deduced from conclusions that the temperature ramps did flatten.

The reasons for the lack of conformance of FLHT-1 to the behavior expected if metal/water reaction proceeded with no hydrogen blanketing is speculative.

It is possible the hydrogen blanketing model should not be completely eliminated. There may be local steam saturation that terminates the reaction on a local basis. There may be heat removal mechanisms that are not modeled in the calculation, perhaps those that might result from the liner breach; or too much credence may be placed on the behavior of the two TCs in question. However, regardless of the source of the discrepancy, the results covered in this discussion and those presented for the PSAR will bracket the course of the test.

Quantification of the effects of these uncertainties is not possible. In fact, the test is being performed in part to supply this information. The arguments presented in this section point out that the effects are not large. They should be less than the 300^{o}C (540^{o}F) effect of removing the hydrogen blanketing. In any case, the discussion concerning bypass cooling shows that the shroud tubes more than adequately protect against these concerns.

An argument presented earlier leads to the conclusion that a trip based on saddle temperatures gives a huge conservatism in protection from the hazards of failure of the shroud to contain the high temperature and radioactive products of the test. There is additional conservatism in this conclusion arising from the use of a thermal resistance of the saddle to outside shroud deduced from the earlier stages of FLHT-1. During this time, the mechanical loading (from the inside test assembly to shroud cavity pressure difference) could be transmitted to the saddle and cause good saddle to inner tube contact. Later in the test, after the liner has breached, the equalization of test assembly and shroud cavity pressures would relieve the mechanical loading. This, in turn, could cause an increase in the thermal resistance.

FLHT-1 data were examined to assess the change in thermal contact resistance between saddles and the inner tube of the MMPD. A lower confidence is placed on the thermal contact information in the later stages of the test because of failure of many of the TCs and because of the linearizing effects of the steam injection. However, the data indicated that the thermal resistance may have doubled after the liner breached.

Doubling the thermal contact resistance between the saddles and the shroud inner tube increases the sensitivity of the saddle temperature trips. For example, if the saddle trip temperature were 980°C (1800°F), the arguments presented earlier show that a four-fold increase in the radial heat flow would cause a reactor trip and would do so after a negligible rise in the outside shroud temperature. Doubling the thermal contact resistance would mean that half of the radial heat flow would cause a reactor trip for a 980°C (1800°F) saddle trip temperature. This raises more questions about unwarranted reactor trips than it does about reactor safety.

The discussion thus far is concerned with saddle trip temperatures with respect to MMPD tube temperatures. It is concluded that an increase in the radial heat flow by a factor of two to four will cause the saddle TCs to reach their trip temperatures but that the outer tube temperature would be no more than 64°C (148°F) and the inner tube temperature would be no more than 390°C (735°F). These temperatures are the results of increased radial heat flow rates and are independent of the cause of the increase.

A brief comparison between Urbanic and Prater Zircaloy oxidation rates was made by Prater (PNL); his remarks made July 22, 1985, are as follows.

Recent oxidation studies at PNL suggest that Urbanic's kinetics data on Zircaloy oxidation may be low. Comparison of the Urbanic data with the most recent PNL and KfK data on oxide thickness clearly suggest that Urbanic is underestimating the ZrO_2 layer thickness. However, it is not clear whether Urbanic's weight gain measurements, which were used in the model calculations for FLHT, are also in error. At 1600°C, there is good correlation between the Urbanic and KfK data—PNL currently has no weight gain data for comparison. The discrepancy in ZrO_2 thickness may have to do with averaging techniques. Urbanic may have underestimated the variations in thickness that occur due to hot spots that develop in induction heated samples.

The best oxidation data set is believed to be that recently obtained at PNL; the temperature measurements are the most reliable. The only uncertainty is due to the unknown distribution of the heat of oxidation within the sample.

This could result in the measured temperatures at the exterior surface of the oxide to be lower than at the interior where most of the oxidation heat is evolved. This error could potentially overestimate the oxidation kinetics at a reported temperature. Urbanic's data should be considered a lower limit. The limited amount of other data (from KfK, Baker-Just, and Lemmon and Bostrom) fall between these two data sets. Thus, calculations based on the Urbanic rate constants are probably underestimating the oxidation kinetics. At 1600°C, the error in oxide layer thickness if fairly small: PNL is a factor 1.4 higher than Urbanic. However, extrapolating the Urbanic data to 2200°C results in rates that are a factor of 3.6 too low. Assuming the worst, that the errors in oxide thickness are also reflected in the weight gain data that are used in the computer model, similar errors could be potentially present in the code calculations.

Such an error would accelerate the rate of temperature rise during the early stages of runaway oxidation. However, the overall effect both on maximum temperature and hydrogen production would not be significant since steam supply would still limit the thermal runaway. Roughly the same transient would be followed, the major difference being that it would be translated to slightly earlier times.

SAFETY CONCERNS DURING ACCIDENTS

Several specific unexpected, potentially unsafe conditions could occur during the FLHT-2 experiment. These unplanned events raise safety concerns mainly about test hardware integrity and performance should any of these events occur. The following areas are addressed in this section:

* loss of bypass coolant flow
* loss of bundle pressure
* ECM pipe rupture
* loss of all power.

LOSS OF BYPASS COOLANT FLOW

The integrity of the two Zircaloy tubes of the shroud must be maintained to retain the hazardous products of the test. This requirement is easily achieved if the tubes remain cool. A continuous flow of bypass coolant is therefore essential for the safe operation of the FLHT-2 test. High and low bypass coolant flow reactor trips are provided to protect against a loss of coolant flow. The information given earlier confirms a large flow safety margin. A reactor trip will be set at a flow of about 75% of normal. The results of calculations show that a flow reduction to 4% of normal would be necessary to cause bulk boiling of the bypass coolant, which in itself is a conservative definition of a hazardous condition.

Even though the reactor trips automatically on a high/low flow signal, it is important that at least stagnant water be maintained in the reactor pressure tube. Calculations indicate that vaporization of only about 7 kg (15 lb) of stagnant bypass coolant would remove the stored energy in the fuel rods, the insulation, and the shroud components if the loss of bypass flow occurred when the peak temperature was 2200°C (4000°F).

A question has been raised concerning the need to remove additional heat above that of the stored energy if debris should fall into the water pool at the time of the trip. Calculations were therefore made starting with a debris defined by Cronenberg. A mass of coherent (molten) debris of 2844 g (6.3 lb) with a temperature of 2130°C (3861°F) and a heat capacity of 394 J/kg-K (0.094 Btu/lb-°F) could fall into the pool.

The assumption was made that all the energy of the debris vaporized water from the pool. Further conservative assumptions were made that all of the steam reacted with Zircaloy and all of the energy released passed into the bypass. The calculations show that 0.9 kg (1.98 lb) of water would vaporize from the pool, and it would generate 14.6 MJ (13,890 Btu) when reacted with Zircaloy. Putting this much heat into the bypass annulus would vaporize an additional 6 kg (13 lb) of coolant to give a total of 13 kg (29 lb). Because there is about 20 kg (44 lb) of water in the bypass annulus above the test assembly, there is about 50% more available than is necessary to remove even these conservatively calculated heat sources.

LOSS OF BUNDLE PRESSURE

The consequences of a loss of fuel bundle pressure would depend on the cause of the pressure loss. If such a loss occurred, the reactor would automatically trip on low pressure in the ECM tubing. Should the loss of pressure be due to a failure of the bundle inlet tubing, the small amount of coolant water in the bundle would be discharged into the upper service space or ECM secondary containment system and be replaced by nitrogen from the ECM. The test fuel would then be slowly cooled to the bypass coolant system temperature. Any escaping fission products would be drawn into the ECM. The charcoal absorber in the ECM ventilation system would retain at least 99% of the radioiodines, resulting in only a small release to the reactor stack.

If a loss of pressure were due to a failure of the ECM control system, reduction of the system pressure from 1.38 to 0.1 MPa (200 to 14.7 psia) would result in approximately 18% of the water below the liquid-steam interface flashing to steam. The time taken to flash would depend on the nature of the failure and the time taken to reach atmospheric pressure. The increased steam production could conceivably increase the Zircaloy-steam reaction temporarily. Calculations were performed to provide information concerning the possible magnitude of the increased chemical reaction.

First, if the loss of pressure were sudden, the steam in the test assembly would undergo a nearly isentropic expansion. The temperature would drop by about 350°C (660°F), which would inhibit the degree of the Zircaloy/steam reaction. Moreover, a sudden drop in the test assembly pressure implies removal of most of the gaseous contents. In other words, in this case the steam flashed from the pool would almost instantaneously be swept from the test assembly and would not react.

However, if the conservative assumption is made that all the steam reacts and it occurred at the end of the test when temperatures were highest, approximately 0.24 kg (0.5 lb) of steam would flash. Its reaction would release 3.9 MJ (3718 Btu). Instantaneously placing this much heat into the assembly above the water level would raise the average assembly temperature by 125°C (280°F).

ECM PIPING RUPTURE

Fission products that are released from ruptured FLHT-2 fuel rods may flow through the effluent piping to the ECM from the test train head closure. A postulated rupture of that pipe could release fission products to the reactor building.

Rupture of the effluent pipe is extremely unlikely. The high-pressure piping that connects the ECM to the test train head closure and other ECM components is designed for a pressure rating of 2.41 MPa (350 psia), which is 75% greater than the planned operating pressure of 1.28 MPa (185 psig). In addition, the ECM is anchored to the NRU deck with seismic anchors designed to withstand a 0.25-g horizontal load, precluding rupture of outlet piping due to unplanned ECM movement. The piping is also protected by 10-cm (4-in.) thick lead shielding and 3.2-mm (0.125-in.) thick sheet steel, which provide protection from a seismic event and falling objects. Further, the piping will be leak tested during commissioning. No credible failure mechanism is foreseen.

To provide an additional safety measure for the postulated accidental release of the fission products contained in the ECM and its piping, a container was fabricated of 3.2-mm (0.125-in.) thick sheet steel to prevent any leakage and to direct liquid, gas, or vapor to the NRU radioactive waste disposal systems. A double-ended rupture of the effluent pipe was analyzed to estimate the maximum discharge flow rate.[a] About 86 g/s (0.19 lbm/s) could be released from the ECM, and about 149 g/s (0.33 lbm/s) could be released from the test train outlet. Both flow rates decrease as the system depressurizes through the rupture pipe and the NRU reactor would be tripped by the effluent low-pressure sensor. The reactor building ventilation and exhaust system is designed to process any accidentally released vapor or gas from the ECM and upper service space through the activated charcoal filter system and dispose of it through the plant stack. Any leaking liquids would be piped directly to the loop catch tanks.

LOSS OF ALL POWER

During the course of the FLHT-2 experiment, it is possible that all offsite power to the NRU facility could be lost. The worst possible time for that to occur would be at the completion of the transient when fuel cladding temperatures are about 2200°C (4000°F). The NRU Reactor would be tripped by the loss of power, but the energy stored in the fuel bundle and test train would have to be dissipated without endangering NRU operating limits.

The analysis[b] of this accident assumed that a 95°C (200°F) overshoot resulted after the NRU Reactor was tripped and that a temperature distribution

(a) Memo: G. M. Hesson to F. E. Panisko. April 24, 1984. "Approximate Discharge Rates from Broken Inlet Pipe to ECM for FLHT-1."

(b) Memo: G. M. Hesson to F. E. Panisko. May 10, 1984. "Boiloff of Water from FLHT Bypass After Loss of All Coolant."

with a peak at 1980°C (3600°F) was used to initiate the calculations. The enthalpy was calculated for each of 11 axial segments using the saturation temperature--194°C (382°F)--of the bypass coolant as the base. The total stored energy included contributions from the fuel bundle and shroud. Dividing the total stored energy by the latent heat of vaporization gives an estimate of the water mass that must be boiled away to remove the stored energy. Less than 5 kg (11.3 lbm) of water is required, which is about 25% of that in the bypass region above the fueled core region. The stored energy of a core debris bed was also estimated to be about 10% greater than that of a structured fuel bundle. However, much more water is available around and above the test train than is needed to dissipate the stored energy.

Whether the fuel rod bundle is structured or a debris bed, the maximum stored energy in the FLHT-2 test assembly will be easily converted into a small fraction of the bypass coolant vaporization. The temperature of the shroud exterior and the pressure tube will also remain below the bypass coolant saturation temperature. Loss of all power is not a significant safety hazard.

FIGURES

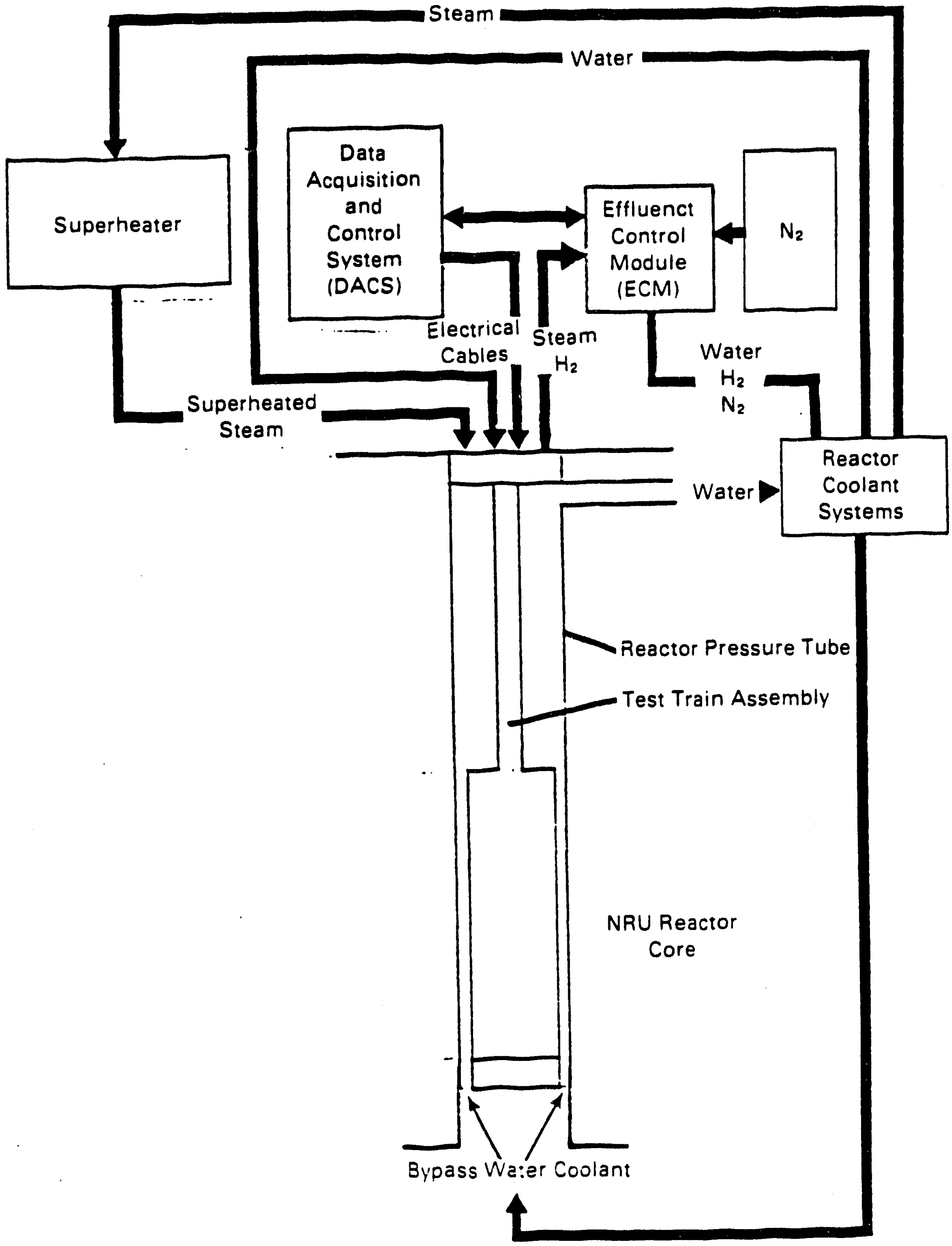


FIGURE 1. FLHT-2 Experiment Hardware Arrangement

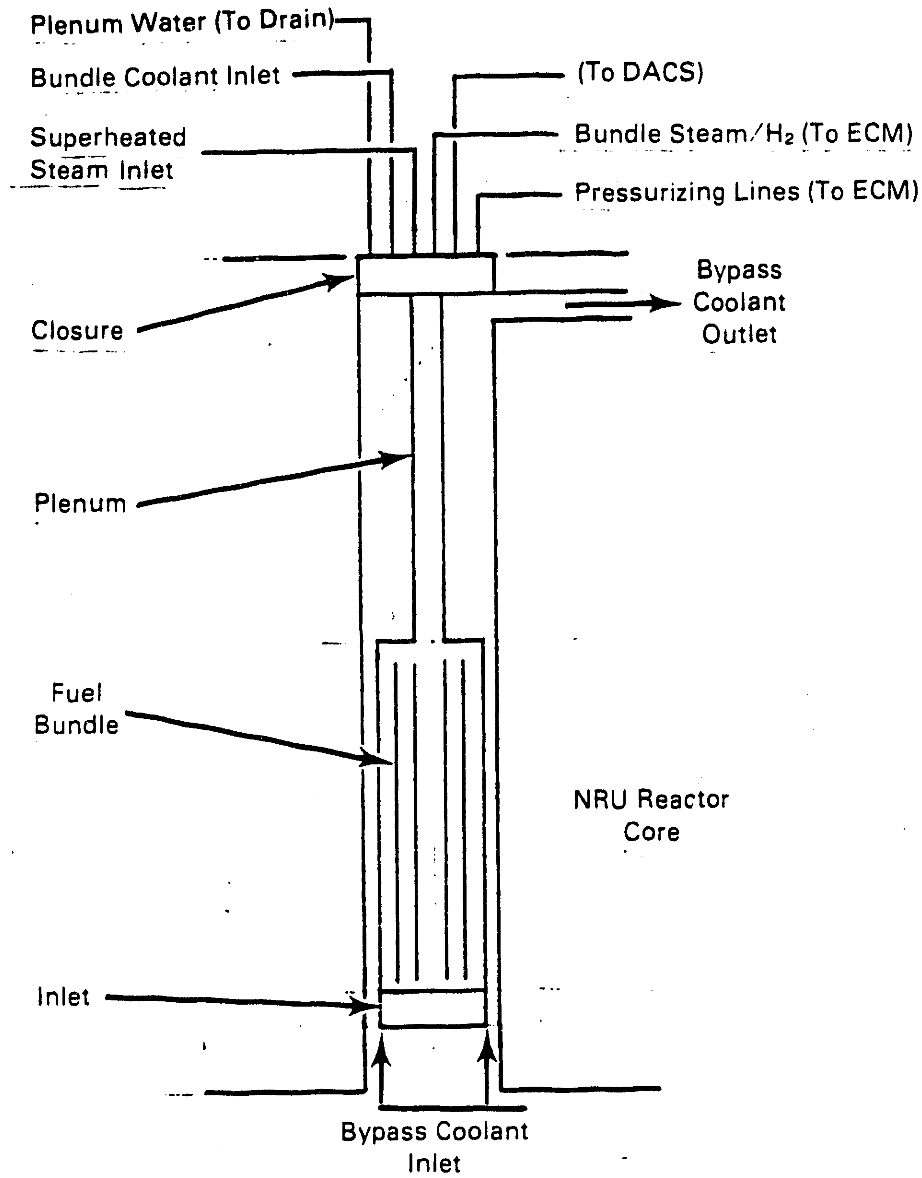


FIGURE 2. FLHT-2 Test Train Assembly

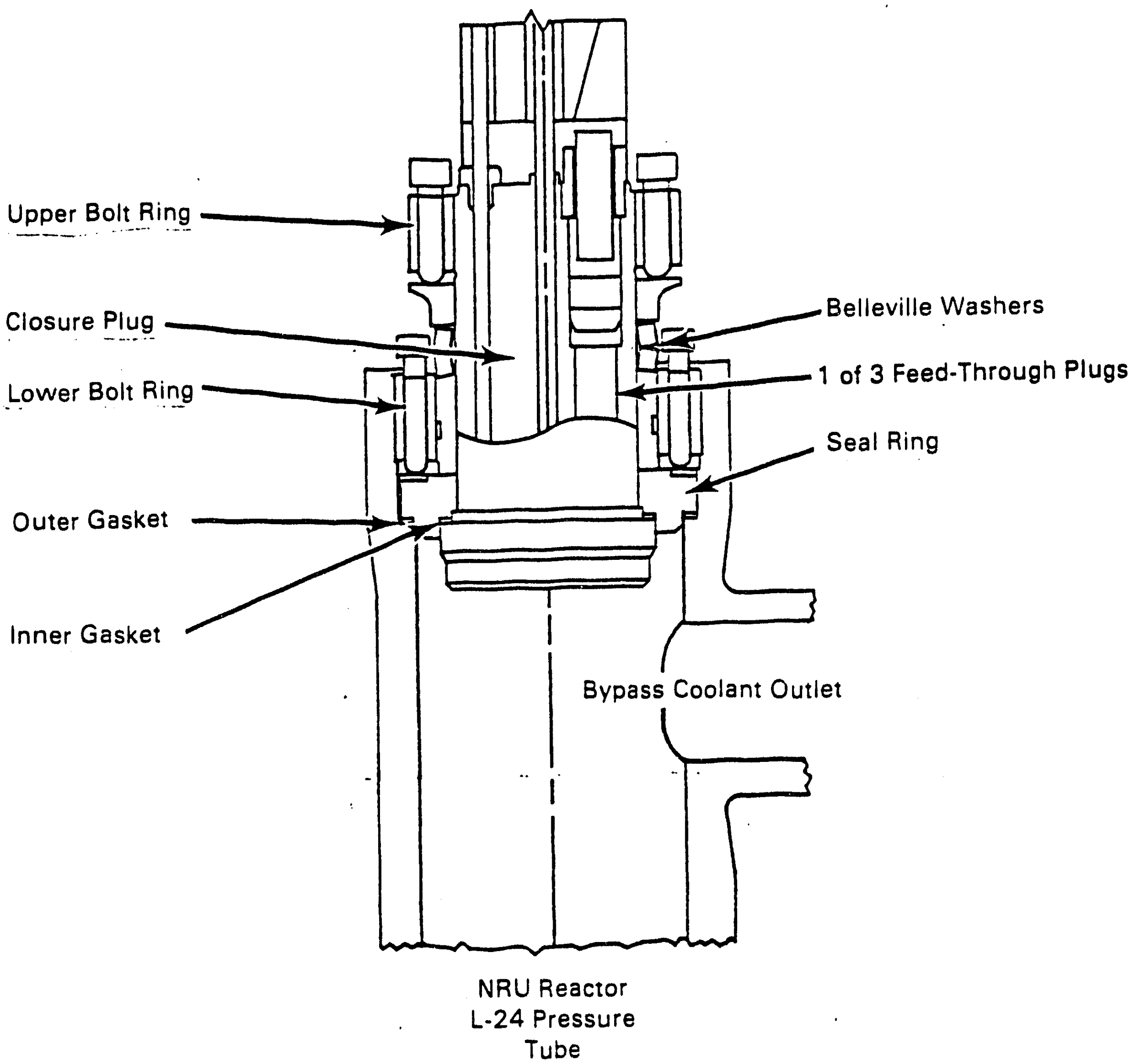


FIGURE 3. FLHT-2 Test Train Closure Region

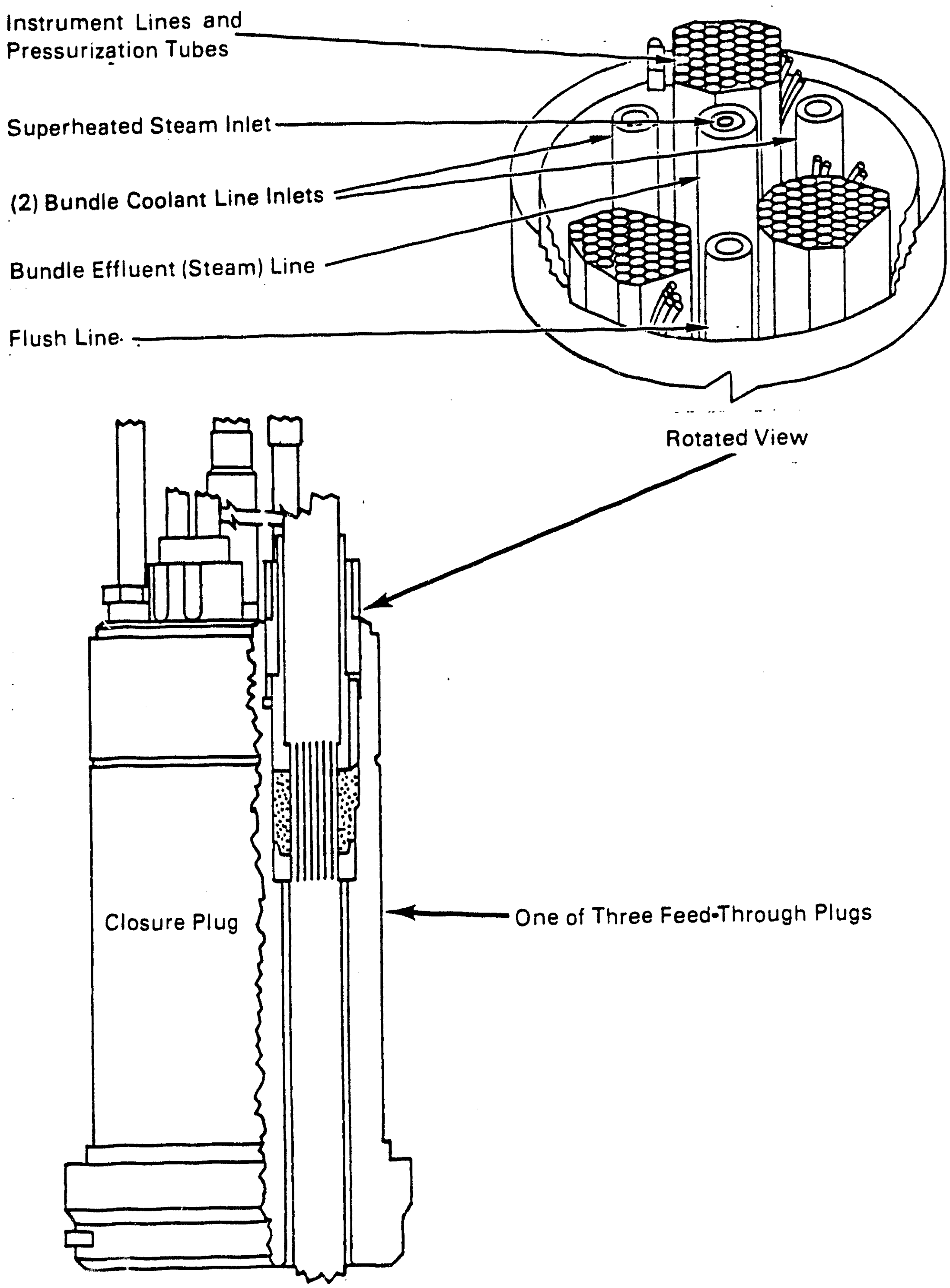


FIGURE 4. FLHT-2 Closure Plug Penetrations

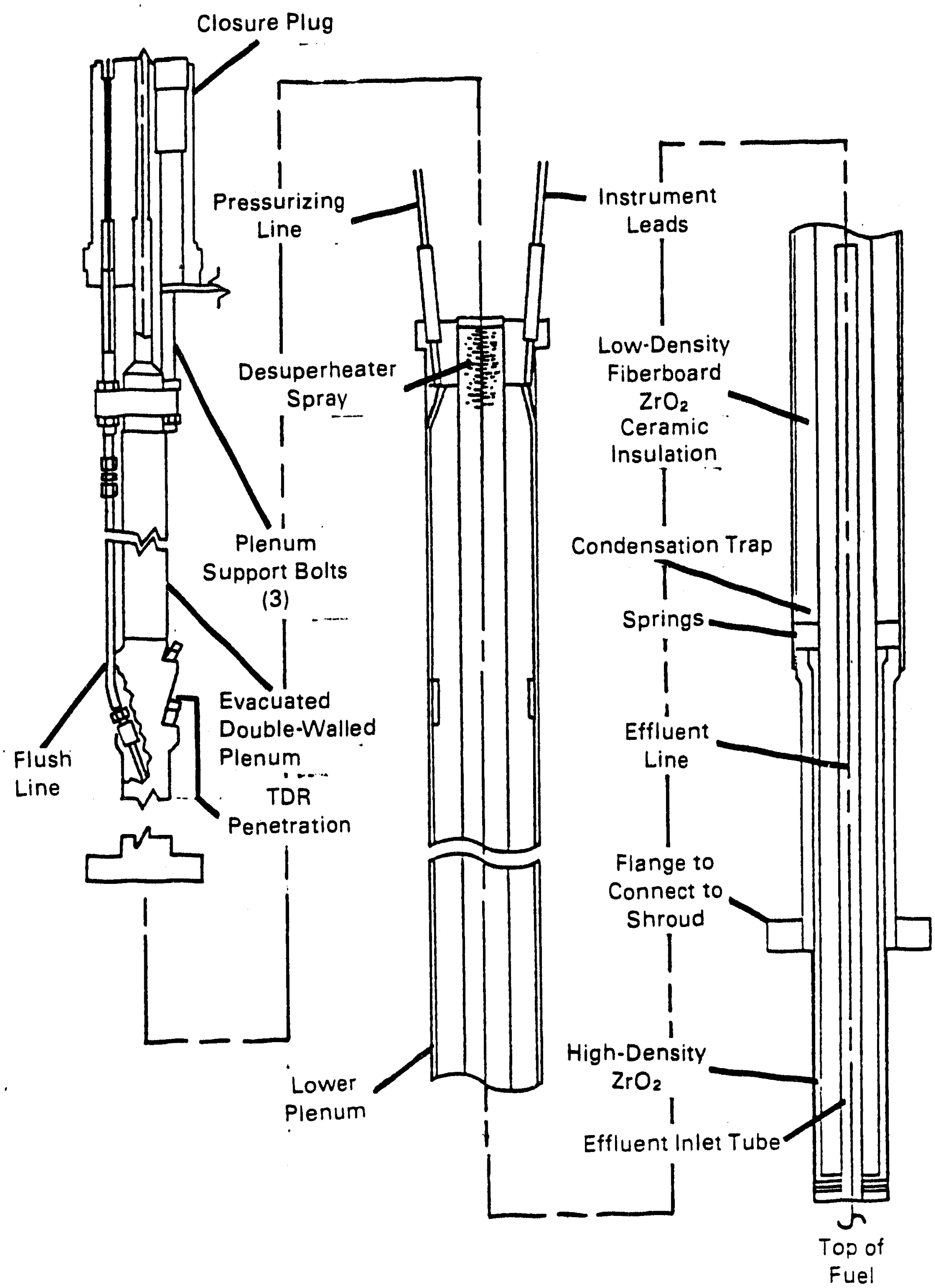
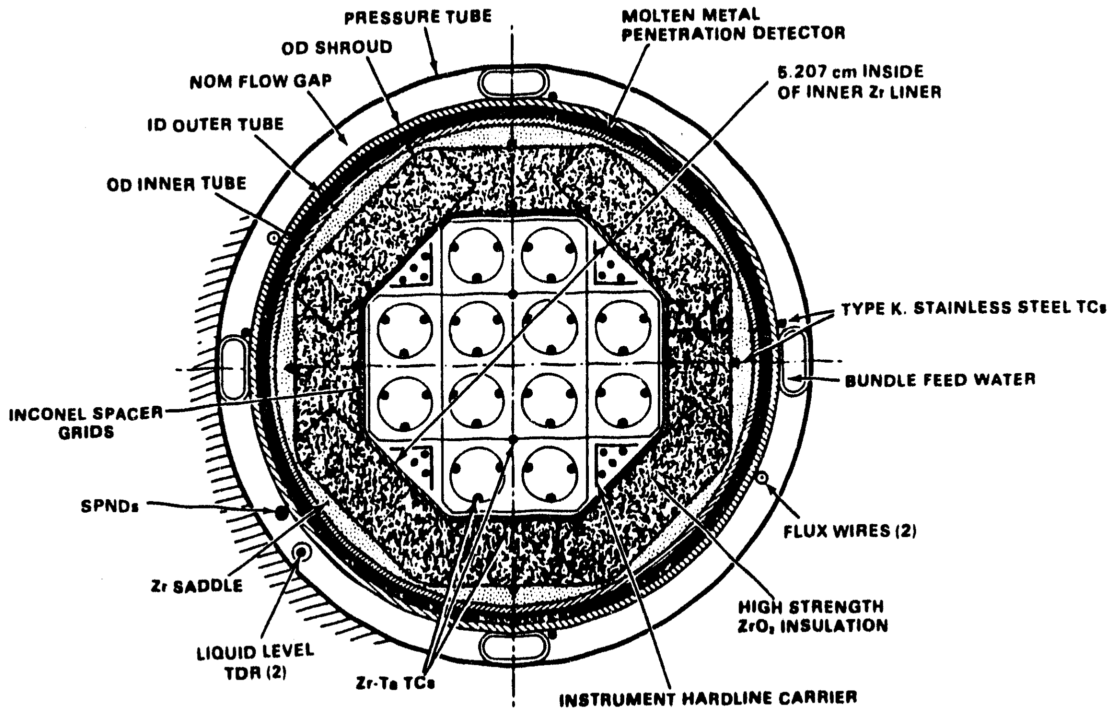


FIGURE 5. FLHT-2 Test Train Plenum Region



Number of Fuel Rods	12
Enrichment	2% $^{235}U/U$
Fill Gas Pressure	To be Determined
Rod Length	156.665 in. (397.929 cm)
Rod Outside Diameter	0.379 in. (0.963 cm)
Rod Inside Diameter	0.331 in. (0.841 cm)
Wall Thickness	0.024 in. (0.061 cm)
Pitch	0.502 in. (1.275 cm)
Pellet Diameter	0.325 in. (0.826 cm)
Gap Diameter	0.006 in.
Active Length	142.875 in. (362.902 cm)
Total Shroud Length	171.5 in. (435.6 cm)
Shroud Outside Diameter	3.650 in. (9.271 cm)
Shroud Wall	0.050 in. (0.127 cm)
Inner Tube Outside Diameter	3.430 in. (8.458 cm)
Liner Flat to Flat Outside	2.160 in. (5.484 cm)
Liner Wall	0.030 in. (0.076 cm)
Zircaloy-2 Pressure Tube Inside Diameter	4.070 in. (10.338 cm)
Zircaloy-2 Pressure Tube Wall	0.200 in. (0.508 cm)

FIGURE 6. Cross Section of FLHT-2 Test Train Reactor Core Region

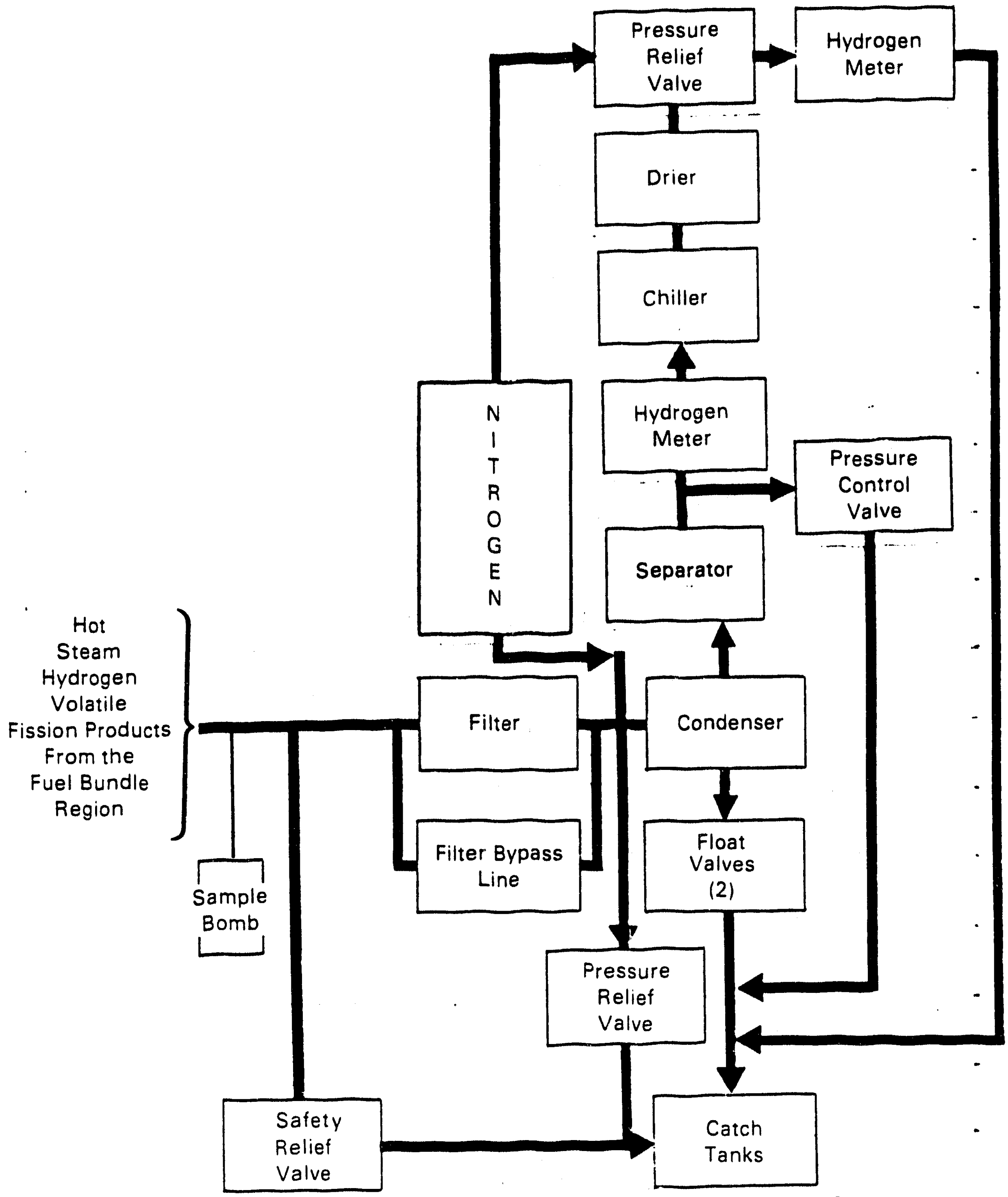


FIGURE 7. Flow Paths in the Effluent Control Module

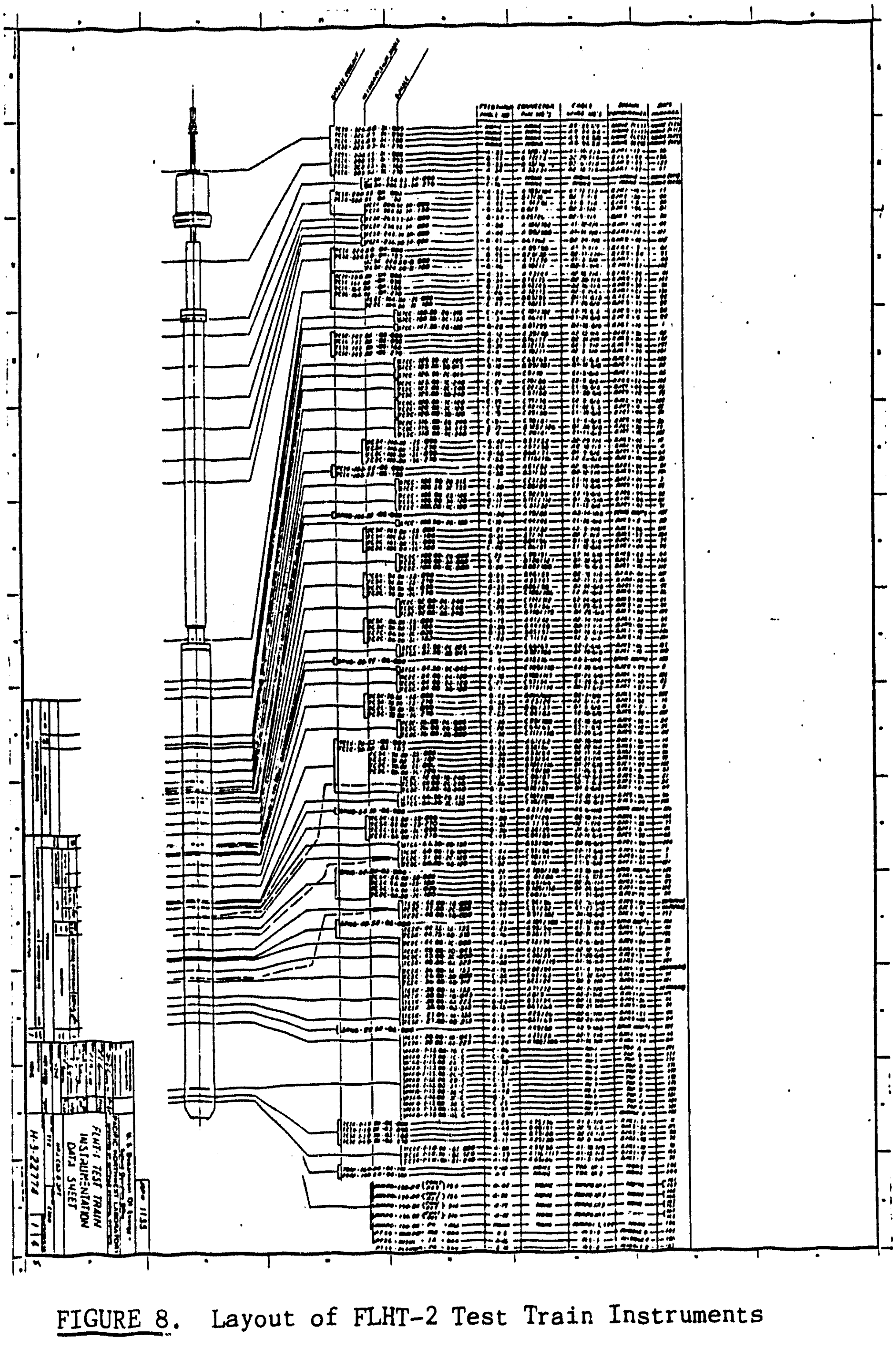


FIGURE 8. Layout of FLHT-2 Test Train Instruments

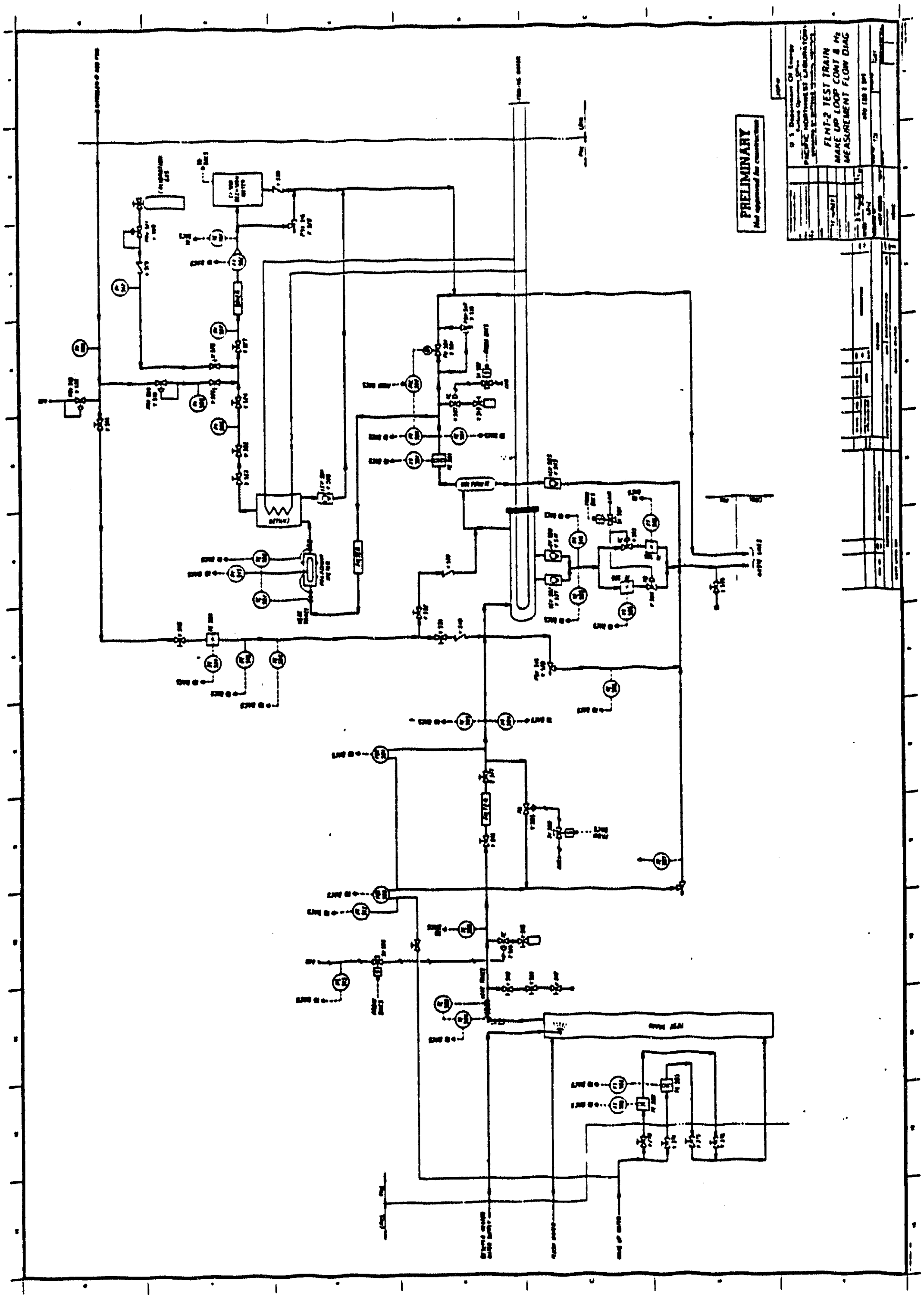
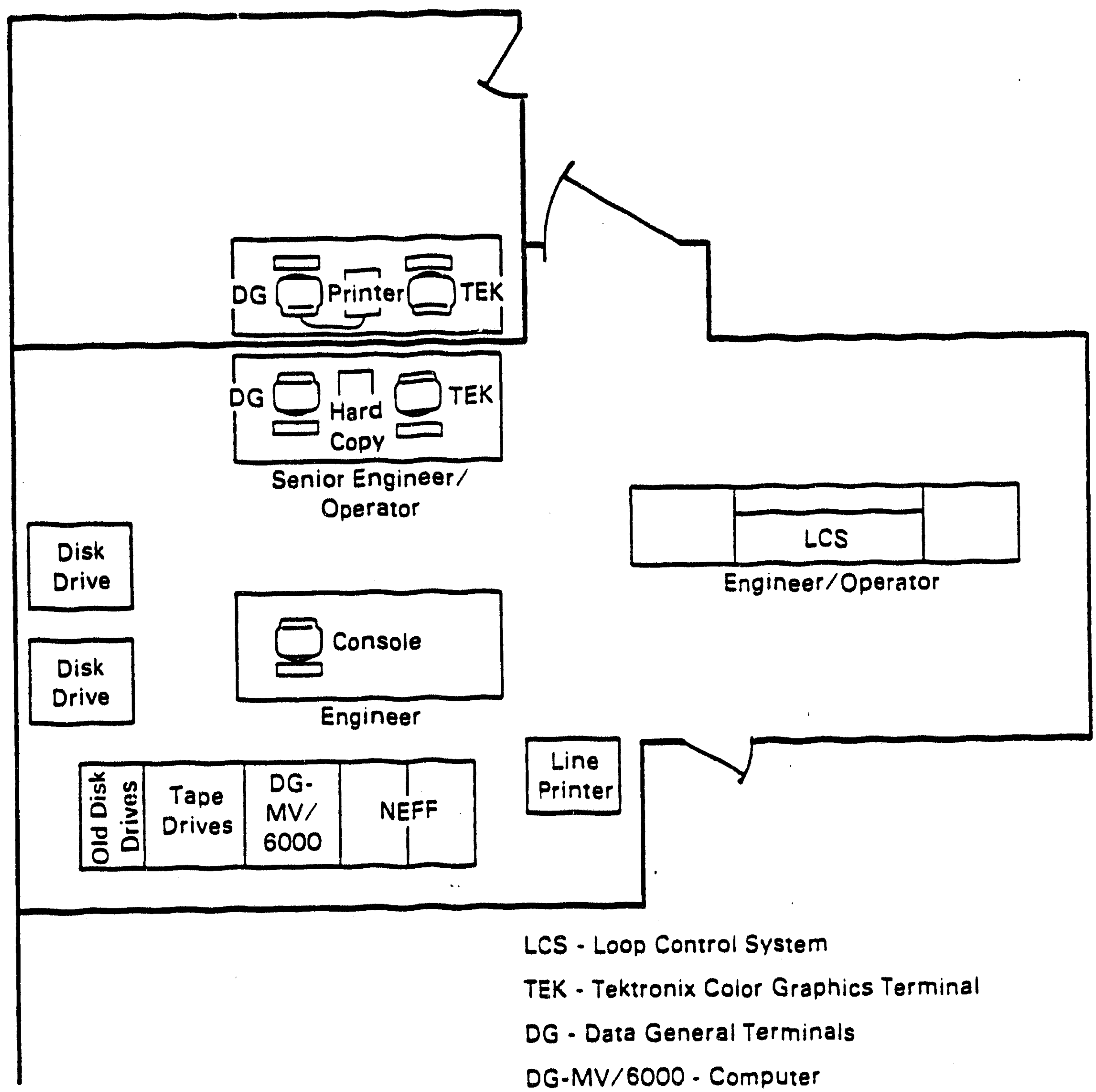


FIGURE 9. Layout of Effluent Control Module Instruments



LCS - Loop Control System

TEK - Tektronix Color Graphics Terminal

DG - Data General Terminals

DG-MV/6000 - Computer

NEFF - Analog-to-Digital Data Acquisition System

FIGURE 10. Floor Plan of the Data Acquisition and Control System

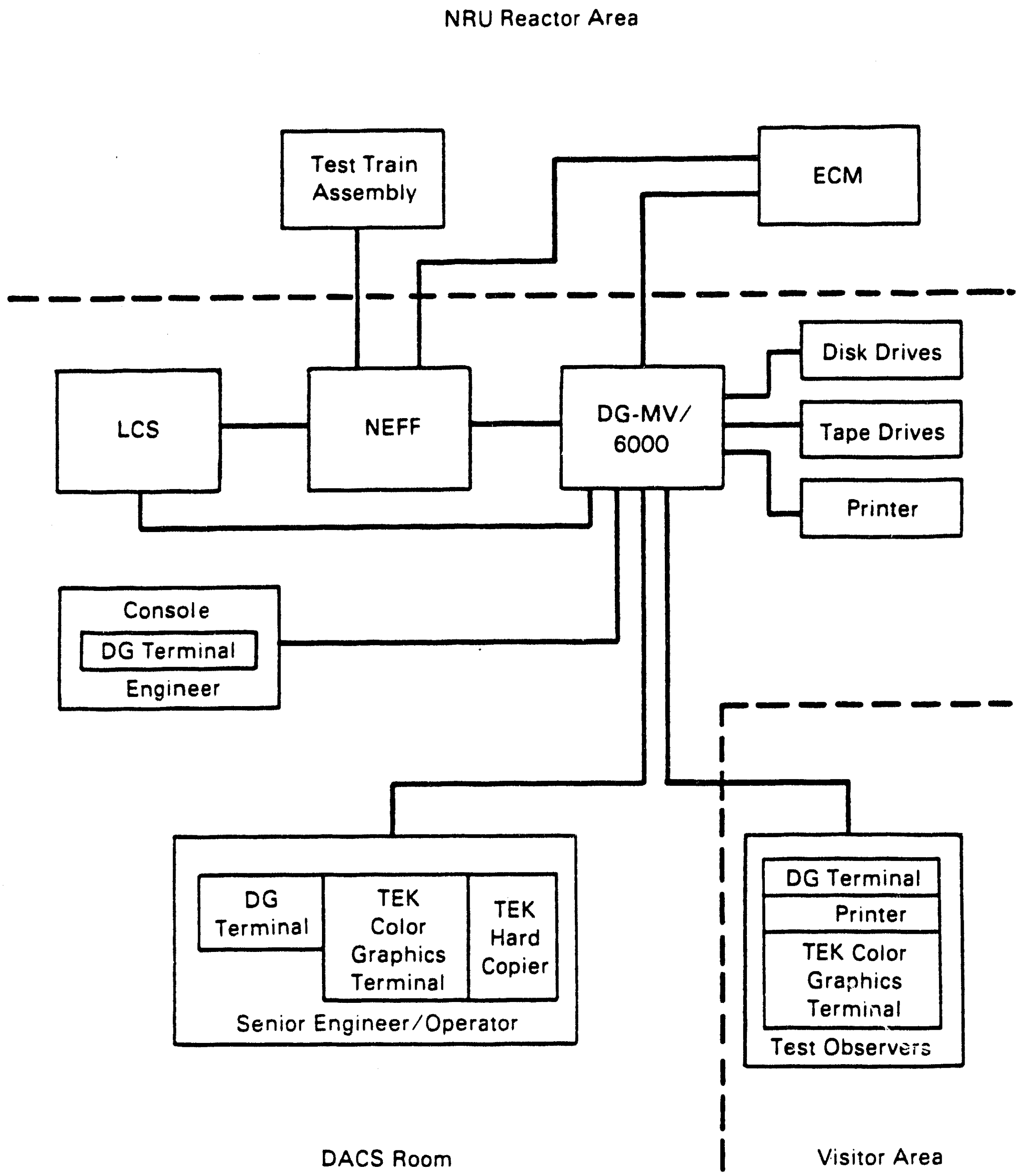


FIGURE 11. Electrical Connections for the DACS, LCS, ECM, Test Train, and Data Stations

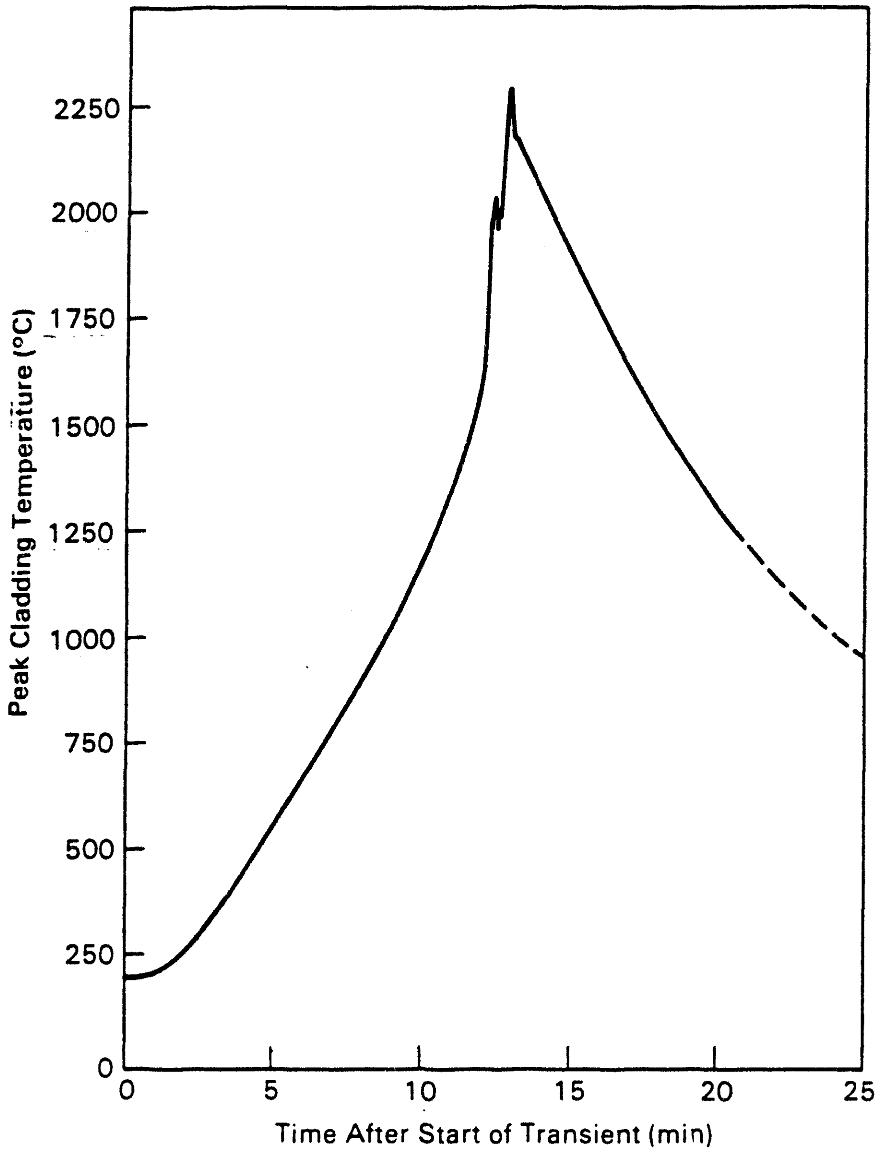


FIGURE 12. Predicted FLHT-2 Peak Cladding Temperatures

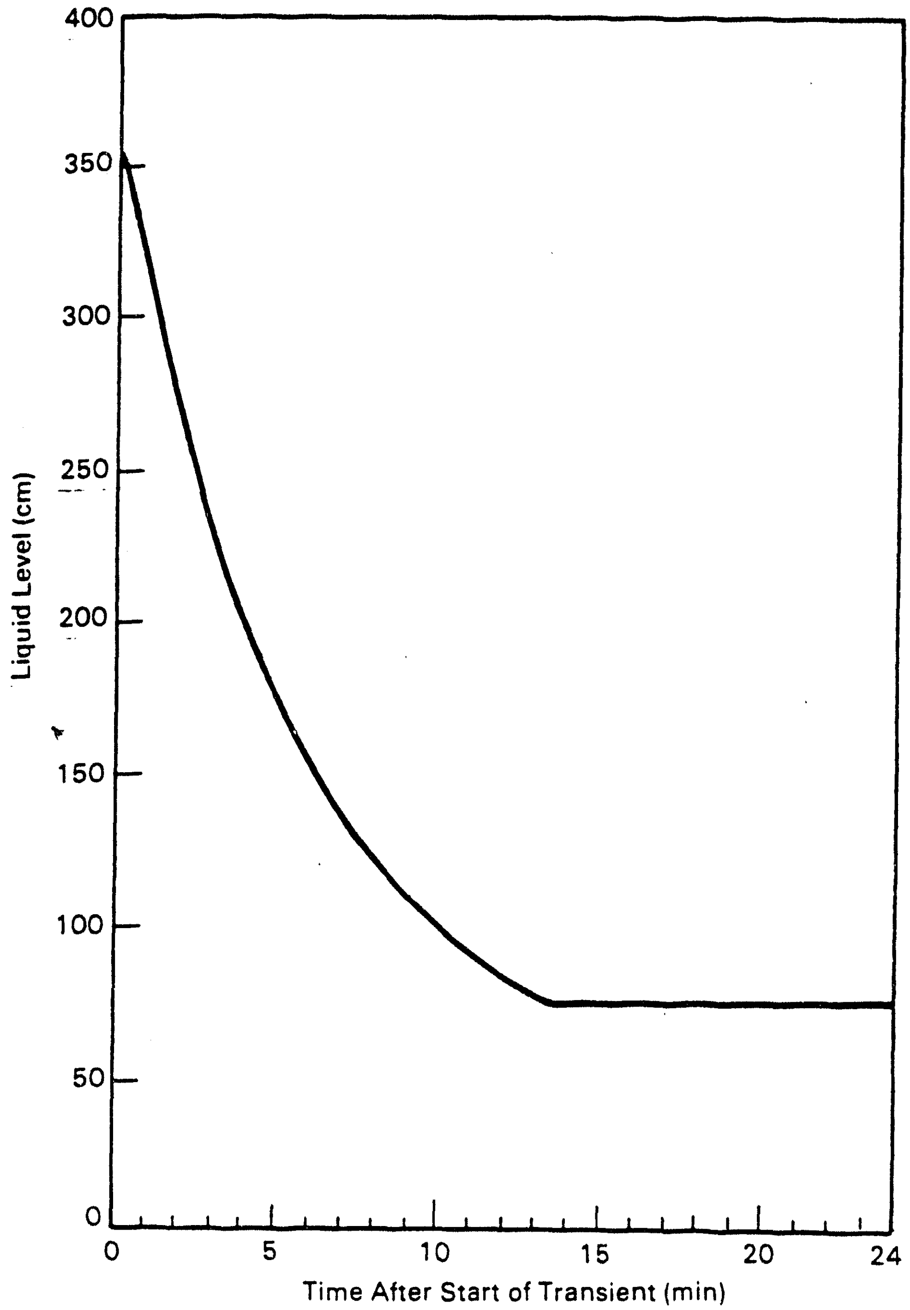


FIGURE 13. Predicted FLHT-2 Liquid Level Decrease Due to Fuel Bundle Coolant Boilaway

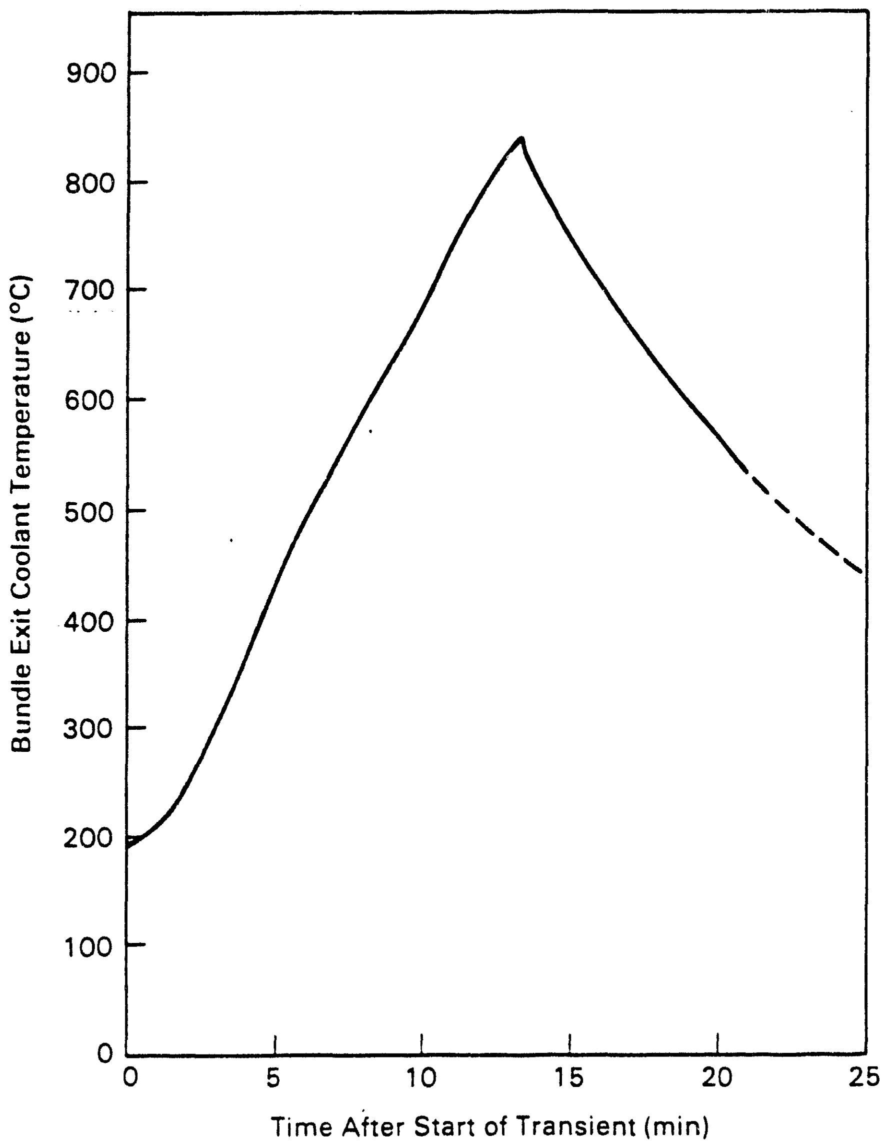


FIGURE 14. Predicted FLHT-2 Bundle Coolant Exit Temperatures

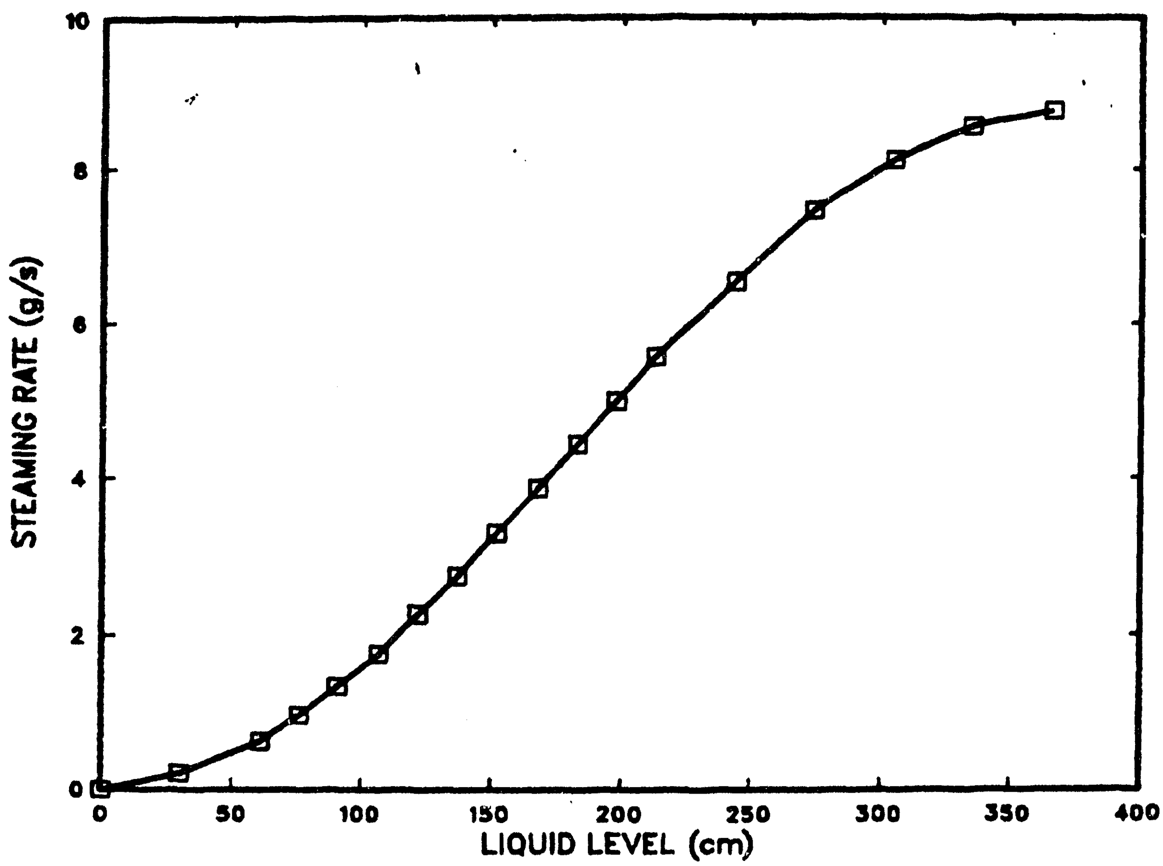


FIGURE 15. Predicted FLHT-2 Steam Generation Rates

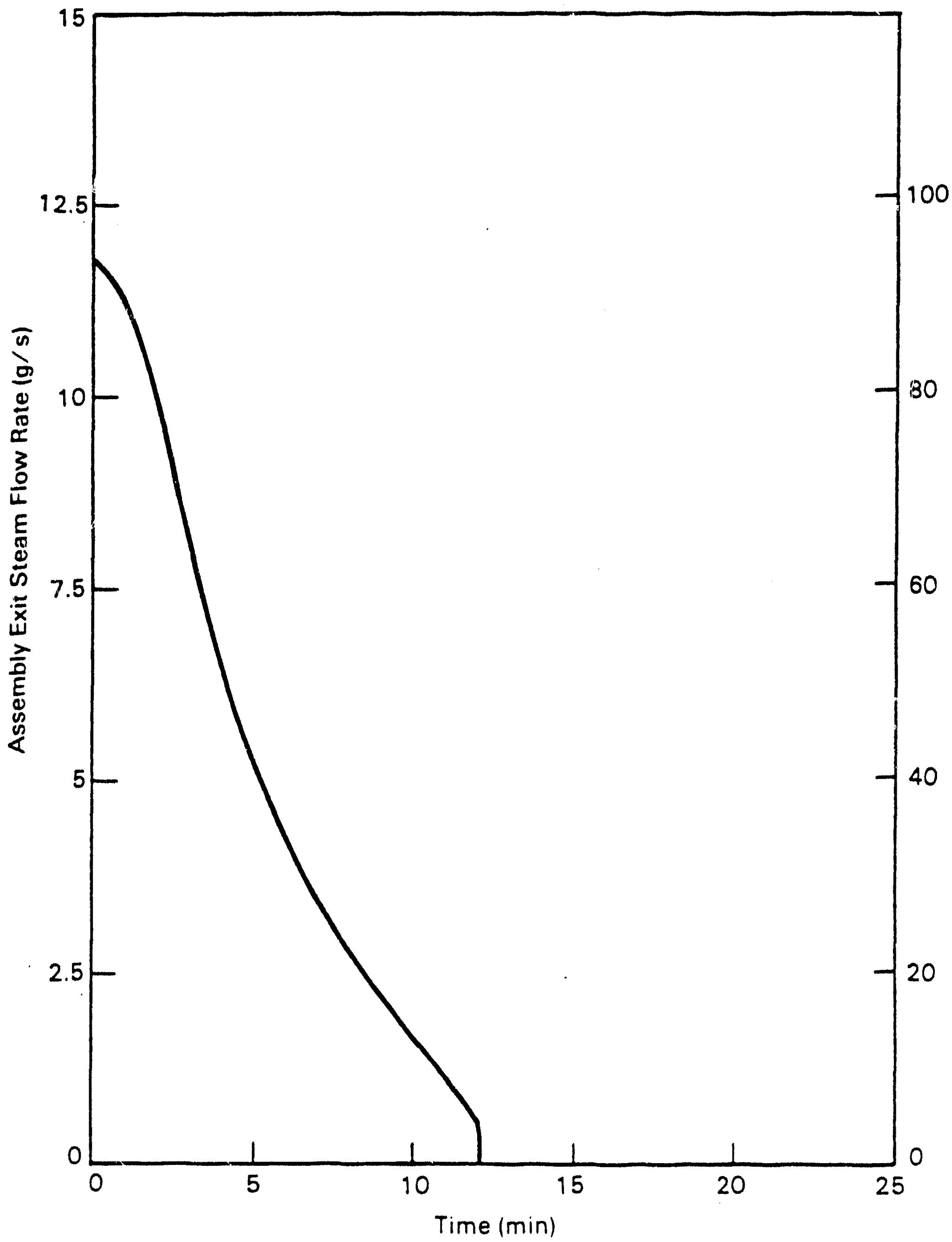


FIGURE 16. Predicted FLHT-2 Bundle Exit Steam Flow Rate

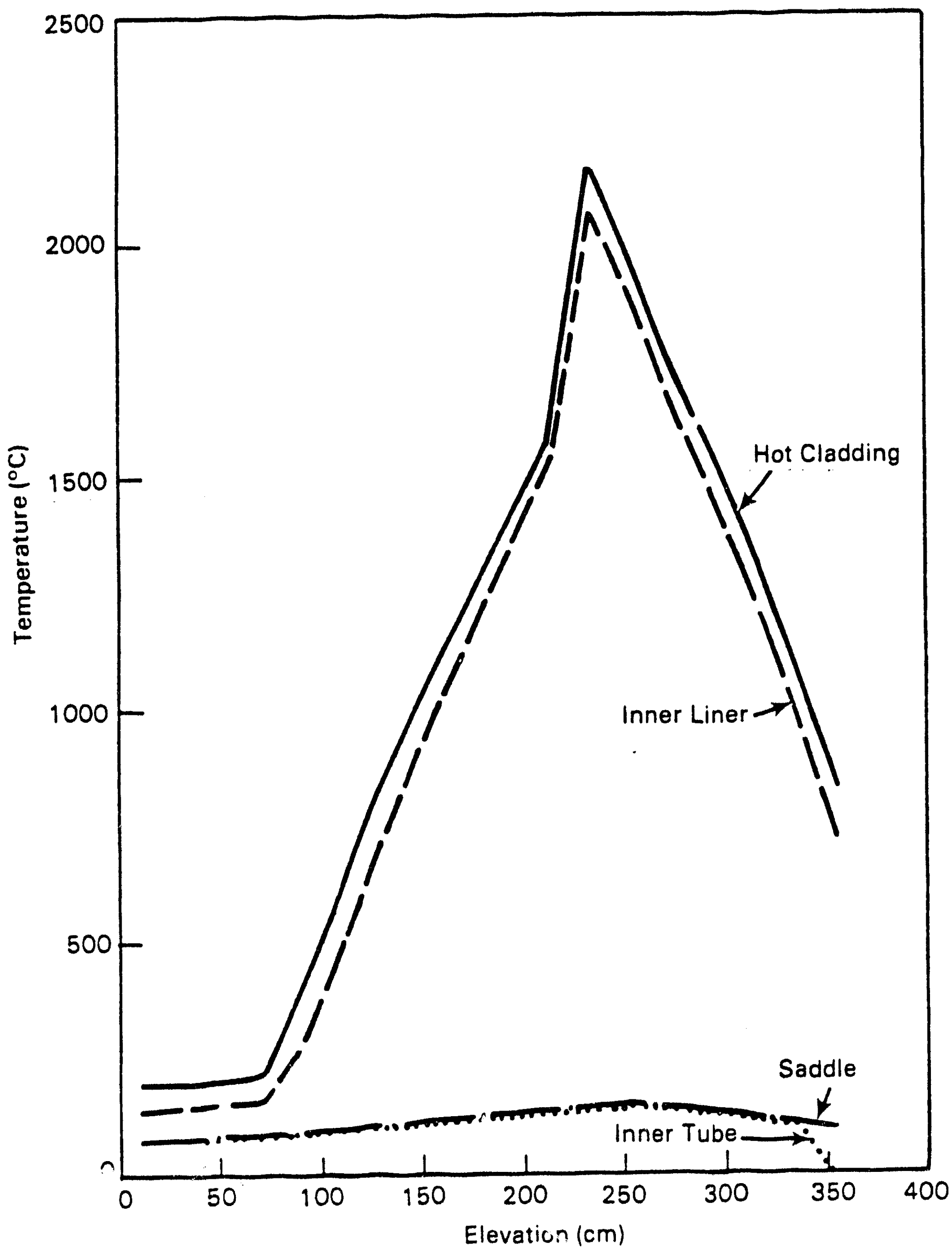


FIGURE 17. Predicted FLHT-2 Cladding, Liner, Saddle, and Inner Tube Peak Axial Temperatures

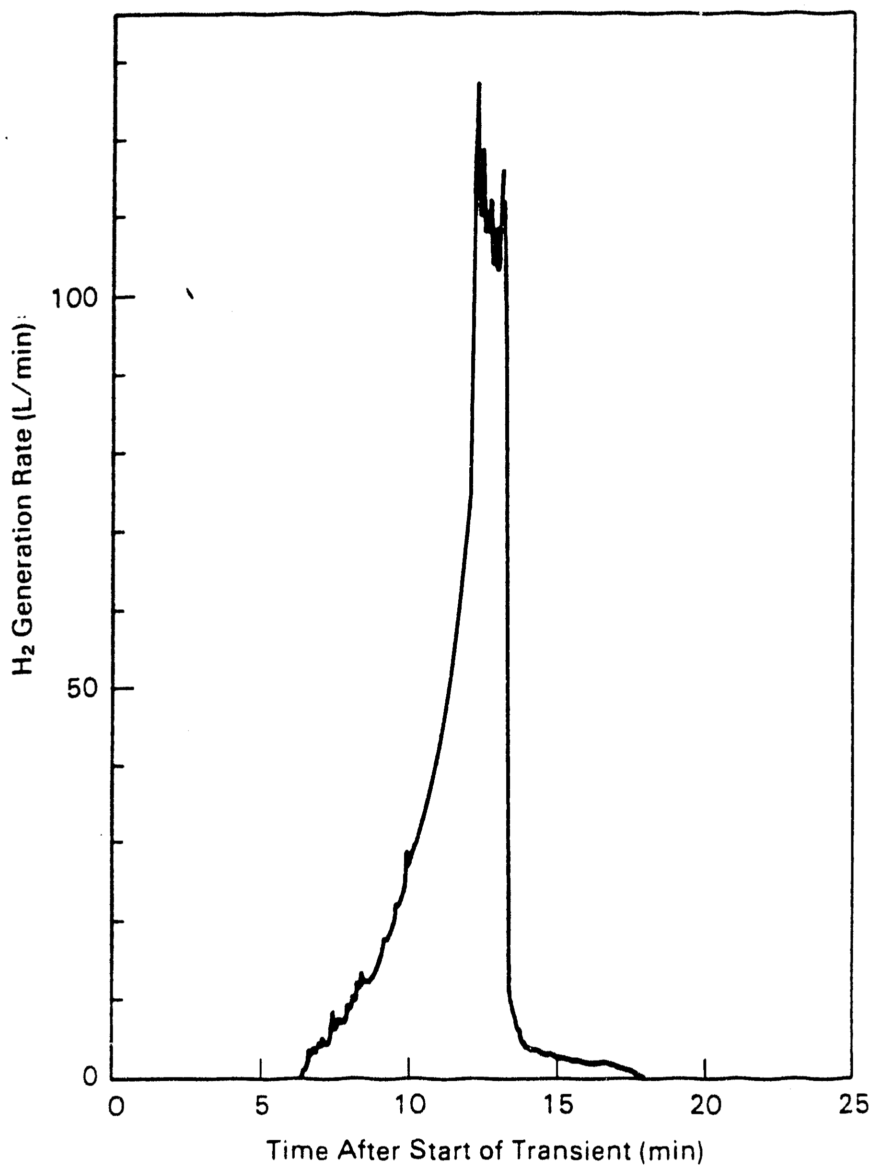


FIGURE 18. Predicted FLHT-2 Hydrogen Generation Rates

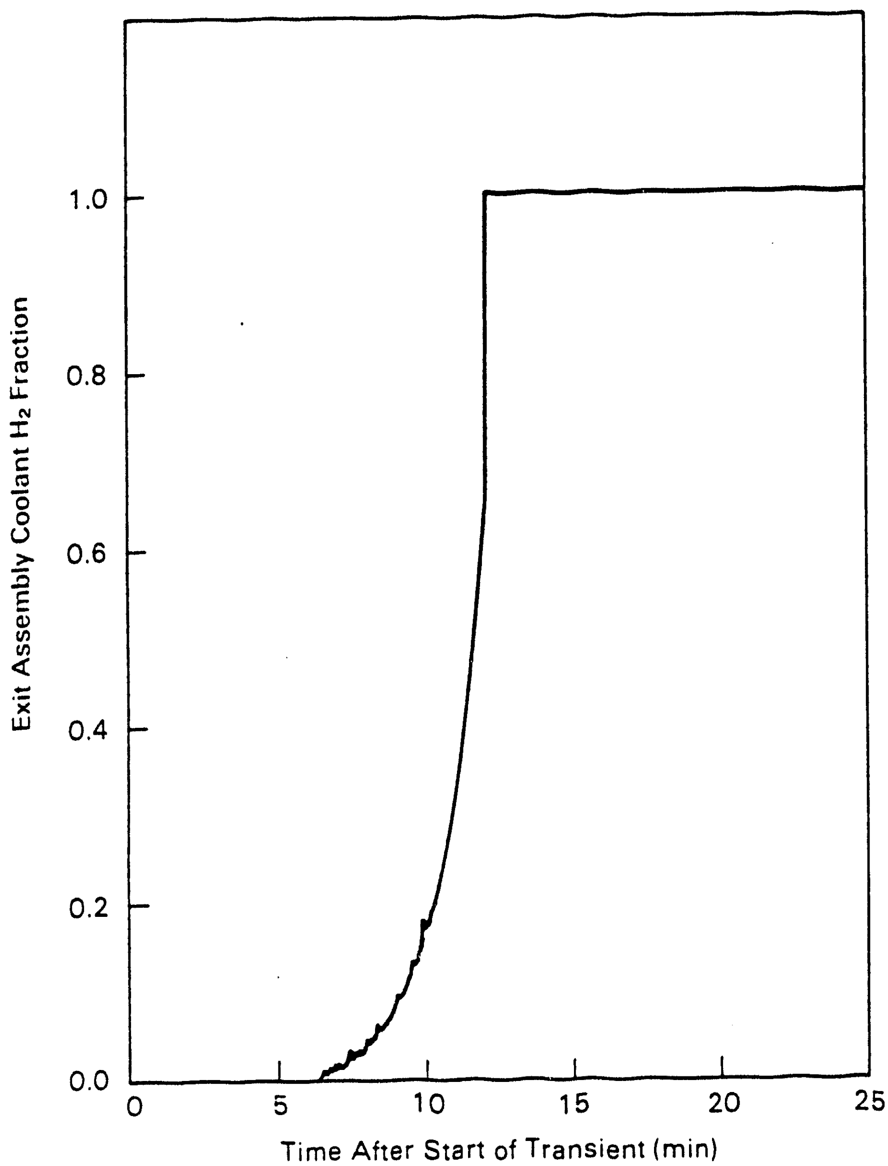


FIGURE 19. Predicted FLHT-2 Exit Coolant Hydrogen Fraction

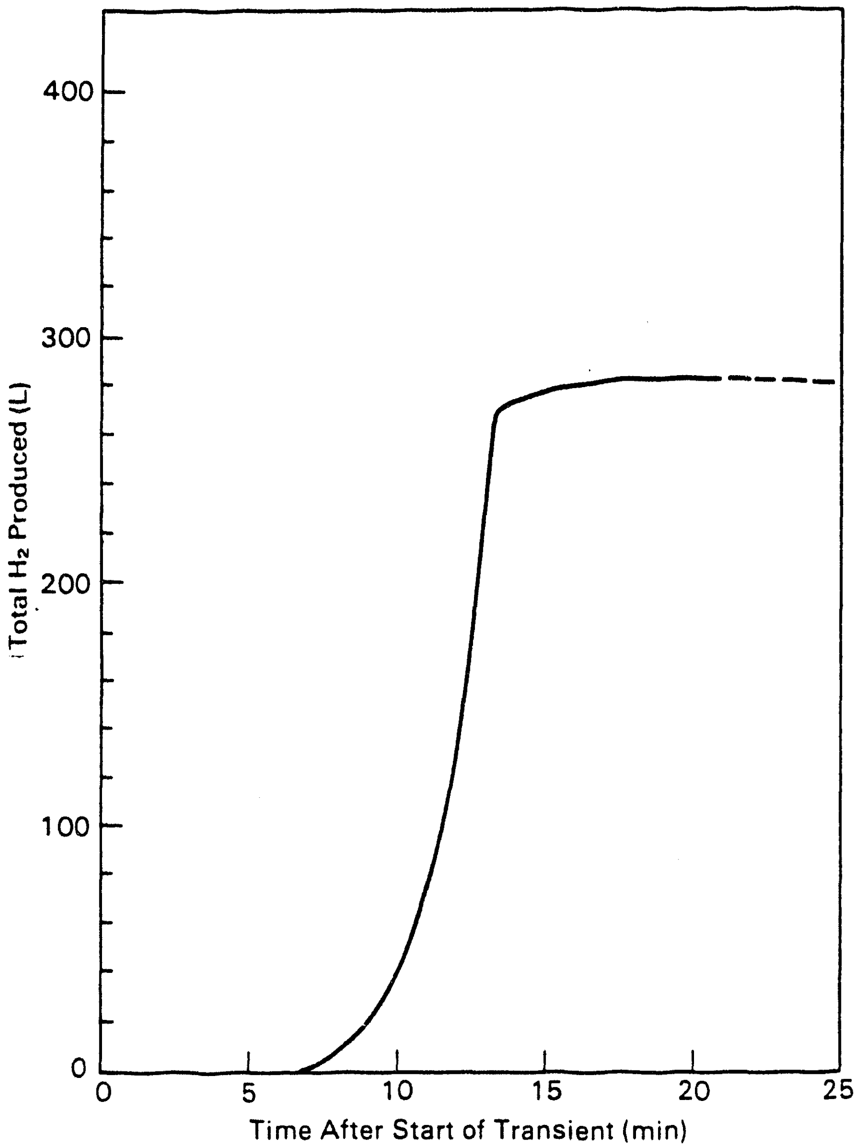


FIGURE 20. Predicted FLHT-2 Cumulative Hydrogen Production

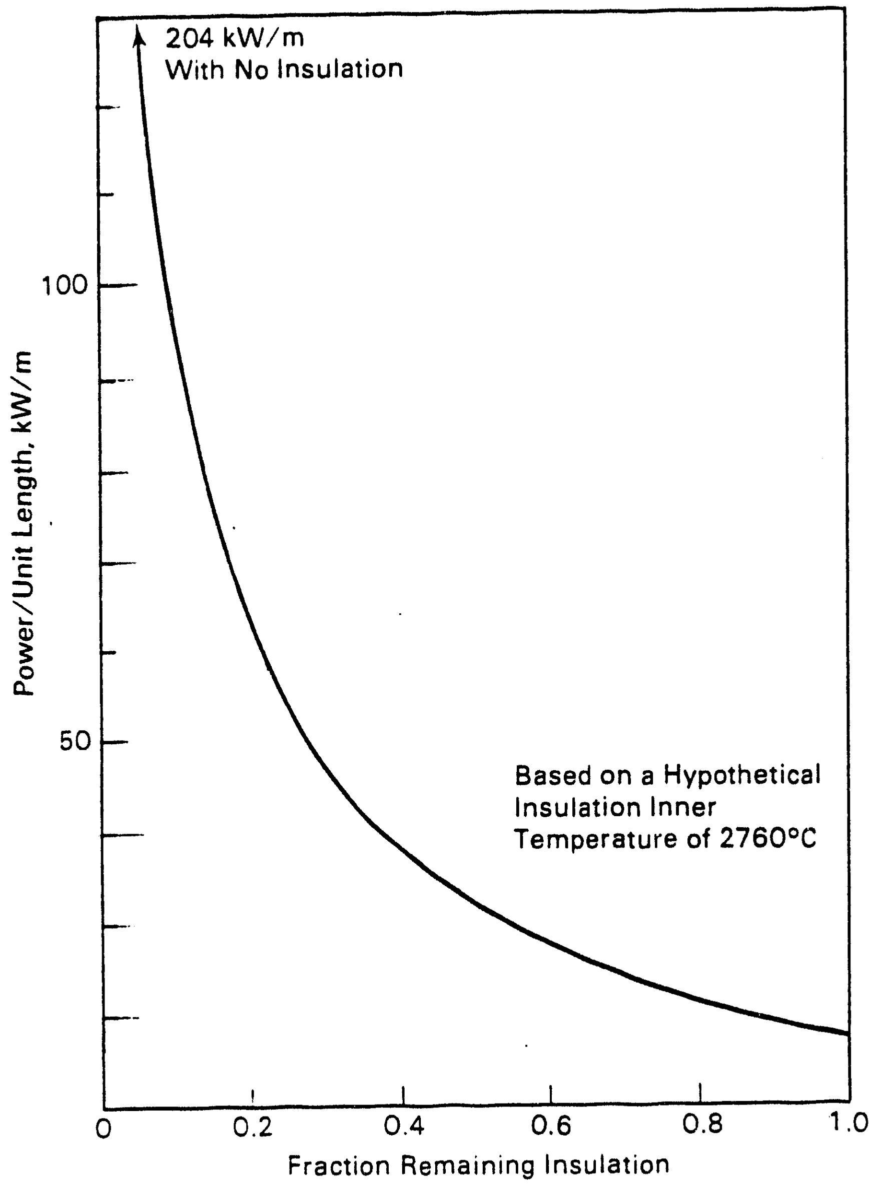


FIGURE 21. Radial Heat Flow Versus Fraction of Remaining Insulation

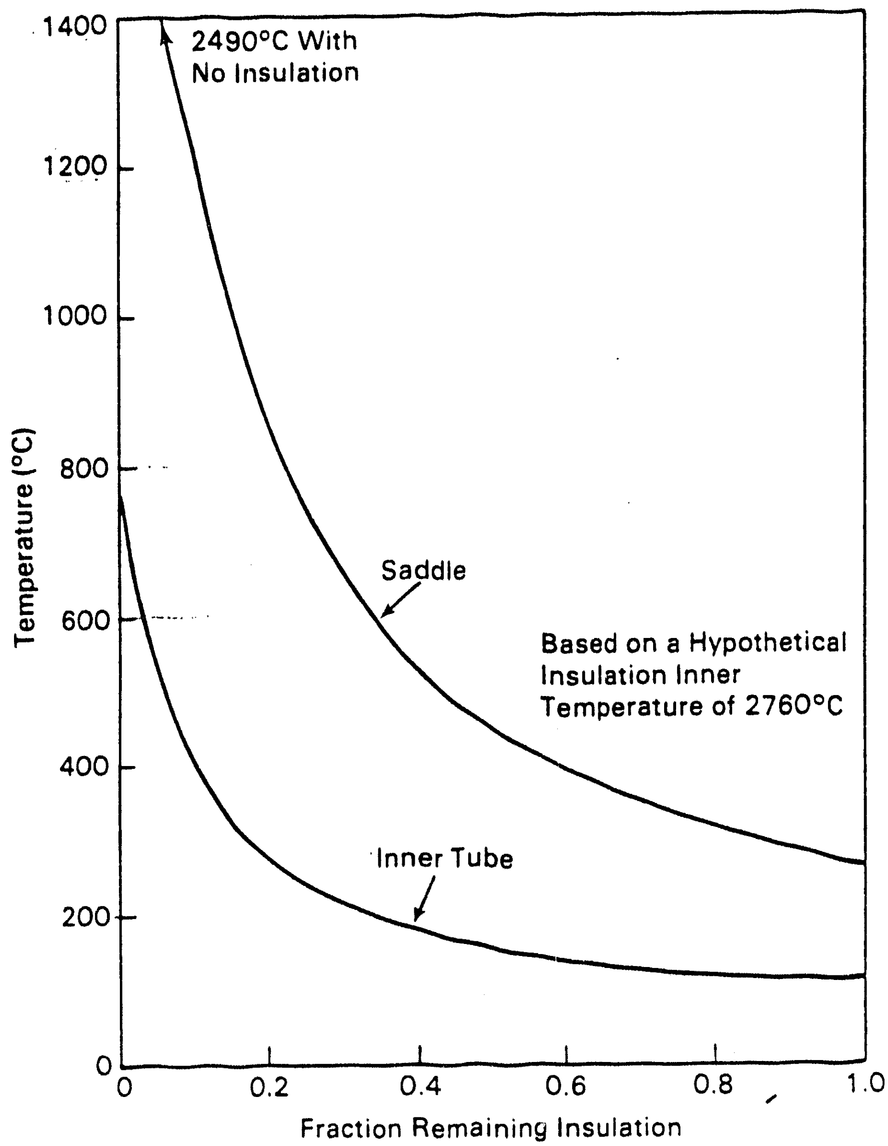


FIGURE 22. Saddle and Inner Zircaloy Tube Temperatures as a Function of the Fraction of Remaining Insulation

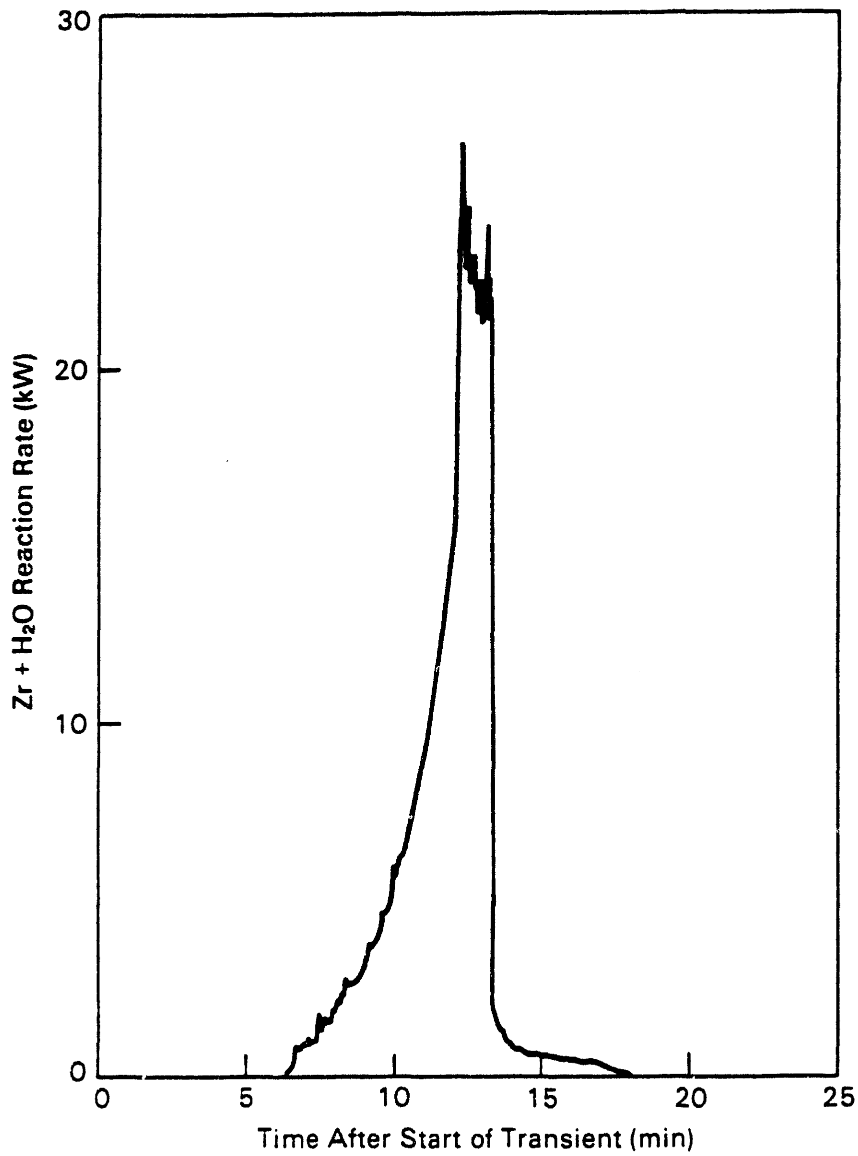


FIGURE 23. Zircaloy/Steam Chemical Heat

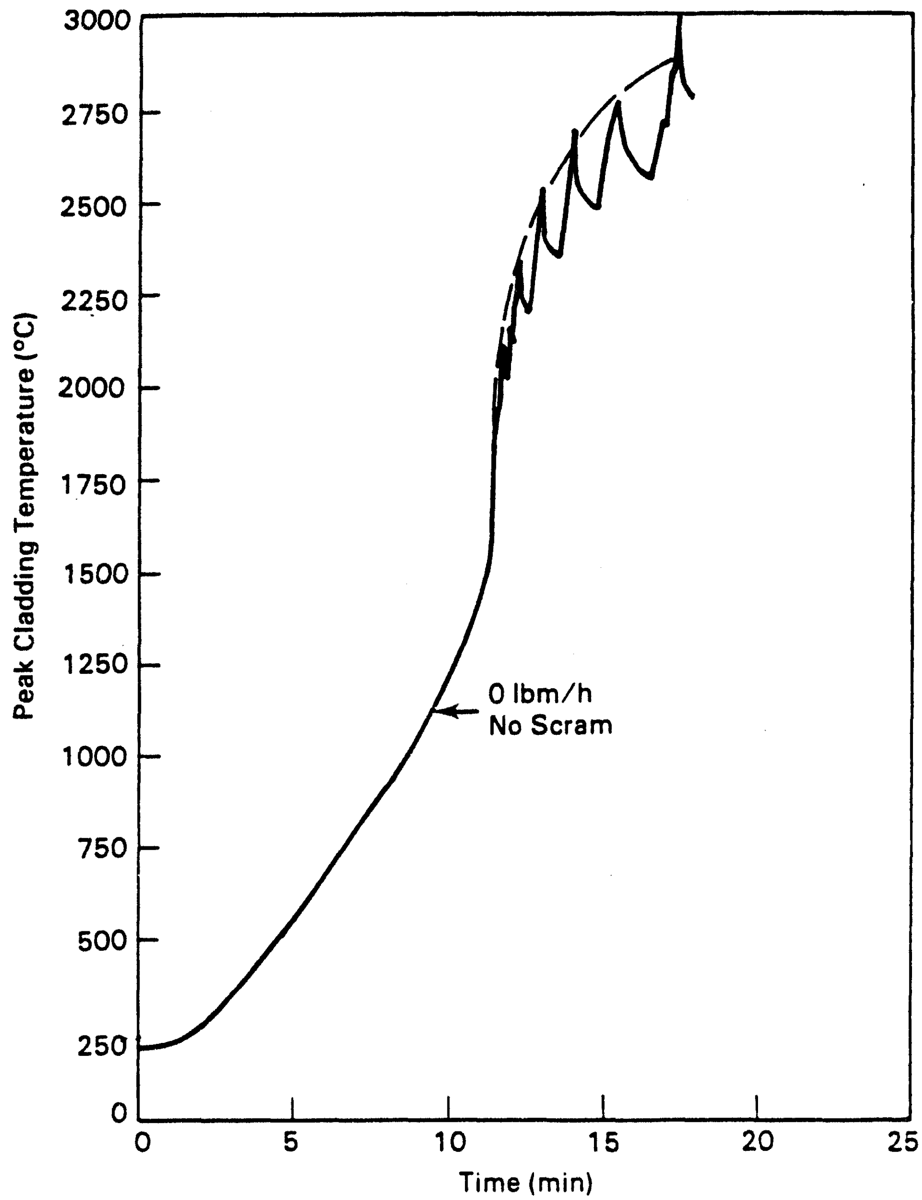


FIGURE 24. Peak Cladding Temperatures Without a Reactor Scram

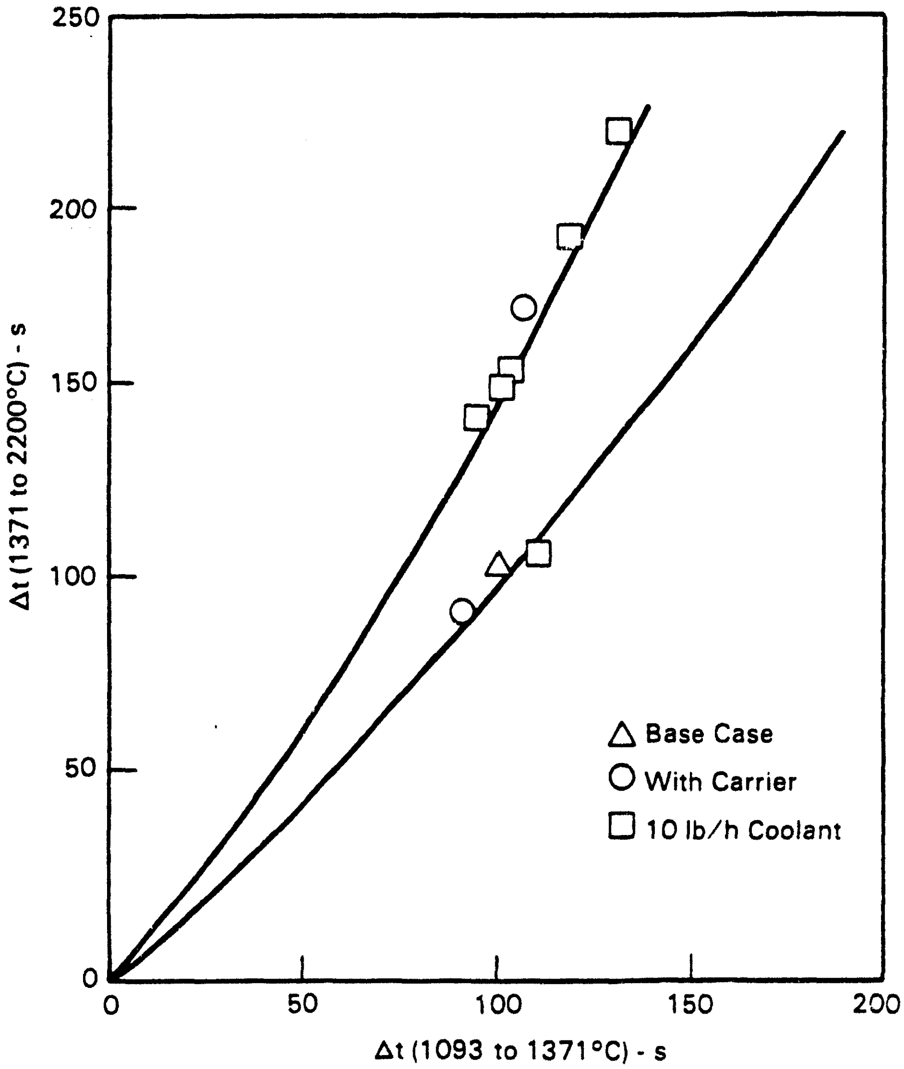


FIGURE 25. Calculated Time Intervals Required to Achieve Specified Peak Cladding Temperature Intervals for FLHT-2

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