CA8205537

AECL-7153

ATOMIC ENERGY
OF CANADA LIMITED

L'ÉNERGIE ATOMIQUE
DU CANADA LIMITÉE

FINAL SAFETY AND HAZARDS ANALYSIS FOR THE BATTELLE LOCA SIMULATION TESTS IN THE NRU REACTOR

Analyse finale de sécurité et des risques pour les essais de simulation LOCA Battelle dans le réacteur NRU

D.J. AXFORD, I.C. MARTIN and S.J. McAULEY

Chalk River Nuclear Laboratories — Laboratoires nucléaires de Chalk River

Chalk River, Ontario

April 1981 avril

ATOMIC ENERGY OF CANADA LIMITED

FINAL SAFETY AND HAZARDS ANALYSIS FOR THE BATTELLE LOCA SIMULATION TESTS IN THE NRU REACTOR

by

D.J. Axford, I.C. Martin and S.J. McAuley

Reactor Technology Branch
Chalk River Nuclear Laboratories
Chalk River, Ontario, KOJ 1JO
Canada

1981 April

AECL-7153

L'ENERGIE ATOMIQUE DU CANADA, LIMITEE

Analyse finale de sécurité et des risques pour les essais de simulation LOCA Battelle dans le réacteur NRU

par

D.J. Axford, I.C. Martin et S.J. McAuley

Résumé

Il s'agit du rapport final de sécurité et des risques pour les essais de simulation LOCA Battelle devant être effectués dans le réacteur NRU. Une brève description de l'expérimentation de l'équipement et des procédures de fonctionnement précède une revue du projet concernant l'analyse de la sécurité et des risques. La revue des risques touche les défaillances possibles d'équipement ainsi que le potentiel pour une réaction métal/eau et elle évalue les conséquences.

Le fonctionnement des essais proposés ne présente aucun risque inacceptable pour le réacteur NRU, le personnel de Chalk River ou le grand public.

Département de technologie des réacteurs
Laboratoires nucléaires de Chalk River
Chalk River, Ontario, K0J 1J0
Canada

Avril 1981

AECL-7153

ATOMIC ENERGY OF CANADA LIMITED

FINAL SAFETY AND HAZARDS ANALYSIS FOR THE BATTELLE LOCA SIMULATION TESTS IN THE NRU REACTOR

by

D.J. Axford, I.C. Martin and S.J. McAuley

ABSTRACT

This is the final safety and hazards report for the proposed Battelle LOCA simulation tests in NRU. A brief description of equipment test design and operating procedure precedes a safety analysis and hazards review of the project. The hazards review addresses potential equipment failures as well as potential for a metal/water reaction and evaluates the consequences.

The operation of the tests as proposed does not present an unacceptable risk to the NRU Reactor, CRNL personnel or members of the public.

Reactor Technology Branch
Chalk River Nuclear Laboratories
Chalk River, Ontario, K0J 1J0
Canada

1981 April

AECL-7153

TABLE OF CONTENTS

TABLE OF CONTENTS

FINAL SAFETY AND HAZARDS ANALYSIS FOR THE BATTELLE LOCA SIMULATION TESTS IN THE NRU REACTOR

by

D.J. Axford, I.C. Martin and S.J. McAuley

1. INTRODUCTION

As part of an ongoing program concerned with the performance characteristics of nuclear reactor fuel in light water reactors (LWR) under accident conditions, Battelle Pacific Northwest Laboratories (PNL) have proposed to do a series of loss of coolant accident (LOCA) simulations in an additional loop position in the NRU reactor. NRU was chosen because of its capability for testing multi-element bundles up to four metres long under thermal hydraulic conditions similar to LWRs and because of its ability to achieve requisite power densities using fuels with normal commercial enrichment.

The series is composed of six separate tests scheduled to begin in the fall of 1980, with a test being carried out at up to two month intervals as NRU operating schedule allows. The experimental program will concentrate on the heatup, reflood and quench phase of the LOCA. The system enthalpy and decay heat normally available to drive a LOCA transient will be simulated by low level fission heat from NRU operating at relatively low power (about 10% power).

This report is the final safety and hazards analysis of the proposed test series. It first considers the tests to be performed, followed by a design description of the test assembly, test loop, and associated instrumentation and controls. The safety aspects of the tests are then considered, as well as the normal and accident hazards, to ensure that the tests do not present an unacceptable risk to the safety of the reactor, operating personnel or members of the public.

2. TEST OPERATION

2.1 General

The program consists of a series of six fuel tests designed to generate data on nuclear fuel performance during a loss of coolant transient. This data will be used to verify existing computer models in codes used by the U.S. Nuclear Regulatory Commission (NRC) in the licensing of light water reactors.

Each of the six test assemblies is composed of 31 fuel elements with a 3.66 m long fuel section in a 6 x 6 matrix contained within a stainless steel shroud (see Section 3.2.1 for a detailed description).

Two types of tests are planned. The first test fuel assembly will consist of non-pressurized fuel elements on which a series of thermal hydraulic test runs will be done to provide an experimental data base of heatup-reflood information for thermal hydraulic code verification, and to provide a calibration of the system to define necessary operating conditions for the subsequent materials tests. The first run will be made under mild conditions and successive runs under more and more extreme conditions. Up to 45 runs are planned and no significant fuel cladding deformation is expected.

The remaining five test fuel assemblies will be used for materials tests which will provide information concerning the fuel failure modes and geometry. The central eleven elements in the 31 element bundle will be pressurized to an extent that they will balloon and rupture during the course of some of the tests. The remaining elements, functioning as guard elements, are unpressurized and may be reused in successive tests.

The object of the tests is to simulate varying conditions in a LOCA heatup and reflood. The variance is simulated by changing delay time to start reflood, and reflood rate, from test to test. Tentative variable values for each test are shown in Tables 1 and 2.

2.2 Test Installation

Final assembly of the first test assembly will be done in the NRU main reactor hall. The assembly will be gauged by inserting it into a prototype reactor pressure tube prior to installation in the reactor. The assembly will then be installed in the pressure tube in reactor position L-24 with the main crane. The position will be sealed, instrument leads connected to the data acquisition system (DAS) and a final checkout procedure carried out prior to loop startup. Subsequent test assemblies will be assembled in the rod bays on the disassembly-examination-reassembly-machine (DERM) and transferred via the elevator and J-rod flask to reactor position L-24. This shielded sequence of operations is necessary because the guard heater elements will have been previously irradiated.

2.3 Sequence of Operations

The tests will be carried out in three phases in order to best simulate conditions encountered during the heatup, reflood and quench phases of a LOCA. The three phases, which are described in detail in the following sections, are the preconditioning phase, pretransient phase and transient phase. Briefly, the preconditioning phase will cause fuel pellet cracking and condition the fuel to approximate the condition of power reactor fuel. The pretransient phase will bring the fuel up to the temperature and decay heat power which would be encountered prior to reflood in a real LOCA. The transient phase will involve adiabatic heatup of the fuel for a predetermined length of time followed by reflood and quench at a predetermined rate.

TABLE 1: Proposed operating conditions for thermal hydraulic test runs.

Heat rate: 8°C/s
Peak element power: 2.1 kW/m
System pressure: 0.28 MPa

Test Number	Reflood Rate (cm/s)	Reflood Delay Time (s)	Maximum Cladding Temperature - °C	Test Number	Reflood Rate (cm/s)	Reflood Delay Time (s)	Maximum Cladding Temperature - °C
101	10.2	3[1]	< 540	124[3]	10.2	37	760
102	10.2	20	650	125	10.2	51	870
103	10.2	29	705	126	10.2	70	980
104	10.2	37	760	127[3]	10.2	37	760
105	5.1	7	760	128	3.3	3[1]	980
106	5.1	19	815	129	3.8	20	980
107	5.1	30	870	130[3]	10.2	37	760
108	4.8	3[1]	760	131	5.1	50	980
109	3.8	3[1]	870	132	20.3	71	980
110[2]	3.8[2]	11	925	133[3]	10.2	37	760
111[2]	3.8[2]	11	925	134	25.4	72	980
112	7.6	32	760	135	2.8	3[1]	1040[4]
113	20.3	39	760	136[3]	10.2	37	760
114	20.3	46	815	137	3.8	32	1040[4]
115	20.3	53	870	138	5.1	60	1040[4]
116	25.4	40	760	139[3]	10.2	37	760
117	25.4	47	815	140	10.2	76	1040[4]
118	25.4	54	870	141	20.3	77	1040[4]
119[2]	15.5[2]	52	870	142[3]	10.2	37	760
120[2]	15.5[2]	52	870	143[2]	3.8[2]	53	1040[4]
121	7.6	48	870	144[2]	3.8[2]	53	1040[4]
122[2]	7.6[2]	53	925	145[3]	10.2	37	760
123[2]	7.6[2]	53	925				

(1) Minimum delay time (less than 3 s) is necessary for the reflood water to arrive at the bottom of the fuel column after steam flow is stopped.

(2) First test in pair uses a fast fill rate up to the 30 cm level of the fuel column. The second test uses a constant reflood rate.

(3) Repeat of test number 104.

(4) Cladding temperature may exceed 980°C due to uncertainty in predictive capability.

TABLE 2: Proposed operating conditions for materials tests.

Heating rate	8°C/s
Peak element power	2.1 kW/m
System pressure	0.28 MPa
Initial element pressure	3.1 MPa

Test Number	Reflood Rate (cm/s)	Reflood Delay Time (s)	Maximum Cladding Temperature °C
2	12.7	32	760
3	5.1	12	760
4	2.5-3.6	0	760
5	5.1	25	870
6	5.1	40	980

2.3.1 Preconditioning Phase

The purpose of this phase is to cause some fuel pellet cracking and relocation within the fuel tubes so their condition approximates that of power reactor fuel. The test section will be connected to the U-2 loop cooling circuit and the test assembly cooled in the pressurized water mode. The reactor will be started up and power raised in steps until test assembly design power or maximum reactor power is reached. Power calibration data will be taken at each step. Rate of rise of power will be determined by NRU Operations. Approximately one hour after startup the reactor will be shutdown. This period is limited to one hour to minimize the build-up of the fission product inventory. For the thermal hydraulic test this phase will be carried out for the first run only since the same test bundle is used for all the runs in the test. Preconditioning conditions are shown in Table 3.

TABLE 3: Preconditioning operating limits.

Coolant	Pressurized water
Flow	16.3 kg/s
Outlet pressure	6.2 MPa
Test section pressure drop	0.168 MPa
Coolant inlet temperature	222°C
Coolant outlet temperature	252°C
Total power	2.6 MW
Maximum heat flux	143 W/cm^2
Maximum linear element power	43.3 kW/m
Average linear element power	23.0 kW/m
Peak cladding surface temperature	286°C
Peak fuel temperature	1394°C
Maximum shroud temperature	273°C
Maximum pressure tube temperature	300°C
Axial peaking factor	1.51
Radial peaking factor	1.24
Minimum DNBR ratio	3.43

2.3.2 Pretransient Phase

The purpose of this phase is to provide the test assembly with the appropriate initial conditions needed for the start of the transient test. On completion of the preconditioning phase the test section will be disconnected from the U-2 loop circuit and connected to the test loop circuit and the test assembly will be cooled with superheated steam. The reactor will be started up and power raised in steps until the test assembly reaches its design power for the transient, which should result in a peak pretransient cladding temperature of about 430°C (see Figure 1). The power will be held at this level until the transient is initiated. The operating conditions for this phase are listed in Table 4.

TABLE 4: Pretransient operating limits.

Coolant	Superheated steam
Average coolant flow	.378 kg/s
Coolant inlet temperature	163°C
Average coolant outlet temperature	349°C
Test section outlet pressure	0.276 MPa
Test section pressure drop	0.058 MPa
Total power	141 kW
Maximum heat flux	7.03 W/cm^2
Maximum linear element power	2.1 kW/m
Average linear element power	1.2 kW/m
Peak cladding surface temperature	427°C
Peak fuel temperature	473°C
Maximum shroud temperature	354°C
Maximum pressure tube temperature	
At test assembly location	300°C
Above test assembly	315°C

2.3.3 Transient Phase

From the steady state steam cooled condition the fuel temperature transient will be initiated by quickly shutting off the steam flow while maintaining a constant reactor power (see Figure 1). At a predetermined time after steam shutoff, reflood water will be introduced to turn around the temperature transient at a predetermined temperature. This predetermined time and reflood rate are experimental variables. Introduction of reflood will quench the fuel and the reactor will be shut down after quench and temperature turn-around. The values of the transient variables are listed in Table 5.

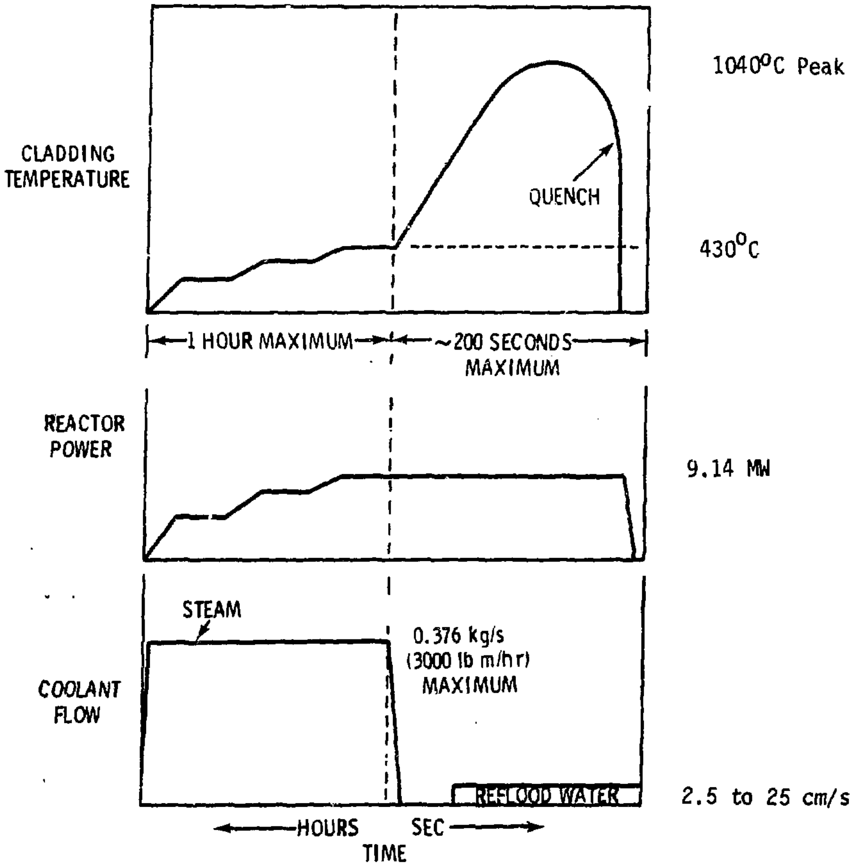

FIGURE 1: PRETRANSIENT AND TRANSIENT PHASE IN TEST LOOP.

TABLE 5: Transient variables.

Reflood rate	2.5-25 cm/s
Reflood temperature	52°C
Maximum linear heat rating per test element	2.1 kW/m
Temperature ramp rate	8.3°C/s
Delay time to start of reflood	3 - 77 s
Design peak cladding temperature	980°C (1)
Maximum peak temperature at trip point	1200°C
Maximum design coolant outlet temperature	454°C
Maximum pressure tube design temperature	454°C

(1) Cladding temperature may reach 1040°C due to uncertainty in predictive capability.

The exact course of events that will take place during the experimental transients is not known; in fact, the experiments are being performed to supply this information. Hazardous conditions will be avoided by careful selection of the test sequence. The first tests will be done under mild conditions, i.e. early start of reflood and high reflood rates. The information from the early tests will be used to define subsequent test conditions to avoid hazardous temperatures. Results of calculations from the COBRA-TF and THERM codes by Battelle will further define the test parameters.

The expected histories of several of the dependent variables for a transient experiment are given in Figure 2. The information for this figure was calculated using reflood heat transfer coefficients from the FLECHT correlation by Battelle as input to the TRUMP heat transfer code. The transient calculated was one for which a peak cladding temperature of 980°C was desired. For the calculations, it was conservatively assumed that conditions were not exact and that the peak cladding temperature reached 1080°C. The figure gives the peak cladding temperatures of both the test and guard elements, the peak fuel centerline temperature, and the peak temperatures of the shroud and pressure tube.

2.4 Test Removal and Examination

On conclusion of a test the fuel will be allowed to cool in reactor for a minimum of one hour. After disconnection of instrument lines the assembly will be removed with the J-rod flask and transferred to the rod bays. The assembly will be installed horizontally in the DERM where it will be dismantled and examined by Battelle personnel.

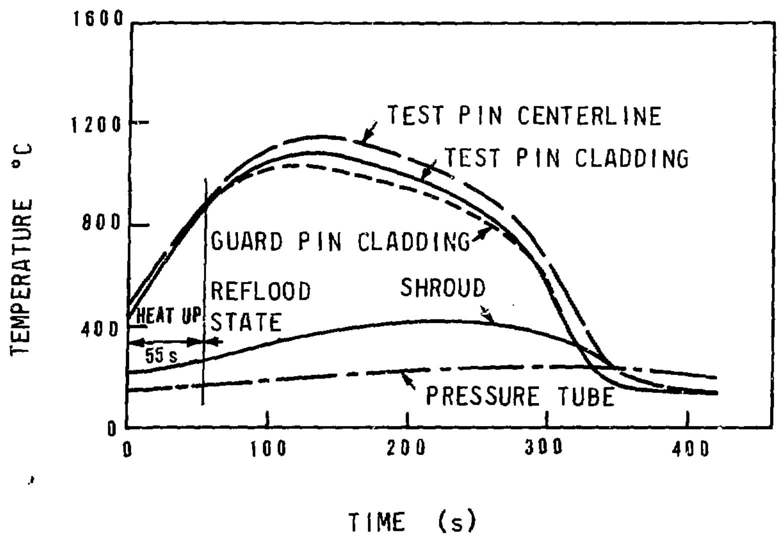

FIGURE 2: MAXIMUM TEMPERATURES EXPECTED DURING LOCA SIMULATION TESTS

3. TEST DESIGN

3.1 General

This section contains a description of the test assembly and its component parts and also reviews the instrumentation used for monitoring the flow, temperature, pressure and neutron flux. The data acquisition system for retrieving and recording monitored parameters is also described. The test loop provided to supply superheated steam and reflood water is described, including its control circuit. The estimated reactor powers required to achieve the desired test assembly power are listed.

3.2 Test Assembly

3.2.1 Mechanical

The overall test assembly length including both closure region and test region is 9.18 m (see Figure 3).

The fuel bundle consists of a 6 x 6 segment from a 17 x 17 LWR assembly, with the four corner elements removed for insertion in a shroud, resulting in a basic test array of 32 elements, as shown in Figure 4. The outer row of elements, including the corner elements of the next inner ring, will not be pressurized and will serve as guard heaters during the test. The remaining central group of twelve elements will contain eleven fuelled test elements and one unfuelled instrument thimble, arranged in a cruciform pattern, as shown in

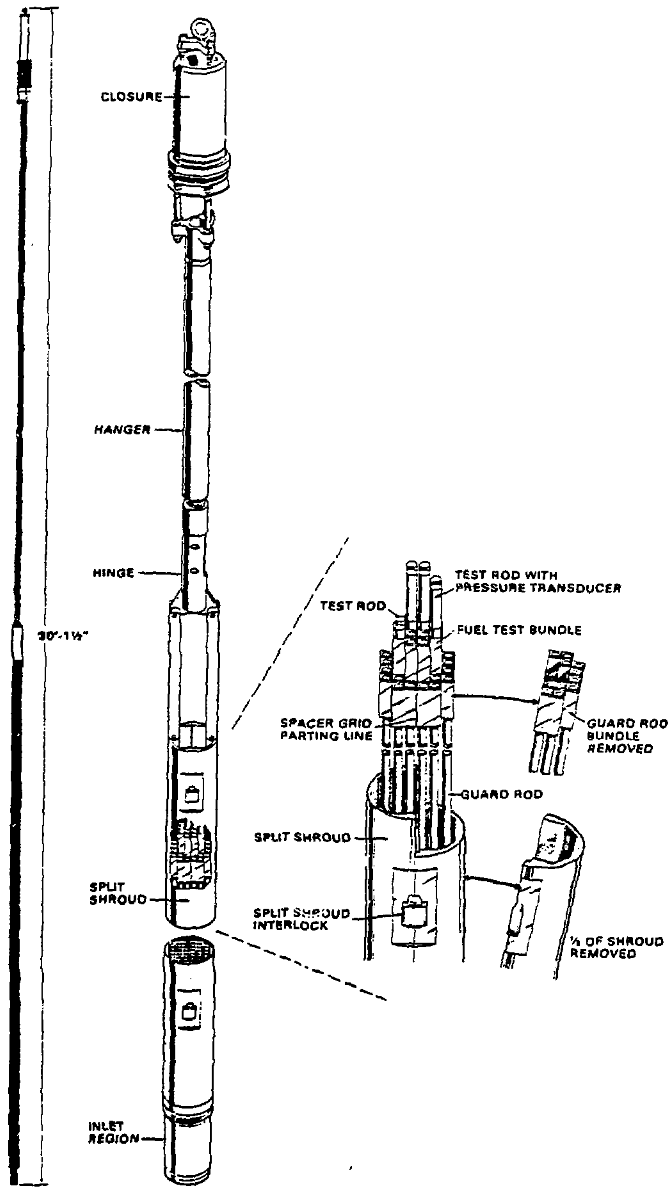

FIGURE 3: SCHEMATIC OF BATTELLE LOCA SIMULATION ASSEMBLY.

Figure 4. The test elements will be unpressurized for the first test series (thermal hydraulic test) and pressurized for the subsequent five tests (materials deformation tests). The bundle is designed to enable reuse of the guard element heaters, and the guard element array can be separated into two sections. The cruciform test element array can also be divided into segments for inspection and also for removal of the instrumented thimble tube. The eleven test elements will be replaced after each test by unirradiated fuel elements.

A stainless steel shroud will provide the support structure for the fuel bundle and serve as a liner during various stages of the experiment. The shroud, which is approximately 4.3 m long, will consist of two halves split along its length, clamped together at intermittent locations and attached at end fittings.

A hanger tube will suspend the test bundle and shroud from the top closure seal block. The closure seal block will provide the pressure boundary at the top end of the test assembly, between the test assembly and the loop closure. It will also provide a leak tight penetration for up to 183 instrument leads from the test assembly. Battelle will build a mock-up of the instrument leads in the closure seal block for quality control leak testing.

The fuel will consist of 8.27 mm diameter by 9.53 mm long sintered UO_2 pellets at 3 wt% U-235 enrichment in a 8.43 mm ID x 0.56 mm wall thickness Zircaloy sheath. The overall length of the active fuel is 3.66 m. A complete 31 element fuel bundle will contain about 61.28 kg of uranium dioxide and 1.62 kg of U-235.

3.2.2 Instrumentation

3.2.2.1 General

The test assembly will be highly instrumented with temperature, flux and pressure detection devices (a possible total of 183) in order to glean the maximum amount of information possible from each test. The test assembly is sectioned, from an instrument point of view, into 21 levels with the degree of instrumentation from level to level varying according to the information required at that level (see Figure 5). Output from these devices will be fed to the data acquisition system as well as to indicators on the test loop control panel.

3.2.2.2 Flow

Flow measuring instruments will not be part of the test assembly. Flow will be measured by loop instrumentation outside of the reactor core. The standard U-2 loop flow indications will be used during the preconditioning. Separate flow control and indication will be provided in the test loop for steam flow and for reflood flow.

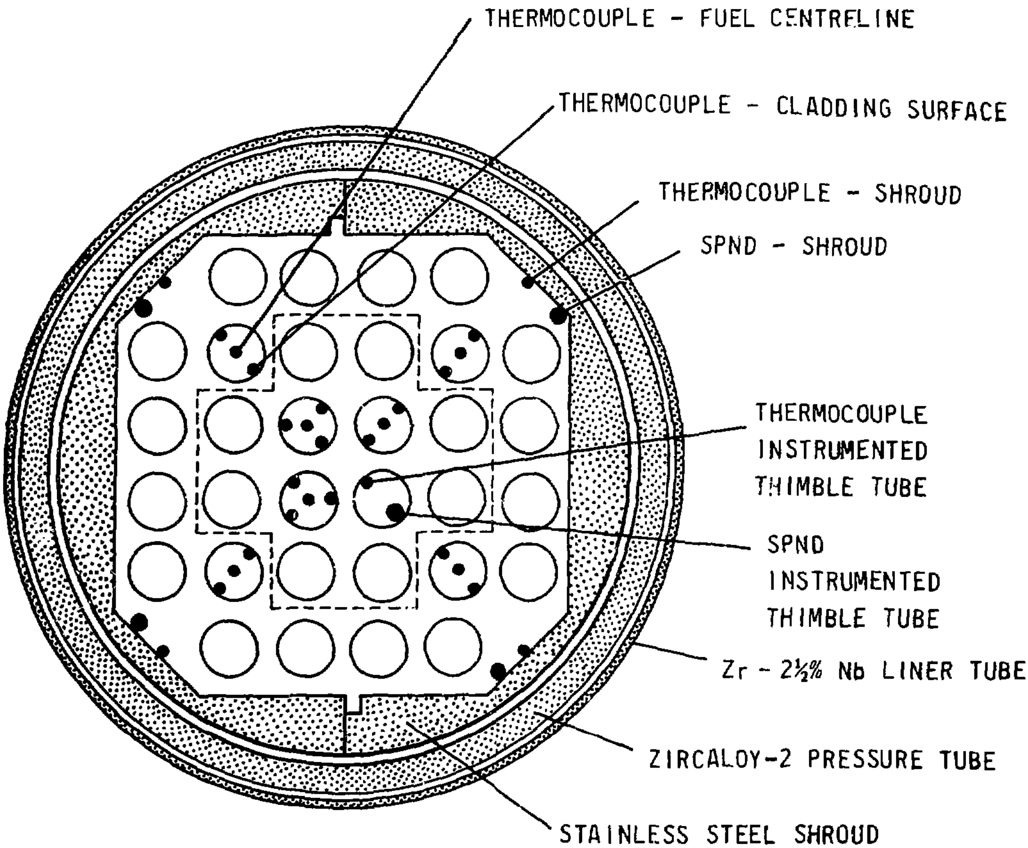

FIGURE 4: CROSS SECTION OF ASSEMBLY

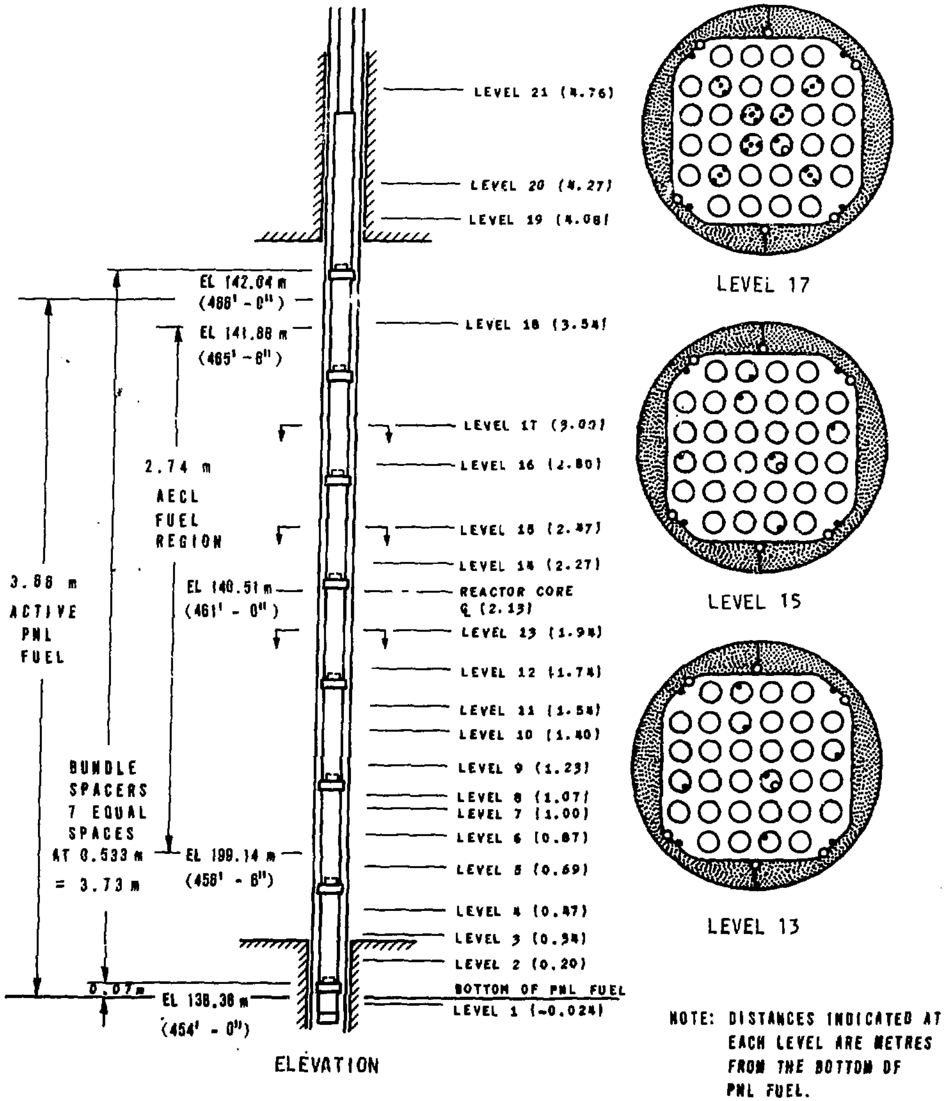

FIGURE 5: BATTELLE LOCA SIMULATION TEST SECTION INSTRUMENT LOCATIONS

There will be a small flow bypassing the test section, in the annulus between the outside of the shroud and the inside of the pressure tube, to cool the pressure tube. This bypass flow will not be measured during the tests. Out-of-reactor mockup hydraulic tests will be performed by Battelle to ensure that the bypass flow is adequate.

3.2.2.3 Temperature

All temperature measurements are made with thermocouples having Inconel-600 sheaths and chromel-alumel thermoelements insulated with magnesia. Thermocouple locations and numbers are indicated in Table 6.

TABLE 6: Test train thermocouples.

Location	Number of Devices	Purpose
Fuel rod		
Fuel centreline	7	Fuel temperature
Cladding ID	32	Fuel/cladding gap temperature
Cladding OD	27	Cladding temperature
Steam probes	18	Quench front location
Shroud	44	Temperature, axial and radial location of quench front
Thimble tube	8	Temperature, quench front location
Hanger bar	4	Pressure tube temperature protection

3.2.2.4 Pressure

In test numbers 2 to 6, where elements in the central test sub-bundles are pressurized to 3.1 MPa, internal element pressure will be measured on a pre-selected element with a Kaman model 1921 pressure transducer. The remaining elements in the test sub-bundle will contain on/off pressure actuated switches which will signal the reduction in fuel rod pressure, thereby signalling that a cladding failure has occurred.

3.2.2.5 Neutron Flux

Self-powered neutron detectors (SPND) with cobalt emitters are distributed radially and axially along the bundle to measure the neutron flux levels during the steady state and transient operation of the test. Total power generated during steady state operation will be determined by calorimetry. Fission power is obtained by correcting total power for the estimated gamma power component. The axial distribution is determined from the signal output from the axially distributed SPND's.

3.3 Reactor Power Estimate

The reactor powers required to obtain the test assembly power levels in the test bundles based on estimated reactor loading at the time are listed in Table 7. These values will be re-examined when the reactor loading for the test has been set.

TABLE 7: Reactor powers required to obtain required test bundle powers for LOCA tests.

Phase	Thermal Hydraulic Test		Materials Tests	
	Bundle Power (MW)	Reactor Power (MW)	Bundle Power (MW)	Reactor Power (MW)
Preconditioning	2.6	127	2.6	127
Pretransient	.14	9	.14	9

3.4 Test Loop

3.4.1 Test Loop Description

For the preconditioning phase the test section will be connected to U-2 loop and cooled by pressurized water. Measurement and control of conditions within the loop will be provided by U-2 loop instrumentation and equipment. Test bundle instrumentation signals as well as selected signals from the U-2 loop instrumentation will be connected to the DAS and test loop control panel.

On completion of the preconditioning phase the test section will be disconnected from the U-2 loop circuit and connected to the test loop circuit. Figure 6 illustrates the connect/disconnect arrangement between the two circuits.

The test loop circuit consists basically of two sub-circuits. A steam circuit provides cooling with dry steam during the pretransient phase with a quick steam shutoff to initiate the transient, and a reflood circuit provides an adjustable reflood flow to terminate the transient. The test circuit is a once through system with cooling flow from the source going via the test section to the loop catch tank.

During the pretransient phase superheated steam from the U-1 loop is fed to the bottom of the test section. When the transient is initiated the steam is isolated from the test section and dumped to the U-1 condensers.

SP# - SPOOL PIECE NO.

PRECONDITIONING CIRCUIT
SP#1,2,5,7,8,10,11,12,
13,14 REMOVED.
SP#3,4,6,9 INSTALLED.

TRANSIENT CIRCUIT
SP#3,4,6,9 REMOVED.
SP#1,2,5,7,8,10,11,12,
13,14 INSTALLED

DECONTAMINATION CIRCUIT
SAME AS FOR TRANSIENT
EXCEPT SP#7 REMOVED.

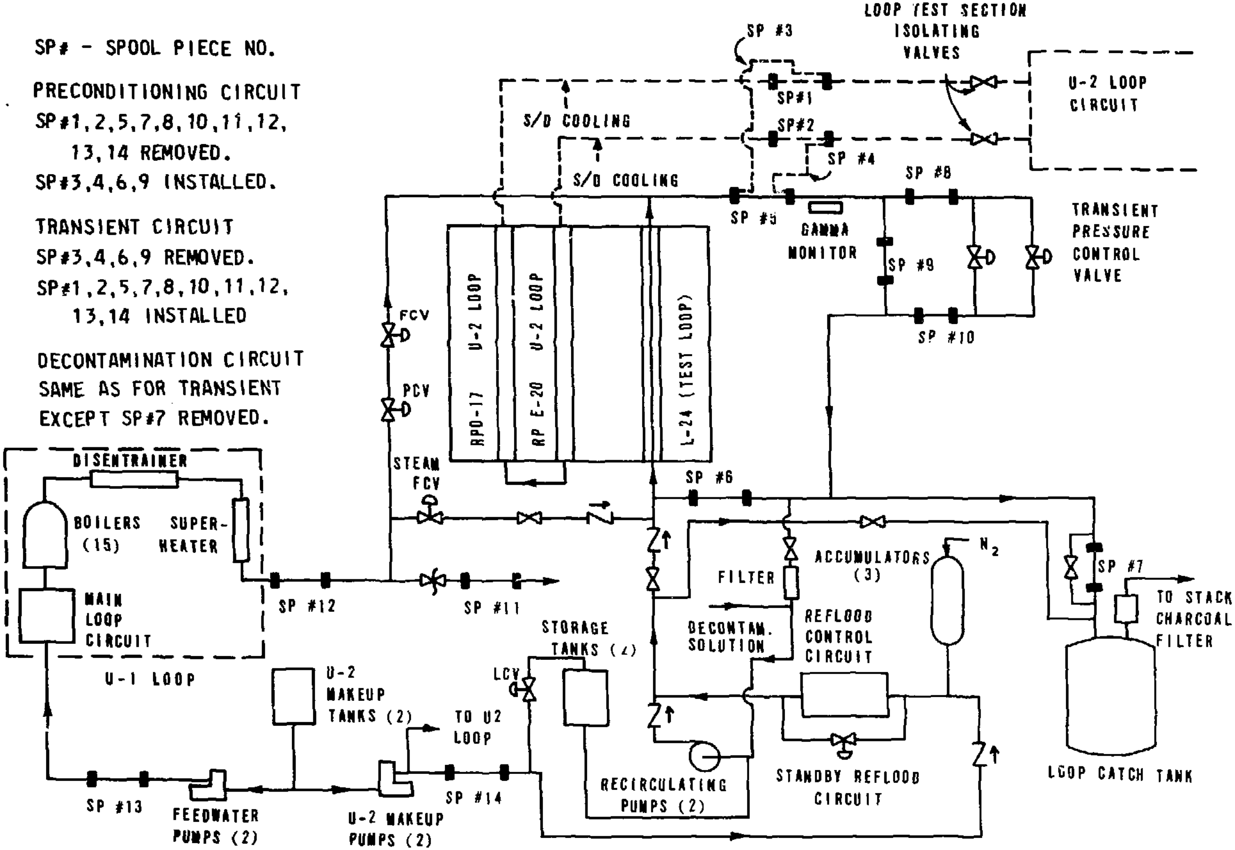

FIGURE 6: SCHEMATIC - BATTELLLE LOCA SIMULATION TEST FLOWSHEET

After a predetermined time delay water is introduced to the test section at a predetermined rate from the reflood circuit. This terminates the transient. Reflood water is supplied from accumulators in Room 222 which are pressurized with nitrogen. The rate of reflood flow is controlled by the programmed control system through fast acting control valves. A standby reflood circuit which bypasses the normal system is automatically actuated on failure of the normal reflood circuit. It can also be actuated manually if required. Failure of the standby reflood circuit will trip the reactor.

A strainer will be installed at the bottom of the test section to retain any fuel fragments released during the more severe transients. If any fuel fragments are carried over to the catch tank and the normal catch tank ion exchange system will not clean up the activity, the fuel fragments will be removed by dissolving them and then storing the resulting solution in tanks until such time as the Waste Treatment Centre is operational. The volume of liquid involved should be quite small and special tanks are available for this purpose. This procedure has been discussed with the CRNL Environmental Authority and is acceptable to them.

Should it be necessary to decontaminate the piping after a test, provision has been made to recirculate decontamination solution through the test section and piping, as shown in Figure 6. The decontamination solution would be transferred to the catch tanks in Room 110, and then to special tanks prior to processing at the Waste Treatment Centre.

As discussed in Section 5.1.2, charcoal filters will be installed on the loop catch tank vent line to the reactor stack, to retain any radioiodines released during the transients. A schematic of the loop catch tank ventilation system, as modified for the Battelle LOCA simulation tests, is shown in Figure 7.

3.4.2 Test Loop Control Circuit

3.4.2.1 Pretransient Phase

Steam at 0.38 kg/s will be provided from the U-1 loop boilers via a flow control valve to maintain the sheath temperature at about 430°C [1]. The steam from the loop at about 2.0 MPa will be dried by passing it through the U-1 superheaters. The outlet pressure from the test section will be controlled at 0.28 MPa by feeding a small quantity of bypass steam flow to the outlet of the test section. The test section outlet pressure will be regulated by varying the size of the restriction in the outlet line (via air operated pressure control valves) to maintain a constant test section outlet pressure under varying flow conditions. The total steam flow during the pretransient will be steady. The U-1 boiler controls are listed in Table 8.

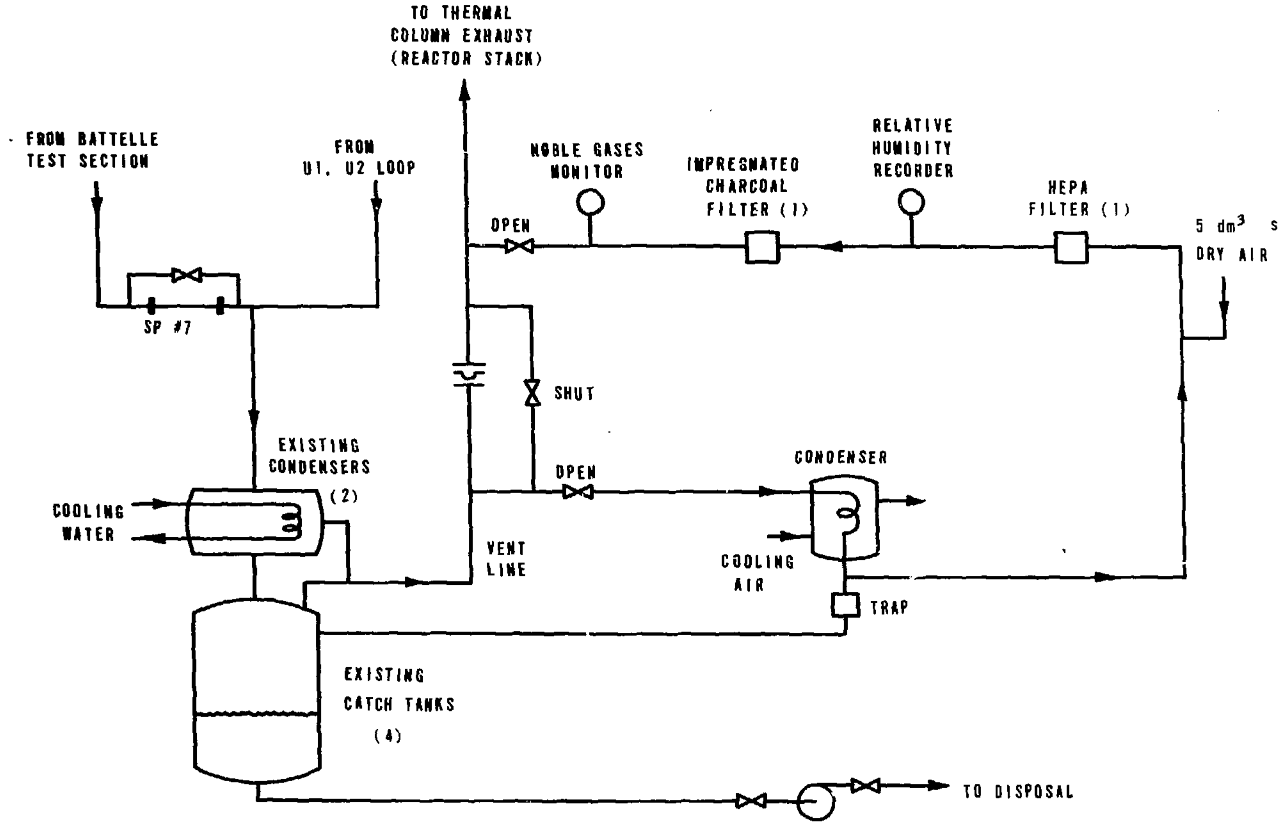

FIGURE 7: SCHEMATIC - LOOP CATCH TANK VENTILATION SYSTEM FOR BATTELLE LOCA SIMULATION TESTS

TABLE 8: **Boiler controls.**

Feed water	-	0.28 to 0.40 kg/s controlled on steam flow to the test section.
Pressure	-	0 to 2.0 MPa controlled by modulating boiler heater power in the U-1 loop.
Superheat	-	6 to 22^{o}C controlled by modulating the electrically heated superheaters in the U-1 loop.
Steam flow	-	0.09 to 0.38 kg/s to the test section inlet. 0 to 0.025 kg/s to the test section outlet.

3.4.2.2 Transient Phase

The transient phase will be controlled from an automatic sequence programmable controller [1]. The control variables (time delay to reflood and reflood rate) will be entered into the controller before each test. The transient start push button starts the automatic sequence when pushed. When the transient is initiated the main steam flow to the test section will be shut off and a steam bypass circuit in U-1 loop will open fully to dump the excess steam to the U-1 condensers. Reflood water to the test section will be supplied from the three accumulators via fast acting solenoid valves (opening time of 45 ms). Operation of the valves is governed by the predetermined delay time and reflood rate. Flow rates will be measured and control signals fed to the control valve by turbine type flowmeters (response time 4 to 6 ms). The flow rates will be indicated and recorded in the DAS control room. A standby reflood system will be automatically actuated if the normal reflood system fails. Failure of the standby system will cause a reactor trip. The reflood control system specifications are listed in Table 9.

TABLE 9: **Reflood controls.**

Fast fill rate		-	up to about 1.07 kg/s (valve approx. wide open) for 2 s
Reflood rate	High	-	up to 0.84 kg/s for up to 20 s.
	Low	-	0.04 to 0.19 kg/s for up to 300 s.
Accumulator pressure		-	3.4 MPa maximum.
Accumulator pressure control		-	Nitrogen gas pressure through regulating valve.
Accumulator heater control		-	38 to 65^{o}C ± 3^{o}C ON-OFF control.
Total accumulator capacity		-	227 kg

On completion of the transient the reactor will be shut down manually unless actuation of the accumulators low inventory trip, reflood low flow trip, transient termination time trip or high sheath temperature trip shut it down automatically.

3.5 Data Acquisition System

Each instrument read by the DAS will be immediately recorded on magnetic tape and magnetic disk concurrently. While reading and recording, a low nominal scan rate of 10 samples/second will be used in the steady state mode, and a high scan rate nominally set at 40 samples per second will be used in the transient mode. If an instrument is recorded by the DAS, its immediate or any historical value in engineering units may be displayed on the control console cathode ray tube or the computer line printer, or it may be graphically presented on the graphic terminal or graphic hardcopy unit. The list of instruments which will be recorded and which could be displayed by the DAS is given in Table 10.

TABLE 10: DAS monitored instruments.

Reactor Loop
Coolant inlet and outlet temperatures.
Coolant differential and outlet pressures.
Main loop coolant flow rate.
Steam flow rate.
Steam pressure.
Steam temperature.
Reflood flow rate.
Reflood coolant temperature.
Test Assembly
Cladding temperatures.
Shroud temperatures.
Local coolant temperatures and differential temperatures.
Local neutron fluxes.
Fuel center line temperatures.
Internal fuel rod pressures.
Thimble tube temperatures.
Hanger bar temperature above fuel assembly.

4. SAFETY

4.1 Design Procedures

The following design procedures will be used for new components and loop modifications necessary for conducting the Battelle LOCA simulation tests:

- Vessels in the low pressure steam and reflooding circuit will be designed and manufactured to the ASME Boiler and Pressure Vessel Code, Section VIII, Division I.

- All piping will be designed to ANSI Power Piping Code B31.1.

- The fuel string sealing block will be designed to the ASME Boiler and Pressure Vessel Code, Section III, Class 1.

- The cold-worked Zircaloy-2 pressure tube will be designed specifically to meet the test requirements. Zircaloy-2 was chosen for the pressure tube rather than Zr-2.5 wt% Nb due to its superior high temperature creep properties and also to avoid distortions during the reflood portion of the tests. The safety factor of the nominal 0.508 cm wall thickness pressure tube is greater than 3 based on burst strength vs. pressure relief stress. As is the case with all NRU in-reactor loops, the pressure tube will be surrounded by a cold-worked Zr-2.5 wt% Nb liner which would protect the reactor if the pressure tube should fail.

Fuel assembly components designed and manufactured by Battelle have undergone a rigorous analysis as indicated in Table 11. Battelle have a quality assurance program in place which will be used during the design, procurement, fabrication, construction and assembly of each fuel assembly that will be tested in NRU. This plan governs all activities that affect product quality commencing with design and development and continuing through procurement, materials handling, fabrication, testing, inspection, storage, and shipment [2].

4.2 Trips and Alarms

The following trips will be in service during the various phases of the test.

(a) Preconditioning Phase:

During this phase the test section will be connected to the U-2 loop and the following standard U-2 reactor trips will be in service.

Test section outlet temperature high.
Pump sub-cooling ΔT low.
Test section water flow low.
Surge tank level low.
Surge tank pressure high.
Manual.

In addition the following two reactor trips will be in service.

Pressure Tube High Temperature: Thermocouples at the bottom of the fuel string hanger bar (level 21 on Figure 5) will actuate the trip to protect the pressure tube from overheating.

TABLE 11: REQUIRED STRUCTURAL INTEGRITY EVALUATION.

COMPONENTS, SUBASSEMBLIES	STRESS							VIBRATION - HIGH CYCLE FATIGUE			LOW CYCLE FATIGUE	OXIDATION	EROSION	THERMAL EXPANSION	FIT-UP	LOAD LIMITS
		OPERATIONAL														
		MECHANICAL			THERMAL											
	HANDLING	PRE-COND.*	PRE-TRANS.**	TRANSIENT AND REFLOOD	PRE-COND.	PRE-TRANS.	TRANSIENT AND REFLOOD	PRE-COND.	PRE-TRANS.	TRANSIENT AND REFLOOD						
TOTAL TEST TRAIN ASSEMBLY	●													●	●	●
CLOSURE REGION		●	●	●	●	●	●		●		●			●	●	●
INLET REGION								●	●					●	●	
HANGER	●	●	●	●			●	●	●		●	●		●	●	●
SHROUD	●	●					●		●		●	●		●	●	●
SPACER GRIDS	●								●		●			●	●	
FUEL RODS	●			●			●		●			●		●	●	●
FEED THRUS		●			●	●	●				●			●	●	
BUNDLE REGION	●								●			●		●	●	●
INSTRUMENTS (THERMOCOUPLE PROBES, SPND's)					●	●	●	●	●	●		●	●	●		

* PRECONDITIONING

** PRETRANSIENT

Outlet Piping High Temperature: Thermocouples on the outlet piping will trip the reactor to minimize thermal expansion stresses in the outlet piping.

(b) Pretransient Phase:

During this phase the test section will be connected to the steam-reflood water supply and will discharge to the loop catch tanks in Room 110. The following four trips will be in service:

Pressure tube high temperature.
Outlet piping high temperature.
Low steam flow. This trip will actuate if steam flow drops below an acceptable level during the pretransient phase. The trip signal will come from the steam control circuit logic.
Manual.

(c) Transient Phase:

This phase will immediately follow the pretransient phase with the following seven trips in service:

Pressure tube high temperature.
Outlet piping high temperature.
Fuel cladding high temperature. The reactor will trip if the temperature of the fuel cladding at the hottest point exceeds a specified temperature. The trip signal will be computed by the DAS. Allowance will be made to compensate for the difference between the maximum temperature and the measured temperature.
Reflood low flow. This trip will actuate if the normal reflood fails and the standby reflood flow is insufficient. The trip signal will be actuated from the reflood control circuit logic.
Accumulators low inventory. This trip will automatically shut the reactor down when the accumulators are nearly empty.
Transient termination time. This trip will automatically shut the reactor down at a predetermined time for each transient, if shutdown has not occurred by other means.
Manual.

All the above reactor trip circuits will be triplicated, but five of them (low steam flow, fuel cladding high temperature, reflood low flow, accumulators low inventory, transient termination time) will be single sensor trips in that a single signal will actuate the triplicated trip.

Alarms are also provided for the test loop as listed in Table 12.

TABLE 12: Test circuit alarms.

Sub-system	Alarm
Feedwater pumps	Power off.
Steam pressure	High.
	Low.
Accumulators	High temperature.
	Heater tripped.
	Pressure low.
	Inventory low.
Test section coolant	Outlet pressure high.
	Outlet temperature high.
Reflood	Fast reflood - low flow.
	Slow reflood - low flow.
Standby cooling	On.
Programmable controller	Fault.
Reactor	Tripped.

4.3 Reactivity Considerations

4.3.1 Reactivity Effect of Fuel String Assembly

Due to the thickness of the stainless steel shroud surrounding the fuel, the fuel string assembly will have a negative reactivity effect and will act as a load on the reactor.

4.3.2 Voiding of the Test Section

The worst situation from a reactivity ramp point of view would be voiding of the test section during the preconditioning phase with the reactor at full power and the test section full of pressurized hot water. The estimated maximum reactivity value of this hot water is 2.6 mk. Since this is below the approved limit [3] for allowable reactivity holdup due to light water loop coolant these tests are satisfactory from a reactivity point of view.

4.3.3 Relocation of the Fuel

Significant dispersal of the fuel within the test loop is not likely to occur during the test. If dispersal did take place the limiting case would involve the compaction of the fuel within the shroud. This compaction would result in an estimated reactivity increase of 5.8 mk above the steam cooled condition. Since this dispersal and compaction would necessarily take a significant length of time the rate of reactivity increase would be well within the range of the NRU control and safety system's ability to safely cope.

5. HAZARDS

5.1 Normal Operation

5.1.1 Reflood Effectiveness

The largest body of information bearing on fuel rewetting or quench is that of the Westinghouse FLECHT experimental series. Reports have been written that describe the experiments and results that cover the same range of reflood rates as in the tests proposed for NRU [2]. The FLECHT tests used a 10 x 10 array of 3.66 metre long electrically heated elements as a representation of a 15 x 15 element bundle. Qualitatively the tests showed a brief period after reflood was initiated at the bottom of the fuel when the temperature in the upper regions continued to rise. However, steam caused by vigorous boiling of the reflood water in the lower sections of the fuel then resulted in some cooling in the upper sections and the temperature rise began to decrease. The temperature then peaked and began decreasing before the quench front arrived at the upper sections of the fuel. The tests showed the adequacy of the rewetting to cool the fuel elements. They also provided the basis of correlations that quantitatively describe its occurrence. The basic differences between the tests are listed in Table 13.

TABLE 13: Difference between Battelle LOCA simulation test and FLECHT tests.

Item	Battelle	FLECHT
Heat source	Nuclear	Electrical
Rod cladding	Zircaloy	Stainless steel
Peak-to-average axial power distribution	1.51	1.66
Number of elements (array)	32 (6 x 6)	100 (10 x 10)

In addition, pretransient steam cooling in the Battelle tests will distort the initial temperature distribution. Also, fuel elements in the materials tests are expected to distort to a significant degree.

The Battelle tests should show as effective rewetting as the FLECHT tests; however, the timing of the temperature increase, turnaround and quench may be different. The stepwise approach to the performance of the tests should avoid the problem of excessive temperatures up to the time quench occurs.

The maximum temperatures expected in the Battelle LOCA simulation tests as determined by Battelle are shown in Figure 2. These temperatures were determined for conditions that would give peak cladding temperatures somewhat higher than the maximum planned (980^{o}C).

5.1.2 Radioactivity Releases

No fuel cladding failure is anticipated during the thermal hydraulic test and therefore no releases are expected. During the materials tests significant fuel cladding failure is expected. Because of the low fuel burnup for these tests the quantity of iodines and noble gases released is expected to be less than 1% of the fission product inventory [4,5]. These releases will be vented to the atmosphere via the catch tank vent and reactor stack. An impregnated activated charcoal filter will be installed on the loop catch tank vent line to reduce any radioiodines released to an acceptable level (see Figure 7). Prior to each test the charcoal filter will be tested to ensure that its efficiency is no lower than 99.5%. A HEPA filter will be installed in the vent line upstream from the charcoal filter. A cooler to condense any moisture in the air will also be installed. A continuous flow of about 5 dm^3/s of dry air will be drawn through the charcoal while it is in service, to further ensure dry conditions. The relative humidity of the air entering the charcoal filter will be recorded.

A monitor will be installed downstream from the charcoal filter to record the amount of noble gases released. Any I-131 released will be detected by the normal stack monitoring system.

The fission product inventory, the amount releasable and the amount released to the atmosphere are shown in Table 14. The figures shown are conservative because:

(a) They assume 1% of fission product inventory releases whereas actual releases should be less than 1%.

(b) They assume releases based on all 31 elements whereas only the centre 11 elements are expected to fail.

(c) There is no allowance for any plateout.

TABLE 14: Estimated radioactive releases for materials tests.

Fission Product	Inventory	Releasable	Released
Equivalent I-131 (Ci)+	317	3.17	.032*
Noble Gases (Ci·MeV)	3140	31.40	31.4

+ 1 Ci = 37 GBq.
* Assuming charcoal filters in the vent line will reduce release by 99%.

The estimated iodine release represents about 0.5% of the weekly derived release limit for I-131. The estimated noble gas release is insignificant when compared with the derived release limit.

5.2 Accidents

5.2.1 General

Accidents considered for the hazards analysis include a loss of coolant accident during the preconditioning phase and three failures of reflood accidents during the transient phase. The first reflood accident postulated assumes there is no reflood at the specified time and that a total delay of 80 seconds from time of steam shut-off occurs before the reactor trips. In the second case it is assumed there is no reflood at the specified time and that a total delay of 80 seconds from time of steam shut-off occurs before standby reflood is introduced. In the third case a partial reflood is considered, which will result in high outlet steam temperatures and some metal water reaction.

5.2.2 Loss of Coolant During Preconditioning

In this case the concern is during the preconditioning phase when the reactor is at full power and the test section is connected to the U-2 loop. The sudden loss of coolant with a resulting increase in reactivity is similar to the case already addressed for U-2 loop, but much less severe. There will be only one test section instead of two and the fuel will be a low burnup fuel of approximately 0.05 MWh/kg versus up to 200 MWh/kg burnups for an average reactor test fuel string. The operating period will be of short duration and closely monitored. The normal loop automatic emergency cooling system will be in service during this portion of the test.

5.2.3 Loss of Reflood and Trip

In this case it is assumed that the high temperature transient is underway, that normal or standby reflood does not come on as scheduled and that a total time delay of 80 seconds from time of steam shut-off occurs before the reactor trips. It is noted that in actual fact the reactor will trip automatically on failure of standby reflood at the end of the specified delay period if reflood does not occur. Since the specified delay times vary from 3 to 77 seconds, a trip, if required, should always have occurred at no later than 80 seconds. The thermal power of the test fuel will drop as illustrated in Figure 8. This thermal power is based on the measured neutron decay following a conditional trip in addition to the fission product decay power as calculated for the accumulated power history of the fuel.

Test assembly temperatures calculated by Battelle using the TRUMP code are illustrated in Figure 9. The cladding temperature stops increasing about 20 seconds after the trip and a peak of 1070°C is reached. Heat is removed through the shroud and pressure tube to the D_2O moderator. The shroud and pressure tube temperatures in the area of the fuel will continue to increase for 8 or 10 minutes after the reactor trip, peaking at about 500°C and 250°C, respectively. The corresponding fuel centreline temperature will be about 1085°C.

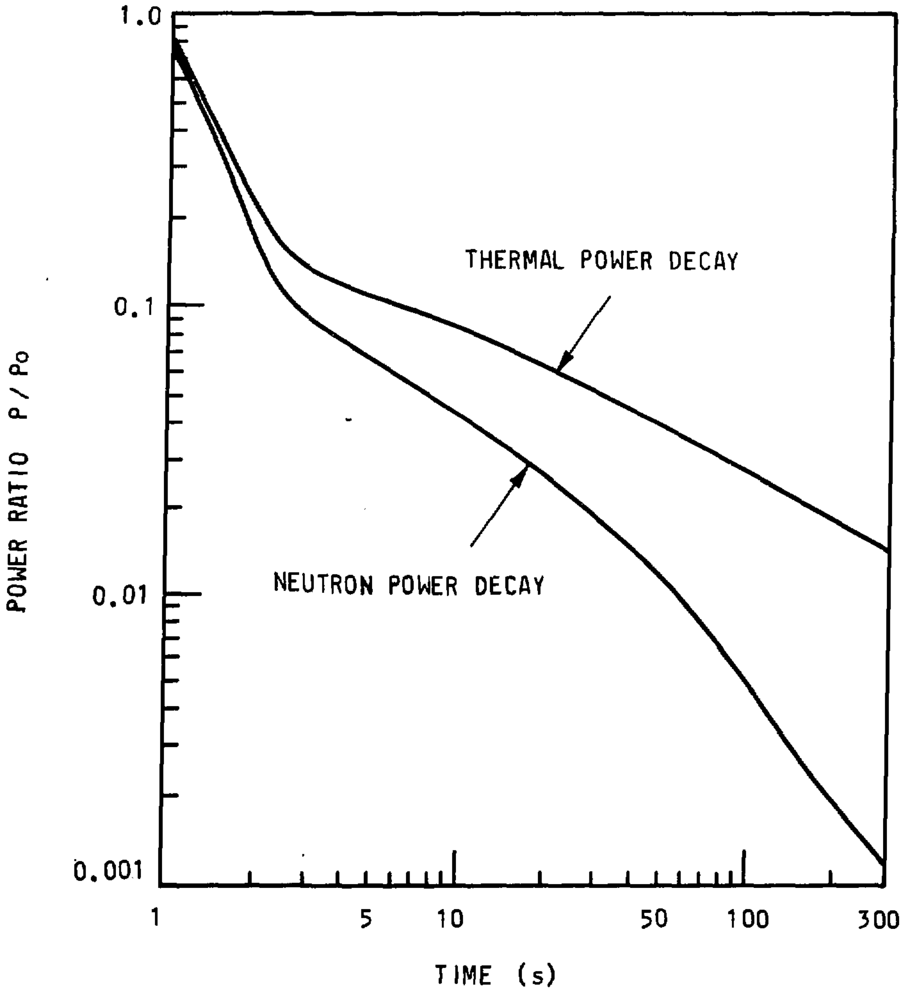

FIGURE 8 : POWER DECAY AFTER TRIP

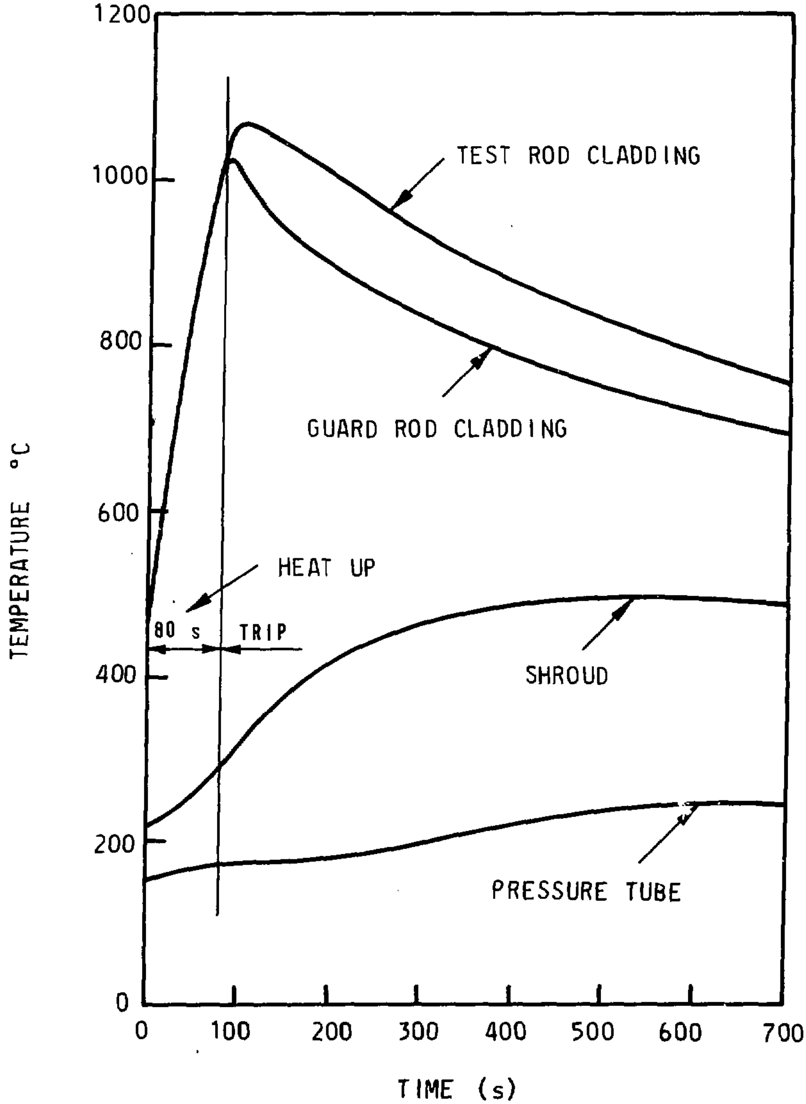

FIGURE 9: TEST ASSEMBLY TEMPERATURES AFTER A TRIP WITH NO REFLOOD

With no steam flow as a result of no reflood, the calculated pressure tube temperature above the fuel will drop off as is illustrated in Figure 10. The initial pressure tube temperature of 371°C used for the calculated temperatures shown in Figure 10 is conservatively high, since it corresponds to an average rod power approximately 30% higher than that expected in the actual test.

5.2.4 Loss of Reflood and Standby Reflood Initiated

In this case it is assumed that normal reflood does not come on after the specified time delay and that a total time delay of 80 seconds from time of steam shut-off occurs before standby reflood is introduced at a rate of 18 cm/s, with no reactor trip. Since the specified delay times vary from 3 to 77 seconds, standby reflood, if required, should always have occurred at no later than the 80 seconds considered here. Here again it is noted that in actual fact the reactor will trip automatically on failure of standby reflood at the end of the specified delay period if reflood does not occur.

Data from FLECHT tests (see Figure 11) show that for a reflood rate of 18 cm/s with highly sub-cooled water, cladding temperatures will rise about 45°C. If standby cooling is introduced at the 80 second point in Figure 9 a peak cladding temperature of 1090°C will be reached. The corresponding fuel centreline temperature will be about 1105°C. The FLECHT test data also shows that for reflood rates similar to this, steam exiting from the test assembly has a temperature not much greater than saturation, and therefore it will cool down the pressure tube above the fuel assembly faster than in the no reflood case.

5.2.5 Partial Loss of Reflood

The metal-water reaction (Zr-H_2O) at the elevated temperatures possible in this test will contribute to test heating. The net contribution will be the difference between the heat addition by the reaction and the heat removal by the water available for the reaction. Normally this value is negative. The basic concern is that under accident conditions this net contribution might be positive and large enough to cause assembly temperatures to continue to rise after a reactor trip.

Flecht test data indicate that conditions are most severe at very low reflood rates of about 1.2 cm/s. For this accident case it is assumed that the reflood is initiated after the transient but the reflood rate is one half the programmed rate at a level just above the standby reflood initiation point. The reactor will trip when the cladding temperature reaches 1200°C.

For the analysis Battelle considered accident reflood rates of 2.5 cm/s and 1.2 cm/s. The GAPCON-3 computer code was used to compute the value of the element power contributed by the energy release from the Zr-H_2O reaction. The 1.2 cm/s reflood rate case was found to be the most severe. Both Baker-Just and Cathcart correlations were used for comparison and the results are illustrated in Figure 12.

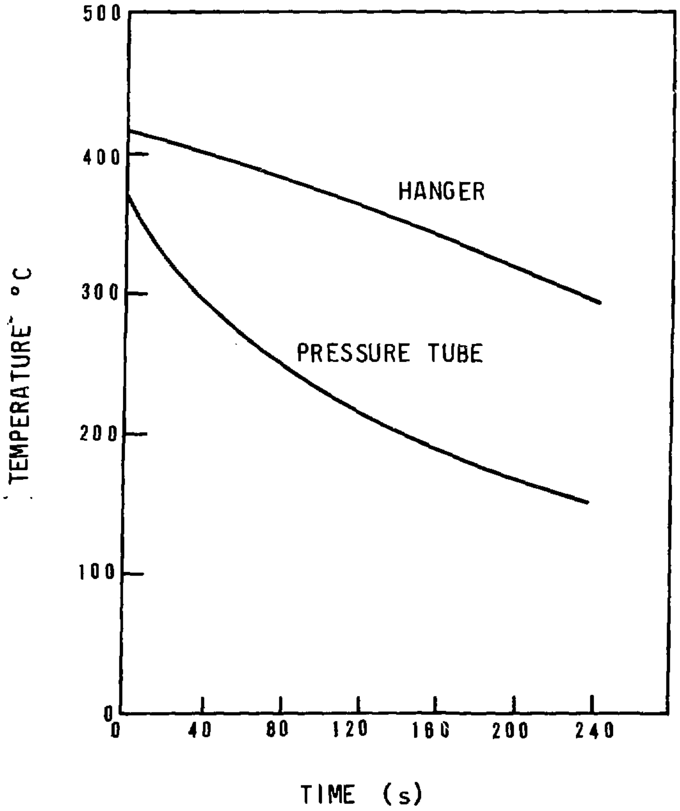

FIGURE 10: PRESSURE TUBE TEMPERATURE ABOVE TEST ASSEMBLY WITH NO STEAM FLOW AFTER A TRIP.

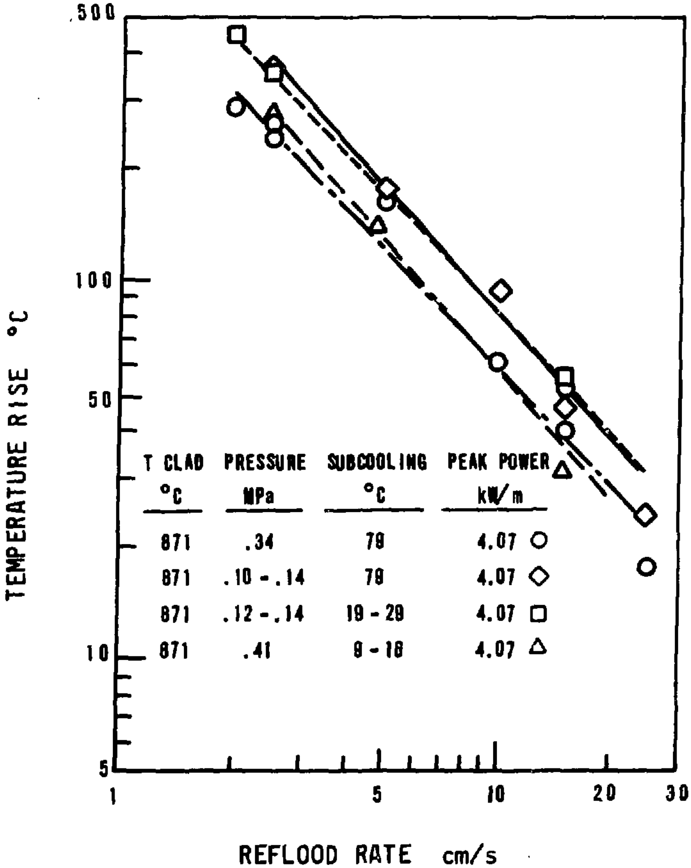

FIGURE 11: EFFECT OF REFLOOD RATE ON TEMPERATURE RISE

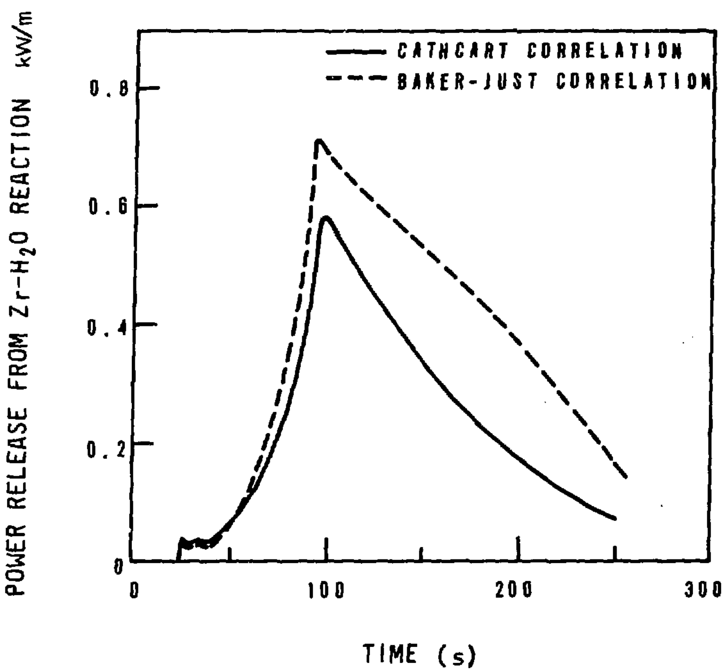

FIGURE 12: CONTRIBUTION OF METAL-WATER REACTION TO ROD POWER

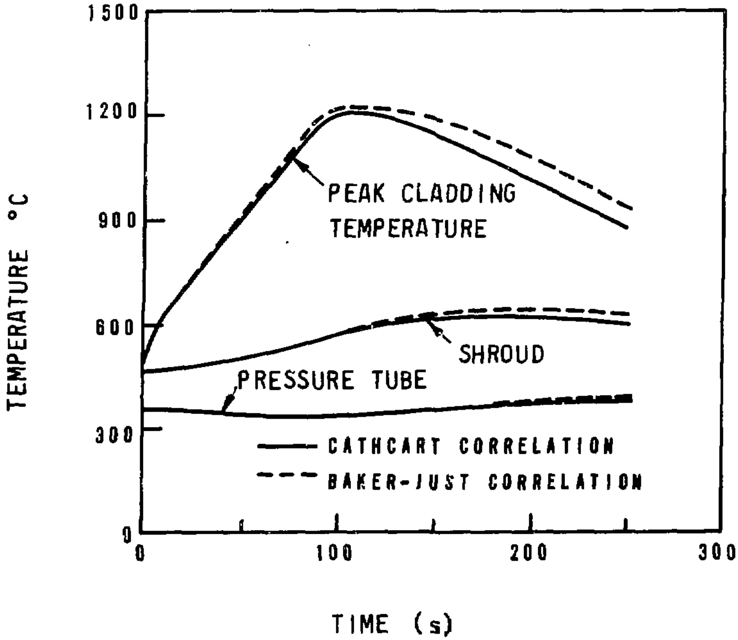

FIGURE 13: TEST ASSEMBLY TEMPERATURES FOR 21 s HEATUP TO 700°C, 1.27 cm/s REFLOOD, AND TRIP AT 1200°C

The results were incorporated into input to the TRUMP computer code to calculate test assembly temperatures during the accident. The results are illustrated in Figure 13.

As is indicated the peak cladding temperature rises only slightly above 1200°C (the temperature at which the trip occurs) before turning around. The shroud and pressure tube temperatures continue to increase for a short period of time before turning around at a level well below any problem temperatures.

5.2.6 Accident Radioactivity Releases

Since the tests are designed to cause fuel cladding failure the release of radioactivity has been anticipated and charcoal filters have been installed in the loop catch tank vent line to limit the releases of radioiodine to less than 1% of the derived release limit. The greatest release would occur in an accident case in which all 31 fuel sheaths experienced failure in the last thermal hydraulic test when fission product inventory is highest. The estimated releases are as indicated in Table 15.

TABLE 15: Estimated accident radioactive releases.

Fission Product	Inventory	Releasable	Released
Equivalent I-131 (Ci)+	610	6.1	0.061*
Noble Gases (Ci·MeV)	12370	123.7	123.7

+ 1 Ci = 37 GBq.
* Assuming charcoal filters in the vent line will reduce release by 99%.

The conservative assumptions made in Section 5.1.2 for the radioactive release are applied in this case as well. Because of the low burnup of the fuel and the resultant low fission product inventory the amount released in this accident condition is insignificant. Estimated I-131 release represents 1% of the weekly derived release limit for stack releases. The estimated noble gas release is insignificant when compared with the derived release limit.

6. SUMMARY AND CONCLUSIONS

The proposed LOCA simulation test series for NRU has been reviewed and found to be acceptable from a safety point of view.

The uncertainties inherent in this type of an experimental program have been compensated for by a stepwise approach from moderate to more severe operating conditions from test to test. Measured results from each test will influence the operating conditions for the succeeding test.

The safety systems installed are adequate to ensure that the safety of the reactor and operating personnel are not jeopardized.

On this basis it was concluded that the operation of the proposed Battelle LOCA simulation tests in NRU reactor do not present an unacceptable risk to the safety of the reactor, operating personnel or members of the general public.

7. REFERENCES

[1] Kendrick, S.W., "Control System for the NRC Transient Tests", unpublished report AECL-MISC-205.

[2] Mohr, C.L. and the Battelle Pacific Northwest Laboratory NRU Project Staff, "Loss-of-Coolant Accident Simulations in the National Research Universal Reactor - Safety Analysis Report", 1981 February, NUREG/CR-1208, PNL-3093.

[3] "Principles and General Rules for the Operation of AECL Nuclear Facilities", unpublished report AECL-MISC-191.

[4] Lorenz, R.A., D.O. Hobson and G.W. Parker, "Final Report on the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT", ORNL-4635, 1971, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

[5] Hastings, I.J. and M.J.F. Notley, "Prediction of Fission Gas Release from High Burnup Oxide Fuel", AECL-6682, 1979 November.

ISSN 0067 - 0367

To identify individual documents in the series we have assigned an AECL- number to each.

Please refer to the AECL- number when requesting additional copies of this document

from

Scientific Document Distribution Office
Atomic Energy of Canada Limited
Chalk River, Ontario, Canada
KOJ 1JO

Price $3.00 per copy

ISSN 0067 - 0367

Pour identifier les rapports individuels faisant partie de cette série nous avons assigné un numéro AECL- à chacun.

Veuillez faire mention du numéro AECL- *si* vous demandez d'autres exemplaires de ce rapport

au

Service de Distribution des Documents Officiels
L'Energie Atomique du Canada Limitée
Chalk River, Ontario, Canada
KOJ 1JO

Prix $3.00 par exemplaire