NUREG/CR-2599
PNL-4190

Experiment Operations Plan for the MT-3 Experiment in the NRU Reactor

Prepared by G. E. Russcher, C. L. Wilson, L. J. Parchen, M. D. Freshley

Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory
Commission

NOTICE

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, or any of their employees, makes any warranty, expressed or implied, or assumes any legal liability of responsibility for any third party's use, or the results of such use, of any information, apparatus, product or process disclosed in this report, or represents that its use by such third party would not infringe privately owned rights.

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555
2. The NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission,
   Washington, DC 20555
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the NRC/GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

GPO Printed copy price: $3.75

NUREG/CR-2599
PNL-4190
R3

Experiment Operations Plan for the MT-3 Experiment in the NRU Reactor

Manuscript Completed: May 1983
Date Published: June 1983

Prepared by
G. E. Russcher, C. L. Wilson, L. J. Parchen, M. D. Freshley

Pacific Northwest Laboratory
Richland, WA 99352

Prepared for
Division of Accident Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN B2277

ABSTRACT

A series of thermal-hydraulic and cladding materials deformation experiments were conducted using light-water reactor fuel bundles as part of the Pacific Northwest Laboratory Loss-of-Coolant Accident (LOCA) Simulation Program. This report is the formal operations plan for MT-3--the third cladding materials deformation experiment conducted in the National Research Universal (NRU) reactor, Chalk River, Ontario, Canada. The major objective of MT-3 was to simulate a pressurized water reactor LOCA that could induce fuel rod cladding deformation and rupture.

SUMMARY

The Loss-of-Coolant Accident (LOCA) Simulation Program was conducted by Pacific Northwest Laboratory (PNL) to evaluate the thermal-hydraulic and mechanical deformation behavior of full-length light-water reactor (LWR) fuel bundles under LOCA conditions. The test conditions were designed to simulate the heatup, reflood, and quench phases of a large-break LOCA, and the tests were performed in the National Research Universal (NRU) reactor using nuclear fission to simulate the low-level decay power that is typical of these conditions.

The formal experiment operations plan for the third materials test (MT-3) is presented in this document. The experiment simulated a LOCA with peak cladding temperatures from 1033 to 1089K (1400 to 1500°F) for up to 200 s. To assure the successful operation of the loop steam and reflood control system, a preliminary thermal-hydraulic test (TH-3) was conducted to verify the system and to replicate the operating conditions produced in the previous materials experiment (MT-2).

An approved draft copy of the experiment operations plan for TH-3 and MT-3 was sent to Chalk River Nuclear Laboratories (CRNL), Chalk River, Ontario, in late October 1981; and the TH-3/MT-3 test series was conducted from November 9 through November 14, 1981. This report represents the formal documentation of the experiment operations plan for MT-3.

CONTENTS

TABLES

INTRODUCTION

The Loss-of-Coolant Accident (LOCA) Simulation Program was conducted in the National Research Universal (NRU) reactor at Chalk River Nuclear Laboratories (CRNL),[a] Chalk River, Ontario, Canada, by Pacific Northwest Laboratory (PNL).[b] The program was sponsored by the U.S. Nuclear Regulatory Commission (NRC) to evaluate the thermal-hydraulic and mechanical deformation behavior of full-length, 3% enriched light-water reactor (LWR) fuel rod bundles during the heatup, reflood, and quench phases of a LOCA. Low-level nuclear fission heat was used to simulate the decay heat in the fuel and the cladding that is typical of a LOCA.(1)

Several thermal-hydraulic and cladding materials deformation experiments were conducted. Previous experiments emphasized higher temperature LOCA fuel failure. The second thermal-hydraulic test (TH-2) characterized peak cladding temperatures for a series of reflood delay times and variable reflood flow rates. The test assembly used for TH-2 was also used in another test (TH-3) to verify the loop control system (LCS) instrumentation and operation. In addition, TH-3 verified test assembly radial power profiles, repeatability of TH-2.14, and preprogrammed control performance; it also allowed tuning the LCS for first and second derivative (reflood flow rate) control.

This report represents the formal experiment operations plan for the third materials test (MT-3), which was performed in November 1981. The experiment was jointly funded by the NRC and the United Kingdom Atomic Energy Authority (UKAEA). A new test assembly with pressurized rods was preconditioned and tested using the same boundary conditions as the TH-2.14 test. The data acquisition and control system (DACS) and the LCS controlled the reflood flow to produce peak cladding temperatures of 1033 to 1089K (1400 to 1500°F) for up to 200 s. The 12 fuel rods were pressurized with helium to 3.10 MPa (450 psi) to provide cladding ballooning and rupture.

The remainder of this report consists of:

- a summary of the design differences between the MT-2 and MT-3 experiments
- a safety hazards statement

(a) Operated by Atomic Energy of Canada Ltd. (AECL).

(b) Operated for the U.S. Department of Energy (DOE) by Battelle Memorial Institute.

- the operating procedure for MT-3
- a sample Test Parameter Log (Appendix A) and an Experiment Conditions Log (Appendix B).

DESIGN DIFFERENCES

Because the new unirradiated MT-3 assembly was quite similar to the MT-2 assembly, only design differences are described in this report. Detailed descriptions can be found in the safety analysis report(2) and the experiment operations plan addendum for MT-1 and MT-2.(3) The NRU reactor and LOCA systems were basically unchanged.

MT-3 TEST ASSEMBLY

Because the MT-3 test assembly was new, all instrumentation was operational.(2) The only instrumentation differences were 1) the addition of two thermocouples (TCs) to fuel rod braze junctions at Level 19, 2) the addition of two TCs on the hanger tube at Level 20, and 3) the substitution of a test fuel rod for the instrument tube with its associated TCs and neutron detectors.

A desuperheater system that was used to insure the integrity of fuel rod instrument brazes consisted of a water supply and spray nozzles that directed cooling sprays of water onto the instrument braze region. CRNL provided a water supply system from the accumulators to the top of the reactor, where water supply requirements were 3.10 MPa (450 psi) with a maximum flow of 0.0163 kg/s (129 lbm/h). A single 0.00635-m (0.250-in.) line carried water through the reactor deck plate to the top of the breach block. Before the water line entered the breach block, it was connected to a Swagelok® assembly containing a check valve and a CONAX 5 probe gland. The CONAX gland provided a transition seal from the single line to five 0.00159-m (0.0625-in.) lines. The five lines penetrated the breach block through the C-position seal and extended down to the shroud upper region where they were directed through five major fuel assembly subchannels. The bottoms of the spray lines were attached to the upper spacer grid, and a series of holes extending over several inches of the lines constituted the "nozzle" that directed atomized water onto the fuel rod instrument braze region. A description of the desuperheater system is shown on PNL drawing H-3-52017.

A test fuel rod bundle with new fuel rods and instrumentation constituted the cruciform fuel region of the test assembly. The 12 fuel rods were pressurized to 3.90 MPa (565 psia) with helium. These rods were instrumented as in the MT-2 assembly except for fuel rod 4D, which replaced the instrument tube. Rod 4D had one TC mounted in the fission gas plenum. A summary of the test fuel rod instrumentation is presented in a previous report,(3) where the DACS terminal number, its name, and its sensor location in the test train are given.

® Registered trademark of Crawford Fitting Company, Solon, OH 44139.

NRU FACILITIES

The NRU reactor and loops were basically unchanged from previous LOCA experiments.(3,4) The reactor core loading that was used provided the required power distribution.(5) NRU operations shuffled fuel as required to assure less than a 5% radial power skew across the L-24 position. The reactor loading used for TH-2 was not acceptable for this test series.

EXPERIMENT OPERATIONS

The primary objective of the MT-3 experiment was to provide peak test fuel cladding temperatures of 1033 to 1089K (1400 to 1500°F) for at least 150 s. This objective required the proper radial power profile and good control of peak cladding temperature. To achieve proper power and temperature control, a preliminary thermal-hydraulic test series (TH-3) was conducted with the unpressurized assembly used in TH-2. Adiabatic tests measured the proper radial power profile, and a series of characterization tests produced the desired peak cladding temperatures. Operating conditions for TH-3 are described in Reference 6.

MT-3 Operating Conditions

After TH-3 was completed, the thermal-hydraulic assembly was removed from the NRU reactor and the MT-3 test assembly was inserted into the L-24 position. After piping changes, the MT-3 assembly was preconditioned using the U-2 loop. After preconditioning and subsequent piping changes that connected the L-24 position to the reflood system, reflood calibrations were performed at 0.0254 and 0.0508 m/s (1.0 and 2.0 in./s). Finally, the best technique was chosen to run MT-3, using experience gained from TH-3. Operating conditions for MT-3 are summarized in Table 1.

Protective and Safety Trips

TH-2 transient tests qualified the protective trip set points, overshoot, and the protective trip circuitry. Protective trips were designed to protect the assembly during the tests so that numerous tests can be run. Protective trips were used to insure the best possible test results for MT-3. TH-3 tests further refined the protective trip system.

Protective trip logic limited test fuel temperatures in several ways: maximum allowable cladding temperature, maximum and minimum allowable heatup rates, time limitations for both cladding temperatures and fast fill reflooding to preselected levels, and maximum reflood delay times. These protective trip

TABLE 1. MT-3 Operating Conditions

Parameter	Preconditioning	Reflood Calibration	Pretransient	Transient
Reactor power, MW	127	0	7.5	7.5
Coolant	U-2 water	U-1 steam/ reflood water	U-1 steam	U-1 steam/ reflood water
Coolant flow, kg/s (lbm/h)	0 to 16.3 (0 to 129,400)	0.378 (3000)	0.378 (3000)	0.378 (3000)
Reflood delay, s	NA[a]	0 to 40	NA	0 to 40
Variable reflood rate, m/s (in./s)	NA	0.0254 to 0.0508 (1.0 to 2.0)	NA	0.0089 to 0.15 (0.35 to 6.0)
Fast fill level, m (in.)	NA	NA	NA	0.9 to 1.2 (36 to 48)
Peak cladding temperature, K (°F)	700 (800)	444 (340)	728 (850)	1256 (1800)
Reactor trip criteria	Safety	Safety	Protective and safety	Protective and safety

---

(a) Not applicable.

options and their set points were selected at various times during the course of the experiment. The protective trip signal was generated for several operating condition criteria, and the LCS and the NRU reactor always responded the same: 1) the reactor was conditionally tripped and 2) emergency reflood flow was initiated.

The protective trip logic was based on sensor sampling as frequently as 10 times per second; however, the safety trip logic and circuitry were the same as designed for previous experiments.(3) Safety trip set points are summarized in Tables 2 and 3; protective trip set points were developed during qualifications tests. Both types of reactor trips use the same hard-wired circuitry to achieve a trip; however, the program logic that dictated a trip condition was completely independent.

TABLE 2. Pretransient and Transient Trip Set Points

Parameter	Location	Use	Operating Limits	Trip Set Point
Hanger tube temperature--high	LCS(a)	Pretransient and transient	691K (785°F)	839K (1050°F)
Outlet pipe temperature--high	LCS	Pretransient and transient	672K (750°F)	700K (800°F)
Fuel cladding temperature--high(b)				
Level 13 305 PSD-13-IR-1	DACS(c)	Transient	1305K (1890°F)	1361K (1990°F)
Level 15 308 PSD-15-IR-1	DACS		1305K (1890°F)	1361K (1990°F)
Level 17 315 PSD-17-IR-4	DACS		1278K (1840°F)	1333K (1940°F)
Low steam flow trip	LCS	Pretransient	0.378 kg/s (3000 lbm/h)	0.277 kg/s (2200 lbm/h)
Standby reflood flow--low	LCS	Transient	1.78 cm/s (0.7 in./s)	1.27 cm/s (0.5 in./s)(d)
Accumulator inventory--low	LCS	Transient	22.7 kg (50 lbm)	11.3 kg (25 lbm)

(a) LCS = loop control system.
(b) Standard trip set point criterion; see Table 3 for nonfunctional TC criteria.
(c) DACS = data acquisition and control system.
(d) 0.64 cm/s (0.25 in./s) to be used during reflood loop calibration.

TABLE 3. Standard and Defaulted Safety Trip Set Point Criteria for Operating and Nonoperating Fuel Cladding Thermocouples

Summary of TC Operating Conditions	Criteria and Safety Margins (SM)	Sensors at Level 17	Sensors at Level 15	Sensors at Level 13
		DACS Thermocouple Numbers		
		315,PSD-17-IR-4[a] (99,101,107,109)	308,PSD-15-IR-1 (71,72,73,74, 98,100,106,108)	305,PSD-13-IR-1 (56,57,58,59)
		Trip Set Point Temperatures, K (°F)		
$2 \leq$ Number of operating TCs on each of Levels 13, 15, 17	Standard SM = 56K (100°F)	1333 (1940)	1361 (1990)	1361 (1990)
$2 \leq$ Number of operating TCs on each of Levels 15 and 17	Standard SM = 56K (100°F)	1333 (1940)	1361 (1990)	
$2 \leq$ Number of operating TCs on each of Levels 15 and 17 or Levels 13 and 17	Alternate SM = 84K (150°F)	1305 (1890)	1333 (1940)	1333 (1940)
$2 \leq$ Number of operating TCs on only Level 15, 17, or 13	Fallback SM = 111K (200°F)	1277 (1840)	1305 (1890)	1305 (1890)

(a) These pseudo sensor data (PSD) are the calculated time-average of the following DACS sensor-numbered data: 99, 101, 107, 109.

SAFETY HAZARDS REVIEW

The MT-3 experiment was much like MT-2 (July 1981) except that it operated at a lower peak cladding temperature--1033 to 1089K (1400 to 1500°F)--for 150 to 200 s. MT-2 had cladding temperatures ranging from 1047 to 1186K (1425 to 1675°F) for 155 s. During the period when the MT-3 cladding ballooned, its temperature dropped and the fuel temperature rose. Steady-state heat transport calculations provided a conservative estimate of the maximum credible fuel temperatures that might be possible. These calculations indicated that even if the fuel rods were fully deformed and ruptured (the worst case with steam replacing helium), the maximum credible fuel centerline temperature would be less than 2144K (3400°F).(3)

Thermal-hydraulic test fuel rod cladding had reached temperatures as high as 1366K (2000°F) without experiencing unstable cooling or steam/Zircaloy reactions. MT-3 operations were limited to less than 1361K (1990°F) under all conditions. Protective trip circuitry as well as the designed experiment operations provided that limitation.

MT-3 contained 12 test fuel rods (compared with 11 in MT-2), and both assemblies contained 20 guard rods. Consequently, MT-3 could provide up to 3.2% more fission products than MT-2, but the safety margins used to predict the fission product availability and releasability for MT-2 were at least 100%. The possible decrease in the margin is insignificant.

Other safety analysis reports(2,7) are valid for the MT-3 experiment because its operating conditions are well within the envelope of cases analyzed and reported there.

MT-3 OPERATING PROCEDURE

This section describes the operating procedures that were followed for the MT-3 experiment. Experiment operating conditions are summarized in Table 4.

TABLE 4. Experiment Operating Conditions

Parameter	Preconditioning	Reflood Calibration	Transient
Reactor power, MW	127	0	7.5
Coolant	U-2 water	U-1 steam/ reflood water	U-1 steam/ reflood water
Coolant flow[a] initial condition, kg/s (lbm/h)	0 to 16.3 (0 to 129,000)	0.378 (3000)	0.378 (3000)
Reflood delay, s	NA[b]	0 to 40	0 to 40
Variable reflood rate, m/s (in./s)	NA	0.0254 to 0.508 (1.0 to 2.0)	0.0089 to 0.15 (0.35 to 6.0)
Fast fill level, m (in.)	NA	NA	0.9 to 1.2 (36 to 48)
Peak cladding temperature, K (°F)	700 (800)	444 (340)	1256 (1800)
Reactor trip criteria	Safety	Safety	Protective and safety

(a) Transient initiated by termination of steam flow.
(b) Not applicable.

MT-3 PRECONDITIONING OPERATIONS

Test Configuration

1. Install MT-3 test assembly in L-24 NRU reactor position.

2. Connect U-2 loop to L-24 NRU reactor position.

3. Pressure test the test train head seal.

Loop System Preparations

1. Fill and electrically preheat the U-2 loop.
2. Provide water chemistry as shown in Table 5.
3. Calibrate U-2 loop instruments (see Table 6).
4. Confirm trip circuit operability.
5. Implement trip set points (see Table 7).

TABLE 5. Water Chemistry Requirements

Requirement	Applicability	Recommended Limits
Deionized supply	Preconditioning water coolant	$\leq 1 \times 10^{-5}$ Mho
Impurity concentrations	Halides Oxygen All other elements	<1 ppm $\leq$100 ppm $\leq$100 ppm

TABLE 6. U-2 Loop Instrument Calibration

Sensor	Loop Parameter	DACS	Instrument Range	Acceptable Accuracy
TE-78	Inlet temperature	221	311 to 533K (70 to 500°F)	±1K (±2°F)
TE-79	Outlet temperature	222	323 to 589K (122 to 600°F)	±1K (±2°F)
FT-4D	Flow	215	3.9[(a)] to 19.3[(b)] kg/s (36,400 to 131,400 lbm/h)	±0.4 kg/s (±2000 lbm/h)
PDT-90	Outlet pressure	204	5.52 to 8.96 MPa (800 to 1300 psia)	±0.3 MPa (±50 psia)
PDT-90	Test assembly pressure drop	205	0.021 to 0.172 MPa (3 to 25 psi)	±0.002 MPa (±0.3 psi)

(a) At a temperature of 589K (600°F).
(b) At a temperature of 394K (250°F).

TABLE 7. Preconditioning Safety Trip Set Points

Parameter	Nominal Operating Limits	Trip Set Point
Outlet coolant temperature, K (°F)	552 (534)	561 (550)
Pump subcooling temperature--low, K (°F)	83 (150)	28 (50)
Coolant flow--low, kg/s (lbm/h)	15.5[(a)] (123,000)	13.4[(a)] (106,300)
Surge tank level--low	TBD[(b)]	30%
Surge tank pressure--high, MPa (psig)	TBD[(b)]	8.90 (1275)

(a) At a temperature of 517K (472°F).
(b) To be determined by CRNL.

NRU Reactor Preparations

1. Load NRU reactor fuel assemblies and absorber assemblies as required.

2. Input NRU reactor linear and log rate trip set points as required by CRNL.

3. Adjust the neutron detector scatter plug as required.

4. Establish mean power trip set points as required.

5. Confirm that all trip set points are activated and report to the experiment director when ready for operation.

DACS Computer Preparations

1. Load labeled, certified disk packs and mount a labeled, certified tape on the tape drive.

2. Start a dummy test and set the DACS mode to idle.

3. Set the steady-state scan rate at 1 s.

4. Set the immediate display scan rate at 4 s.

5. Set the graphic display scan rate at 5 s.

6. Format the steady-state immediate display.

7. Identify and remove failed sensors from both displays.

8. Verify trip circuit operability.

9. Set Keithley amplifiers to the 1.0 scale and verify that self-powered neutron detector (SPND) coefficients are correct.

Preconditioning Operating Procedure (to be followed in sequence). NOTE: Total time at full reactor power is to be limited to 1.0 EFPH (equivalent full-power hour).

1. End the dummy test and start a new test on the DACS.
2. Start the loop, using sparge pumps to provide an initial flow of 16.3 kg/s (129,400 lbm/h) and initial pressurization of 8.62 MPa (1250 psia).
3. Change the DACS mode to steady state and turn on the video tape recorder.
4. Adjust the inlet temperature to 517 ±3K (472 ±5°F).
5. Print the DACS sensor status report and the REDACE print-out for review of ΔT heat loss check.
6. Increase the reactor power to 63.5 MW. Request REDACE print-out.
7. Perform a power calibration, using a REDACE print-out and DACS data (loop flow rate times test assembly ΔT).
8. Print the DACS sensor status report and REDACE print-out for review.
9. Increase the reactor power to 127 MW.
10. Adjust test assembly inlet temperature to 517 ±3K (472 ±5°F).
11. Print the DACS sensor status report and the REDACE print-out.
12. Perform a power calibration.
13. Conditionally trip the reactor.

14. Increase the reactor to full power (127 MW), maintaining test assembly inlet temperature at 517 ±3K (472 ±5°F).

15. Print the DACS and REDACE sensor status reports.

16. Recheck power calibrations.

17. Conditionally trip the reactor.

18. If time permits, repeat steps 14 through 17.

19. Make a hard copy of the CRT immediate and graphic displays showing the hottest centerline TCs. Change DACS to idle mode. Make a tape copy when ending the test, a disk image copy, and a historical request for all the data required to run the transient test.

20. Shut down the loop facilities to prepare for piping rearrangement. Return DACS to steady state and scan the sensors once per minute until reflood flow tests are initiated.

21. Place one copy of all materials collected in Test Parameter Log and have data coordinator sign.

MT-3 PRETRANSIENT AND TRANSIENT OPERATIONS

Test Configuration

1. Connect reflood system to the L-24 NRU reactor position.

2. Connect U-1 steam supply to the test assembly.

Loop System Preparations

1. Start up the U-1 loop.

2. Insure that U-2 makeup tanks (which supply water to the U-1 loop) are full.

3. Preheat the steam/reflood loop to 408K (275°F).

4. Fill reflood accumulators at ~311 ±6K (100 ±10°F). Check water temperature in the three accumulators.

5. Verify that the nitrogen supply for accumulator pressurization is adequate.

6. Calibrate reflood system instruments; enter conversion factors and units in the test log.

7. Implement the safety trip set points as shown in Tables 2 and 3 for the pretransient and transient phases of the experiment. The trip pseudo sensors used on DACS to represent the high cladding temperature trip circuits and sensors are identified in Table 8.

NRU Reactor Preparations (CRNL)

1. Confirm that two linear rate and two log rate neutron flux detectors (ion chambers) are set and being recorded in the NRU reactor control room.

2. Confirm that the REDACE data will be taken on demand or at a 30-s frequency when requested.

3. Adjust the neutron detector scatter plug as required. Establish mean power trip set points as required.

4. Confirm that all trip set points are activated, and report to the experiment director when ready for experiment operation.

TABLE 8. Cladding High-Temperature Trip Sensors

Level	Sensor Thermocouples	DACS Sensor Number	DACS Pseudo Sensor	DACS Sensor Number
13	TC-13-6C-IR-2	56	PSD-13-IR-1	305
	TC-13-4F-IR-4	57		
	TC-13-1D-IR-2	58		
	TC-13-3A-IR-4	59		
15	TC-15-6C-IR-4	71	PSD-15-IR-1	308
	TC-15-4F-IR-2	72		
	TC-15-1D-IR-4	73		
	TC-15-3A-IR-2	74		
	TC-15-5B-IR-3	98		
	TC-15-2E-IR-1	100		
	TC-15-5B-IR-1	106		
	TC-15-2E-IR-3	108		
17	TC-17-5E-IR-4	99	PSD-17-IR-4	315
	TC-17-2B-IR-2	101		
	TC-17-5E-IR-2	107		
	TC-17-2B-IR-4	109		

DACS Computer Preparations

1. Load the DACS with labeled, certified tape and disks.
2. Start a dummy test and set the DACS mode to idle.
3. Set the steady-state scan rate at 1 s.
4. Set the transient scan rate at 40 ms.
5. Set the immediate display scan rate at 4 s.
6. Set the graphic display scan rate at 5 s.
7. Format the steady-state immediate display with the sensors listed in Table 9.
8. Format the transient graphic display.
9. Identify and remove failed sensors from displays, pseudo sensors, and trip circuits.
10. Reset Keithley amplifiers to the 0.1 scale and change SPND coefficients; reduce by a factor of 10.
11. Report to the experiment director when ready for pretransient experiment operation.

TABLE 9. DACS Immediate Display Sensors

Sensor Name	DACS Sensor Number	Sensor Name	DACS Sensor Number
TC-13-3B-IR-2	55	TC-17-3D-IR-5	90
TC-15-3C-IR-2	91	TC-17-3D-IR-C	93
TC-15-3C-IR-C	94	TC-17-3D-IR-4	138
TC-15-3C-IR-7	97	TC-17-2D-OR-1	120
TC-15-3B-IR-4	70	TC-17-2D-IR-2	126
TC-17-4C-IR-3	88	TC-17-5D-IR-4	123
TC-17-4C-IR-C	92		

Reflood Calibration Test Operating Procedure (to be followed in sequence)

1. Calibrate the reflood prefill controls to fill the test nozzle annulus up to Level 0.

2. Place the DACS in the steady-state mode.

3. Increase test section steam flow to 0.378 kg/s (3000 lbm/h), and control test section backpressure at 0.276 MPa (40 psia).

4. Enter the reflood rates, duration, and delay time (selected from the TH-3 tests) for DACS and LCS controls.

5. Print a sensor status report to insure that the test assembly and all TCs are ≥422K (300°F).

6. Reproduce the DACS immediate and graphic displays.

7. Turn on the video tape recorder, and zero the counter when loading a new tape.

8. Prefill and drain a total of three times; display on graphic terminal.

9. Switch the DACS to the transient mode 20 s before issuing the verbal command "BEGIN THE TRANSIENT" (directed to the LCS operator), and record the time.

10. LCS operator initiates the transient.

11. Reproduce the DACS immediate and graphic displays as required.

12. Stop the test when reflood water passes the TCs at Level 20.

13. Change the DACS mode to steady state for 5 min and then to idle.

14. Make a historical request on the DACS graphic display and reproduce copies of the following data:

    - reflood rate (DACS sensors 201 and 202)
    - steam flow rate (DACS sensor 221)
    - TCs at each level (DACS sensors 289, 291, 297, 299, 302, 304, 305, 307, 308, 310, 313, 317, and 320).

15. Print all DACS data throughout the transient at 5-s intervals.

16. Repeat steps 1 through 15 for calibrating the assembly, using the designated reflood rate and delay time selected in TH-3.

17. Turn off the video tape recorder, and record the counter reading.

18. Make a tape copy on the DACS and a disk image copy on tape as time permits.

19. As necessary, repeat Loop System Preparations and DACS Computer Preparations sections before proceeding.

Pretransient Operating Procedure (to be followed in sequence)

1. Set the protective reactor trip set points for DACS and LCS control (values must be approved and recorded by the test director).

2. LCS operator will set timers for reflood delay times (values must be approved and recorded by the test director).

3. Check accumulator water levels (weights); record the values in the Test Parameter Log.

4. Input the following information in the Test Parameter Log (this step does not apply for adiabatic transients):

   - name (DACS number) and level of sensor to be used for reflood water quench measurement
   - maximum allowed time for quench at the level selected
   - list of sensors (maximum 20) for hot spot search (no pseudo sensors)
   - time-temperature pairs (maximum 20 pairs) for low-limit window, high-limit window, and operator-controlled temperatures
   - DACS-controlled reflood rates (yes - to control; no - not to control)
   - if DACS-controlled, specify reflood rate numbers and their corresponding reflood coolant rates to be preset in the LCS
   - control function coefficients weighting flow rates and delays for the DACS control algorithm (as determined during the TH-3 tests).

5. Start a new test on the DACS; change DACS mode to steady-state.

6. Insure that the REDACE scan frequency for NRU data is 30 s.

7. Increase the NRU reactor power to the low neutron level.

8. Set the inlet steam flow rate to 0.378 kg/s (3000 lbm/h) and the test section backpressure to 0.28 MPa (40 psia).

9. Before proceeding, the NRU reactor operator must acknowledge that the neutron power will not exceed the neutron power level requested by the test director.

10. Increase the NRU neutron power to 4.0%. With the reactor power at nominally 50% of the pretransient power, scan the DACS immediate display for the hottest TC and reproduce the display.

11. Insure that the test assembly inlet temperature stabilizes at 436 ±3K (330 ±10°F).

12. Adjust the NRU neutron power level to obtain a steady-state 178K (320°F) temperature increase across the test assembly.

13. Check the peak cladding temperature, steam flow, test assembly inlet temperature, and outlet pressure.

14. Reproduce the DACS immediate and graphic displays.

15. Activate the video tape recorder.

Transient Operating Procedure (to be followed in sequence)

1. Prefill and drain the inlet annulus a total of three times. Display test assembly temperatures on the DACS graphic display.

2. Change to the transient operating mode on the DACS; 20 s later, issue the verbal command "BEGIN THE TRANSIENT" (directed to the LCS operator); and record the time in the test log.

3. LCS operator begins the transient.

4. Shut down the reactor when the test assembly quench is completed or when the transient time exceeds 200 s at the desired cladding temperature range.

5. Shut off reflood water flow only after upper TCs have quenched (this step does not apply for adiabatic transients).

6. Record reflood water used (accumulator weight difference) in the Test Parameter Log (this step does not apply for adiabatic transients).

7. Insure that tripping the reactor has returned control to the DACS (transient forcing signal #257 = 0).

8. Return the DACS mode to steady state for 5 min and then to idle, ending the data record.

9. End the test on the DACS; verify that the Test Parameter Log is completed.

10. Turn off the video tape recorder, and record the location.

11. Copy the following historical data on the DACS:

    - reproduce data from the hottest pseudo sensors at Levels 15 and 17

    - plot data for sensors 289, 290, 211, 201, 202, 70, 91, 97, 137, and 314.

12. Make a tape copy of the DACS data.

13. Make a disk image copy of the DACS data.

14. Terminate the experiment and remove the test train to the bay, using established CRNL procedures.

REFERENCES

1. Hann, C. R. 1979. Program Plan for LOCA Simulation in the National Research Universal (NRU) Reactor. PNL-3056, Pacific Northwest Laboratory, Richland, Washington.

2. Mohr, C. L., et al. February 1981. Safety Analysis Report: Loss-of-Coolant Accident Simulation in the National Research Universal Reactor. NUREG/CR-1208, PNL-3093, Pacific Northwest Laboratory, Richland, Washington.

3. Russcher, G. E., et al. September 1981. Experiment Operations Plan for Loss-of-Coolant Accident Simulation in the National Research Universal Reactor, Addendum 1: Materials Test 1 and 2. NUREG/CR-1735, PNL-3765, Pacific Northwest Laboratory, Richland, Washington.

4. Russcher, G. E., et al. April 1981. Experiment Operations Plan for a Loss-of-Coolant Accident Simulation in the National Research Universal Reactor. NUREG/CR-1735, PNL-3551, Pacific Northwest Laboratory, Richland, Washington.

5. Atfield, M. D., memorandum to NRU Superintendent. October 2, 1980. "Reactor Loading for Battelle LOCA Simulation Tests." Chalk River Nuclear Laboratories, Chalk River, Ontario, Canada.

6. Russcher, G. E., C. L. Wilson, and L. J. Parchen. 1983. Experiment Operations Plan for the TH-3 Experiment in the NRU Reactor. NUREG/CR-2660, PNL-4232, Pacific Northwest Laboratory, Richland, Washington.

7. Axford, D. J., I. C. Martin, and S. J. McAuley. 1980. Final Safety and Hazards Analysis for the Battelle LOCA Simulation Tests in the NRU Reactor. AECL 7153, Chalk River Nuclear Laboratories, Chalk River, Ontario, Canada.

APPENDIX A

TEST PARAMETER LOG

APPENDIX A

TEST PARAMETER LOG

LOCA Simulation in NRU

Run number: ____________________

Date: ____________________

1.0 PRECONDITIONING ACTIVITIES

Date: __________ Time: __________

Data stored on data tape number: ________ Disk number: ________

1.1 Power calculations (attach to log)

2.0 PRETRANSIENT ACTIVITIES

Tape number: __________

Disk configuration: DPO __________

DP1 __________

DP1F __________

2.1 Operation summary

PARAMETER/UNITS	SPECIFIED VALUE/ ACCEPTED RANGE	ACTUAL VALUE
Steam flow rate, lbm/h	3000 (±5%)	
Steam inlet temperatures, °F	325 (±15)	
Maximum fuel cladding temperatures, °F	800 (NA)	
Sensor name ____________________		
Sensor name ____________________		
Sensor name ____________________		
Total test assembly ΔT, °F	320 (±10)	
Outlet pressure, psia	40 (±5%)	

2.2 Print all sensor data with DACS in steady-state mode; review.

2.3 Protective trip set points (pretransient and transient)

PARAMETER/UNITS	VALUE
Hanger tube temperature--high, °F	
Outlet pipe temperature--high, °F	
Steam flow--low, lbm/h	
Fuel cladding temperature, °F	
Level 17--high	
Level 15--high	
Level 13--high	
Level __ time to quench	

3.0 SPECIAL COMMENTS ON RUN CONDITIONS:

3.1 Preparations:

3.2 Pretransient:

3.3 Transient:

3.4 Post-transient:

4.0 CONDITIONS CAUSING RUN TERMINATION:

5.0 SPECIAL CONDITIONS TO BE CONSIDERED IN THE ANALYSIS OF THE TEST RUN:

6.0 CONDITIONS THAT MAY CAUSE THE TEST TO BE INVALID:

7.0 INSTRUMENTATION FAILURES BEFORE TEST:

8.0 INSTRUMENTATION FAILURES AFTER TEST TERMINATION:

9.0 GENERAL COMMENTS ON TEST:

APPENDIX B

EXPERIMENT CONDITIONS LOG

APPENDIX B

EXPERIMENT CONDITIONS LOG

A. Protective trip temperatures for Levels 13, 15, and 17 ____________________

B.

MT Run Number	Delay Time, s	Reflood Rate	
		in./s	Duration, s

C. Maximum allowed time (______ s) for reflood water to quench sensor ______ at Level ______

D. DACS control ____

E. Time-temperature for test control (if needed)

F. Control functions coefficients (if needed) ______________________________

G. Temperature increase across test assembly for pretransient ________________

Test Director ___________________________

NUREG/CR-2599
PNL-4190
R3

DISTRIBUTION

No. of
Copies

OFFSITE

315 U.S. Nuclear Regulatory
Commission
Division of Technical
Information and Document
Control
7920 Norfolk Avenue
Bethesda, MD 20014

5 R. Van Houten
U.S. Nuclear Regulatory
Commission
Fuel Systems Research Branch
Washington, DC 20555

FOREIGN

2 R. R. Lewis
Chalk River Nuclear Laboratories
Chalk River, Ontario
Canada

6 D. T. Nishimura
Chalk River Nuclear Laboratories
Chalk River, Ontario
Canada

No. of
Copies

ONSITE

50 Pacific Northwest Laboratory

W. J. Bailey
L. W. Cannon
S. K. Edler
M. D. Freshley
R. L. Goodman
C. R. Hann
G. M. Hesson
U. P. Jenquin
L. L. King
R. K. Marshall
P. N. McDuffie
L. J. Parchen (5)
W. N. Rausch
G. E. Russcher (18)
B. J. Webb
N. J. Wildung
C. L. Wilson (5)
M. C. Wismer
Publishing Coordination (2)
Technical Information (5)

NRC FORM 335
(11-81)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER *(Assigned by DDC)*
NUREG/CR-2599
PNL-4190

4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)*
Experiment Operations Plan for the MT-3 Experiment in the NRU Reactor

2. *(Leave blank)*

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)
G. E. Russcher, C. L. Wilson, and L. J. Parchen

5. DATE REPORT COMPLETED
MONTH May | YEAR 1983

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Pacific Northwest Laboratory
P.O. Box 999
Richland, Washington

DATE REPORT ISSUED
MONTH June | YEAR 1983

6. *(Leave blank)*

8. *(Leave blank)*

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Division of Accident Evaluation
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

10. PROJECT/TASK/WORK UNIT NO.

11. FIN NO.
B2277

13. TYPE OF REPORT
Topical

PERIOD COVERED *(Inclusive dates)*

15. SUPPLEMENTARY NOTES

14. *(Leave blank)*

16. ABSTRACT *(200 words or less)*

A series of thermal-hydraulic and cladding materials deformation experiments were conducted using light-water reactor fuel bundles as part of the Pacific Northwest Laboratory Loss-of-Coolant Accident (LOCA) Simulation Program. This report is the formal operations plan for MT-3--the third cladding materials deformation experiment conducted in the National Research Universal (NRU) reactor, Chalk River, Ontario, Canada. The major objective of MT-3 was to simulate a pressurized water reactor LOCA that could induce fuel rod cladding deformation and rupture.

17. KEY WORDS AND DOCUMENT ANALYSIS
loss-of-coolant accident
cladding deformation and rupture
light-water reactor fuel rod bundles
reflood control
data acquisition and control system

17a. DESCRIPTORS
experiment operations plan
National Research Universal (NRU) reactor
cladding materials deformation experiments

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS *(This report)*
Unclassified

21. NO. OF PAGES
40

20. SECURITY CLASS *(This page)*
Unclassified

22. PRICE
$

NRC FORM 335 (11-81)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

120555078877 1 IAN1R3
US NRC
ADM-DIV OF TIDC
POLICY & PUB MGT BR-PDR NUREG
W-501
WASHINGTON DC 20555

NUREG/CR-2599 EXPERIMENT OPERATIONS PLAN FOR THE MT-3 EXPERIMENT IN THE NRU REACTOR JUNE 1983