NUREG/CR-2881
PNL-4293

Experiment Operations Plan for the MT-4 Experiment in the NRU Reactor

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Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory Commission

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Manuscript Completed: May 1983
Date Published: June 1983

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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-4--the fourth materials deformation experiment conducted in the National Research Universal (NRU) reactor, Chalk River, Ontario, Canada. A major objective of MT-4 was to simulate a pressurized water reactor LOCA that could induce fuel rod cladding deformation and rupture due to a short-term adiabatic transient and a peak fuel cladding temperature of 1200K (1700°F).

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 experiments 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 seventh experiment in this program--materials experiment 4 (MT-4)--is presented in this document. The MT-4 experiment simulated a LOCA that produced a peak cladding temperature of 1200K (1700°F) during a relatively rapid (adiabatic) heatup transient. Experiment operations closely followed the operating conditions used in TH-1.16, where reflood cooling was used to turn around the heatup transient caused by a long reflood delay. Afterward, thermal-hydraulic measurements evaluated the effects of fuel cladding deformation and rupture.

An approved draft copy of the experiment operations plan for MT-4 was sent to Chalk River Nuclear Laboratories (CRNL), Chalk River, Ontario, in May 1982 just prior to the MT-4 experiment. This report represents the formal documentation of the experiment operations plan for MT-4.

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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 cladding that is typical of a LOCA.[1]

Previous materials deformation experiments in the LOCA program emphasized higher temperature LOCA fuel failure and extended operating time at cladding deformation conditions. The MT-2 experiment produced ruptured fuel rods during a simulated LOCA transient in which peak fuel rod cladding temperatures exceeded 1200K (1700°F). MT-3 produced maximum ballooned and ruptured fuel rods during an extended LOCA transient in the Zircaloy high alpha temperature regime.

The MT-4 experiment is most similar to the MT-2 experiment, except for three differences: 1) the MT-4 heatup was a short-term adiabatic transient rather than an extended one, 2) after the adiabatic heatup transient and temperature turnaround, peak cladding temperatures were stabilized to measure heat transfer characteristics of the deformed (and ruptured) fuel rods, and 3) self-powered neutron detectors (SPNDs) mounted on the shroud were relocated to grid elevations to minimize axial neutron power distortion.

The major objectives of the MT-4 experiment were to:

- evaluate simulated LOCA fuel rupture during adiabatic heatup conditions
- evaluate LOCA fuel rupture in a carefully monitored neutronic environment
- compare MT-4 results with those from the MT-2 and MT-3 experiments (all other variables were minimized)
- simulate LOCA fuel rupture (12 rods) during a delayed reflooding transient with a peak cladding temperature of 1200K (1700°F)

(a) Operated by Atomic Energy of Canada Ltd. (AECL).
(b) Operated for the U.S. Department of Energy (DOE) by Battelle Memorial Institute.

- measure heat transfer coefficients for deformed fuel rods at 1089K and 1200K (1500°F and 1700°F).

The MT-4 experiment used a new cruciform bundle of 12 pressurized test fuel rods and the 20 guard fuel rods and the shroud that were previously used in MT-3. Test operations most closely followed the operating conditions of TH-1.16, where reflood cooling was used to terminate the heatup transient temperature at ~1200K (1700°F). Stabilized post-transient operation closely followed the operating conditions used in the MT-3 experiment to investigate high-temperature thermal-hydraulic effects of the LOCA.

This report presents the formal experiment operations plan for the fourth materials deformation experiment in the program (MT-4), which was performed in May 1982. The remainder of the report consists of:

- a summary of the design differences for MT-4 as compared with MT-3

- a safety hazards statement

- the operating procedure for MT-4

- a sample Test Parameter Log (Appendix A) and an Experiment Conditions Log (Appendix B).

DESIGN DIFFERENCES

The MT-4 test assembly was quite similar to the MT-3 test assembly; therefore, only design differences are described in this report. Detailed descriptions are given in the safety analysis report[2] and the experiment operations plan for MT-3.[3]

TEST ASSEMBLY

The MT-4 test assembly used a new cruciform bundle of 12 test fuel rods and the MT-3 guard fuel rod bundle and shroud. The 12 fuel rods were pressurized to 4.62 MPa (670 psia) just prior to the transient and were monitored by pressure transducers located just above the reactor head. Guard fuel rods were pressurized to 0.101 MPa (14.7 psia).

Because the MT-3 guard fuel rod assembly had been used in only one previous experiment, most instrumentation was still operational for this test.[2] In addition, three new types of instrumentation were included in MT-4: 1) a new liquid level sensor that sampled the coolant at Level (18) 139.9;[a] 2) a time domain reflectometer (TDR) sensor that monitored the coolant liquid level in the test assembly; and 3) cumulative neutron flux (fluence) monitor wires. Most of the instrumentation that had been added for the MT-3 experiment remained: 1) two thermocouples (TCs) at fuel rod braze junctions at Level (19) 155.7; 2) two more TCs on the hanger tube at Level (20) 168.7; 3) one TC in a fuel rod plenum region; and 4) the desuperheater spray coolant system.

The MT-4 test rod bundle with new fuel rods and instrumentation constituted the cruciform test fuel assembly. The 12 test rods were pressurized to 4.62 MPa (670 psia) with helium just before the LOCA experiment and were instrumented the same as those in the MT-3 assembly. Fuel rod 4D, which replaced the instrument tube, also had one TC mounted in the fission gas plenum. The test rod instrumentation is summarized in Table 1; new data acquisition and control system (DACS) terminal numbers, names, and sensor locations in the test train are given.

NRU FACILITIES

The NRU reactor and loops were basically unchanged from previous LOCA experiments.[3,4] The same reactor fuel loading was used to provide the

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(a) Instrument levels are reported with the old designation in parentheses followed by the new number (which indicates the inches above the top of the tie plate of the test assembly).

TABLE 1. DACS and Instrumentation Array for MT-4

Instrument Level (Old)	Instrument Level New	DACS Channel Number	Sensor Name	Notes
(1)	-0.2[a]	108	TC-(-)0.2-6A-IN	
(1)	-0.2	106	TC-(-)0.2-1F-IN	
(2)	8.7[a]	74	TC-08.7-6F-C-4	
(2)	8.7	4	TC-08.7-6F-S-C	
(2)	8.7	82	TC-08.7-6A-C-3	
(2)	8.7	5	TC-08.7-1A-S-C	
(3)	13.9[a]	83	TC-13.9-6F-C-4	
(3)	13.9	11	TC-13.9-6A-C-3	
(3)	13.9	86	TC-13.9-6A-S-C	
(3)	13.9	12	TC-13.9-1F-S-C	
(4)	19.2	13	TC-19.2-6F-S-C	
(4)	19.2	14	TC-19.2-1A-S-C	
(5)	28.0	15	TC-28.0-6A-S-C	
(5)	28.0	16	TC-28.0-1F-S-C	
(6)	34.9	19	TC-34.9-6F-S-C	
(6)	34.9	18	TC-34.9-6A-S-C	
(6)	34.9	21	TC-34.9-1A-S-C	
(6)	34.9	20	TC-34.9-1F-S-C	
(7)	40.2	35	TC-40.2-5F-IR-4	
(7)	40.2	33	TC-40.2-2A-IR-2	Failed in MT-4 preconditioning[b]
(8)	42.9	30	TC-42.9-6F-S-C	
(8)	42.9	31	TC-42.9-1A-S-C	
(9)	49.0	36	TC-49.0-6A-S-C	
(9)	49.0	37	TC-49.0-1F-S-C	
(10)	55.9	40	TC-55.9-6F-S-C	
(10)	55.9	39	TC-55.9-6A-S-C	
(10)	55.9	42	TC-55.9-1A-S-C	
(10)	55.9	41	TC-55.9-1F-S-C	
(11)	61.2	47	TC-61.2-6F-S-C	
(11)	61.2	104	TC-61.2-6B-IR-3	

TABLE 1. (contd)

Instrument Level (Old)	New	DACS Channel Number	Sensor Name	Notes
(11)	61.2	34	TC-61.2-3A-IR-2	
(11)	61.2	48	TC-61.2-1A-S-C	
(11)	61.2	102	TC-61.2-1E-IR-1	
(11)	61.2	132	TC-61.2-4F-IR-4	
(12)	69.0	59	TC-69.0-5E-SP-4	
(12)	69.0	58	TC-69.0-5B-SP-3	
(12)	69.0	57	TC-69.0-2B-SP-2	
(12)	69.0	56	TC-69.0-2E-SP-1	
--	73.9	186	TC-73.9-5D-IR-2	Relocated[c]
--	73.9	69	TC-73.9-4D-IR-2	Relocated
--	73.9	95	TC-73.9-4C-IR-2	Relocated
--	73.9	90	TC-73.9-3D-IR-2	Relocated
(13)	76.9	61	TC-76.9-6F-S-C	
(13)	76.9	60	TC-76.9-6A-S-C	
(13)	76.9	63	TC-76.9-1A-S-C	
(13)	76.9	62	TC-76.9-1F-S-C	
(13)	76.9	101[d]	TC-76.9-6C-IR-3	PS-76.9-GR-IR-1 safety circuit
(13)	76.9	52[d]	TC-76.9-4A-IR-3	PS-76.9-GR-IR-1 safety circuit
(13)	76.9	99	TC-76.9-1D-IR-1	
(13)	76.9	50[d]	TC-76.9-3F-IR-1	PS-76.9-GR-IR-1 safety circuit
(13)	76.9	53	TC-76.9-5E-IR-4	
(13)	76.9	67[d]	TC-76.9-5B-IR-1	PS-76.9-GR-IR-1 safety circuit
(13)	76.9	23	TC-76.9-2B-IR-2	
(13)	76.9	127	TC-76.9-2C-IR-3	
(13)	76.9	135	TC-76.9-2D-IR-3	
(13)	76.9	32	TC-76.9-2E-IR-3	
(13)	76.9	93	TC-76.9-3E-IR-3	
(13)	76.9	89	TC-76.9-4E-IR-8	
--	78.9	123	TC-78.9-5D-IR-3	
--	78.9	17	TC-78.9-4D-IR-3	
--	78.9	88	TC-78.9-4C-IR-3	
--	78.9	96	TC-78.9-3D-IR-3	
--	80.9	91	TC-80.9-2C-IR-2	Failed in MT-4 preconditioning[a]
--	80.9	188	TC-80.9-2D-IR-2	
--	80.9	38	TC-80.9-3E-IR-2	
--	80.9	64	TC-80.9-4E-IR-2	

TABLE 1. (contd)

Instrument Level (Old)	Instrument Level New	DACS Channel Number	Sensor Name	Notes
--	83.9	133	TC-83.9-5C-IR-2	
--	83.9	117	TC-83.9-4B-IR-2	
--	83.9	54	TC-83.9-4D-IR-8	
--	83.9	137	TC-83.9-3C-IR-2	Failed in MT-4 preconditioning[a]
(14)	90.0	65	TC-90.0-5E-SP-4	
(14)	90.0	51	TC-90.0-5B-SP-3	
(14)	90.0	44	TC-90.0-2B-SP-2	Failed in MT-3.06
(14)	90.0	43	TC-90.0-2E-SP-1	
--	93.9	115	TC-93.9-3B-IR-2	
--	93.9	49	TC-93.9-4E-IR-3	
(15)	97.9	76[e]	TC-97.9-6F-S-C	PS-97-SH-5-2 protective circuit
(15)	97.9	75[e]	TC-97.9-6A-S-C	PS-97-SH-5-2 protective circuit
(15)	97.9	78[e]	TC-97.9-1A-S-C	PS-97-SH-S-2 protective circuit
(15)	97.9	77[e]	TC-97.9-1F-S-C	PS-97-SH-S-2 protective circuit
(15)	97.9	103[f]	TC-97.9-1C-IR-2	PS-97.9-GR-IR-1 safety circuit
(15)	97.9	25[f]	TC-97.9-5E-IR-2	PS-97.9-GR-IR-1 safety circuit
(15)	97.9	10	TC-97.9-5E-IR-C	
(15)	97.9	45[f]	TC-97.9-5B-IR-3	PS-97.9-GR-IR-1 safety circuit
(15)	97.9	46	TC-97.9-5B-IR-C	
(15)	97.9	24[f]	TC-97.9-2B-IR-4	PS-97.9-GR-IR-1 safety circuit
(15)	97.9	9	TC-97.9-2B-IR-C	
(15)	97.9	55	TC-97.9-2C-IR-8	
(15)	97.9	119	TC-97.9-2D-IR-8	
(15)	97.9	2[f]	TC-97.9-2E-IR-1	SD-97.9-IR-1 safety circuit
(15)	97.9	26	TC-97.9-2E-IR-C	
(15)	97.9	122	TC-97.9-3E-IR-8	
(15)	97.9	105	TC-97.9-6D-IR-4	Failed in MT-3 preconditioning
--	101.9	135	TC-1D1.9-5C-IR-3	
--	101.9	114	TC-101.9-4B-IR-3	
--	101.9	120	TC-101.9-3B-IR-8	
--	101.9	190	TC-101.9-3C-IR-3	
--	104.9	116	TC-104.9-5C-IR-8	
--	104.9	1	TC-104.9-4B-IR-8	
--	104.9	94	TC-104.9-3C-IR-8	
(16)	111.0	109	TC-111.0-6F-C-4	
(16)	111.0	100	TC-111.0-6A-C-3	

TABLE 1. (contd)

Instrument Level (Old)	New	DACS Channel Number	Sensor Name	Notes
(16)	111.0	107	TC-111.0-1A-C-2	Failed in MT-3.06
(16)	111.0	98	TC-111.0-1F-C-1	
(16)	111.0	85	TC-111.0-5E-SP-4	
(16)	111.0	84	TC-111.0-5B-SP-3	
(16)	111.0	80	TC-111.0-2B-SP-2	
(16)	111.0	72	TC-111.0-2E-SP-1	
(16)	111.0	98	TC-111.0-1F-C-1	
(17)	118.9	111[e]	TC-118.9-6F-S-C	PS-118.9-SH-1[b] protective circuit
(17)	118.9	110[e]	TC-118.9-6A-S-C	PS-118.9-SH-1[b] protective circuit
(17)	118.9	113[e]	TC-118.9-1A-S-C	PS-118.9-SH-1[b] protective circuit
(17)	118.9	112[e]	TC-118.9-1F-S-C	PS-118.9-SH-1[b] protective circuit
(17)	118.9	68	TC-118.9-6E-IR-4	Failed in MT-3.04
(17)	118.9	81[g]	TC-118.9-1B-IR-2	PS-118.9-GR-IR-1 safety circuit
(17)	118.9	126[g]	TC-118.9-3B-IR-3	PS-118.9-GR-IR-1 safety circuit
(17)	118.9	92[g]	TC-118.9-4C-IR-8	PS-118.9-GR-IR-1 safety circuit
(17)	118.9	138[g]	TC-118.9-3D-IR-8	PS-118.9-GR-IR-1 safety circuit
(18)	139.9	129[e]	TC-139.9-6F-S-C	PS-139.9-SH-1[b] protective circuit
(18)	139.9	128[e]	TC-139.9-6A-S-C	PS-139.9-SH-1[b] protective circuit
(18)	139.9	131[e]	TC-139.9-1A-S-C	PS-139.9-SH-1[b] protective circuit
(18)	139.9	130[e]	TC-139.9-1F-S-C	PS-139.9-SH-1[b] protective circuit
(18)	139.9	87	TC-139.9-6E-IR-2	
(18)	139.9	73	TC-139.9-5A-IR-3	Failed in MT-3.06
(18)	139.9	66	TC-139.9-1B-IR-4	
(18)	139.9	71	TC-139.9-2F-IR-1	
--	148.2		TC-148.2-4E-SP-1	
--	148.2		TC-148.2-3B-SP-3	
--	148.2		TC-148.2-3E-SP-4	
--	151.5	6	TC-151.5-5D-IR-C	
(19)	156.0	79	TC-156.0-5D-B	
(19)	156.0	70	TC-156.0-3B-B	
(20)	168.7	7	TC-168.7-0T-1	
(20)	168.7	27	TC-168.7-0T-2	
(20)	168.7	3	TC-168.7-0T-3	
(20)	168.7	8	TC-168.7-0T-4	

TABLE 1. (contd)

Instrument Level (Old)	Instrument Level New	DACS Channel Number	Sensor Name	Notes
(21)	187.4	22	TC-187.4-HT-1	
(21)	187.4	CRNL	TC-187.4-HT-2	
(21)	187.4	CRNL	TC-187.4-HT-3	
(21)	187.4	CRNL	TC-187.4-HT-4	
--	HM(h)	173	PT-M1-3B	Pressure transducer
--	HM	174	PT-M1-4B	Pressure transducer
--	HM	175	PT-M1-2B	Pressure transducer
--	HM	176	PT-M1-3B	Pressure transducer
--	HM	177	PT-M1-4B	Pressure transducer
--	HM	178	PT-M1-5B	Pressure transducer
--	HM	179	PT-M1-2B	Pressure transducer
--	HM	180	PT-M1-3B	Pressure transducer
--	HM	181	PT-M1-4B	Pressure transducer
--	HM	182	PT-M1-5B	Pressure transducer
--	HM	183	PT-M1-3B	Pressure transducer
--	HM	184	PT-M1-4B	Pressure transducer
--	139.9	169	LL-D-DP	Liquid level detector(j)
--	FL(i)	172	LL-D-TDR	Liquid level detector(j)
(4A)	25.5	167	ND-25.5-6F-S-3	Failed in MT-3.04
(4A)	25.5	143	ND-25.5-1A-S-1	SPND
(8A)	46.5	145	ND-46.5-6A-S-2	SPND
(8A)	46.5	146	ND-46.5-1F-S-4	SPND
(11A)	67.5	142	ND-67.5-6F-S-3	Failed (reading high)
(11A)	67.5	148	ND-67.5-6A-S-2	Failed in MT-3.1
(11A)	67.5	151	ND-67.5-1A-S-1	SPND
(11A)	67.5	150	ND-67.5-1F-S-4	SPND
(12)	69.0	163	ND-69.0-1F-C	Failed (reading high)
(13A)	88.5	155	ND-88.5-6F-S-3	SPND
(13A)	88.5	154	ND-88.5-6A-S-2	SPND
(13A)	88.5	157	ND-88.5-1A-S-1	Failed in MT-3 preconditioning
(13A)	88.5	156	ND-88.5-1F-S-4	Failed in MT-3.04

TABLE 1. (contd)

Instrument Level (Old)	Instrument Level New	DACS Channel Number	Sensor Name	Notes
(15A)	109.5	160	ND-109.5-6F-S-3	SPND
(15A)	109.5	159	ND-109.5-6A-S-2	SPND
(15A)	109.5	162	ND-109.5-1A-S-1	Failed (reading low)
(15A)	109.5	161	ND-109.5-1F-S-4	Failed (reading low)
(16)	111.0	152	ND-111.0-1F-C	SPND
(17A)	130.5	166	ND-130.5-6F-S-3	SPND
(17A)	130.5	165	ND-130.5-6A-S-2	SPND
(17A)	130.5	168	ND-130.5-1A-S-1	SPND
(17A)	130.5	149	ND-130.5-1F-S-4	SPND
(18A)	151.5	170	ND-151.5-6F-S-3	Failed in MT-3 preconditioning
(18A)	151.5	171	ND-151.5-1A-S-1	SPND
	U-1L(k)	201	SRCS-FR-LO-W	FE-4 reflood rate low
	U-1L	202	SRCS-FR-HI-GH	FE-3 reflood rate high
	U-1L	203	STBY-FL-OW	FE-9 reflood emergency
	U-2L(l)	204	U2LP-PR-ES-S-1	PDT-90 loop back pressure
	U-2L	205	U2LP-TA-PS-DR-1	PDT-90 in/out pressure drop
	U-1L	206	SRCS-S-TC-IN-1	TE-2 in steam temperature
	U-1L	207	SRCS-S-TC-OT-1	TE-3 out steam temperature
	U-1L	208	SRCS-S-PS-IN-1	PT-5 in steam pressure
	U-1L	209	SRCS-S-PS-OT-1	PT-6 out steam pressure
	U-1L	210	SPCS-S-FR-1	FY-6 steam flow rate (failed)
	U-1L	211	SRCS-S-FR-IN-1	FV-1 in steam rate (failed)
	U-1L	212	SRCS-S-FR-OT-1	FI-2 out steam rate (failed)
	U-1L	213	SRCS-TC-RF-LP-1	TE-17 - Reflood loop temperature
	U-1L	214	SRCS-TC-RF-TA-1	TE-18 - Reflood assembly temperature
	U-2L	215	U2LP-FL-OW-1	FT-4D - Water flow rate
		216	UNDEFINED	
	U-1L	217	SRCS-S-PS-OT-2	PT-4 Outlet steam pressure
	U-1L	218	ACUM-WT	WIS Reflood weight
	U-2L	221	U2LP-TC-IN-1	TE-78 Inlet water temperature
	U-2L	222	U2LP-TC-OT-1	TE-79 Outlet water temperature

TABLE 1. (contd)

Instrument Level (Old)	Instrument Level New	OACS Channel Number	Sensor Name	Notes
U-1L		223	SRCS-FR-HI-B	FE-3B Reflood flow high
U-1L		224	SRCS-FR-LO-B	FE-4B Reflood flow low
U-1L		257	SRC-S-OE-LT-A	LCS LCS control switch
NRU[m]		258	SRC-RF-TR-IP	LCS[m] NRU trip switch
(1)	-0.2	289	PS-(-)0.2-In-1	
(2)	8.7	290	PS-08.7-SH-1	
(3)	13.9	291	PS-13.9-SH-1	
(3)	13.9	292	PS-13.9-CR-2	
(4)	19.2	293	PS-19.2-SH-1	
(5)	28.0	294	PS-28.0-SH-1	
(6)	34.9	295	PS-34.9-SH-1	
(7)	40.2	296	PS-40.2-GR-IR-1	
(7)	42.9	297	PS-42.9-SH-1	
(8)	42.9	298	PS-42.9-SH-1	
(9)	55.9	299	PS-55.9-SH-1	
(10)	55.9	300	PS-55.9-SH-1	
	61.2	301	PS-61.2-GR-IR-1	
	69.0	302	PS-69.0-CR-SR-1	
	73.9	303	PS-73.9-CR-IR-1	
(13)	76.9	304	PS-76.9-SH-1	
	118.9	305	PS-118.9-SH-2	
(13)	76.9	306	PS-CR-IR-1	
	78.9	307	PS-78.9-CR-IR-1	
	80.9	308	PS-80.9-CR-IR-1	
	83.9	309	PS-83.9-CR-IR-1	
(14)	90.0	310	PS-90.0-CR-SP-1	
(15)	97.9	311	PS-97.9-SH-1	
	97.9[o]	312	PS-97.9-GR-1R-1[o]	
	97.9	313	PS-97.9-GR-IR-2	103, 53, 45, 24, 2[o]
	97.9	314	PS-97.9-GR-C-1	
(14)	97.9	315	PS-97.9-CR-IR-1	
	101.9	316	PS-101.9-CR-IR-1	
	104.9	317	PS-104.9-CR-IR-1	
(16)	111.0	318	PS-111.0-CA-1	
	118.9	319	PS-118.9-SH-1	
	118.9	320	PS-118.9-SH-1	
(17)	118.9	321	PS-118.9-CR-IR-1[o]	138, 126, 92, 81[o]

TABLE 1. (contd)

Instrument Level (Old)	New	DACS Channel Number	Sensor Name	Notes
(18)	139.9	322	PS-139.9-SH-1	
	139.9	323	PS-139.9-GR-IR-1	
	148.2	324	PS-148.2-CR-SR-1	
(20)	155.7	324	PS-155.7-CR-8-1	
	168.7	326	PS-168.7-OT-1	
	76.9	327	PS-76.9-GR-IR-1	

(a) Distance above top of the tie plate, e.g., -0.2 in., 8.7 in., and 13.9 in.
(b) Failed during preconditioning.
(c) These sensors are located at a different axial location than in previous tests.
(d) Pseudo sensor (DACS No. 327) PS-76.9-GR-IR-1 is composed of DACS Nos. 1D1, 52, 50, and 67.
(e) Protective trip sensors and pseudo sensors.
(f) Safety pseudo sensor (DACS No. 312) PS-97.9-GR-IR-1 is composed of DACS Nos. 103, 25, 45, 24, and 2.
(g) Safety pseudo sensor (DACS No. 321) PS-118.9-GR-IR-1 is composed of DACS Nos. 81, 126, 92, and 138.
(h) Head-mounted pressure transducers that were connected to fuel rods with high-pressure tubes.
(i) Full-length TDR liquid level detector.
(j) Sampled coolant and orifice pressure drop.
(k) Located in the U-1 loop of the NRU (steam- and reflood-cooled system).
(l) Located in the U-2 loop of the NRU (water-cooled system).
(m) NRU safety trip circuit indicator.
(n) Safety trip circuit pseudo sensors and "averaged" DACS numbers.
(o) Protective trip sensors.

required power distribution.[5] NRU reactor fuel operations reloaded specified[6] fuel as required to assure less than a 5% radial power skew across the L-24 position.

EXPERIMENT OPERATIONS

The primary objective of the MT-4 experiment was to evaluate cladding ballooning and rupture during adiabatic heatup in the range from 1033 to 1200K (1400 to 1700°F). To meet that objective, the unirradiated test fuel rods were first preconditioned.

After installation, the test rod pressure system was purged, sampled, and maintained at relatively low pressure--0.21 MPa (30 psia). Two approaches to full-power NRU operation and two conditional reactor trips assured fuel pellet cracking and good fuel/cladding mechanical contact. This preconditioning operation provided up to 1 equivalent-full-power-hour (EFPH) of NRU irradiation with operating conditions similar to those of previous experiments. The operating conditions are summarized in Table 2.

After preconditioning operations were completed and the L-24 piping was connected to the U-1 loop for steam supply and the reflood system, reflood system calibrations began (see Table 2). Reflood calibrations focused on the mode of operation used in the MT-4 transient, preset reflood delay time and flow rates, and computer-controlled flow after the transient. Initial operations checked out the prefill and drain sequence that delivered reflood water to the bottom of the fuel column. Constant reflood flow rates were calibrated from 0 to 0.0508 m/s (2.0 in./s), at 0.0965 m/s (3.8 in./s), and at 0.1219 m/s (4.8 in./s). Pretransient power coupling measurements were made using adiabatic heatup and standard calorimetric techniques with steam cooling at a power level of ~6 to 8 MW. During these measurements, the protective trip circuitry was also verified using a fuel cladding trip set point at ~922K (1200°F) to terminate the calibration tests.

When it was confirmed that the test assembly instrumentation and reflood loop control system (LCS) were ready, the 12 test rods were pressurized to ~4.62 MPa (670 psia) using the pressure system mounted on the NRU reactor head. The pressure tubes were subsequently sealed and independently monitored during the remainder of the experiment. MT-4 transient operation repeated the operating conditions used in TH-1.16.[4] After steam flow stopped, reflood flow was delayed for ~57 s. Reflood flow at a rate of ~0.0965 m/s turned the peak cladding temperature (1200K) around at ~70 s into the transient.

After the peak fuel cladding temperature was reached, reflood flow was stopped briefly and then re-established (with computer control) at ~0.00508 m/s (0.2 in./s) to stabilize the cladding temperature at ~1089K (1500°F) for post-LOCA heat transfer coefficient measurements (2 to 3 min). Reflood flow was again reduced to establish a stable cladding temperature at ~1300K (1700°F) and to enable similar heat transfer coefficient measurements (2 to 3 min). The experiment was terminated with a conditional NRU reactor trip and test assembly quench using reflood cooling.

TABLE 2. MT-4 Operating Conditions

Parameter	Preconditioning	Calibrations Steam, Reflood, Water		Pretransient	Transient[a]
Reactor power, MW	127	0 to 8		8	8
Coolant	U-2 water	U-1 steam	U-2 water	U-1 steam[a]	Reflood water
Inlet temperature, K (°F)	518 (472)	435 (325)	311 (100)	435 (325)	NA
Coolant flow, kg/s (lbm/h)	0 to 16.3 (0 to 129, 400)	0.378 (3000)	0.441 (3500)	0.378 (3000)	0
Reflood delay, s	NA	0 to 51		NA	51
Reflood rate, m/s (in./s)	NA	0 to 0.1219 (0 to 4.8)		NA	0 to 0.0965 (0 to 3.8)
Reflood temperature, K (°F)	NA	311 (100)		NA	311 (100)
Peak cladding temperature, K (°F)	700 (800)	444 (340)		728 (850)	1200 (1700)
Reactor trip criteria[b]	Safety	Protective and safety		Protective and safety	Protective and safety

(a) Transient initiated by termination of steam flow.
(b) See Table 3 for trip set points and operating limits.

PROTECTIVE AND SAFETY TRIPS

Previous transient tests qualified the protective trip set points, overshoot, and the protective trip circuitry. Protective trips were designed to limit fuel temperatures to that range compatible with the experiment objectives and operations.

Protective trip logic limited test fuel temperatures in several ways: maximum allowable fuel cladding temperature, maximum and minimum allowable heatup rates, and time limitations for both fuel cladding temperatures and coolant reflooding to preselected levels. These protective trip options and their set points were selected by the experimenter at various times during the course of the tests. The protective trip signal was designed for several operating condition criteria, and the LCS and NRU reactor controls 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. The safety trip logic and circuitry were nearly all the same as designed for previous experiments.[3] The only revision was the removal of the minimum reflood flow rate limit to permit the use of zero and very low reflood flow rates during post-transient operation. During that period, temperature stabilization was provided by computer-controlled reflood flow rates that included zero flow intervals. Safety trip set points are summarized in Tables 3 and 4. Protective trip set points were selected and activated by the experimenter at various times during the experiment. Both protective and safety trips utilized the same hard-wired circuitry to achieve an NRU reactor trip. Each trip condition had its own independent program logic; thus, safety trips were not influenced by any change in protective trips.

TABLE 3. Pretransient and Transient Safety Trip Set Points[a]

Parameter	Location	Use	Operating Limits	Safety Trip Set Point
Hanger tube temperature - high	LCS	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[a]				
Level (13) 76.9 305 PS-76.9-GR-IR-1	OACS	Transient	1305K (1890°F)	1361K (1990°F)
Level (15) 97.9 308 PS-97.9-GR-IR-1	DACS		1305K (1890°F)	1361K (1990°F)
Level (17) 118.9 315 PS-118.9-GR-IR-1	DACS		1278K (1840°F)	1333K (1940°F)
Low steam flow trip	LCS	Pretransient (3000 lbm/h)	0.378 kg/s (2200 lbm/h)	0.277 kg/s
Accumulator inventory - low	LCS	Transient	22.7 kg (50 lbm)	11.3 kg (25 lbm)

(a) Standard trip set point criteria; see Table 4 for nonfunctional TC criteria.

TABLE 4. Standard and Defaulted Safety Trip Set Point Criteria for Operating and Nonfunctional Fuel Cladding Thermocouples

Operating Conditions	Criteria and Safety Margins (SM)	Sensors at Level (17) 118.9	Sensors at Level (15) 97.9	Sensors at Level (13) 76.9
		DACS Thermocouple Numbers		
		315, PS-118.9-GR-IR-1[a] (81,126,92,138)	308, PS-97.9-GR-IR-1 (103,25,45,24,2)	305, PS-76.9-GR-IR-1 (101,52,50,67)
		Safety Trip Set Point Temperatures, K (°F)		
$2 \leq$ Number of operating TCs on each of Levels 76.9, 97.9, and 118.9	Standard SM = 56K (100°F)	1333 (1940)	1361 (1990)	1361 (1990)
$2 \leq$ Number of operating TCs on each of Levels 97.9 and 118.9	Standard SM = 56K (100°F)	1333 (1940)	1361 (1990)	
$2 \leq$ Number of operating TCs on each of Levels 97.9 and 118.9 or Levels 76.9 and 118.9	Alternate SM = 84K (150°F)	1305 (1890)	1333 (1940)	1333 (1940)
$2 \leq$ Number of operating TCs on only Level 97.9, 118.9, or 76.9	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: 81, 126, 92, 138.

SAFETY HAZARDS REVIEW

The MT-4 experiment was similar to previous experiments (TH-1.16, MT-2, and MT-3). Both the MT-4 operating conditions and the expected fuel failure mechanisms had been tested in previous experiments.

The same operating conditions that were used in TH-1.16 and MT-3 were used in MT-4. Fuel preconditioning was produced by up to 1 EFPH in the NRU reactor. The simulated LOCA transient was produced by a reflood delay of ~57 s followed by reflood coolant flow at ~0.0965 m/s (3.8 in./s) (same as in TH-1.16). For MT-4, however, the 12 test rods were pressurized just prior to the transient that caused ballooning and rupture, which precluded premature ballooning and rupture during earlier transients. After the transient, fuel cladding high temperatures stabilized and the procedures and operating ranges established in MT-2 and MT-3 were used. Computer-controlled reflood flow enabled thermal-hydraulic data to be collected on the deformed test assembly during oscillatory operating conditions.

The 12 test rods were expected to balloon and rupture ~60 to 80 s into the transient when the fuel cladding reached ~1144K (1600°F). The neutronic environment and the fuel rod internal pressure were carefully monitored in MT-4. The MT-4 experiment was the first experiment to use pressure transducers located outside the NRU reactor to provide more reliable rupture times. The 12 pressure transducers were mounted above the reactor head but were connected to the test rods with high-pressure Type 321 stainless steel tubing.[a] The lowest yield strength pressure rating of the (weakest) pressure system component outside the reactor (high-pressure tubing) that communicated with fission gas in the test fuel rods was 106.9 MPa (15,500 psi) at a temperature of 700K (800°F). This compares with a maximum credible hoop stress of 15.03 MPa (2180 psi) due to the fission gas pressure at the time of fuel cladding rupture. During transient operations, the valves on these high-pressure tubes were sealed to preclude fission gas leakage.

Other safety analysis reports[2,7] are valid for the MT-4 experiment because the operating conditions, expected fuel failure mechanisms, and failure effects are within the envelope of cases analyzed there and reported for subsequent experiments.

(a) 1.65 mm (0.065 in.) outside diameter and 0.635 mm (0.025 in.) inside diameter.

MT-4 OPERATING PROCEDURE

PRECONDITIONING OPERATIONS

Test Configuration

1. Install MT-4 test assembly in L-24 NRU reactor position.
2. Purge test fuel rod pressure system and collect helium samples before preconditioning.
3. Install traveling fission chamber in J-22 reactor position.
4. Connect U-2 loop to L-24 test position.
5. 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 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	≤ 1 ppm
	Oxygen	≤ 100 ppm
	All other elements	≤ 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 (100 to 500°F)	±1K (±2°F)
TE-79	Outlet temperature	222	323 to 589K (122 to 600°F)	±1K (±2°F)
FT-40	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	552K (534°F)	561K (550°F)
Pump subcooling temperature--low	338K (150°F)	283K (50°F)
Coolant flow--low	15.5 kg/s[a] (123,000 lbm/h)	13.4 kg/s[a] (106,300 lbm/h)
Surge tank level--low	TBD[b]	30%
Surge tank pressure--high	TBD[b]	8.90 MPa (1275 psig)

(a) At a temperature of 518K (472°F).
(b) To be determined by CRNL.

NRU Reactor Preparations

1. Load NRU reactor fuel assemblies and absorber assemblies as required.[6]

2. Adjust the neutron detector scatter plug as required.

3. Establish mean power trip set points as approved.

4. 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 disks 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 SPND coefficients are correct.

Preconditioning Operating Procedure (Total time at full NRU reactor power is to be limited to 1.0 EFPH.)

1. End the dummy test, and start a new test on the DACS.

2. Start the loop using sparge pumps to provide initial flow [6.3 kg/s (129,400 lbm/h)] and initial pressurization [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 518 ±3K (472 ±5°F).

5. Print the DACS sensor status report and REDACE print-out for review for 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 ±5%.

10. Print the DACS sensor status report and the REDACE print-out.

11. Perform a power calibration, including neutron flux data records (unless done at low power) using the traveling fission chamber in the J-22 reactor position.

12. Conditionally trip the reactor.

13. Repeat Steps 3 and 5.

14. Increase the reactor to full power (127 MW ±5%), and maintain test assembly inlet temperature at 518 ±3K (472 ±5°F).

15. Print the DACS and REDACE sensor status reports.

16. Recheck power calibrations and traveling fission chamber data.

17. Conditionally trip the reactor.

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

19. 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.

20. Place one copy of all data records collected in system log and have data coordinator sign.

REFLOOD CALIBRATION, PRETRANSIENT, AND TRANSIENT OPERATIONS

Reflood calibration, pretransient, and transient operations are discussed in this section. MT-4 operating conditions are summarized in Table 8.

Test Configuration

1. MT-4 assembly installed in L-24 NRU reactor position.

2. Fuel rod pressure system connected to the MT-4 test assembly.

3. Traveling fission chamber installed in J-22 reactor position.

4. Reflood loop connected to the L-24 NRU reactor position.

5. U-1 steam supply connected to the reflood loop.

6. NRU reactor fuel and absorber assemblies loaded as required.(6)

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 loop instruments as shown in Table 9. Enter conversion factors and units in the test log.

7. Implement the safety trip set points as shown in Tables 10 and 11 for the pretransient and transient phases of the experiment. The safety pseudo sensors used by the DACS to represent the high cladding temperature trip set points and sensors are identified on Table 12.

TABLE 8. MT-4 Operating Conditions

Parameter	Calibrations Steam, Reflood, Water		Adiabatic Test	Pretransient	Transient
Reactor power, MW	0 to 8		6 to 8	6 to 8	6 to 8
Coolant	U-1 steam	U-2 water	U-1 steam[a]	U-1 steam[a]	Reflood water
Inlet temperature, K (°F)	435 (325)	311 (100)	435 (325)	435 (325)	NA
Coolant flow, kg/s (lbm/h)	0.378 (3000)	0.441 (3500)	0.378 (3000)	0.378 (3000)	0
Reflood delay, s	0 to 51		NA	NA	51
Reflood rate, m/s (in./s)	0 to 0.1219 (0 to 4.8)		NA	NA	0 to 0.0965 (0 to 3.8)
Reflood temperature, K (°F)	311 (100)		NA	NA	311 (100)
Peak cladding temperature, K (°F)	444 (340)		853 (1075)	728 (850)	1200 (1700)
Reactor trip criteria[b]	Protective and safety		Protective[c] and safety	Protective and safety	Protective and safety

(a) Transient initiated by termination of steam flow.
(b) See Table 3 for trip set points and operating limits.
(c) Protective trip set point was 825K (1025°F) for adiabatic tests.

TABLE 9. Steam/Reflood System Calibration (CRNL)

Sensor	Parameter	DACS	Instrument Range	Acceptable Accuracy
TE-2	Inlet coolant temperature	206	394 to 700K (250 to 300°F)	±1K (±2F)
TE-3	Outlet coolant temperature	207	422 to 973K (300 to 1300°F)	±6K (±10°F)
FY-6	Steam flow rate, reflood boiling	210	0 to 0.378 kg/s (0 to 3000 lbm/h)	±0.014 kg/s (±100 lbm/h)
FV-1	Inlet steam system control flow rate	211	0 to 0.378 kg/s (0 to 3000 lbm/h)	±0.014 kg/s (±100 lbm/h)
FI-2	Outlet steam system control flow rate	212	0 to 0.032 kg/s (0 to 250 lbm/h)	±0.0013 kg/s (±10 lbm/h)
PT-5	Steam inlet pressure	208	0 to 0.69 MPa (0 to 100 psia)	±0.034 MPa (±5 psia)
PT-6	Steam outlet pressure	209	0.069 to 0.345 MPa (0 to 50 psia)	±0.017 MPa (±2.5 psia)
PT-4	Steam pressure control, outlet region	217	0 to 0.69 MPa (0 to 100 psia)	±0.034 MPa (±5 psia)
FI-4	Reflood coolant, low flow rate	201	0 to 0.254 m/s (0 to 10 in./s)	±5%
FI-3	Reflood coolant, high flow rate	202	0.013 to 0.305 m/s (0.5 to 12 in./s)	±5%
FI4b	Reflood coolant, low flow rate (backup)	224	0 to 0.254 m/s (0 to 10 in./s)	±5%
FI3b	Reflood coolant, high flow rate (backup)	223	0.013 to 0.305 m/s (0.5 to 12 in./s)	±5%
FE-9	Standby reflood coolant flow	203	0.013 to 0.305 m/s (0.5 to 12 in./s)	±10%
TE-17	Reflood coolant temperature control valve inlet	213	305 to 322K (90 to 120°F)	±5%
TE-18	Reflood coolant temperature control valve outlet	214	305 to 322K (90 to 120°F)	±5%

NRU Reactor Preparations

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. Establish mean power, log rate (at 10%/s), and auxiliary log rate (at 10%/s) trip set points as required.

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

DACS Computer Preparations

1. Load the DACS with a 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 13.

8. Format the transient graphic display.

9. Identify, record (log), and remove the 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 calibration test.

TABLE 10. Pretransient and Transient Safety Trip Set Points

Parameter	Location	Use	Operating Limits	Trip Set Point
Hanger tube temperature - high	LCS	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[(a,b)]				
Level (13) 76.9, 327, PS-76.9-GR-IR-1	DACS	Transient	1305K (1890°F)	1361K (1990°F)
Level (15) 97.9, 312, PS-97.9-GR-IR-1	DACS		1305K (1890°F)	1361K (1990°F)
Level (17) 118.9, 321, PS-118.9-GR-IR-1	DACS		1278K (1840°F)	1333K (1940°F)
Low steam flow trip	LCS	Pretransient	0.378 kg/s (3000 lbm/h)	0.277 kg/s (2500 lbm/h)
Accumulator inventory - low	LCS	Transient	22.7 kg (50 lbm)	11.3 kg (25 lbm)

(a) Standard trip set point criteria; see Table 11 for nonfunctional TC criteria.
(b) In use during transient operation only.

TABLE 11. Standard and Defaulted Safety Trip Set Point Criteria for Operating and Nonfunctional Fuel Cladding Thermocouples

Operating Conditions	Criteria and Safety Margins (SM)	Sensors at Level (17) 118.9	Sensors at Level (15) 97.9	Sensor at Level (13) 76.9
		DACS Thermocouple Numbers		
		321, PS-118.9-GR-IR-1[(a)] (81,126,92,138)	312, PS-97.9-GR-IR-1 (103,25,45,24,2)	327, PS-76.9-GR-IR-1 (101,52,67,50)
		Safety Trip Set Point Temperatures, K (°F)		
2 $\leq$ Number of operating TCs on each of Levels 76.9, 97.9, and 118.9	Standard SM = 56K (100°F)	1333 (1940)	1361 (1990)	1361 (1990)
2 $\leq$ Number of operating TCs on each of Levels 97.9 and 118.9	Standard SM = 56K (100°F)	1333 (1940)	1361 (1990)	
2 $\leq$ Number of operating TCs on each of Levels 97.9 and 118.9 or Levels 76.9 and 118.9	Alternate SM = 84K (150°F)	1305 (1890)	1333 (1940)	1333 (1940)
2 $\leq$ Number of operating TCs on only Level 97.9, 118.9, or 76.9	Fallback SM = 111K (200°F)	1277 (1840)	1305 (1890)	1305K (1890)

(a) These pseudo sensor data (PSD) are the calculated time-average of the following DACS sensor-numbered data: 81, 126, 92, 138.

TABLE 12. Cladding High-Temperature Trip Sensors

Level (Old)	Level New	Sensor Thermocouples	DACS Sensor Number	DACS Pseudo Sensor	DACS Sensor Number
(13)	76.9	TC-76.9-6C-IR-3	101	PS-76.9-GR-IR-1	327
		TC-76.9-4A-IR-3	52		
		TC-76.9-5B-IR-1	67		
		TC-76.9-3F-IR-1	50		
(15)	97.9	TC-97.9-1C-IR-2	103	PS-97.9-GR-IR-1	312
		TC-97.9-5E-IR-2	53		
		TC-97.9-5B-IR-3	45		
		TC-97.9-2B-IR-4	24		
		TC-97.9-2E-IR-1	2		
(17)	118.9	TC-118.9-1B-IR-2	81	PS-118.9-GR-IR-1	32
		TC-118.9-3B-IR-3	126		
		TC-118.9-4C-IR-8	92		
		TC-118.9-3D-IR-8	138		

TABLE 13. DACS Immediate Display Test Sensors

Sensor Name	DACS Sensor Number	Sensor Name	DACS Sensor Number
TC-76.9-2C-IR-3	127	TC-118.9-3B-IR-3	126
TC-97.9-2B-IR-C	9	TC-118.9-1B-IR-2	81
TC-97.9-2C-IR-8	55	TC-118.9-6E-IR-4	68
TC-97.9-2D-IR-8	119	TC-118.9-6F-S-C	111
TC-97.9-3E-IR-8	122	TC-97.9-6F-S-C	76
TC-118.9-4C-IR-8	92	TC-139.9-6F-S-C	129
TC-118.9-3D-IR-8	138		

Reflood Calibration Operating Procedure

1. Calibrate the reflood prefill controls to fill the test nozzle annulus 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. For successive calibrations, enter the following reflood rates: 0.00508 m/s (0.2 in./s); 0.0254 m/s (1.0 in./s); 0.0508 m/s (2.0 in./s); and 0.0965 m/s (3.8 in./s).

5. Print a sensor status report to insure that the test assembly and all TCs are $\geq$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 three times and drain twice.

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 calibration transient.

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

12. Stop the test when reflood water passes the TCs at Level (20) 168.7.

13. Drain the MT-4 test train in L-24, and repeat Steps 2 through 12 until all reflood calibrations are completed.

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

15. 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 211)

    - TCs at each level (DACS sensors 289, 291, 296, 299, 302, 304, 327, 307, 312, 310, 321, 319, and 320).

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

17. Turn off the video 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. Pressurize the 12 test rods to 4.62 MPa (670 psia) using helium; monitor the pressure; and remove the charging system and seal the valves.

20. As necessary, repeat DACS Computer Preparations and Loop System Preparations before proceeding to Pretransient Operating Procedure.

Steam and Water Flow Power Calibration Procedure

1. Place DACS in the steady-state mode.

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

3. Manually valve (top of room 501) condensate water from the U-1 boiler feed-water pumps into the reflood accumulator fill line.

4. Bring the NRU reactor to a low neutron power level.

5. LCS operator will set timers and controls for a zero reflood delay and a reflood flow rate of 0.1219 m/s (4.8 in./s).[a] Values must be approved by the test director.

6. Increase NRU reactor power to nominally 5% of full power, scan the DACS immediate display for the hottest TCs, and reproduce the display.

7. Insure that the test assembly inlet steam temperature stabilizes at 435 ±8K (325 ±15°F).

8. Adjust NRU power to obtain a steady-state temperature increase of 178 ±4K (320 ±7°F) across the test assembly.

9. When steady-state operating conditions have been reached, switch the DACS to the transient mode; 20 s later, issue the verbal command "BEGIN THE POWER CALIBRATION TRANSIENT" (directed to the LCS operator).

(a) The reflood water flow rate for calorimetry calibration corresponds to a mass flow rate of 0.441 kg/s (3500 lbm/h).

10. LCS operator begins the power calibration transient to establish water cooling for power calibration measurements; record time in the Test Parameter Log.

11. Insure that the test assembly inlet water temperature stabilizes at 311 ±6K (100 ±10°F).

12. When all fuel cladding TCs have quenched, change the DACS operating mode to steady state. Note the control rod movements necessary to maintain a constant NRU reactor power.

13. When steady-state operating conditions have been reached and measurements are completed, shut down the reactor and shut off reflood water flow.

14. Return valve configuration (top of room 501) to normal position supply for the reflood accumulator fill line.

15. Change the DACS operating mode to idle.

16. Make historical requests on the DACS graphic display and reproduce the basic thermal-hydraulic power calibration data.

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

Adiabatic Test Procedure

1. Set the protective reactor trip set points for DACS and LCS control (values must be approved and recorded by the test director); see Table 14.

2. LCS operator will set timers for reflood delay times; record values (values must be approved by the test director); see Table 14.

3. Check accumulator levels (weights), record values in the Test Parameter Log, and drain the test assembly in preparation for steam cooling.

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

TABLE 14. Adiabatic Experiment Conditions Log

A. Protective trip temperatures for fuel Levels (13) 76.9, (15) 97.9, and (17) 118.9 825K (1025°F)

B. Protective trip temperatures for shroud Levels (15) 97.9, (17) 118.9, and (18) 139.9 825K (1025°F)

C.

MT Run Number	Delay Time, s	Reflood Rate m/s (in./s)	Reflood Rate Duration, s
1	40	0	
2		0.122 (4.8)	Continue until steady state is reached

D. Maximum allowed time (___ s) for reflood water to quench sensor __ at Level ____ not applicable

E. DACS control No

F. Number and settings for DACS control reflood rates not applicable

G. Control function coefficients not applicable

H. Pressurization level 0.101 MPa (14.7 psia), time ______, and temperature Room Temperature.

I. Fuel rods pressurized:

______ Yes ___X___ No ______ Qty

Sign off ______________________________

Test Director ______________________________

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

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

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

8. Before proceeding, NRU reactor operator must acknowledge that the neutron power will not exceed the neutron power level specified by the test director (nominally 5.0%).

9. Increase NRU neutron power to the level specified. With the power at nominally 5.0% of the NRU reactor fuel power, scan the DACS immediate display for the hottest TC and reproduce the display.

10. Insure that the test assembly inlet temperature stabilizes at 435 ±8K (325 ±15°F).

11. Adjust the reactor power to obtain a steady-state 178K (320°F) temperature increase across the test assembly.

12. Reproduce the DACS immediate and graphic displays.

13. Activate the video tape recorder.

14. Prefill a total of three times and drain twice. Display test assembly temperatures on the DACS graphic display.

15. 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 Parameter Log.

16. The LCS operator begins the transient.

17. When the fuel cladding temperature reaches 825K (1025°F) or the protective trip set points on the DACS and LCS, the NRU reactor is shut down or automatically tripped and full reflood flow quenches the test assembly.

18. Repeat Steps 1 through 17 until the steam flow and NRU reactor power provide the required 8.3K/s (15°F/s) full-temperature ramp rate. Leave the test assembly filled with coolant water.

TABLE 15. LOCA Experiment Conditions Log

A. Protective trip temperatures for fuel Levels (13) 76.9, (15) 97.9, and (17) 118.9 ______

B. Protective trip temperatures for shroud Levels (15) 97.9, (17) 118.9, and (18) 139.9 850K (1070°F) for Level 118.9 ______

C.

MT Run Number	Delay Time, s	Reflood Rate m/s (in./s)	Reflood Rate Duration, s
1	57		
2		0.203 (8)	6
3		0.102 (4)	6
4		0.025 (1)	3

D. Maximum allowed time (___ s) for reflood water to quench sensor ___ at Level not applicable

E. DACS control Yes, 72 s after start of transient

F. Number and settings for DACS control reflood rates

Setting	Reflood Rate, m/s (in./s)	Setting	Reflood Rate, m/s (in./s)
1	0	6	0.0152 (0.60)
2	0.0051 (0.20)	7	0.0203 (0.80)
3	0.0076 (0.30)	8	0.0254 (1.00)
4	0.0089 (0.35)	9	0.0508 (2.00)
5	0.0102 (0.40)		

G. Control function coefficients ΔT = 1.0, T' = 10.0, T" = 6.0

ΔT = absolute value of minimum allowable temperature deviation

T' = differential temperature gain coefficient

T" = second derivative of temperature gain coefficient

H. Pressurization level 4.62 MPa (670 psia), time 10:30 p.m., and temperature 296K (73°F)

I. Fuel rods pressurized:

X Yes ______ No 12 Qty

Sign off ______

Test Director ______

Pretransient Operating Procedure

1. Set the protective reactor trip set points for DACS and LCS control (values must be approved and recorded by the test director); see Table 15.

2. LCS operator will set timers for reflood delay times (Table 8); record values (values must be approved by the test director); see Table 15.

3. Check accumulator levels (weights), record values, and drain the test assembly in preparation for steam cooling.

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

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

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

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

8. Before proceeding, NRU reactor operator must acknowledge that neutron power will not exceed percent power level specified by test director (nominally 5.0%).

9. Increase NRU reactor power to the level specified. With the reactor power at nominally 5.0% of NRU full power, scan the DACS immediate display for the hottest TC and reproduce the display.

10. Insure that the test assembly inlet temperature stabilizes at 435 ±8K (325 ±15°F).

11. If necessary, adjust the reactor power to obtain a steady-state 178K (320°F) temperature increase across the test assembly; begin collecting traveling fission chamber data.

12. Check the peak cladding temperature, steam flow, test assembly inlet temperature, and outlet pressure. Complete fission chamber data records and attach them to the Test Parameter Log.

13. Reproduce the DACS immediate and graphic displays.

14. Activate the video tape recorder.

Transient Operating Procedure

1. Prefill the inlet annulus a total of three times and drain it twice. 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 Parameter Log.

3. The LCS operator begins the transient.

4. When fuel cladding temperatures exceed 1200K (1700°F), the LCS will revert reflood flow control to the DACS-selected rates--0 to 0.00508 m/s (0.2 in./s).

5. Complete thermal-hydraulic measurements at stabilized peak cladding temperatures of 1089K (1500°F) and 1200K (1700°F) (~2 to 3 min each).

6. Shut down the NRU reactor when measurements are complete, and use full reflood flow to quench the test assembly.

7. Shut off reflood water flow only after upper TCs have quenched.

8. Record reflood water used (accumulator weight difference) in the Test Parameter Log.

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

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

11. End the test on the DACS. Verify that the Test Parameter Log is completed.

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

13. Copy the following historical data on the DACS:

    - Make a hard copy of the hottest pseudo sensors at Levels (15) 97.9 and (17) 118.9.

    - Plot data for the following sensors: 289, 290, and 211.

REFERENCES

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

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

3. Russcher, G. E., et al. June 1983. Experiment Operations Plan for the MT-3 Experiment in the NRU Reactor. NUREG/CR-2599, PNL-4190, 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. Mohr, C. L., et al. April 1981. Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents. NUREG/CR-1882, PNL-3681, Pacific Northwest Laboratory, Richland, Washington.

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

7. Axford, O. 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)

1.2 Traveling fission chamber data (attach to log)

2.0 PRETRANSIENT ACTIVITIES

Tape number: ____________

Disk configuration: DP0 ____________

DP1 ____________

DP1F ____________

2.1 Operation summary

PARAMETER/UNITS	SPECIFIED VALUE (ACCEPTED RANGE)	ACTUAL VALUE
Steam flow rate, lbm/h	3000 (±5%)	____________
Steam inlet temperature, °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) 118.9 - high	________
Level (15) 97.9 - high	________
Level (13) 76.9 - high	________
Time to quench Level (20) 168.7	________

2.4 Traveling fission chamber data (attach to log)

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 CONOITIONS LOG

APPENDIX B

EXPERIMENT CONDITIONS LOG

A. Protective trip temperatures for fuel Levels (13) 76.9, (15) 97.9, and (17) 11B.9 ______________________________

B. Protective trip temperatures for shroud Levels (15) 97.9, (17) 11B.9, and (18) 139.9 ______________________________

C.

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

D. Maximum allowed time (___ s) for reflood water to quench sensor _____ at Level ______________________________

E. DACS control ______________________________

F. Number and settings for DACS control reflood rates ____________________

G. Control function coefficients ______________________________

H. Pressurization Level __________, time _________, and temperature _________

I. Fuel rods pressurized:

_____ Yes ________ No _________ Qty

Sign Off ______________________________

Test Director ______________________________

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NRC FORM 335
(11-81)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER *(Assigned by DDC)*
NUREG/CR-2881
PNL-4293

4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)*
EXPERIMENT OPERATIONS PLAN FOR THE MT-4 EXPERIMENT IN THE NRU REACTOR

2. *(Leave blank)*

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S) G.E. Russcher, C.L. Wilson, L.J. Parchen, R.K. Marshall, G. M. Hesson, B.J. Webb, M.D. Freshley

5. DATE REPORT COMPLETED

MONTH	YEAR
May	1983

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Pacific Northwest Laboratory
P.O. Box 999
Richland, WA 99352

DATE REPORT ISSUED

MONTH	YEAR
June	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-4--the fourth materials deformation experiment conducted in the National Research Universal (NRU) reactor, Chalk River, Ontario, Canada. A major objective of MT-4 was to simulate a pressurized water reactor LOCA that could induce fuel rod cladding deformation and rupture due to a short-term adiabatic transient and a peak fuel cladding temperature of 1200K (1700°F).

17. KEY WORDS AND DOCUMENT ANALYSIS
thermal-hydraulic
loop control system
data acquisition and control system
light-water reactor fuel bundles
cladding deformation and rupture

17a. DESCRIPTORS
experiment operations plan
loss-of-coolant accident simulation
National Research Universal (NRU) reactor

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS *(This report)*
Unclassified

20. SECURITY CLASS *(This page)*
Unclassified

21. NO. OF PAGES
51

22. PRICE
S