MASTER

HEAT TRANSFER FROM ELECTRICALLY HEATED RODS
DURING A SIMULATED LOSS-OF-COOLANT ACCIDENT*

C. G. Lawson
Oak Ridge National Laboratory
Oak Ridge, Tennessee

To Be Presented at the American Institute
of Chemical Engineers Annual Meeting,
San Francisco, California
November 29-December 2, 1971

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

* Research Sponsored by the U.S. Atomic Energy Commission
Under Contract With Union Carbide Corporation.

Abstract

The loss of primary coolant from a pressurized water reactor as a result of an instantaneous rupture of a main circulating line is called a LOCA and is perhaps the most serious accident in terms of radioactivity release from the fuel to the outer containment system that can be realistically postulated. Emergency Core Cooling Systems (ECCS) are provided to maintain continuity of energy removal from the reactor core following such an event. Paramount to the evaluation of the ECCS is knowledge of the fuel and fuel-clad temperatures at the end of the coolant loss period of 8 to 14 seconds. Heat-transfer data have not been available for the conditions of heater power (heat flux) and of transient coolant pressure, velocity, and temperature comparable to the environment around a fuel-rod bundle during a loss-of-coolant transient. Appropriate transient heat-transfer data have now been obtained with single Zircaloy clad heaters (0.5-in.-OD x 2-ft-long) and a seven-rod cluster in a closed cycle, circulating, 1500 psi pressurized, water loop at power levels of 17.5 kw/ft ($450,000 \text{ Btu/hr}\cdot\text{ft}^2$) with cooling water near 500°F inlet and 1500 psi during a 12-second depressurization and coolant loss. Flow regimes associated with bulk-nucleate boiling and with thin film surface evaporation are clearly discernible where the temperature difference and heat flux data are considered. The high heat-transfer coefficients were observed for times greater than the heat-removal time constants of the heater rods and suggest that assumptions made in current calculations are conservative.

Key Words

Unsteady Flow-Multi Phase, Heat Transmission-Unsteady Flow, Nuclear Fuel Assemblies, Nuclear Fuel Cladding-Zirconium, Nuclear Reactor Accidents, Nuclear Reactor Engineering, Pressurized Water Reactors, and Pressure Reduction

HEAT TRANSFER FROM ELECTRICALLY HEATED RODS
DURING A SIMULATED LOSS-OF-COOLANT ACCIDENT*

In a Pressurized Water Reactor

C. G. Lawson
Oak Ridge National Laboratory

→ The Loss-of-Coolant Accident (LOCA) is a design basis accident that is evaluated as part of the information presented to the Atomic Energy Commission in support of an application for a nuclear power plant construction permit.^{1,2} *Pressurized water* The LOCA is hypothetical in that it presupposes *a complete and* the instantaneous rupture of one primary coolant circulating pipe line, typically ~2.5 ft in diameter, with the nuclear steam generator at full power. *As* A consequence of the instantaneous large pipe rupture, *is* the coolant and coolant pressure are released from the primary system in about ten seconds and the coolant is contained in the containment shell. Loss of the coolant, which moderates the neutrons, stops all energy production except the energy associated with the fission products contained in the fuel pins. This remaining energy release rate is about five percent of the operating power ten seconds after the reactor power is turned off and decreased *roughly* in proportion to the one *power of time* *proportionally to (time)^{0.2}*. Emergency Core Cooling Systems (ECCS) are included in all water reactor power plants to maintain the continuous cooling of the core by the rapid addition to the reactor vessel

of water poisoned against neutrons from pressurized storage tanks and pumped pools and by the contained circulation of this coolant through a special heat removal system. Successful operation of the ECCS is achieved within the context of the LOCA by assuring that the thermal energy released from the core during the accident does not damage the plant containment system. This assurance is achieved by maintaining the integrity of the reactor core fuel pins. We now discuss briefly the core geometry, the core materials and some thermal effects that may occur in order to demonstrate the need for rapid heat removal.

By integrity do you mean no perforation, or structural shape?

The fuel pins of pressurized water reactors are made of cylindrical uranium dioxide ceramic pellets housed in a Zircaloy tube. The pins have an active length of 12 ft and Zircaloy tubes are typically 0.420-in. OD by 0.026-in. wall. The surface heat flux has an average value of about 150,000 to a maximum of about 550,000 Btu/hr-ft². The pins are arranged into a fuel assembly as a 15 x 15 square array. Assemblies are joined together to form a reactor core approximately 12 ft diameter. There is a coolant gap of about 0.125-in. between pins resulting in a matrix that is slightly more than 50% coolant by volume. The coolant flow through the core is bottom to top and parallel to the fuel pin axis. The fuel pins contain pressurized inert gas, helium or argon, added

during fabrication as a purge for oxygen, nitrogen and moisture.

The fission process also generates gaseous fission products xenon and krypton which are released to the fuel pin cavity such that the internal pressure of the fuel pins may vary during reactor operation up to the reactor coolant pressure of 2200 psig. During normal operating conditions this would result in a tangential stress that is compressive. During a LOCA, the tangential stress becomes tensile. The tangential tensile stress is typically one-seventh of the outward directed pressure difference due to geometry.

In Table 1 is a listing of temperatures, in ascending order, that relate the thermal conditions that exist in the reactor fuel with the significant temperature boundaries describing the physical occurrences in the cladding and fuel.

There are two potential situations for retrograde type cladding failure that are of concern even though they both require multiple and coincidental malfunctions that make their occurrence unlikely, since

The first of these occurs during or shortly after the coolant depressurization caused by the failure of a large circulating line; a time period from 10 to 30 seconds. Should a pipeline break and the emergency coolant injection not function, or should the core remain steam blanketed and uncooled for several seconds

Table I. Critical Temperature Limits of Fuel Rods

Temperature (°F)	Related Conditions
650	Pressurizer Saturation Pressure; approximate temperature of cladding surface.
1300	Approximate average fuel temperature in core at normal operating condition
1500 $\pm$ ³⁰⁰ / ₂₀₀	Cladding swells and ruptures due to excessive tangential stress. About 45% tangential stress blocks flow channel.
$\sim$ 1600 $\pm$ 100	Zirconium α to β phase transportation, anneal is rapid.
$\sim$ 1900	Eutectic formation between Zircalloy and Ferrous alloys. (Normal oxide coatings act as barrier to interaction.)
$\sim$ 2100	Exothermic Zirconium-steam reaction becomes significant, heat generated $\approx$ decay heat rates.
$\sim$ 2500	Zirconium Oxide diffuses into Zircaloy and embrittles base metal at rapid rate.
$\sim$ 2600	Average fuel temperature at high flux locations.
$\sim$ 2800	Cooling possible by water quench only.
3300	Zirconium melts. Fission product release from UO ₂ .
$\sim$ 5000	Matrix increases as temperature increases. Uranium Oxide and Zirconium Oxide melt.

much of
before the thermal energy stored in the fuel is removed, then many of the rods could reach surface temperature in the range of 1200 to 2200°F in several seconds.^{1,2} In this temperature range the ultimate stress of Zircaloy drops from about 11,500 to about 300 psig⁴ which corresponds to rod internal pressures of 1600 and 43 psig. Above about 1700°F, and with internal pressures below 150 psi, the cladding is ductile and expands rapidly and extensively before rupturing when heated at rates between 10 and 200°F/second.⁵ For fuel rods with an internal gas pressure below about 200 psi that are subjected to this temperature range, there is a distinct possibility of blocking off several adjacent flow channels due to excessive clad swelling.¹² Relatively small blisters are formed with longitudinal cracks at 1500°F and 600 psi internal pressures. Above 600 psi the Zircaloy expands at temperatures below 1500°F in a ductile manner.

The second mode of fuel failure might exist if the cooling is marginal and the Zircaloy cladding is maintained for several minutes above about 2400°F in contact with steam. Under these conditions, the Zircaloy is oxidized. The presence of zirconium oxide embrittles the metal which may be easily shattered, cracked, or broken when the temperature of the fuel

one temperature on
2600°F in Table 1

Is bulk of
gas at the
ends of the
pins, away
from the
increased temp

6

rods is finally lowered. There is some concern that the rubble produced could block flow channels and be difficult to cool.

The experimental work currently being performed in the PWR Full Length Emergency Cooling Heat Transfer Program (FLECHT) is being performed mostly with stainless steel clad heater rods that are not internally pressurized.^{7,8} These tests are designed as "proof tests" to demonstrate the cooling capability of the emergency systems. Tests are conducted at a constant pressure near one atmosphere, with the cladding heated to an initial temperature from 1300 to 2100°F and decay heat power levels. The tests conducted at high initial temperature conditions resulted in cladding temperatures above 2400°F and heater failure. Since the tests were performed with non-pressurized heater rods, the effects of swollen or blistered rods was not determined.

The FLECHT tests have shown, however, that more information is required on thermal energy removal during the coolant loss period of a loss-of-coolant accident. This information will help establish more realistic bounds on the temperature of the fuel cladding when the emergency coolant is injected. In addition, it is apparent now that the temperature range of interest to the Emergency Cooling System performance requires the use of Zircaloy cladding in experimental

(1)

programs because: (1) it is the material of interest, and (2) it has a high ductility and relatively low burst pressure compared to the stainless steel used in the past, and (3) the chemical-metallurgical-mechanical properties interaction is unique above about 1500°F.

The experimental and analytical effort required to define the failure mode for Zircaloy cladding subjected to the above sets of conditions is different for the first and second mode. The first failure mode requires a study of the transient thermal and mechanical stresses on the clad associated with the blowdown and with the heatup following the blowdown. The second failure mode requires information of stress-strain fracture in Zircaloy clad embrittled by hydrogen, oxygen, and possibly other gases.

This paper is a report of some work currently in progress at the Oak Ridge National Laboratory to obtain data on simulated fuel rods in a typical depressurization accident. The data of particular interest are the thermal responses of the rods, since if the cladding is kept cool the rods will undergo only small dimensional changes.

Previous experimental work on transient blowdown heat transfer concluded ^{that} steady state correlation could be utilized. The tests were performed with pressure

Table II. Comparison of Test Variable to Reactor Condition

Variable	Test Condition	Reactor Condition (PWR)
Inlet water pressure, psia, p	1515	2250
Inlet water temperature, °F	500	545
Inlet water velocity, ft/sec	2 and 5*	~15
Inlet water saturation pressure, psia, $p_{i, sat}$	680	1000
Inlet ($p = p_{i, sat}$)	835	1250
Outlet water temperature, °F	515 and 570	600
Outlet water saturation pressure, psi, p_o	780 and 1226	1543
Outlet ($p - p_{o, sit}$) psi	735 and 250	707
Rod internal pressure, psi (max)	Not done	2400
Rod length, in. heated	24	144
Rod diameter, in.	0.5	0.422
Gas plenum	~10% of rod volume	~10% of rod volume
Type of tubing	Zircaloy IV	Zircaloy IV
Wall thickness, in.	0.035	0.026
Linear heat rating, kW/ft, max.	~18 uniform distribution	~18 maximum cosine dist.
Temperature rise rate at 18 kW/ft, (Adiabatic, °F/sec)	~300	400
Average rod temperature at 18 kW/ft, °F	1900	~2500
Geometries	Both single rod and 7-rod clusters in a triangular array	Square array
Coefficient of linear exp 1/°C of filler	~11 × 10 ⁻⁶ (80% MgO)	~10 × 10 ⁻⁶ (UO ₂)
Fuel time constant	7 seconds (no gas gap) Up to 15 seconds (with gas gap)	4.5 to 5.5 seconds (no gap) Unknown (with gap)
Effective heat conductance, Btu/hr·ft ² ·°F	~285	~330

* The second number refers to condition with a cluster of 7 heater rods.

transients that are slow relative to a postulated reactor depressurization.¹¹

Heater rods clad with Zircaloy and operable at a heat flux and a stored thermal energy comparable to fuel rods were developed as was a closed cycle loop in which the depressurization tests were performed.

These two items are discussed below.

The Depressurization Loop

The depressurization loop is shown schematically in Fig. 1. The operating conditions are listed in Table II, and compared with a PWR plant. The loop consists of a high-pressure closed loop and a low-pressure or energy absorption tank. The high-pressure and low-pressure systems are isolated from each other through a pair of rupture discs confining a buffer pressure zone.

F1

TII

The high-pressure (1500 psia) systems contain a 100-gpm canned rotor pump, a pressurizer, a heat exchanger, and a bypass line as well as a location for flanging in a test section. The test section utilizes a power supply of seven separately controllable variacs of 1800 amps, 230 volts each. Flow, pressure, temperature, and other normal operating instrumentation were included. Steady-state flow data are determined in the test section and the bypass line. Transient data are obtained from pressure transducers and

9

thermocouples located throughout the loop and on the test rods. The depressurization time to 100 psia is controlled from about 7 to 60 seconds by varying the diameter of a critical flow orifice located near the rupture disks. The pressure transducer output is recorded on an FM recorder, and the thermocouple data are converted from an analogue to a digital signal and recorded in turn on a magnetic tape at a combined speed of 600 data points per second. The low-pressure system is essentially a large tank with cold water to absorb the blowdown energy; a 100 gpm pump circulated cold water to condense the steam. In addition, a water injection pump supplies water for cooling the Zircaloy rod after the blowdown by means of a spray and a flood line from the water supply at 100 psi.

A typical test included the following steps:

1. Installation of test section with Zircaloy rod.
2. Heatup of loop and establishment of steady-state, including
 - a. test rod operating at ~ 19 kW/ft of rods,
 - b. temperature across test section, inlet at 500°F , outlet at 515°F ,
 - c. loop pressure at 1500 psia.
3. Blowdown initiation by increasing the gas buffer pressure between rupture disks and pump power cut out.
4. High-speed recording of pressure and temperature

5. Power level reduced to simulate decay heat.
6. Water injection initiated by signal of Zircaloy-cladding temperature or at a preset time.
7. Continued recording on moderate speed recorders until rod is quenched.
8. Loop cooldown and removal of test section.
9. Photographic examination of rods; measurement of rods.
10. Analysis of data.

The Heater Rods

The 1/2-in.-diam Zircaloy-clad rods had a heated length of 2 feet; the internal heaters were made from Kanthal A-1 wire and were contained with magnesium oxide insulation. A heater sketch is shown in Fig. 2. A Pt to Pt-10% Rh thermocouple measured the centerline temperature. The magnesium oxide of the heater rod has a thermal diffusivity and a product of specific heat X density which are .038 and 52 as compared with .053 ft²/hr and 45.7 Btu/ft³ for UO₂. Thus the transient thermal behavior of the heater is comparable to the fuel rod. This stored thermal energy in the heated rods is comparable to that in the fuel rods, since the weighted mean temperature for the heater rod ~~is~~ ^{above the surface temperature by} about 3/4 of the maximum radial temperature difference rather than 1/2 as in the case for a fuel rod which has a distributed heat source.

F2

$$\begin{aligned} T_c - t &= a r^2 \\ T_c - \bar{t} &= \frac{2\pi}{\pi R^2} \int_0^R R \cdot a r^2 dr \\ &= \frac{a R^2}{2} = \frac{T_c - T_w}{2} \end{aligned}$$

ORNL-DWG 70-1237

FIG. 2. SPECIAL HEATER WITH HOLLOW CERAMIC CORE TO PERMIT GAS TRANSPORT WITHIN HEATER ROD.

(1w)

These heaters have been operated for short times with centerline temperatures up to 2300°F. The rods were operated at heat fluxes to 5×10^5 Btu/hr·ft² (19 kW/ft) in subcooled boiling water at atmospheric pressure and to 4.75×10^5 Btu/hr·ft² (18.5 kW/ft) in 500°F water at 1500 psi. These heat flux and temperature levels are higher than the values for the hot rod average in a water reactor, but slightly lower than the hot spot values. We believe, therefore, that these heated rods may be used to simulate the thermal behavior of fuel rods during a LOCA. (The radiation embrittlement effects appear to be annealed out as the metal approaches the phase transformation of about 1500°F.)

Tests wherein the rod-internals were repeatedly heated, while the cladding surface was kept cool, indicate that the clad expanded plastically as a result of either long-term operation or operation at increasing power levels.⁶ This was determined by measuring the clad diameter before and after operation and comparing these measurements with the calculated rod expansion. This gap increases the resistance to heat flow. The temperature of the rod internals increased during prolonged steady-power operation with a new rod as the gap developed. The temperature also showed an increase when the rod was returned to the initial power

~~Doesn't this merely mean that the pellets stay hot enough to keep the gap open.~~

level following a power increase and decrease; this is shown in Fig. 3.

F3

During a decrease-in-power transient, the heat removal rate decreased as a result of the associated lower heat-transfer coefficient in the rod. This latter effect is shown in Fig. 4, where the reduced temperature is plotted versus time for different rods at the beginning and end of use. The thermal decay times of the rods was about 7 seconds for a new and 15 seconds for an old rod. Fuel rods in PWRs have a smaller diameter and a time constant with no gas gap present of 4 to 5 seconds.

F4

would "half-decay times" a "c-le" timer" be more explicit?

A sample heater rod (see Fig. 2) designed for internal pressurization has been tested in atmospheric pressure steam with an internal argon gas pressure of 400 psi to determine whether the gas could diffuse through the heater and impose its stress on the Zircaloy cladding. The rod ruptured at 1450°F and a stress of about 3000 psi. This result is in agreement with previously published data.⁴ Diametral expansion measured after the rupture varied from 0.01 in. to 0.113 in. (Fig. 5) along the heated length (from 2 to 22%) with the maximum at the break; the diametral expansion was greater than 6% in 60% of the heated zone. This test demonstrated the ability of the gas to spread rapidly through the ceramic filler of the heater rod. Thus, we believe we can use this rod to simulate the

F5

ORNL-DWG 69-6464

FIG. 3. CENTERLINE TEMPERATURE RISE VERSUS ROD

ORNL - DWG 70-234

FIG. 4. COMPARISON OF TIME CONSTANT DATA OBTAINED DURING DEPRESSURIZATION TESTS WITH LOW PRESSURE DATA.

FIG. 5. EXPANSION TEST OF INTERNALLY PRESSURIZED HEATER ROD.

swelling and rupture characteristics of Zircaloy cladding by pressurizing the heater rods internally to simulate the fission gas internal pressure. (The radiation embrittlement is annealed out if rupture occurs above about 1500°F.⁴⁾)

Depressurization Tests and Results

Seven single-rod tests and one test with a seven rod cluster have been completed wherein time traces were measured on the coolant pressure, the heater centerline temperatures and the heater cladding temperatures.

The coolant discharge time was 12 seconds in four of these tests, including the seven rod cluster, and about 60 seconds in the remainder.

In the 60 second tests the rod power, initially set at steady levels ^{that} which varied from 13 to 17.5 kW/ft (342,000 to 450,000 Btu/hr·ft²·°F), was maintained when the loop rupture was initiated. About 4 to 8 seconds after rupture, the power was reduced to about 10% of the original level. All of these tests showed that the cladding temperature and the measured water temperature in the rod flow channels followed each other closely until the loop pressure was near atmospheric pressure; these two temperatures then diverged rapidly. We interpreted this to mean that

heat-transfer coefficients greater than ~ 10000 Btu/hr·ft²·°F existed until the flow section was out of water at which time the surface temperature increased rapidly while the remaining stored energy in the rod was being redistributed.⁶ Typical data are shown in Fig. 6, wherein the thermal response of the clad-surface and the rod-centerline temperatures and of the loop pressure are depicted as a function of time after rupture. Saturation temperatures of the coolant are also plotted. We observe that the coolant and clad temperatures are close together until the loop is near atmospheric pressure at which time the two temperatures diverge. The effect of the gap between the clad and filler is clearly seen as follows: The temperature difference between the cladding and the centerline at 25 to 50 seconds corresponds closely to the temperature difference which would be estimated from Fig. 3 for a rod operating at 3 kW (1.5 kW/ft) after operation at 28 kW. These data had been recorded on a strip chart and the resolution was limited to $\pm 25^\circ\text{F}$.

The data recording system was improved to obtain the temperature difference between the cladding and the coolant. The results recorded with this system for a 12 second depressurization test are shown in Fig. 7. The test section coolant pressure and the

F6

F7

1/2

12

temperature difference between the clad and the coolant at the heater midplane are plotted on the right hand scale. In this test the power input was held at 17.5 kW/ft during the first 8 seconds after pipe rupture and the power was then lowered to 1.87 kW/ft (450,000 and 48,000 Btu/hr·ft², respectively). Both the cladding temperature and the heater central temperature underwent only slight changes in the first 8 seconds. The plot of ΔT versus time suggests at about 0.8 seconds the development of bulk boiling from subcooled boiling concurrent with the loop pressure reaching saturation pressure. In addition, the relatively rapid decrease in clad-to-coolant ΔT starting at 5 seconds and continuing through about 14 seconds is associated in time with the rapid decrease in system pressure starting at about 450 psi some 5 seconds after rupture initiation. This sudden decrease in ΔT may be associated with the development of a thin liquid film and annular two-phase flow around the rod due to the large steam volume and high ratio of steam-to-liquid volume associated with the rapid depressurization regime. The rapid rise in ΔT from about 15°F at 12 seconds to about 300°F two seconds later occurs when the loop coolant pressure is close to atmospheric pressure and the water has been discharged. The drop

(24)

Temperature

in coolant ^{AT} at 42 seconds is associated with the spray addition of emergency coolant.

Heat-transfer coefficients of ~6000 to 3500 Btu/hr·ft²·°F have been inferred for the pressurized portion, about 10,000 Btu/hr·ft²·°F for the saturation bulk boiling portion, and about 15,000 Btu/hr·ft²·°F in the annular flow or preburnout region of the depressurization curves. The heat-transfer coefficient at about 35 seconds is estimated at 10 Btu/hr·ft²·°F and is about 20 to 25 Btu/hr·ft²·°F after the water spray was turned on. These estimates were made from a two-region transient analyses assuming a radial temperature distribution in the ceramic element the same as in the steady state. These results are typical of this particular system only and are not typical of any particular reactor. The results of this and other experiments are to be examined by and compared with the calculated results of computer programs such as RELAP3⁹ and THETA-1B¹⁰ used in computing reactor transients and thermal results. These tests were performed to assure the adequate development of both the heater rods and the experimental facility.

Data have been obtained on one cluster of 7 rods that show the thermal response and the pressure history during a depressurization. The heaters were identical to the single rod test heaters. The 7 rod array is shown in cross section in Fig. 8.

ORNL-DWG 70-5280

FIG. 8. PLAN VIEW OF TEST SEVEN-ROD CLUSTER.

The system was operated with a power input to the test rods of 166.2 kW. The power distribution to the rods was as given in Table III. The average linear power rating for all rods was 11.9 kW/ft, which is comparable to the rating of the nominal high-power rods in a PWR. During the depressurization of the loop there was full power to the rods until interrupted by the loop safety circuits (~15 sec of full power after start of depressurization). Since the power was not reduced to simulate shutdown, the experiment was conservative in terms of heat flux from the rods during the simulated LOCA.

T-III

Flow-resistance orifices were positioned in the loop circulating system (see Fig. 1) to minimize coolant flow through the test section during the depressurization. The check valve and all flow resistances in the line between the mixing tank, heat exchanger, pressurizer, and blowdown line were removed, and all load orifices were placed in the line between the mixing tank, pump, test section bypass, and the test section. Power to the pump was turned off when the depressurization was initiated to further minimize flow through the test section.

The temperature on the center heater rod, rod 7, and coolant temperature and pressure are shown on Fig. 9. Curves are shown for the temperature of the center

Table 1
11 Rods

Rod	1	2	3	4	5	6	7	Total
-----	---	---	---	---	---	---	---	-------

to 2 sec

ORNL-DWG 71-1172

Fig. 9
 Fig. 10. Experimental Data from Seven-Rod Depressurization Test.

Table II:
in Rods

Rod	1	2	3	4	5	6	7	Total
1								1

Fig. 9. Experimental Data from Seven-Rod Depressurization Test.

of the rod and the cladding surface and the temperature difference between the cladding and the coolant. Heat transfer coefficients calculated from these data were $3600 \text{ Btu/hr}\cdot\text{ft}^2\cdot^\circ\text{F}$ before the simulated LOCA, approximately $4500 \text{ Btu/hr}\cdot\text{ft}^2\cdot^\circ\text{F}$ for up to 2 sec after the LOCA was initiated, and between 11,000 and 45,000 $\text{Btu/hr}\cdot\text{ft}^2\cdot^\circ\text{F}$ during the period 2 sec to dryout. Dryout occurred ~9.5 seconds after pipe rupture.

This information is important because it indicates that (1) heat transfer coefficients during the depressurization may be several times greater than estimates using steady state data, and (2) good cooling continued for a period of time somewhat greater than the time constant (the time to remove $1/e$ times the initial stored energy of the rods) of the heater rods. If (1) and (2) are true in the reactor accident case, the temperature of the fuel rod cladding at the time of initiation of emergency cooling could be as low as 1400 to 1500°F at the hot spot.

Fig. 10 shows cladding temperatures of the other heaters in the 7-rod array versus time during and after the depressurization. (Note that on the time scale of Fig. 10 and 11 the LOCA was initiated at 4.2 sec rather than the 0.23 sec indicated in Fig. 9.) The temperatures of rods 1 and 4 (with zero heat flux)

End Tap

F-10

F-11

Fig. 1.11. Cladding Surface Temperature Traces as a Function of Time After Blowdown.

Fig. 1.12. Additional System and Heater Rod Temperature Data Obtained During Blowdown Test.

follow the coolant temperature closely. All the cladding surface temperatures (on the powered rods) decreased about 100°F during the first 3 sec of blowdown as bulk boiling started and then remained within about 30°F of the coolant temperature until dryout occurred. During this period the heat flux remained constant. Apparently dryout was followed by a period of flow reversal, or sequential wetting as evidenced by the cladding temperature spikes between 13 and 16 sec on the time scale. (Thermocouples not reported did not record.)

Cladding temperature traces at two additional stations, the temperature of the coolant upstream of the discharge orifice, and rod axis temperatures are shown on Fig. 11. Power cutoff to rod 3 was at 18.2 sec and to rod 5 at 19.2 sec; that is, approximately 1 sec before each reached maximum temperature. The very rapid rise in the temperature of the blowdown coolant upstream of the rupture disks marks the initiation of depressurization and is to be noted as a reference point for pipe rupture.

On the basis of these preliminary tests it appears that energy removal from fuel rods during the period of depressurization associated with the LOCA may be considerably greater than currently assumed. However, the data are not to be applied directly to

any reactors. The conditions of these tests are not the same as the conditions of a reactor. This may be seen by examination of Table II. In addition, the boundary conditions of a large fuel bundle array are different than that of a single rod or a 7-rod array; the flow in these tests is more unidirectional and does not develop the radial flow expected in a large reactor core. Most important, the reversal of flow direction in a 12-ft long assembly that would occur in a reactor plant if an inlet pipe were to rupture was not examined. This flow reversal could result in a *(film boiling?)* critical heat flux condition as soon as depressurization started ^s because of the high ~~coolant~~ *of the coolant that would be* enthalpy ~~now~~ entering the heated region.

The most important single aspect of these tests results is that data may be obtained using electrically heated rods under conditions that approach the environment of a Loss-of-Coolant Accident. Thus it is possible to test our ability to describe "what occurs" under these circumstances up to the destruction of the heaters.

The lack of instruments for measuring rapid two-phase transient flow in pipes is a great handicap in interpreting results mechanistically. The shape and similarity of the transient temperature difference curves in the single and 7-rod tests suggests strongly

(3)

the development of an annular flow ^(regime) situation with decreasing pressure and increasing steam quality. This situation would result in high heat transfer coefficients that would increase until dryout occurred.

Conclusions

An experiment facility has been built ^{that can} which may be used to perform tests to measure the thermal response of heaters in pressure blowdown transient ^{that} similar in time scale to a Pressurized Water Reactor Loss-of-Coolant Accident. Electrical heaters have been developed ^{that} which approximate the thermal transport characteristics of a nuclear fuel pin and these rods have been operated at heat flux levels comparable to a nuclear full pin.

^{from this facility} ~~These~~ test results indicated ^{that the} heat transfer coefficients during depressurization ^{are} ~~that~~ were greater than 3000 Btu/hr-ft²·°F for time periods larger than the heater heat transport time constants.

^{behavior} The ~~shape~~ of the surface to coolant temperature difference suggested a rapid development of annular two-phase flow ^{that} which lasted until surface dryout occurred.

Additional testing is required under geometrical and thermodynamic conditions more nearly like a reactor ^{from} before reactor-relevant conclusions ^{can be} ~~are~~ drawn.

These variables include heater geometry, coolant inlet and outlet flow and enthalpy, the coolant pressure transient and the heater internal pressure.

Ref. Code

List of Figures

1. Depressurization Loop
2. Special Heater with Hollow Ceramic Core to Permit Gas Transport within Heater Rod
3. Centerline Temperature Rise Versus Rod Power or Heat Flux
4. Comparison of Time Constant Data Obtained During Depressurization Tests with Low Pressure Data
5. Expansion Test of Internally Pressurized Heater Rod
6. Thermal Response of Heater Rod to LOCA
7. Thermal Response of Heater Rod to LOCA
8. Plan View of Test Seven-Rod Cluster
9. Experimental Data from Seven-Rod Depressurization Experiment Including Central Rod Thermal Data
10. Cladding Surface Temperature Traces
11. Additional Cladding and Heater Axis Temperature with Steam Discharge Temperature

Ref. Code

References

1. Tennessee Valley Authority, "Browns Ferry Nuclear Power Station, Preliminary Safety Analysis and Design Report," Docket Nos. 50-259 and 50-260 (1966).
2. Duke Power Company, "Oconee Nuclear Power Station, Units 1 and 2 Preliminary Safety Analysis Report," Docket Nos. 50-269 and 50-270 (1966).
3. C. G. Lawson, "Emergency Cooling Systems of Light Water Cooled Reactors," ORNL-NSIC-24, Oak Ridge National Laboratory (October 1968).
4. J. F. White, "AEC Fuels and Materials Development Program Progress Report No. 76," GEMP-1000, pp. 229-280 (September 1968).
5. D. O. Hobson, M. F. Osborne, and G. W. Parker, "Comparison of Rupture Data from Irradiated Fuel Rod and Cladding," *Nucl. Technol.* 11(4), 479-491, August 1971.
6. C. G. Lawson, Cladding Behavior in Loss-of-Coolant Accident, pp. 23-26, "ORNL Nuclear Safety Research and Development Program Bimonthly Report for May-June 1969," ORNL-TM-2663 (September 1969).
7. J. O. Cerman, A. S. Kitzex, F. F. Cadec, R. H. Leyse, D. P. Dominicio, "PWR Full Length Emergency Cooling Heat Transfer," USAEC Report WCAP-7435, January 1970.

8. F. F. Cadck, D. P. Dominicio, and R. H. Leyse,
"PWR Full Length Emergency Cooling Heat Transfer,
Group II Test Report, USAEC Report WCAP-7544,
September 1970.
9. W. H. Rettig, G. A. Jayne, K. V. Moore, C. E.
Slater, M. L. Uptmor, "RELAP3---A Computer Program
for Reactor Blowdown Analysis," IN-1321, June
1970.
10. C. J. Hocevar and T. W. Wineinger, "THETA1-B, A
Computer Code for Nuclear Reactor Core Thermal
Analysis," IN-1445, February 1971.
11. J. O. Cermak, L. S. Tong, J. E. Casterline et
al., "DNB in Rod Bundles During Pressure Blow-
down," ASME Paper No. 70HT12.
12. B. E. Bingham and A. L. Lowe, Jr., "Application
of Experimental Data to Analytical Evaluation of
Cladding Failure Distribution," *Nucl. Tech.*,
11(4), 521-525, August 1971.

End Tape 2

Table III. Power Distribution
in Rods of Seven-Rod Cluster

Rod	Power (kW)	Linear Heat Rating (kW/ft)
1	0	0
2	32.3	16.15
3	29.4	14.7
4	0	0
5	32.3	16.15
6	37.2	18.6
7	34.9	17.45
Total	166.1	