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SAFETY ANALYSIS OF THE C-1 LOOP

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SAFETY ANALYSIS OF THE C-1 LOOP

By

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December 31, 1964

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SAFETY ANALYSIS OF THE C-1 LOOPI. INTRODUCTION

The C-1 Loop is a small in-reactor water loop located in the 105-C Building. The loop was originally designed and constructed for nonfueled testing of the effects of reactor radiation on sample corrosion, crud deposition, and coolant radiolysis.⁽¹⁾ The loop was modified in 1964⁽²⁾ to permit the irradiation of aluminum-clad fuel elements and subsequently used for about eight months for the in-reactor testing of aluminum-clad, plutonium-aluminum fuel assemblies at high coolant temperatures and pressures. The facility is currently inactive but is available for the performance of special testing.

Prior to nuclear operation of the C-1 Loop the mechanical equipment was carefully checked and the backup coolant system was experimentally verified to perform as designed. A safety analysis was also made with results summarized briefly in Reference (3) and in more detail in the production test which authorized nuclear operation of the C-1 Facility.⁽⁴⁾ This report updates those earlier documents with improved descriptions of the C-1 Loop and its operation including analyses of postulated accidents. The consequences of the maximum credible accident are described with emphasis on subsequent releases of radioactive material to the environs. The analyses are based on the fueling of the loop with the plutonium-aluminum fuel elements used in the most recent series of experiments. Future tests in the facility may use these or oxide type fuel elements. The conclusions reached in this report on operation with plutonium-aluminum fuels apply conservatively to operation with oxide fuels.

-
1. R. K. Marshall. Production Test IP-414-D, Installation and Operation of A Loop Facility at C Reactor, HW-69838. June 5, 1961. (Unclassified).
 2. T. L. Deobald. Irradiation Processing Department Design Change No. 767 - C-1 Loop Test Facility. August 22, 1963. (Unclassified).
 3. Staff, Irradiation Processing Department. Hazards Summary Report - Safety Analysis and Hazards Evaluation, Six Oldest Hanford Production Reactors, HW-74094, Volumes 1, 2, and 3. June 2, 1963. (Secret).
 4. T. L. Deobald. Production Test IP-623-D, Operation of the C-1 Loop Facility With Aluminum Clad Plutonium-Aluminum Alloy Fuel Elements, HW-79097. October 10, 1963. (Secret).

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II. SUMMARY AND CONCLUSIONS

The C-1 Loop is an in-reactor water loop designed to irradiate small fuel assemblies for testing fuel cladding materials in high temperature coolant environments. The loop piping and equipment are located on the X-1 level at the C Reactor, and the instrumentation is located on the X-0 level. The design pressure of the loop is 2530 psi at a temperature of 650 F. The maximum power capability is estimated to be about 100 kw.

The in-reactor portion of the C-1 Loop is installed in the C-test hole between process tube rows twenty and twenty-one. The in-reactor section consists of two concentric stainless steel tubes extending about twelve feet into the C Reactor. The outer tube serves as the pressure shell; the fuel elements are housed in the inner tube. Coolant enters the test section in the annulus formed between the fuel housing and the fuel assembly, makes a 180° turn at the end of the test section, and exits in the annulus formed between the pressure shell and the fuel housing. The fuel elements used in the most recent tests in the C-1 Facility consisted of 8 wt/o plutonium-aluminum cores clad in aluminum. Operation with these elements was assumed in this report.

A major objective in the design of the C-1 Facility was to achieve a high degree of safety. Sound engineering practice has been used throughout the design. The primary hazard associated with operation of the facility is that of fission product release from the loop to the surrounding environs. Since the best fission product container is the clad, cooled, plutonium-aluminum fuel element itself, particular attention has been given to preventing accidents which might result in the release of radioactivity by the provision of highly reliable coolant systems. Mechanical features built into the design to prevent individual equipment failures from developing into accidents include high quality and high reliability loop components, dump circuits that automatically transfer the system to once-through emergency cooling in the event of incipient difficulty, scram circuits that shutdown the fission process, cooldown circuits that automatically reduce the loop coolant temperature when certain abnormal conditions occur, and annunciator circuits that warn of impending difficulty.

The facility was designed for automatic operation without continuous operator coverage. The loop is instrumented, therefore, to automatically transfer from circulating cooling by the primary coolant system to single-pass cooling by the backup coolant system under conditions which may indicate corrective action but do not require a reactor scram. A reactor scram occurs only if the backup coolant system is insufficient to maintain adequate cooling flow throughout the transition period and the following period of once-through operation.

A number of conceivable accidents have been studied to determine the sequence of events, the probability of occurrence, and outcome. The situations studied included loss or reduction of cooling, reactor power surge, loss of heat removal capability, loss of primary pressure control, inadequate shutdown cooling, charge-discharge accidents, fuel element failure, and breaks in the high pressure system. It was concluded from these studies that, with the exception of major breaks in selected runs of high integrity piping, at

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least two, simultaneous, independent failures must occur for the C-1 fuel elements to be inadequately cooled. One of these failures must be in the primary coolant system causing loss of that system and the second must be failure of one of the inlet check valves of the backup coolant system to open. Assuming the proper operation of these check valves, three, simultaneous, independent failures must occur before the fuel elements would be inadequately cooled, and the accident must include failure of both dump valves of the backup coolant system.

Breaks in the high pressure piping could lead to serious outcomes depending on the location and size of the failure. A break in the in-reactor pressure shell would release high pressure water into the reactor which could conceivably result in forces large enough to lift the graphite stack in a local region and distort upper aluminum process tubes. The fuel elements in both the C-1 Loop and distorted process tubes would remain adequately cooled throughout this accident. Potentially the most serious accident would be complete failure of the inlet piping common to both the primary and backup coolant systems. This accident, which is considered the maximum credible accident, would result in complete loss of water to the test section, melting of the fuel elements, and release of a portion of the fission products to the environs. The potential fission product release and offsite consequences of the release have been estimated. Because of the relatively small quantities of radionuclides in the plutonium-aluminum fuel assembly (about 6 per cent of the fission product inventory of a normal C-Reactor fuel column) the estimated release would not result in serious radiation exposures to offsite personnel. The maximum whole body exposure from inhalation as the fission product cloud passed by would occur three miles from the point of release and is estimated at less than 0.1 rem; the maximum thyroid exposure is estimated at less than 0.4 rem. The deposition of I-131 may exceed one microcurie per square meter, which is the control level, depending upon wind speed. The deposition is estimated to extend 6.2 miles from the C-Reactor ventilation stack and covers an area of only one-third square mile. No control measures are likely to be imposed in this small area since the I-131 will decay to less than the control level in four days. The potential deposition of strontium-90 is estimated to be less than 10^{-7} microcuries per square meter at the maximum point.

The General Electric Company respectfully submits that the design and operating information contained in this report, including the safety and hazards analyses, demonstrates with reasonable assurance that the C-1 Loop facility can be operated without undue risk to the health and safety of the public.

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III. DISCUSSION

A. Description of the C-1 Loop and Associated Equipment

1. General

The C-1 Loop is an in-reactor water loop designed for the irradiation of small fuel assemblies in the C Reactor to evaluate fuel cladding materials and water quality at high coolant temperature and pressure. The loop piping and equipment is located on the X-1 level of the reactor and instrumentation is located on the X-0 level. The loop is designed for a coolant flow rate of 15 gpm at maximum operating conditions of 650 F and 2530 psi and has a heat removal capacity of about 240 kw. The test section is 0.87-inches inside diameter and 14-feet long, about 12 feet of which is in the reactor. The loop is designed for unattended operation. Safety circuits include annunciator circuits that warn of incipient difficulty, scram circuits that shutdown the fission process, dump circuits that automatically initiate a transfer to process water cooling, and cool-down circuits that automatically reduce the coolant temperature.

A line diagram of the C-1 Loop is shown in Figure 1. The principle features of the loop equipment are summarized below.

2. In-Reactor Assembly

The in-reactor portion of the C-1 Loop is installed in the C-test hole in C Reactor. This test hole runs from the far side of the reactor to the thermal shield on the near side between process tube rows 20 and 21 and passes over normal process channels about 15 inches downstream of their midpoints. The test hole is 4 3/8-inches square. The in-reactor test assembly consists of an outer pressure shell which is made of 1 1/2-inch, Schedule-160 (1.90" OD x 1.338" ID), Type-316, stainless steel pipe, and extends about 12 feet into the C-test hole. The end is closed by a butt welding cap. The fuel housing is centered in the pressure shell by tubes, 5/32 inches outside diameter, which also serve as thermocouple and flux monitor wells. The fuel housing is made of 304-L stainless steel tubing which is one-inch outside diameter with a 0.065-inch wall thickness. The test section is the re-entry type. Cooling water flows down the inner annulus formed between the fuel housing and fuel assembly, makes a 180° turn at the end of the fuel assembly, and returns through the outer annulus formed between the pressure shell and the fuel housing. The geometry of the test section is shown in Figure 2.

The fuel elements most recently irradiated in the facility and upon which this report is based is shown in Figure 3. The fuel core is 8 ± 0.5 weight per cent plutonium in aluminum. The fuel cladding is generally an aluminum alloy being tested in the high temperature environment. Typical cladding materials are X-8001, A-288, or KYZ (iron-nickel) aluminum alloys. The fuel core and cladding are coextruded to form a bonded fuel element. End caps, which are welded to the cladding, are not bonded to the fuel cores. The fuel elements are examined for unbonded areas by ultrasonic techniques

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DESCRIPTION

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FIGURE I RL-REA-316 C-1 LOOP FLOW DIAGRAM

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NOTE:
1. PUMP COOLING WATER LO FLOW & HI TEMP ARE WIRED TO SHUT DOWN PUMPS
2. DESIGN CONDITION: 2530 PSIG @ 650°F
3. HYDRAULIC PRESSURE TEST AT 4175 PSIG
4. SEE DWG NO. H-1-12421 SH 1
AS PER IPID INSTRUCTIONS
FOR INST LEGEND

LEGEND
--- SIGNAL LINES
--- CONTROL LINES

PRESSURE INSTRUMENTS
PRESS RECORDER
DIFF PRESS RECORDER
PRESS CONTROLLER
PR-1 0-4000 PSI
PR-2 0-4000 PSI
PR-3 0-4000 PSI
PT-1
PT-2
PT-3
DP-1
DP-2
DP-3
DP-4
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C-1 LOOP FUEL ELEMENTS

FIGURE 3

All Dimensions in Inches

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and are autoclaved 48 hours at 660 F. A minimum cladding thickness of 50 mils is assured by radiographic tests. Up to eight of these four-inch fuel elements are interlocked to form a fuel assembly. The upstream fuel element in the assembly is equipped with a thermocouple to measure core temperatures to give an indication of the rate of buildup of surface crud during irradiation.

3. Primary Coolant System

The primary coolant system circulates liquid water through the C-1 Loop at 15 gpm at operating temperatures. The coolant enters the reactor section of the loop into the annulus formed between the fuel housing and fuel assembly, makes a 180° turn at the outlet of the fuel assembly, and exits through the outer annulus formed between the pressure shell and fuel housing. The pumping system consists of pumps, piping, valves, fittings, and heat exchangers. A line diagram of the primary coolant system is shown in Figure 1.

The loop contains two canned-motor pumps connected in parallel, both of which are capable of delivering 15 gpm against 400 feet of head at 68 F. One pump is driven by a 5-hp motor and the other by a 10-hp motor. Either pump will provide the required coolant flow, but the loop is operated with both pumps except in periods when either pump may be down for repairs. Electrical power for the pumps is obtained from the #3 Bus at C Reactor.

Heat is removed from the system by three water-to-water heat exchangers, each rated at about 80 kw and operating in parallel. As shown in Figure 1, a portion of the loop coolant flow is by-passed through the heat removal system to maintain a set coolant temperature at the fuel assembly outlet. Flow through the heat removal system is from the outlet to the inlet of the pumps. A two-way, 1/2-inch valve controls the amount of coolant bypassed through the system.

Loop pressure is maintained by the addition of makeup water by either of two pumps. One pump is a 3-gph, positive-displacement, electric pump and the other is a 15-gph, air-operated pump. The loop pressure is controlled by bleeding excess water from the loop through either of two bleed valves, each with its own controller. A three gallon accumulator, normally operating about one-half full of water and one-half full of nitrogen, is provided to even out pressure fluctuations.

Makeup water is produced by passing water from the C-Reactor filtered water system through an anion exchange column, a cation exchange column, a deoxygenator, a mixed-bed ion exchange column, and thence to a 155 gallon covered storage tank. Additions of chemicals for pH adjustment or corrosion control are made to the water in this tank. This batch makeup system, located on the X-0 level, provides water for three days of loop operation.

Cooling water for the heat exchangers and circulating pumps is provided at 90 psig from the X-1 level test hole water supply (normal reactor process water). Manually operated valves allow valving to the filtered water supply of the reactor.

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Rupture safety heads are installed near the in-reactor test section outlet and near the pump outlet. A 1/8-inch orifice is located in the inlet to each of these systems to prevent excessive blow-down rates. Rupture discs, rated to burst at 2530 psig at ambient temperatures, are installed in the safety heads. The safety head located near the pump outlet is used for overpressure protection during non-heat generating and out-of-reactor tests when the 15-kw heaters are used. The safety head located at the in-reactor test section outlet, is used for overpressure protection for irradiation tests.

The piping in the primary coolant system is stainless steel. All welded joints were radiographed.

4. Instrumentation

The C-1 Loop instrumentation is located on the X-0 level. Information for the calculation of the fuel assembly power is obtained by measuring the total flow of water and the temperature rise of that water in its passage through the fuel assembly. The coolant flow rate is obtained by measuring the pressure drop across metering Gentile tubes located in the piping at the test section inlet. The temperature rise across the fuel assembly is obtained from thermocouples which measure the coolant temperatures at the inlet and outlet of the fuel assembly. Loop pressure is measured near the pump inlet.

The outlet coolant temperature, coolant pressure, and coolant flow rate are automatically maintained at set values by control instrumentation. The outlet coolant temperature is maintained by a controller which positions the two-way valve which limits coolant flow through the heat exchangers. The input to the controller comes from a thermocouple located at the test section outlet. Either of two pressure controllers will activate the bleed valves to control loop pressure. The controllers also shut off the positive-displacement makeup pump when the operating pressure exceeds the control point and start the air-operated, rapid-fill pump when loop pressure is lower than the control point. The coolant flow rate is automatically maintained at the set point by controllers which establish the position of the flow control valve. The input to the controller comes from one of the flow metering devices located at the test section inlet.

Other instrumentation, besides the safety circuits described below, include: 1) water quality control instruments, 2) a miscellaneous temperature recorder which will also turn off the electrical heaters in the event that heater wall temperature approaches an unsafe value, 3) instruments to turn off the recirculating pumps on loss of cooling water, and 4) instrumentation for monitoring the routine operation of the equipment.

5. Emergency Backup Coolant System

The backup coolant system consists of piping, dump valves, check valves, and a three-gallon accumulator. A line diagram of the emergency backup coolant system is incorporated in Figure 1. Short runs of inlet and out-

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let piping are common to both the primary and backup coolant systems. The single-pass emergency coolant flow is provided by the C-Reactor coolant supply system and originates at the B and C risers of the reactor. A design change has been approved⁽⁵⁾ to also tie into the A and D risers to provide a backup coolant supply from both sides of the reactor cooling system. Coolant is supplied at 420 psig, passes through the test section, and exits through the dump valves installed in parallel in the outlet piping (See Figure 1).

From the dump valves, the coolant flows into a dump tank and thence into a trench drain located in the C Reactor exhaust tunnel. The backup coolant system is isolated from the primary coolant system by two check valves installed in series in the inlet piping.

The backup accumulator serves to maintain pressure and coolant flow after the dump valves open and until all the hot water has been displaced from the test section. The accumulator contains about three gallons of cold water which is forced through the loop by high pressure gas at about twice the normal circulation flow rate when the dump valves are opened. After the accumulator flow stops, the saturation pressure of the water in the test assembly is less than the emergency coolant supply pressure; emergency cooling water enters through the two check valves, flows through the test assembly, and out the dump valves to the drain.

6. Safety Circuits

Three types of safety circuits are provided. One reduces the temperature of the circulating coolant, a second initiates a transition from circulating to emergency, single-pass cooling, and the third initiates a reactor scram. In addition, an annunciator system warns of unsafe or abnormal conditions. In each case, the abnormal condition causes an audible alarm which can be silenced, but a visual alarm will remain lighted. The signals which result in annunciated alarms are given in Table 1.

In addition, the C-1 Loop annunciators will actuate one of four different annunciators in the reactor control room. The control room annunciators are:

- No. 35 - C-1 Loop critical off-normal
- No. 36 - C-1 Loop dump valve bypass
- No. 8 - C-1 Loop scram
- No. 95 - C-1 Loop non-critical off-normal

The C-1 Loop annunciators which will actuate these annunciators are shown in Table 1. Placing the C-1 loop scram bypass switch (located in the C-Reactor control room) in the bypass position will actuate C-Reactor Annunciator No. 12 - any scram critical bypassed.

5. K. B. Sowa. Design Change 906, Alternate Process Water Supply System for C-1 Loop. July 31, 1964. (Unclassified).

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TABLE 1
C-1 LOOP ANNUNCIATOR CIRCUITS

<u>Ann. No.</u>	<u>Title</u>	<u>Trip Point</u>	<u>C-Reactor Annunciator</u>
1	No. 1 flow scram	11 gpm	35
2	No. 1 flow trip	13 gpm	35
3	No. 1 low pressure trip	2150 psi	35
4	No. 1 high temperature trip	590 F max.	35
5	No. 1 recirculating pump	off	35
6	No. 2 recirculation pump	off	35
7	No. 2 flow scram	11 gpm	35
8	No. 2 flow trip	13 gpm	35
9	No. 2 low pressure trip	2150 psi	35
10	No. 2 high temperature trip	590 F max.	35
11	No. 1 pump cooling water	3 gpm	95
12	No. 2 pump cooling water	3 gpm	95
13	No. 3 flow scram	11 gpm	95
14	No. 3 flow trip	13 gpm	35
15	No. 3 low pressure trip	2150 psi	35
16	No. 3 high temperature trip	590 F max.	35
17	High temperature trip warning	+ 10 F	35
18	Low flow heat exchanger cooling water	18 gpm	95
19	Loop high temperature	+ 10 F	95
20	Dump flow trip by-pass	On	35
21	Dump pressure trip by-pass	On	35
22	Dump valve by-pass	On	36
23	Loop low flow by-pass	On	95
24	Flow scram by-pass	On	35
25	Low pressure backup accumulator	2100 psig	95
26	High radiation	95%	35
27	Low flow trip warning	14 gpm	35
28	Low pressure trip warning	2200 psig	35
29	Water in gas system	---	95
30	Automatic cooldown switch	On	95
31	Backup accumulator valve	Valve partially closed	95
32	High pressure backup accumulator	2200 psig	35
33	High pressure warning	2300 psig	95
34	Block valve not automatic	off	35
35	Makeup tank low	+7 inches	95

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An automatic transition to single-pass emergency cooling is actuated in the event that adequate cooling of the fuel assembly cannot be maintained by the primary coolant system. Three independent sensing systems monitor the circulation coolant flow rate at the test section inlet and a low trip from two of these systems will activate a dump. A failure of one system, either safe or unsafe, will not dump the system. Furthermore, although the sensing systems are considered to be fail-safe, an unsafe failure of one system will not prevent the other two systems from actuating a dump if a signal is received. Similarly, a two-out-of-three sensing and trip system monitors the coolant temperature at the fuel assembly outlet and activates a dump on a high temperature condition, and a two-out-of-three coolant pressure signal originating at the pump inlet activates a dump on a low pressure signal. The signals that initiate a system dump with settings which apply to the operating conditions assumed in this report are given in Table 2. A line diagram of the dump and scram circuits is shown in Figure 4.

A reactor scram is initiated by either a low coolant flow rate or a low coolant pressure at the test section outlet⁽⁶⁾. The scram flow rate is less than the dump flow rate so that a reactor scram is initiated only when the dump system fails to perform its designed function. The test section outlet pressure scram trip is set to scram the reactor when the outlet pressure falls below the pressure that exists with the scram flow rate. The signals which result in a reactor scram are shown in Table 3.

An automatic reduction in loop coolant temperature is activated by gross gamma activity in the loop, fission product activity in the loop, and high or low backup accumulator pressure. Fission product activity is monitored by a scintillation spectrometer located at the discharge to the high-pressure bleed valves. The instrument is adjusted to be sensitive to a particular gamma energy characteristic of fission products (1.43 MEV gamma from 32-minute cesium¹³⁸). A cooldown is accomplished by automatically adjusting the set point of the temperature controller downward. The signals which initiate an automatic cooldown are given in Table 4.

B. C-1 Loop Operation

1. General

The C-1 Loop was designed for automatic operation without continuous operator coverage. The loop is instrumented, therefore, to dump or cool the system to prevent minor failures of equipment from developing into accidents. The loop is dumped or cooled at conditions which indicate corrective action but which do not require a reactor scram. The conditions at which a dump or cooldown is initiated may be compared to conditions which might announce incipient difficulties in an operation with full-time operator coverage, resulting in a reactor scram if the difficulty becomes worse or cannot be corrected. In the C-1 Loop, the reactor scram

6. T. L. Deobald. Design Change 949, C-1 Loop Low Pressure Scram Instrumentation. to be issued. (Unclassified).

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FIGURE 4
C-1 LOOP SCRAM AND DUMP INSTRUMENTATION SCHEMATIC

Legend

T/C I = Thermocouple Indicator
T/C = Thermocouple
SV = Solenoid Valve
A = Air
E = Electrical
PS = Pressure Switch
d/p = Differential Pressure
T = Transmitter
PT = Pressure Transmitter
W = Water
F.E. = Flow Element

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TABLE 2
C-1 LOOP DUMP CIRCUITS

<u>Process Variable</u>	<u>Setting</u>	<u>Actuating Signal or Device</u>
Flow rate	-13%	Two of three pressure switches actuated by independent d/p transmitters monitoring independent flow meters at the test section inlet.
Temperature	590 F	Two of three temperature indicators activated by thermocouples at the test section outlet.
Pressure	-100 psi	Two of three pressure switches activated by a pressure signal originating at the pump inlet.

TABLE 3
C-1 LOOP SCRAM CIRCUITS^(a)

<u>Process Variable</u>	<u>Setting</u>	<u>Actuating Signal or Device</u>
Flow rate	-27%	Two of three pressure switches activated by independent d/p transmitters monitoring independent flow meters at the test section inlet.
Outlet pressure	50 psi	Two of three pressure switches monitoring the pressure at the outlet of the test section.

(a) These settings are in force during both circulation and single-pass operation of the loop.

TABLE 4
C-1 LOOP COOLDOWN CIRCUITS

<u>Process Variable</u>	<u>Setting</u>	<u>Actuating Signal or Device</u>
Fission product activity	95%	Recorder activated by scintillation spectrometer.
Gross gamma activity	95%	Recorder activated by ion chamber.
High backup accumulator pressure	+100 psi	Pressure switch on gas system.
Low backup accumulator pressure	-75 psi	Pressure switch on gas system.

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occurs only if the backup coolant system is not sufficient to maintain adequate coolant flow throughout the dump transient and during ensuing operation under single-pass cooling conditions.

The C-1 Loop has been used primarily to measure the corrosion rates of aluminum-clad fuel elements in a high temperature environment.⁽⁷⁾ Tests include studies of different aluminum alloys, different heat generation rates as determined by the position of the fuel elements in the test section, different temperatures, and different water qualities. A further objective of the tests performed to date has been to measure the rise in the temperature of the cladding due to the buildup of aluminum oxide and deposited crud. This is accomplished with the thermocouple fuel element shown in Figure 3.

The C-1 Facility is currently inactive but is available for tests in high coolant temperature environments as required. The plutonium-aluminum fuel element described above has been used to date because of its irradiation stability, resistance to corrosion by high temperature coolant, and availability. Future tests could employ this fuel, or, perhaps, oxide fuels. The nuclear safety conclusions reached in this report considering Pu-Al fuel elements would apply with more conservatism to oxide fuel elements because of the higher melting point of the latter.

2. Normal Operation

Heat is removed from the fuel assembly by the circulation of water at 15 gpm. Normal operating conditions with eight four-inch Pu-Al fuel elements are: fuel assembly power of about 80 kw of which 72 kw is generated in the fuel and the remainder in the stainless steel fuel housing and pressure shell; coolant temperature at the fuel assembly outlet of 572 F; and an outlet pressure of 2290 psia. The loop is designed for and operated on an automatic bases with technician coverage on one shift of the normal work week and during reactor startup. During normal operation the loop coolant flow rate, temperature, and pressure are automatically controlled as discussed below.

The coolant flow rate is regulated by a diaphragm-operated valve located in the test section inlet piping. The position of this valve is regulated by a flow controller which receives its signals from one of the flow metering devices located at the inlet to the test section.

The coolant outlet temperature is controlled by regulating the amount of coolant which is bypassed through the heat exchangers. The amount of coolant bypassed is determined by the position of a two-way valve. The valve position is controlled by a temperature recorder-controller which receives its signal from a thermocouple located at the test section outlet.

7. D. R. Dickinson. Modification of C-1 Loop for Corrosion Testing With Fuel Elements, HW-77157. April 2, 1963. (Unclassified).

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The coolant pressure is maintained by two makeup pumps which inject high-pressure water into the loop. The loop pressure is controlled by either of two pressure controllers which regulate the rate at which water is bled from the system through two bleed valves. Normally these systems are adjusted so that one controller and valve is controlling at the desired set point, and the other controller-valve combination is controlling at a slightly higher set point. Pressure switches are adjusted so that the positive-displacement makeup pump runs when the loop pressure is at or below the desired set point, and the rapid-fill makeup pump runs when the loop pressure is below the desired set point.

3. Steady-State Operating Conditions and Limits

The maximum fuel assembly power is limited by coolant temperature, heat flux and fuel cladding temperature. Boiling of the bulk coolant in the test channel is prevented by limits on the temperature of the effluent water. The measured bulk coolant temperature at the fuel assembly outlet is restricted such that the maximum bulk coolant temperature is at least 60 F below the coolant saturation temperature. Nucleate boiling does not occur on the fuel surface at these conditions. Exceeding these conditions would result in an automatic transition to single-pass cooling by the backup cooling system.

The limit on heat flux is intended to prevent local film burnout at the fuel surface. The heat flux is limited by placing a minimum of 1.4 on the burnout safety factor (BOSF) as described in Appendix A. The BOSF is defined as:

$$\frac{(\text{hot spot factor})(\text{calculated burnout heat flux})}{(\text{operating heat flux})}$$

The hot spot factor takes into account the errors in flow, temperature, and pressure measurements, and variations in neutron flux and fuel composition. This factor is estimated at 0.74 for the C-1 Loop. The burnout heat flux is estimated from the correlation of Tong, Currin, and Thorp^(8,9), as discussed in Appendix A. The factor of 1.4 takes into account the uncertainties in applying this equation to the C-1 Loop.

Limits on the fuel temperatures are imposed as required by the test design and are given in the production test authorization. The fuel temperature is obtained by measuring the core temperature of the first fuel element in the fuel assembly. The limit is imposed to guard against unexpected crud buildup on the fuel surface which increases fuel temperatures.

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8. L. S. Tong, H. B. Currin, and A. G. Thorp II. New Corellations Predict DNB Conditions, Nucleonics, Vol. 21, No. 5, pgs. 43-47, May, 1963.
 9. H. B. Currin and L. S. Tong, Modification of W-2 Corellation for Annular Data, January 6, 1964.

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The low pressure limit is 100 psi below the normal pressure. The total pressure in the C-1 loop is limited to the design pressure of 2530 psi by a rupture disc located near the test section outlet.

For operations described in this report, the circulating coolant flow rate is limited to 13 gpm which provides a BOSF of 1.45. Indicated flow rates below this value results in an automatic transition from circulation to single pass cooling by the emergency automatic dump system. A flow rate of 11 gpm results in a reactor scram.

Figure 5 shows the fuel temperature distribution and burnout safety factors obtained during the steady-state operation of the C-1 Loop with a fuel charge of eight weight per cent Pu-Al fuel elements. The maximum fuel temperature is determined by the crud buildup on the fuel surface during irradiation as a result of corrosion of the fuel cladding material. Temperatures are shown for a clean surface and for a crud coefficient of 5000 BTU/hr ft² F which could occur in C-1 operation⁽¹⁰⁾. The BOSF obtained during normal operation is 1.54. Operation at any one of the dump trip points reduces the BOSF to 1.45.

4. Operation of the Automatic Dump System

A low pressure, high temperature, or low flow trip signal from the coincident trip dump instrumentation described in Section III.A.6 will activate an automatic depressurization of the loop and transition to single-pass cooling by the backup coolant system. Activation of the dump system is as important from a personnel, reactor, and test safety standpoint as a reactor scram, and for this reason a highly reliable dump system has been incorporated which includes the following:

- a. Parallel dump valves with independent controls.
- b. Series isolation valves with independent controls.
- c. Coincident trip dump instrumentation.

The sequence of events following receipt of a dump signal are:

- a. Parallel dump valves are opened allowing the high temperature loop coolant to exit to a steam condenser dump tank and flows from there to the drain.
- b. The inlet flow control valve, V-1, and the isolation block valve, V-3, which are in series, close and isolate the loop equipment from the emergency coolant system.
- c. Recirculation and makeup pumps are turned off.

10. D. R. Dickinson, R. J. Lobsinger, and R. S. Richman. Corrosion of Aluminum Clad fuel Elements, HW-SA-2661. August 24, 1961. (Unclassified).

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FIGURE 5

C-1 FUEL TEMPERATURE DISTRIBUTION
AND HEAT FLUX

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- d. The piston in the backup accumulator, driven by high-pressure gas, forces cold water through the test section. Cold water also flows from the loop accumulator.
- e. After the backup accumulator flow is exhausted, the fuel assembly has been cooled to the point where the saturation pressure of the water in the test section is well below the pressure of the backup coolant system. Inlet check valves open and allow C-Reactor process water to flow through the test section to drain.
- f. Two check valves in series near the exit to the test section close to prevent hot water from the out-of-reactor portion of the loop from discharging through the dump valves.

When the fuel assembly is cooled by the backup coolant system, once-through water flows from the reactor risers through inlet check valves to the test section and exits through two dump valves (or the dump valve by-pass) to drain. Low flow and high temperature activate annunciators during single-pass operation, and low-low flow or low outlet pressure will result in a reactor scram. The scram trip set points are normally the same as for circulation operation and are set so that the coolant at the surface of the fuel element is always sub-cooled.

5. Operation of the Automatic Cooldown System

The radiation level of the loop coolant is kept at the minimum possible. High radiation levels of the loop coolant which might result from increased power input to the in-reactor test section, low quality water, crud burst, or test element failure will actuate an automatic cooling down of the loop. When an automatic cooldown cycle is initiated, the control point on the heat exchanger controller is reduced at the rate of 400 F per hour. During and after the cooldown, loop pressure remains constant, loop flow rate increases approximately 10 per cent, and the loop temperature is reduced to the minimum attainable with the heat removal system, about 240 F.

The gas pressure in the back-up accumulator is normally kept at 2150 psi. Pressures above the operating point would empty the accumulator. High pressures which could result from a faulty pressure regulator or other causes would first ring an alarm (2200 psi) and then actuate an automatic cooldown (2250 psi). Pressures below the operating point could render the accumulator inoperative. Low pressure which might result from failure of the gas supply system or leaks in the accumulator gas system will first operate an alarm (2100 psi) and then actuate an automatic cooldown (2075 psi).

6. Shutdowns and Scrams

When the reactor is shutdown, the C-1 Loop will continue to be cooled by its primary coolant system. During shutdown, the loop can be cooled by either the primary or backup coolant system depending on choice. Coolant flow must be provided to the test section at all times during shutdown when the test section contains fuel elements. When the loop is cooled with the

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circulating system during shutdown, the pumps can be cooled by either the process water or filtered water systems of C Reactor. If the pumps are not cooled, they will shut-off automatically. There are maintenance operations of the reactor during shutdown which may require that the B and C risers, which normally provide cooling water to C-1 pumps and backup cooling for the loop operations, be valved off. In this event the fuel elements must be cooled by the circulation system and the pumps cooled by the filtered water system. The transfer to the filtered water system is a manual valving operation. Procedures and a tag system have been established to assure that the B and C risers are not valved off unless the loop is being cooled by circulation and the pumps cooled by the filtered water supply. A Design Change⁽⁵⁾ has been approved to also tie the system into the A and D Risers which, when accomplished, will eliminate the need for the procedures and assure that uninterrupted shutdown cooling flow will be provided to the C-1 facility without the need for manual valving operations.

Following a scram of the reactor, the loop will continue with circulation cooling. The power decay in the C-1 fuel assembly following a scram is shown in Figure 6. The fuel assembly heat generation rate is reduced rapidly by the scram beginning one and one-half seconds following initiation of the event if the scram is initiated by C-1 instrumentation.

7. Charge-Discharge

The C-1 Loop fuel elements contain less than five grams of plutonium per element, or a total of less than 40 grams per charge. Forty elements were fabricated for the first series of tests. The unirradiated elements are stored in the Irradiation Testing storage facility at KW Reactor. The fuel elements are not stored near other enriched material. The fuel assemblies are hand charged directly into the test section. Irradiated specimens are discharged by pulling them into a cask. The cask has nine inches of lead shielding and will accommodate a four-foot fuel assembly. During the discharge, the cask is coupled directly to the test section inlet connector with a discharge coupling. The coupling is attached to the inlet connector and is bolted to the face of the cask. A ball valve in the coupling allows the test section to be closed off and water flow to be maintained during the time that the cask is being positioned and as other discharge preparations are made. When discharge preparations are complete, the coupling valve and cask valve are opened allowing cooling water to flow out of the test section and through the cask. A retrieving tool is then pushed through the cask (an O-Ring seal prevents leakage of coolant around the retrieving tool), through the discharge coupling, and into the test section where it engages the fuel assembly. The fuel assembly is pulled by hand into the cask. The cask valve is then closed and cooling water flows in a jacket around the cavity of the cask.

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FIGURE 6
C-1 LOOP SCRAM TRANSIENT

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After a short cooling period, and when time permits, the cask is moved to the reactor storage basin and the fuel assembly is discharged from the cask. The assembly is disassembled in the storage basin and after a cursory examination the individual fuel elements are either stored for future examination or loaded in shipping casks for transfer to the Radiometallurgy Laboratory in the 300 Area.

8. Control of C-1 Loop Operations

All phases of the operation of the C-1 Loop are covered by documents that receive formal review, approval, and authorization. These documents consist of: 1) Production Test Authorizations, 2) Design Change Authorizations, and 3) Operating Procedures.

The production test authorization document is prepared by the Testing Subsection for review and approval by technical and management staffs. The document describes the proposed test in detail with primary emphasis being placed on the safety of the experiment. Formal approval from the Manager, Research and Engineering and the Manager, Manufacturing signifies that the risks involved in performing this experiment are known, understood, and reasonable.

The modification of the C-1 Loop is accomplished only under formal documented authorizations of a "Design Change" or as a project action for major modifications. Design Changes are authorized by the managers of the participating operations, and the Supervisor, Plant Engineering Operation, and are reviewed by the Manager, Process Technology to ensure reactor safety during and following the modification and by the Manager, Facilities Engineering for code compliance.

In written documentation authorizing modifications to loop components, an acceptance test procedure is required. This procedure includes a functional test of all safety circuit equipment and components affected by the modification.

The operating procedures are prepared by experienced supervisory personnel in the Testing Subsection. The following procedures are in force in C-1 Loop operations.

a. C-1 Loop Operating Procedures

These procedures serve as a guide to the operating technician during normal operation of the loop. The procedure includes: hydrostatic testing, startup and shutdown of the loop, transfer to and from process water, safety circuit and startup checks, normal operation procedures, miscellaneous equipment procedures, and others.

b. Recirculation Without Technician Coverage

These procedures describe the operation, the monitoring instruments and their set points, and the course of action for reactor personnel to take when an annunciator sounds during the time when the in-reactor test section is being cooled by recirculating coolant.

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c. Process Water Without Technician Coverage

These procedures outline the monitoring instruments and set points when the in-reactor test section and its contents are being cooled by single-pass process water. The corrective action required by reactor personnel when an annunciator sounds is also included.

d. Extended Reactor Outages

This procedure describes the flow monitoring required during extended reactor outages.

e. Discharge Procedure

Discharge procedures are available for the normal discharge of test specimens and for the emergency discharge of the in-reactor test section.

f. Valve Lock-Out Procedure

All valves critical to the operation of the C-1 Loop, including valves in instrument sensing lines, are locked or sealed in the correct position to assure that the valves are not inadvertently operated.

9. Personnel

An experienced and well-trained technical and operating staff is available in the Testing Subsection for the C-1 Loop operations. An exempt engineer is assigned the responsibility for the operation and maintenance of the loop. Normally, four technicians are trained and assigned for routine operation, sampling, and other work. Only competent technicians with considerable experience are assigned to the loop. A technician fully qualified for all phases of operation of the facility must have at least one year of experience in the operation of high-temperature, high-pressure systems, at least one year experience in irradiation testing, and be fully checked out and familiar with the C-1 Loop facility and procedures.

Personnel protection is provided through designs in accordance with applicable codes, radiation detection systems, operating procedures, and job hazards breakdowns. All high-pressure piping and equipment is confined to the X-1 level and the majority of this equipment is contained within the C-1 Loop pump cubicle. Only occasional access to this area is required during reactor operation (instrument adjustment, sampling, equipment inspection) and work on the high-pressure equipment is prohibited during loop operation.

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C. Safety Analysis

1. General

The safety of the C-1 Loop is determined by safety features built into the design and those that are written into the procedures for the proper operation of the equipment. The way that these safety features interact in emergency situations to limit the magnitude of possible accidents is discussed in this section. A number of conceivable nuclear and mechanical accidents are postulated. In each case, the situation in the loop, the action of the safety system, and the subsequent actions of the backup facilities are followed to their logical outcomes. Since operation of the C-1 Loop with fuel elements adds only a small amount of positive reactivity to the C Reactor, the magnitude of possible nuclear accidents resulting from loop operations is negligible. Therefore, the majority of the discussion in this section deals with mechanical accidents.

2. Nuclear Accidents

a. General

A complete loop charge of eight plutonium-aluminum fuel elements described in Section III.A.2 adds only 0.007 per cent $\Delta k/k$ of reactivity to the reactor and causes no serious flux peaks to develop. In addition, the operation of the loop has no effect on the ability of the control systems to satisfactorily control the nuclear process in the reactor. Thus, from a reactivity standpoint, the effect of the C-1 Loop operation on reactor operations is so small as to be indistinguishable. There are two types of nuclear accidents, however, which warrant discussion. The first is possible overheating of the C-1 Loop fuel elements caused by a rapid reactor power excursion introduced, for example, by an improper withdrawal of control rods. In this case, it is of interest to note whether operation of the C-1 Loop significantly adds to the accident outcome. The second type is the loss-of-coolant accident with its attendant nuclear and physical outcomes. Since the initiating events are predominantly mechanical failures, the complete safety analysis is covered below in Section C.3.c.v. Discussion of the nuclear aspects of loss-of-coolant accidents is included in this section, however.

b. Overheating Accidents

The C-Reactor Hazards Summary Report (3) discussed the overheating accidents which are considered credible and the measures and devices used to prevent these accidents. Further analysis shows that the C-1 Loop could experience a 25 per cent surge in power following improper control rod withdrawals in the reactor with a flattened flux distribution, and a 50 per cent surge for the case where the reactor flux is badly skewed. The latter accident is less credible, but is considered here because it is the worst case. The incremental effect of the C-1 Loop on the outcome of a reactor overheating incident of this magnitude is examined in this section.

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It is assumed that the increase in power would be rapid enough so that the outlet temperature control system of the C-1 Loop would not operate in time to maintain the coolant temperature at the fuel column outlet at the normal operating point of 572 F. When the coolant temperature reaches 590 F, an automatic transition to the backup cooling system would be initiated. Test results show that cold water would be forced over the fuel elements from the backup accumulator within one second following initiation of the high temperature trip (See Figure 7). The power level would reach 150 per cent of initial power before the excursion is terminated by a scram initiated by C-Reactor instrumentation. The maximum fuel temperatures and heat fluxes reached in the C-1 Loop during the accident are shown in Table 5.

TABLE 5

C-1 LOOP TEMPERATURES AND HEAT FLUXES FOLLOWING
A 50 PER CENT POWER SURGE

Maximum heat flux, BTU/hr ft ² x 10 ⁻⁶	0.866
Burnout heat flux, BTU/hr ft ² x 10 ⁻⁶	1.14
Maximum outlet coolant temperature, F	590
Saturation temperature at outlet, F	655
Subcooling at the outlet, F	65
<u>Fuel Temperatures, F</u>	
Surface	673
Maximum Core	
clean surface	786
with crud buildup	960
Fuel melting temperature	1220

As indicated in Table 5, a sudden increase of 50 per cent in the C-1 Loop fuel power could be sustained without damage to the C-1 fuel elements. Nucleate boiling would take place at the surfaces of downstream fuel elements, but burnout would not be expected to occur. The maximum fuel surface temperatures would be at least 500 F below the melting temperature of the cladding. The analysis shows that operation of the C-1 Loop will not increase the consequences of reactor power surge accidents.

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c. Loss-of-Water

The nuclear aspects of the C-1 Loop loss-of-water accident were studied using the CLUMSY I⁽¹⁰⁾ computer code. Two situations were examined. The first assumed the fuel elements remained in their as-charged positions, and the second assumed the fuel elements were displaced to the exact center of the reactor channel. The latter situation is not credible but serves as a worst case.

The calculations indicate that the C-1 fuel power would increase less than five per cent following water loss. The power of adjacent C-1 Reactor fuel columns would increase about 50 kw for the situation in which the fuel elements remain fixed and about 100 kw for the situation where the fuel elements are displaced. Outlet coolant temperatures in adjacent tubes would increase about 4 F for the former case and 8 F for the latter, which is insufficient to cause overheating of the fuel elements in these tubes. Further, the total reactor reactivity would increase by no more than 0.005% $\Delta k/k$ which does not jeopardize reactor control.

3. Mechanical Accidents

a. General

The outcomes of possible mechanical accidents are considered in this section. Because of the importance of maintaining adequate cooling of the fuel elements, the major portion of the safety analyses deal with events associated with the coolant systems. The discussion is separated into consideration of the measures and devices in the C-1 Loop to prevent mechanical accidents, and the description of possible mechanical accidents noting the outcomes of failures should the protection provided become inoperative. The mechanical accidents considered are primary coolant system equipment failures, backup coolant system equipment failures, breaks in the high pressure system, fuel element failures, charge-discharge accidents, and inadequate shutdown cooling.

b. Measures and Devices to Avoid Mechanical Accidents

The chief measures taken to avoid mechanical accidents have been through careful design, material selection, quality and reliability of equipment, and provision of a highly reliable backup coolant system. Sound engineering practice was used throughout the design. The backup coolant system includes parallel dump valves with independent controls, series isolation valves with independent controls, and coincident trip instrumentation to achieve a high system reliability.

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10. R. E. Tiller and C. E. Oster. CLUMSY-I - Time and Space Dependent Reactor Kinetics Code for the IBM-7090 Computer, HW-87221. August 1, 1963. (Unclassified).

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The loop piping is well supported, and its location makes it generally inaccessible and not subject to impact by tools and equipment in common use on the reactor. The Schedule-80, Type-316, stainless steel inlet piping easily satisfies the requirements of the American Standards Association Code for Process Piping, ASA B31.1. The piping is designed so that there are no highly stressed points in the system, and the use of deionized coolant during high temperature operation should effectively prevent stress corrosion cracking. The fluid velocity at normal operating flow rates is seven feet per second, or within the accepted range of five to fifteen feet per second for conventional piping practice.

The in-reactor pressure tube has been carefully tested in excess of code requirements by ultrasonic and dye-penetrant methods for both internal and surface defects which could act as stress risers during operation. No internal defects were detected. All outer surface defects were removed as confirmed by repeated dye-penetrant testing following defect removal operations. In addition to assuring the integrity of the loop piping by design and material selection, the loop will be operated within safe pressure limits as specified by applicable codes (ASA B31.1). Pressure relief devices will be employed to prevent a loop over-pressure which might lead to piping failure.

Safety circuits have been provided to prevent individual equipment failures from developing into accidents and include: a) dump circuits that automatically transfer the loop to once-through emergency cooling when abnormal conditions of flow rate, temperature or pressure are detected; b) scram circuits that shutdown the fission process under conditions of inadequate cooling; c) annunciator circuits that warn of incipient difficulty; and d) cooldown circuits that automatically initiate a reduction in coolant temperatures when certain potentially unsafe conditions occur.

c. Outcome of Mechanical Accidents

i. General

The outcome of possible mechanical accidents in the C-1 Loop are described. With the exception of accidents which occur during periods when the reactor is shutdown, the failures are postulated to occur during full power operation of the reactor with the following initial conditions in the C-1 fuel assembly: 1) fuel column power of 80 kw; 2) coolant temperature and pressure at the fuel assembly outlet of 572 F and 2290 psi; and 3) coolant flow rate of 15 gpm.

ii. Failures of Primary Coolant System Equipment

Failures of primary coolant system equipment which result in a decrease in the circulating coolant flow rate, an increase in coolant outlet temperature, and a decrease in loop pressure beyond the set points shown in Table 2 will initiate an automatic dump of the loop coolant and a transition to once-through cooling by the backup coolant system. The operation of the automatic dump system is described in Section

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III.C.4. The coolant flow, pressure, and temperature transients which occur during an automatic dump sequence in the C-1 Loop are shown in Figure 7 as determined in the testing program which was run to checkout the loop equipment prior to nuclear operation of the facility.

The curves shown were obtained by simulating a failure of the flow control system which caused an immediate and complete stoppage of circulating coolant flow. As indicated in Figure 7, the dump valves open immediately after a dump sequence is initiated by a low flow trip (13 gpm). The response of the dump valve system is fast enough to prevent the coolant flow rate from dropping to the scram point (11 gpm). Cold water is forced through the system by the backup accumulator at more than double the normal circulating coolant flow rate during the interval one to five seconds after the flow control system failure. This high rate of flow of cold water serves to cool the system to effect a rapid transition to cooling by the low-pressure backup coolant system. The accumulator water is exhausted in about five seconds, the pressure drops below the backup coolant supply pressure, and emergency coolant begins to flow through the test section. In the tests, the indicated coolant flow rate was observed to oscillate for a period of about ten seconds after which a steady-state, once-through, coolant flow rate of 16 gpm was achieved. The coolant temperatures at the outlet of the test section did not exceed the coolant saturation temperature at this point as indicated by coolant temperature and pressure measurements obtained during the transient. Since the coolant flow rate did not drop below the scram point, a reactor scram would not have been initiated in an actual case. A reactor scram would occur only in the event that adequate cooling was not maintained throughout the transient and subsequent steady-state operating periods. The adequacy of this dump system is considered for various postulated primary system failures below.

A loss of power to the primary coolant pumps, a pump shaft break during periods of single-pump operation, and failures of the flow control valve system constitute single equipment failures which would cause a low flow dump of the C-1 Loop. The transients caused by these failures would be essentially the same as those shown in Figure 7. There would be no boiling in the fuel assembly during the transient and the fuel temperatures would not exceed their initial values despite the fact that the reactor is not scrammed.

The transients caused by failure of the heat removal system are similar. The heat generated in the fuel assembly is removed from the system by three water-to-water heat exchangers operating in parallel. The amount of coolant flow which passes through the heat exchangers is regulated by a two-way valve which is controlled by a temperature recorder-controller on the basis of the coolant temperature at the fuel assembly outlet. Failure of this system could cause the two-way valve to close, thereby causing the coolant to bypass the heat removal system.

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FIGURE 7

**TRANSIENTS FOLLOWING OPENING OF C-1 LOOP DUMP VALVES
- NORMAL OPERATION WITHOUT SCRAM**

At the time of failure, the loop coolant would consist of a cold leg at inlet temperature and a hot leg at normal outlet temperature. There would be no indication of the equipment failure until coolant from the hot leg had passed through the fuel assembly. At this point the outlet coolant temperature would increase from 572 F to 608 F and an automatic transfer to the emergency coolant system would be triggered by a high temperature trip. Cold water from the backup accumulator would rapidly displace the hot water in the fuel assembly and cool the system to effect a transition to cooling by the emergency coolant system as shown above. A reactor scram would not be initiated. The closest approach to burnout during the transient would occur just before the dump valves opened to provide cooling by the emergency system. At this point the burnout safety factor would be 1.38. The maximum fuel temperatures during the transient would be 640 to 740 F depending on the amount of crud which had accumulated on the clad surface.

Loss of primary pressure control would not lead to serious accident outcomes. Loop pressure is maintained by two makeup pumps and controlled by bleeding excess water from the loop through either of two bleed valves, each with its own controller. A three-gallon pressure accumulator is provided to even out pressure fluctuations. If one of the bleed-valve systems fails and the pressure rises, the makeup pumps are instrumented to stop and the second bleed-valve system would tend to reduce pressure to the control point. If this system also fails to operate, the loop pressure could rise to 2530 psi and the rupture disc in the safety head breaks, the loop is depressurized, and an automatic transition to the emergency coolant system is effected by a low pressure trip (100 psi below normal). At the time of transfer to emergency cooling, the coolant in the fuel assembly would still be subcooled and the burnout safety factor would be 1.46. The maximum pressure reached in the loop would be below the design pressure (because of lower temperature coolant), and no damage to the loop piping or in-reactor pressure shell would occur.

A depressurization accident involving, perhaps, the accidental failure of a loop rupture disc, would follow the same course, except that operation near the design pressure rating of the loop would be avoided.

iii. Failures of Backup Coolant System Equipment

As discussed above, single failures in the circulating coolant system cause a loop dump and transition to once-through cooling. As long as the dump system performs as designed, the transition is achieved without scrambling the reactor and without damage to the fuel elements, other loop components, or the parent reactor. This section considers the outcome of failures of backup coolant system equipment simultaneously with failure of the primary coolant system.

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The backup coolant system consists of piping, dump valves, check valves, and a backup accumulator as described in Section III.B.5. The coolant supply is obtained from the C-Reactor risers. In the following discussions it is assumed that a failure has occurred in the circulating coolant system that has initiated a loop dump sequence. As we have seen, once a loop dump is initiated the sole source of cooling for the test section is the backup coolant system. The outcome of additional component failures in this system, excluding piping failures, is considered as follows:

It is assumed that the backup accumulator pressure is lost at the same time the primary coolant system failure occurs. If the backup accumulator pressure had been lost prior to failure of the primary coolant system, instrumentation would have automatically reduced loop coolant temperatures and sounded an alarm as discussed in Section III.C.5. The coolant flow and pressure transients were experimentally determined for this accident in the prestartup tests and are shown in Figure 8.

Both the backup accumulator and the normal pressure accumulator are piped into the emergency coolant system as shown in Figure 1. Therefore, with the loss of the backup accumulator, cold water flow from the pressure accumulator would serve to cool the system to effect a rapid transition to cooling by the low-pressure, backup coolant system. The coolant flow rate during the transition period, however, is about one-third of that achieved with the backup accumulator because of a lower driving pressure and the hydraulic resistance added by the additional piping.

As shown in Figure 8, the coolant flow rate would drop below the scram point about one second after the two failures occurred. Steady-state cooling by the emergency system would be achieved in about 16 seconds. The closest approach to burnout would occur between 1.0 and 2.5 seconds after the failures occurred and before the reactor scram is effective in reducing the heat generation rate. If one assumes a steady-state is achieved during this interval, the steam quality at the fuel assembly outlet is calculated to be five per cent. The burnout correlation in the quality region of Tong, Currin, and Thorp (Reference 6) predicts that film boiling would not take place at the fuel surface at this quality and the conditions shown in Figure 8. Further, if a scram failed to take place as indicated, the calculations indicate that the coolant flow rates during the transient would be sufficient to prevent melting of the fuel elements. It should be noted, perhaps, that at least two additional C-1 facility failures are required for the scram circuit to be inoperative.

Failure of a single dump valve in addition to a primary coolant system failure would not change the dump transients shown in Figure 8 because the isolation valves (locked open during operation) located upstream and downstream of the dump valves determine the rate of depressurization. Failure of both dump valves simultaneously with failure of the primary coolant system would result in complete loss of cooling and eventual melting of the fuel assembly.

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FIGURE 8

TRANSIENTS FOLLOWING OPENING OF C-1 DUMP VALVES
OPERATION WITHOUT THE BACKUP ACCUMULATOR

Test Section Power - kw

Time After Dump Initiated - Seconds

Coolant Flow Rate - gpm

Time After Dump Initiated - Seconds

Coolant Pressure - psig

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It is assumed that an inlet check valve failed to open to allow process water to enter the system following failure of the primary system. The test section would be cooled by cold water flow from the backup accumulator during the first five seconds of the transient and the initial fuel and coolant temperatures would be reduced in this time.

After the accumulator flow was exhausted, the flow rate would drop to zero as a result of failure of the check valve to open and a reactor scram would be initiated. The course of the accident from here to its ultimate outcome is very similar to that described in detail below in Section iv and in Appendix C for a complete inlet piping break. The difference between the two accidents is that the time required for the fuel assembly to melt is slightly less in the accident considered here because of the lower fuel temperatures at the time the coolant is lost. Using the methods discussed in Appendix C, it is calculated that the fuel assembly would be completely melted in about 65 seconds. The stainless steel fuel housing and pressure shell would not melt. The maximum temperature reached during the accident is calculated to be less than 2100 F.

The backup coolant system includes two check valves installed in series between the dump valves and the pump inlet. These valves prevent high pressure water in the out-of-reactor section of the loop from discharging through the dump valves during a dump transient and thereby reducing the coolant flow rate through the test section. Failure of one of these valves to close following opening of the dump valves would have no effect on the dump transients because the second valve would operate. Simultaneous failure of both valves at the time the dump valves open would cause a temporary reduction in the backup coolant flow rate below the reactor scram flow rate. The scram would assure that the fuel assembly would be adequately cooled despite the temporarily reduced coolant flow rate.

iv. Breaks in the In-Reactor Pressure Shell

Assuming a break in the in-reactor pressure shell, adequate cooling would continue to be provided to the fuel assembly because the break would occur downstream of the fuel assembly outlet. A transition to emergency cooling would be initiated by a low-pressure trip shortly after the break in the pressure shell occurred. High-pressure coolant would be discharged into the reactor at rates depending upon the size of the break. The sudden release of high-pressure coolant into the reactor due to large breaks in the pressure shell could conceivably result in forces sufficiently large to lift the graphite stack and distort the upper aluminum process tubes. The amount of lift would not be expected to be large enough to cause rupture of the surrounding tubes or inadequate cooling of the fuel elements in these tubes.

The potential for reactor damage in this accident is considerably reduced by the fact that the pressure shell occupies less than half of the available volume of the C-test hole. Thus, considerable void

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volume is available to dissipate the steam discharge. Calculations of the amount of lift expected following the accident are very uncertain. An experimental evaluation is extremely costly. Therefore, the in-reactor pressure shell has been subjected to testing over and above that required by the codes (see Appendix B) to assure its integrity.

v. Breaks in the Out-of-Reactor High-Pressure System

The only breaks in the out-of-reactor high-pressure system which could conceivably result in a permanent reduction or loss of cooling to the fuel assembly are those in the inlet piping which is common to both the primary and backup coolant systems. Breaks in other sections of the loop would result in a low pressure dump and transition to once-through cooling of the fuel assembly by the backup coolant system. The ability of the backup coolant system to adequately cool the fuel elements would not be compromised. Certain failures in the high-pressure system could result in some spread of contamination within the X-1 level due to the discharge of radioactive coolant within the room. There is an insufficient quantity of radionuclides in the coolant to result in a serious radiation hazard within the building. Personnel are shielded from the high pressure system by a metal wall, 1/8-inch thick, which would prevent their direct contact with hot water steam in the event of a high-pressure system failure.

A major break in the section of piping common to both the primary and backup coolant systems would result in a complete loss of coolant to the fuel assembly. Within one second after the break in the piping occurs the coolant pressure in the fuel assembly drops from 2290 psi to about 900 psi which is the saturation temperature of the coolant, and the heat generation rate in the fuel assembly increases about five per cent. A reactor scram is initiated immediately by a low outlet pressure or a low flow trip depending on the location of the failure. The low outlet pressure scram has not been employed in C-1 operations to date. It is being included in future operations in recognition of the fact that complete failure of a short run (about three feet) of piping between the flow monitors and test section inlet might result in coolant loss without scram. No water is flowing through the system. The water will boil out in a short time, assumed to be one second in this analysis. Fuel cooling was assumed during these first two seconds of the transient. The heat generation rate begins to fall off rapidly, beginning 2.5 seconds after failure as the scram becomes effective. Once the fuel housing has boiled dry, the temperature of the fuel will increase rapidly due to fission product decay heat. Assuming 1) no heat transfer after the tube has boiled dry, and 2) the scram transient shown in Figure 6, the fuel assembly is calculated to reach its melting point in about 10 seconds and to be completely melted in about 50 seconds.

It is assumed that the only mechanism for heat loss from the fuel is radiation to the graphite through the stainless steel fuel housing and pressure shell. There would be little radiation of heat from the

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fuel up to the melting point because the fuel temperature would not be greater than the graphite temperature. As the fuel continues to generate heat the temperature increases to values above the graphite temperature and heat would be lost to the graphite by radiation. After melting, the fuel material flows to the bottom of and is in contact with the fuel housing. The melt continues to generate heat and increases the temperature of the housing. When the housing temperature reaches values significantly above the graphite temperature, which remains very near its initial temperature of 1200 F during the transient, heat from the melt would radiate to the pressure shell, which, in turn, radiates to the graphite moderator. At some point in the transient this heat transfer would be sufficient to arrest the temperature excursion. The calculations indicate that a maximum temperature of about 2100 F would be reached in about 2.7 minutes. This is below the melting temperature of the fuel assembly housing (2550 F). The housing would most likely become distorted but is not expected to melt during the transient. The pressure tube would reach a maximum temperature of about 1580 F and would not be damaged.

The maximum increase in tube power in adjacent columns caused by the water loss would be 50 kw, indicating that the temporarily increased local reactivity in the C-1 Loop would not affect the remainder of the reactor.

A complete break of the inlet piping with simultaneous failure of the scram instrumentation would also result in melting of the stainless steel housing tube and partial melting of the pressure tube. In this case the fuel assembly would melt in about 10-15 seconds. This accident requires failure of at least two of the three scram circuits in addition to the piping failure.

vi. Fuel Element Failure

The C-1 Loop has been used primarily for testing the corrosion resistance of aluminum cladding materials. The minimum cladding thickness of the fuel elements is 50 mils. Test exposures are limited to 15 mils uniform corrosion loss as determined by corrosion rates measured on similar test specimens out-of-reactor. Despite these precautions, fuel cladding failures could occur because of the unknown effects of irradiation on the corrosion rate of the cladding material or fuel element defects missed in the inspection procedures. To reduce the consequences of possible fuel failures, corrosion resistant fuel cores are used. The corrosion resistance of the eight weight per cent Pu-Al alloy has been shown in out-of-reactor tests to be comparable to that of the cladding material. Further, the C-1 fuel elements are coextruded which results in a bond between the jacket and core. This bond keeps the corroding area to a minimum in the event of cladding failure. If a fuel failure occurs, the resulting activity in the loop will be detected and effect an automatic reduction of the loop coolant temperatures so as to reduce the fuel corrosion rate. For these reasons, loop contamination due to fuel failure is held to a minimum and does not constitute a serious personnel or operating problem.

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vii. Charge-Discharge Accidents

Test specimens for the C-1 Loop are stored in the Irradiation Testing storage facilities at the KW Reactor. They are not stored near other enriched material at the reactor. Fuel elements for the first series of tests contained less than 200 grams of plutonium. The test specimens are assembled into a test assembly at KW, after which they are sent to the C Reactor where they are hand charged directly into the test section.

The fuel assembly discharge procedures and equipment are described in Section III.B.7. Discharge operations are not permitted until at least twelve hours after reactor shutdown to allow for the decay of fission product afterheat. Thus, if cooling water is lost to the cask containing the fuel assembly during or after discharge operations are complete, the heat generation rate is low enough so that the fuel elements will not melt.

The fuel assembly is discharged by pulling it by hand from the reactor section using a 20-foot retrieving tool. The cask is mechanically coupled to the inlet connector during the discharge in such a manner that the cask could not become disengaged unless large forces were applied to the retrieving tool. The design of the cask and the retrieving tool prevents inadvertently pulling the fuel element through the cask and into an unshielded position. Immediately after discharge, the cask is uncoupled, a shield plug is installed and bolted in place, and the cask is plainly identified as to its contents. Failure of the coupling or of the design feature which prevents pulling the fuel assembly past the shielding in the cask could result in high radiation exposures to personnel involved in the discharge operation.

A very difficult, but non-hazardous, situation could result if the fuel train were to stick in the discharge coupling between the inlet connector and the cask. This coupling is shielded during a discharge; however, there is a possibility that the fuel train could not be pulled into the cask or returned to the reactor. The only recourse would be to add shielding and remotely disconnect the cask. The fuel assembly, or possibly the entire in-reactor section, would then be discharged using remote unshielded techniques.

The discharge techniques used are similar to those that are used for the discharge of other in-reactor test facilities. A large amount of experience has been gained using these techniques. Procedures have been prepared and mockup discharges are made. Prior to each discharge and after the discharge cask is in place, an alignment bar is used to assure alignment of the cask, discharge coupling, and inlet connectors. The discharge system is designed so that the test assembly is pulled through a progressively larger hole. All edges on the cask and discharge couplings are tapered and edges on the fuel assembly are rounded.

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viii. Inadequate Shutdown Cooling

It is assumed that the process water supply to the C-1 Loop is inadvertently shut off four hours after the reactor has been shutdown on a routine basis. Such action would require a gross personnel error, involving inadvertent closure of the isolation valve (kept locked open under strict procedural control) located in the supply line from the reactor risers. This will be the case following completion of Design Change 906, Reference (5). With the present arrangement, shutdown cooling for the C-1 Loop could also be shut-off by errors by reactor personnel which would result in the coolant to the B and C risers being valved off without following established procedures which are relied on to assure that continued shutdown coolant is supplied to the C-1 Facility (See Section III.B.6). If the loop were being cooled by the circulating system rather than the once-through system, the flow would also stop following the loss of process water to the facility because the loss of cooling water to the pumps would result in their being automatically shut-off. In the latter case, a low flow trip would cause the dump valves to open, so the same loop situation would exist for each case following the valving out of the process water. Water would be trapped in the test section because the exit piping is installed in a vertical orientation.

The fission product decay heat will be transferred to the stagnant water trapped in the test section. The water will reach the boiling temperature in about two minutes. Subsequent steam formation and expansion would expel the trapped water through the outer annulus to the drain. Due to the low heat generation rate (500 watts in whole fuel assembly), the process would be expected to be slow and non-violent in nature. In about two hours after the flow interruption, a steam atmosphere would exist in the fuel assembly. Assuming the water in the outer annulus or inlet section of piping is not available for heat transfer during the transient, the fuel assembly would melt in an additional 20 minutes or about two and one-half hours after the coolant supply was stopped. Flow stoppage during shutdown would result in an annunciator warning which would be expected to result in the cooling being restored well within this two and one-half hour period. The initiation of the accident requires a single failure, the valving error. For fuel melting to occur, however, it is necessary also that the low flow alarm be ignored or inoperative.

4. Conclusions

Analyses of postulated accidents in the C-1 Loop indicates, with the exception of major breaks in selected piping, that at least two simultaneous failures of independent systems are required for the fuel assembly to be inadequately cooled. One of the failures could be a component in the primary coolant system; the second must be the failure of one of the inlet check valves of the backup coolant system. Excluding these check valves, three simultaneous failures of independent systems are necessary

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for fuel melting to occur. The three failures must include the failure of both independent dump valve systems.

Most piping breaks would not have serious outcomes, but a large break in the inlet piping common to both coolant systems would result in complete loss of water in the test section. This accident is considered the maximum credible accident and its consequences are described in detail in Section III.D.2.

The sudden release of high pressure water into the reactor by failure of the in-reactor pressure shell could conceivably result in forces large enough to lift the graphite stack and distort upper aluminum process tubes. The amount of distortion would not be expected to be great enough to result in rupture of these tubes or inadequate cooling of the fuel elements in these tubes. The C-1 fuel elements would be adequately cooled in this accident.

The C-1 test section adds no more than 0.007% $\Delta k/k$ of reactivity to the reactor, and thus, the magnitude of possible nuclear accidents resulting from loop operations is negligible. A complete loss of the water in the C-1 section increases the overall reactor reactivity by no more than 0.005% $\Delta k/k$, and increases adjacent tube powers about four per cent.

The interruption of shutdown cooling could result in melting of the fuel assembly if the situation were not corrected within two and one-half hours after loss. At least two failures would be required for fuel melting to occur during shutdown.

Other accidents studied include accidents during charge-discharge operations and fuel element failure. Accidents during discharge operations could expose personnel involved in the discharge to high radiation levels. The fuel elements used in the C-1 experiments are highly resistant to corrosion by high temperature water. Fuel element failure, therefore, would result in only minor contamination of loop equipment and does not represent a serious personnel or operating problem.

D. Hazards Evaluation

1. Credible Accidents

Safety analyses of several possible nuclear and mechanical accidents in the C-1 Loop have been presented. It was determined that with the exception of a major break in selected piping that at least two simultaneous equipment failures are required for the fuel assembly to be inadequately cooled. One of these failures must be in the primary coolant system, (primary pumping power, for example), and the second must be failure of one of the stainless steel inlet check valves to open when required to allow emergency cooling of the test section. The check valves are highly reliable with few moving parts and open mechanically when the pressure in the test section drops a few psi below the backup coolant supply pressure (420 psi).

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The next combination of failures which could result in fuel melting involves a primary coolant system failure and failure of both dump valves to open, a triple accident. For purposes of assigning credibility to an accident, the convention was used that the occurrence of more than two simultaneous failures of independent systems is not credible. On this basis, the most serious credible accident is the failure of the inlet piping common to both the primary and backup coolant systems. The ultimate consequences are similar to those of the double accident involving failure of the primary coolant system plus failure of an inlet check valve to open. The former accident is considered the maximum credible accident because the fuel melts sooner, the maximum temperature reached during the accident is higher, and the path for fission products to be released from the test section is considerably shorter.

2. Maximum Credible Accident

a. Description

The maximum credible accident is a complete break of the inlet piping of the C-1 Loop common to both the primary and backup coolant systems. The course of this accident is discussed in Section C.3.c.v., and the methods used in determining the sequence of events is given in Appendix C. The accident is assumed to occur when the reactor is operating at full power, resulting in a C-1 test section power level of 80 kw, made up of 72 kw in the fuel assembly and 8 kw in the stainless steel pressure shell and fuel housing. The coolant temperature at the fuel assembly outlet is 572 F and the loop has been operated for 100 days. The accident starts when the inlet piping breaks and water flashes into the room containing the C-1 Loop equipment on the X-1 level of the C Reactor. The loss of water results in a five per cent increase in the fuel assembly power. A reactor scram is initiated by a low flow indication on the flow monitors or a low outlet pressure signal depending on the location of the break. The test section boils dry in about two seconds, and the fuel assembly temperature increases due to fission product decay heat. The rate at which the fuel assembly temperature rises is reduced considerably by the reactor scram which effectively reduces the heat generation rate to less than 10 per cent of its initial value six seconds after initiation of the accident. The fuel assembly is completely melted in about 50 seconds. The maximum temperature that might be reached, assuming the only mechanism for heat loss is radiation to the graphite, is 2100 F which is below the melting points of the fuel housing and pressure shell. The reactor power excursion associated with loss of water in the C-1 Loop would be mild and there would be no overheating of fuel elements in adjacent process tubes.

b. Fission Product Release

To estimate the fission product release following the maximum credible accident, it is necessary to make a number of assumptions. The assumptions made in this analysis are similar to those made in previous safety analyses⁽³⁾ as follows:

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1. All of the noble gas fission products are assumed to escape the molten metal and the reactor by way of the 200-foot ventilation stack.
2. Half of the halogens and half of the volatile solids would escape the molten fuel specimens.
3. One per cent of the remaining fission products (non-volatile solids) would escape the molten fuel specimens.
4. Five per cent of the halogen fission products entering the ventilation system would escape retention by the charcoal filters.
5. Only 0.05 per cent of the fission product solids entering the ventilation system would escape retention by the filters.

The fission product inventory in the plutonium-aluminum fuel assembly and assumed quantities released to the environs following the maximum credible accident are shown in Table 6.

c. Offsite Radiation Hazards

In estimating the radiological effects of fission product release following melting of the C-1 plutonium-aluminum fuel assembly, it was assumed that the fission products are released to the atmosphere from the 200-foot ventilation stack after passing through the confinement system filters of the C Reactor. A moderately stable atmosphere was assumed for the radiological evaluations since this condition maximizes the environmental effects of releasing fission products from the stack. In addition, wind speeds of one meter per second and five meters per second (11 mph) were assumed in the analysis. The latter is a frequent Hanford wind speed. The slower speed, 1 m/sec, was chosen because most environmental consequences are more serious at low wind speeds.

Radiation hazards are greatest at specific distances from the stack, dependent on wind speed. The point of maximum effect at wind speed 1 m/sec is 3 miles, and at 5 m/sec is 12 miles. Interpretation of consequences require relating these distances from the C-Reactor stack to the wind direction and location of Hanford plant boundaries. This information is available in Volume 3 of Reference (3).

Some persons downwind of the C Reactor stack will sustain not only external exposure but exposure from inhalation during passage of the fission product cloud. With the 1 m/sec wind, the maximum whole body exposure would be less than 0.1 rem and the maximum thyroid exposure would be less than 0.4 rem. These exposures would be sustained at the point three miles from the stack. The maximum exposure during a 5 m/sec wind would be at a point twelve miles away and the dose sustained by the whole body and thyroid would be less than 0.01 rem and less than 0.03 rem, respectively. Although these exposures would be considered high for routine discharges of radioactive waste, it is not a serious consequence of a single accidental release.

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An evaluation was also made of the external exposure sustained during the first year after the accident as a result of deposited contamination. Estimates were made at each of the two critical locations corresponding to the two wind speeds. Less than 0.1 rem would be sustained at the 12 mile point and less than 0.3 rem would be sustained at the three mile point. These accumulations of exposure assumed the subject remained at the point of highest deposition for a full year.

At depositions of $0.15 \mu\text{c Sr-90/m}^2$, land use restrictions are to be imposed. These restrictions may include removal of grazing animals and confiscation of crops. There may even be some moderate decontamination in areas of high human occupancy. Following the maximum credible accident in the C-1 Loop, the highest deposition of Sr-90 would be less than $10^{-7} \mu\text{c/m}^2$. Even if the entire inventory of Sr-90 in the assembly were released from the stack, a deposition of $0.15 \mu\text{c Sr-90/m}^2$ could not occur.

The deposition of I-131 is often the most serious environmental effect from a stack release of reactor fission products. The effect is not serious in this accident because of the relatively low inventory of I-131, 2,500 curies. Since the only release is through the charcoal filters less than 70 curies I-131 will be deposited on the ground. The maximum deposition at the twelve mile point, 5 m/sec wind speed, would be less than $0.5 \mu\text{c I-131/m}^2$ and about $1.3 \mu\text{c I-131/m}^2$ at the three mile point if a wind of 1 m/sec was blowing at the time of release. Unless the deposition exceeds $1 \mu\text{c I-131/m}^2$, control measures are not to be imposed. Considering the slow wind speed, the greater than $1 \mu\text{c I-131/m}^2$ deposition will extend 6.2 miles from the stack but will include an area of only one-third square mile. No control measures are likely to be imposed in this small area since the I-131 will decay to less than control levels within four days.

It should be noted that the analysis probably overstates the offsite consequences because 1) no decay of fission product activity was considered prior to release when in the actual case a delay of several minutes would be available; and 2) the assumption that meteorological conditions remain constant tends to maximize the distances from the site that are affected by fission product dispersal. It may be concluded that the potential fission product release resulting from the maximum credible accident in the C-1 Loop would not result in serious radiation exposures to offsite personnel.

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TABLE 6*

QUANTITIES OF RADIONUCLIDES IN A C-1 Pu-A1 FUEL ASSEMBLY AND
ASSUMED RELEASES FOLLOWING COOLANT LOSS

<u>Fission Products</u>	<u>Fission Product Inventory (Curies)</u>	<u>Melt Release Fraction</u>	<u>Fission Products Entering Ventilation System (Curies)</u>	<u>Filter Removal Efficiency (%)</u>	<u>Fission Products Released Through Filters (Curies)</u>
Noble Gases		0		0	
Xe ¹³³	4.7×10^3		4.7×10^3		4.7×10^3
Xe ^{135m}	1.2×10^3		1.2×10^3		1.2×10^3
Xe ¹³⁵	4.7×10^3		4.7×10^3		4.7×10^3
All Noble Gases	3.9×10^4		3.9×10^4		3.9×10^4
Halogens		0.5		95	
I ¹³¹	2.5×10^3		1.3×10^3		65
I ¹³²	3.4×10^3		1.7×10^3		85
I ¹³³	3.5×10^3		1.8×10^3		90
I ¹³⁵	4.6×10^3		2.3×10^3		115
All Halogens	3.2×10^4		1.6×10^4		8.0×10^2
Volatile Solids		0.5		99.95	
Te ¹²⁷	2.4×10^2		1.2×10^2		<1
Te ^{129m}	4.7×10^2		2.4×10^2		<1
Te ¹²⁹	6.1×10^2		3.1×10^2		<1
Te ¹³²	3.4×10^3		1.7×10^3		<1
Cs ¹³⁷	2.4×10^3		1.2×10^3		<1
All Volatile Solids	5.1×10^4		2.6×10^4		12.8
Non-Volatile Solids		0.01		99.95	
Sr ⁸⁹	8.6×10^2		8.6		<1
Sr ⁹⁰	10.3		<1		<1
Zr ⁹⁵	2.6×10^3		26		<1
Ba ¹⁴⁰	3.6×10^3		36		<1
Ce ¹⁴⁴	5.5×10^2		5.5		<1
All Non-Volatile Solids	2.2×10^5		2.2×10^3		1.1
Total All Fission Products	3.4×10^5		8.4×10^4		3.9×10^4
Pu ²³⁹	1.4	0.01	1.4×10^{-2}	99.95	7×10^{-6}
Pu ²⁴⁰	2.0	0.01	2.0×10^{-2}	99.95	1×10^{-5}
Pu ²⁴¹	80	0.01	0.8	99.95	4×10^{-4}
Pu ²⁴²	2×10^{-4}	0.01	2×10^{-6}	99.95	1×10^{-9}

Fuel attained 8 Mwd exposure in 100 days, no decay.

* Computed by RIBD and MOFDA Computer Codes - 12-21-64.

IV. ACKNOWLEDGEMENT

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V. APPENDICESA. Burnout Heat Flux

The burnout heat flux is estimated using the correlations of Tong, Currin, and Thorp as modified for annular data.^(7,8) The basic correlation in the subcooled region, presented in the May, 1963 issue of Nucleonics, is as follows with the nomenclature provided in Table 9:

$$q''_{BO} = (0.23 \times 10^6 + 0.094G) \times (3.0 + 0.01 \Delta T_{sc}) \times (0.435 + 1.23e^{-0.0093L/De}) \times (1.7 - 1.4e^{-a}) \quad (1)$$

$$\text{where: } a = 0.532 [(H_f - H_{in})/H_{fg}]^{3/4} (\rho_g/\rho_f)^{-1/3}$$

This equation applies to circular tube, rectangular channel, and rod bundle geometries whereas the C-1 Loop fuel assembly is an annular geometry with a single-heated surface. Tong and Currin suggest the following modifications to Equation (1) for annular geometries:⁽⁶⁾

1. The equivalent diameter, De , is replaced by a term designated as Dh and defines as: $Dh = 4Ac/Ph$.
2. To account for the postulation that the overall enthalpy rise of the fluid flowing near the heated surface will be much greater than the bulk enthalpy rise for annular geometries, the following ratio, R , is defined: $R = Ph/Ph+Pc$.

When evaluating q''_{BO} , R is used as follows:

$$q''_{BO} = \left[\frac{R}{(1.11)(0.484 + 0.697e^{-0.0109 L/De})} \right] (q''_{BO} \text{ using } Dh \text{ in Equation 1}) \quad (2)$$

3. With these modifications the correlation predicts about 10 per cent too high. Therefore, an added empirical factor of 0.9 is suggested providing a final equation of:

$$q''_{BO} = 0.9 \times \left[\frac{R}{1.11(0.484 + 0.697e^{-0.0109 L/De})} \right] \times (q''_{BO} \text{ using } Dh \text{ in Equation 1}) \quad (3)$$

The authors report that less than five per cent of the experimental data points fell below the -20 per cent deviation line.⁽⁶⁾

* The authors use the nomenclature "departure from nucleate boiling", or q''_{DNB} , in their paper. To be consistent with the terminology used in this report, this is changed to "burnout" or q''_{BO} .

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The ranges of parameters of the correlated experimental data are compared to C-1 Loop parameters as follows:

	<u>Experimental Data</u>	<u>C-1 Loop</u>
Geometry	Annular	Annular
Axial heat-flux distribution	Uniform	Cosine
Mass velocity, lb/hr ft ²	$0.2 \times 10^6 - 8.0 \times 10^6$	3.88×10^6
Pressure, psia	800 - 2750	2290
L/De	21 - 365	188
Inlet subcooling ($H_f - H_{in}$), BTU/lb	0 - 700	174
Subcooling at burnout, °F	0 - 228	60 - 83
Local heat flux, BTU/hr ft ²	$0.4 \times 10^6 - 4.0 \times 10^6$	$0.25 \times 10^6 - 0.87 \times 10^6$
Equivalent diameter, in.	0.1 - 0.54	0.17

The C-1 Loop fuel assembly heat flux is limited to prevent local film burnout at the fuel surface by placing a minimum of 1.4 on the burnout safety factor (BOSF). The BOSF is defined as:

$$\frac{(\text{hot spot factor})(\text{calculated burnout heat flux})}{(\text{operating heat flux})}$$

The hot spot factor takes into account nonidealities in the flux distribution, errors in flow, pressure, and temperature measurements, and deviations in fuel composition from the nominal value.

The burnout heat flux is calculated from measured temperatures, flow rates, and pressures. The fuel assembly flow rate is determined by averaging measurements from three independent flow monitors located near the fuel assembly inlet. The accuracy of this measurement is estimated to be ± 3 per cent. The coolant temperature at the fuel assembly outlet is measured by three independent monitors with an estimated accuracy of ± 3 F. The saturation pressure at the outlet is determined from outlet pressure measurements and is estimated to be within ± 1 F of actual value. The fuel assembly power will vary with the Pu composition in the fuel. The specification is 8 ± 0.5 weight per cent. A plot of heat flux vs. weight per cent Pu in the fuel is shown in Figure 9. Finally, Reference (3) reports that the local neutron flux in the C Reactor could be 1.2 times normal values due to operating variations. These factors are multiplied to provide a C-1 hot spot factor as follows:

<u>Variable</u>	<u>Hot Spot Factor</u>
Neutron Flux	0.83
Pu Composition in fuel	0.97
Flow measurement	0.97
Subcooling measurement	<u>0.95</u>
Total Product	0.74

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FIGURE 9
RELATIVE POWER Vs. PER CENT
PLUTONIUM IN FUEL

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The limiting heat flux is determined using this factor as follows:

$$\text{BOSF} = \frac{(0.74)(\text{calculated burnout heat flux})}{(\text{operating heat flux})} > 1.4$$

The factor of 1.4 takes into account uncertainties in the calculated value of burnout heat flux.

References (5&6) also present a burnout correlation in the quality region which was used to evaluate the heat exchanger accident described in Section III.C.3.c. The reader is referred to the reference documents for the details of this correlation.

TABLE 9

NOMENCLATURE FOR BURNOUT CORRELATION

q''_{BO}	=	heat flux at burnout (BTU/hr ft ²)
G	=	Local mass velocity (lb/hr ft ²)
Δt_{sc}	=	Subcooling = $T_{\text{sat}} - T_{\text{local}}$ (F)
L	=	Length of heated channel (ft)
D_e	=	Equivalent diameter (ft)
H_f	=	Saturated liquid enthalpy (BTU/lb)
H_{in}	=	Inlet enthalpy (BTU/lb)
H_{fg}	=	Heat of evaporation (BTU/lb)
ρ_f	=	Density of saturated vapor (lb/ft ³)
ρ_g	=	Density of saturated vapor (lb/ft ³)
A_c	=	Flow area (ft ²)
P_h	=	Perimeter of heated surface (ft)
P_c	=	Perimeter of unheated surface (ft)

B. Compliance with Engineering Codes

The C-1 Loop facility was designed and fabricated in strict compliance with ASA Code B.31.1, "Code for Pressure Piping" and the ASME Unfired Pressure Vessel Code with two exceptions and these have been documented as code variances in Reference (11), as required by IPD, OPG 33.46. The design features not approved by the codes were: 1) the use of 3/8-inch instrument

11. T. L. Deobald. C-1 Loop Code Variance, HW-72958. April 20, 1962. (Unclassified).

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takeoff connections, and 2) the use of flareless tubing fittings in the C-1 instrument piping. Evaluation of the design and other plant operating experience shows that there will be no reduction in personnel safety as a result of these code variances. An independent engineering audit by Facilities Engineering⁽²⁾, and an inspection by a third party inspector has also shown the facility to be in compliance with applicable codes. The entire facility was pressure tested at 4175 psi or 165 per cent of design pressure. Operation of the facility is well within the confines of the design rating (2530 psi at 650 F).

Information required to show compliance with engineering codes consisting of purchase requisitions, material certifications, material location records, welding procedures, reports of third party inspection, pressure piping completion reports, X-ray films and their analysis, correspondence and other related information are on record in Irradiation Testing files. Reference (12) includes the calculations related to allowable pressures.

The in-reactor pressure shell has been subjected to additional non-destructive testing over and above that required by the codes in light of the potential seriousness of failure of the pressure shell as discussed in Section III.C.3.c. The following additional tests and examinations were performed.

1. 100 per cent longitudinal and transverse ultrasonic testing to less than three per cent wall thickness defect.
2. 100 per cent Viddigauge testing to insure the required wall thickness.
3. 100 per cent external and internal white light examination.
4. 100 per cent external and internal dye-penetrant examination for hidden defects.

The results of these tests were compared to standards established for N Reactor pressure tubes, and the tube was found acceptable.

C. Calculation of Maximum Credible Accident

The maximum credible accident begins when the inlet piping common to both coolant systems suddenly breaks so that the coolant flow to the test section completely stops. The pressure of the coolant in the test section is assumed to fall to the saturation pressure at the loop coolant temperature (~900 psi) and a reactor scram initiated by C-1 instrumentation within one second thereafter. The water remaining in the test section is assumed to boil-out in two seconds. Fuel assembly power is assumed to be 105 per cent of the initial value during this two second interval due to the

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12. R. K. Marshall. C-1 Loop Code Compliance Philosophy and Calculations, HW-70525. July 24, 1961. (Unclassified).

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increased reactivity caused by loss of water. Sufficient cooling to prevent a temperature excursion is assumed available during this period due to the high coolant velocities during the blowdown period. The reactor scram begins to reduce the heat generation rate in 2.5 seconds.

In computing the outcome of the accident, no heat loss was assumed from two seconds after initiation of the event until after the fuel reached temperatures which would provide sufficient driving force for heat loss through radiation to the graphite, which acts as the sink in this case. This would not occur until after the fuel melted since its melting temperature is very near the operating temperature of graphite. The assumptions, thermal properties, and equations used in calculating the course of this accident are summarized as follows:

1. The thermal properties of aluminum were assumed for the Pu-Al fuel cores. The temperature dependence of specific heat and density for aluminum were obtained from curves in Reference (13). The heat of fusion for aluminum included in the calculations was obtained from the Metals Handbook as 170 BTU/lb.
2. The thermal properties of the stainless steel fuel housing and pressure shell were assumed constant with temperature as follows:
 - a. Thermal conductivity - $12.4 \text{ BTU/ft}^2/\text{hr}/^\circ\text{F/ft}$.
 - b. Specific heat - 0.12 BTU/lb F .
 - c. Density - 0.29 lb/in^3 .
 - d. Melting range - $2550 - 2590 \text{ F}$.
3. The graphite temperature remains constant throughout the transient at 1200 F .
4. Once the test section has boiled dry, the temperature of the fuel assembly will increase due to fission product decay heat. Upper estimates of the temperatures that might be reached were made with the assumption that the only mechanism for heat loss is radiation. Because the fuel assembly melting temperature is nearly the same as the graphite sink temperature, no radiation heat loss occurs until after the fuel assembly melts and is in intimate contact with the fuel housing. Following this, heat radiates from the fuel housing to the pressure shell which in turned radiates to the graphite. The radiation heat transfer was determined by first calculating a radiation heat transfer coefficient,

$$h_{\text{rad}} = 4\sigma T^3 \left[\frac{1}{\frac{1}{E_1} + D_1/D_2 (1/E_2 - 1)} \right]$$

13. F. J. Teho. Selected Values of the Physical Properties of Various Materials, ANL-5914. September, 1958. (Unclassified).

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where: σ = Stefan-Boltzman constant
 T = Average temperature between radiating surfaces, R
 D_1 = ID annulus
 D_2 = OD annulus
 E_1 = Emissivity of the D_1 surface
 E_2 = Emissivity of D_2 surface

The resistance of the radiation component is then:

$$R_{\text{rad}} = \frac{1}{h_{\text{rad}} \pi D_1}$$

The heat transfer rate is then computed by the equation:

$$q = \Delta T/R$$

Emissivities of 0.9 for the graphite and 0.3 for the stainless steel surfaces were assumed in determining the rates of heat loss.

Using the heat decay curve (Figure 6), it was determined that the fuel assembly would melt 50 seconds after the piping break occurred. The maximum temperature of the melt would be reached in about 2.7 minutes and would be about 2100 F. This is below the melting point of the stainless steel fuel housing. The temperature of the pressure shell would reach about 1580 F.

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