UNIVERSITY OF MISSOURI, COLUMBIA
MISSOURI UNIVERSITY RESEARCH REACTOR
LICENSE NO. R-103
DOCKET NO. 50-186

HAZARDS SUMMARY REPORT
ADDENDUMS 1-5
IN SUPPORT OF AN APPLICATION
FOR A RESEARCH REACTOR

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUR OR DENOTED BY BRACKETS

University of Missouri
Research Reactor Facility

Hazards Summary Report

Addendums 1 through 5

ADDENDUM NO. 1
HAZARDS SUMMARY REPORT
University of Missouri
Research Reactor Facility

Compiled and Edited by
The Staff
Research Reactor Facility

Submitted by
The University of Missouri
Columbia, Missouri

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1.0 Introduction

The purpose of this document is to make certain revisions to and answer questions resulting from the University of Missouri's application for a Class 104 utilization facility license, submitted to the Commission in July of 1965.

The revisions are concerned mainly with the total inventory of Special Nuclear Material which the University may possess and use at any one time. Specifically, the University wishes to revise the total quantity of Uranium-235 and Plutonium-239 to be included in the facility license.

This document also presents answers to and discussion of questions presented to us as a request for additional information necessary for the safety evaluation of the proposed operation of our research reactor. These questions were sent to us with a letter from Dr. R. L. Doan dated January 5, 1966.

2.0 Revisions to Original Application

In our original application, dated July 1, 1965 and directed to Dr. Richard Doan we stated that our initial fuel inventory would consist of eight fuel assemblies each containing 650 grams of U-235 and four fuel assemblies containing 350 grams of U-235. We wish to amend this quantity to include one additional 650 gram assembly. This will bring our initial U-235 inventory, including neutron detectors, to approximately 7,252 grams.

We would also like to add to our request, authorization to receive, possess and use up to 80 grams of plutonium 239 as sealed plutonium-beryllium neutron sources.

When we purchase our second core we will buy 9 rather than 8 assemblies each containing 775 grams of U-235. Eight fuel assemblies will constitute a core, the ninth assembly will be held as a spare. Table 2.1 lists our anticipated maximum inventory of Special Nuclear Material.

Table 2.1

Maximum Special Nuclear Material Inventory

I.	Uranium-235			
a)	Initial inventory:			
	8 assemblies @ 650 grams each		5,200	gms
	1 spare assembly @ 650 grams		650	
	4 assemblies @ 350 grams each		1,400	
		Total	7,250	gms
b)	Burnup (assume 5%)		(260)	gms
c)	Inventory at end of core life		6,900	gms
d)	New core:			
	8 assemblies @ 775 grams each		6,200	gms
	1 spare assembly		775	
		Total	6,975	gms
e)	Neutron detector		2	gms
f)	Total maximum inventory			
	(a) + (d) + (e) - estimated burnup		13,967	gms
II.	Plutonium-239			
	Sealed Pu-Be sources		80	gms

It should be emphasized that the inventory shown in Table 2.1 represents a maximum. Since a core is exactly 8 fuel assemblies, four of the assemblies listed in the initial inventory will probably not be used and will eventually be returned for reprocessing. In this case our total inventory at the time of receipt of the second core would be reduced by an amount equal to the assemblies already returned.

Also Table 2.1 assumes that the second core will be sized for 10 MW operation. If the modification to the reactor plant required for operation at 10 MW has not yet been completed at the time the second core is procured this new core will also be approximately 5,200 grams.

Reference is made on page 13-24 of the Hazards Summary Report to a 150 psi rupture diaphragm. Recent design changes are incorporating pressure relief valves instead of the rupture diaphragm.

Paragraph 1.2.6 on page 1-5 is to be revised to read, the University of Missouri Research Reactor Facility will be staffed and operated by the University.

3.0 Answers to Questions

In the following paragraphs, each of the 23 questions presented to us will be answered in order. The answer to each question is complete with supporting discussion and illustrations. However, some detailed analysis has been relegated to appendices to avoid making the discussion unnecessarily cumbersome.

3.1 "Provide the assembly drawings of the control rods and control rod drive mechanisms and design drawings showing overall blade dimensions and clearance in the guide structure. Describe the preoperational tests and periodic tests to be performed on the control rod drive system. Discuss the possibility of control rod binding due to thermal distortion of the control blades."

Ten copies each of the General Electric drawings

104B2530 - Regulating Blade
107C4660 - Control Blade
789D880 - Blade Offset Mechanism (Control)
789D884 - Blade Offset Mechanism (Reg.)

are included with this submittal.

The control blades operate in a gap between the outside of the reactor pressure vessel and the inside of the beryllium reflector. The O.D. of the pressure vessel is between 12.546 and 12.566 inches. The I.D. of the beryllium reflector is between 13.680 and 13.699 inches. The gap width is maintained by vertical spacers which are set into the beryllium reflector and cross the gap to the O.D. of the reactor pressure vessel.

The minimum width of the gap in which the rods operate is (13.680 - 12.566) / 2 = .557 inches. The thickness of the control blades is 0.25 inches, leaving a gap of approximately 0.15 inches on each side of the blade.

In order for blade binding due to thermal distortion to occur, some point on a blade must be displaced 0.15 inches with respect to the constrained (upper) end. The most probable way that this amount of displacement could occur would be as a result of differential expansion between the inner and outer longitudinal fibers accompanying a radial temperature gradient through the blade.

Assuming that the inner radius of a blade is at a higher temperature than the outer radius, a calculation was made to determine the differential expansion needed to displace the unconstrained (bottom) end of a control blade 0.15 inches outward, the result was 2.89×10^{-3} inches.

The coefficient of linear expansion of aluminum is 13.0×10^{-6} inches per inch degree fahrenheit. The unconstrained length of a control blade is approximately 26 inches. The differential expansion in inches per inch is:

$2.89 \times 10^{-3} / 26 = 1.11 \times 10^{-4}$

The corresponding temperature change is:

$1.11 \times 10^{-4} / 1.3 \times 10^{-5} = 8.6$ degrees fahrenheit.

The above result implies that the control blade must support a temperature gradient of 8.6 degrees. This temperature difference would require a heat flow radially through the blade of almost 860 BTU/ft^2/hr, which is extremely high in view of the fact that the major heat source in the blades is from the absorption of gamma rays and neutrons.

Additionally, temperature gradients will be suppressed by cooling water which enters the rod gap at 100°F and passes on both sides of the blades. It appears extremely unlikely that control rod binding due to thermal distortion of the control blades will be experienced.

The control rod drive system will receive a thorough inspection in addition to the rod drop checks. During the first year of operation this inspection will be performed twice at approximately six month intervals. The inspection will include a determination of any tendency that each blade is not properly aligned in its respective slot and that adequate clearance exists between the control blade and either the beryllium reflector or the reactor pressure vessel. Currently each blade is checked once every two years with one blade done every six months.

"Rod drop times" will be measured at approximately 90 day intervals. If these tests show any evidence of the rod sticking, immediate corrective action will be taken.

Prior to initial startup, the electrical systems of the control rod drive system will be thoroughly tested as part of the safety system checkout discussed in Section 3.3. In addition to these tests, the control rods will be scrammed 25 times. Rod drop time will be measured for the first and last five scrams on each rod to determine normal statistical variations.

3.2 "Provide the manufacturing tolerances for distances between fuel plates, fuel meat and cladding thickness, fuel inhomogeneity and clad non-bond. Summarize the acceptance checks and tests for fuel subassemblies."

The answer to this question, as presented in the following paragraphs, was excerpted from the fuel procurement specifications.

Fuel plate dimensions and tolerances are:

1.	Cladding Thickness	.015 inches ± .003
2.	Fuel Plate Thickness	.050 inches ± .002
3.	Distance Between Fuel Plates	.080 inches ± .002
4.	Fuel Meat Thickness	.020 inches

Criteria for acceptance are:

1. General
   The completed fuel assemblies shall be inspected for width, straightness, twist, etc., in accordance with the requirements of the contractor's drawings.

2. Cladding Thickness.
   Cladding thickness shall be verified by means of photomicrographs. Photomicrographs shall be taken of one sectioned fuel plate from each melt, each one sectioned transversely at a different location than on a previous plate.

3. Cladding Bond
   The bond between the cladding and the fuel filler shall be verified for each fuel plate by a short time "blister test." This test shall consist of heating the fuel plates to 940°F and holding this temperature for a period of thirty minutes. This test shall be conducted prior to final rolling of the plate. A visual inspection shall be performed on the plate immediately upon removal from the furnace. Fuel plates which exhibit raised or blistered areas shall be rejected.

4. Fuel Filler Location Test

   Each fuel plate shall be given a fluoroscopic examination or radiographed prior to final shearing to establish outline of filler material. The plate shall be sheared such that the fuel filler is centered in the finished plate.

5. Uranium Homogeneity

   A standard radiograph for acceptance of uranium homogeneity shall be established prior to production. One radiograph shall be made of a fuel plate from each melt and compared to the standard. In the event of rejection of one plate from a melt, five additional fuel plates from the same melt shall be radiographed and evaluated. If all of these additional plates are acceptable, then the remainder of these plates are then acceptable. If any of the five additional plates are rejected, each plate from the heat shall be evaluated and accepted or rejected on an individual basis. The radiographs shall be submitted to the buyer after all testing has been completed.

6. Scratches and Marks

   Scratches, pits or other marks on the heat transfer surfaces shall be cause for rejection if they exceed .003 inches in depth. Dents with depth exceeding .012 inches shall be cause for rejection.

7. Mechanical Bond

   The joint design shall be proven with a pull test to determine loading and type of failure.

3.3 "Provide the schematic diagram(s) for the safety system and the process instrument electric control and process control system diagrams and discuss the program for initial and periodic testing of these systems."

Ten copies each of General Electric Drawings

212E760 - Safety System

104R784 - Process Instrumentation and Control Interlock

237E589 - Electric Control

are enclosed with this submittal.

The initial testing of the safety system and process control system consists of the completion of detailed check list of circuit parameters. The performance of each unit is checked using precision external test equipment.

The 34 page document which describes the detailed procedures for initial inspection of the complete safety system is too voluminous to include with this report. The main subject headings are listed below to indicate the depth and detail of the inspection.

Panel Power
Source Range Monitor and Scaler
Intermediate Range Monitors
Wide Range Monitor
Power Range Monitors
Control Rod Drive
Safety System
Rod Run-In System
Regulating Blade and Servo Operation
Startup Counter Drive
Discriminator and High Voltage Settings

As an indication of the detail of the initial test procedures, the operational checkout procedure for the process instrumentation and interlock system has been reproduced and included as Appendix I.

The calibration and checkout procedures will be repeated in detail approximately every six months. A special procedure for the safety system is presented below.

Safety System - Special Tests

I. Visual inspection for loose connections, overheated components, and corroded joints.

II. Circuit Adjustments

Trip Actuator Amplifier

The adjustments listed below will be made when the Trip Actuator Amplifier is initially installed, checked during the periodic maintenance periods and after any component has been replaced.

a) Connect a 24-ohm, 50 watt resistor to connector pins 6 and 7. Connect a 100-ohm, 50 watt resistor to connector pins 7 and 15.

b) Energize the power supply by applying 115 volts a-c at connector pins 1 and 13.

c) Connect 12 volts d-c to input connector pins 4 and 17, ground pin 5.

d) Reset the unit by momentarily jumpering between connector pins 2 and 3, and connector pins 2 and 20.

e) Determine that the load current is 1.0 ampere. If it is not, adjust R-38 for 1 ampere.

f) Reduce the input voltage signal to Input No. 1 to the selected trip point value, and observe that the output current decreases to less than 0.1 ampere in the tripped condition.

g) Adjust potentiometer R1 to cause the circuit to trip at the predetermined voltage level for Input No. 1, if the circuit did not trip in Step f.

h) After this adjustment has been made, lock the potentiometer shaft by tightening the lock nut.

i) Repeat Step f to check trip point.

j) Repeat Step f, g, h, and i for Input No. 2, using potentiometer R21 in place of potentiometer R1.

Because of the redundancy feature transistors Q5 and Q10 will be removed and tested.

A defective power transistor Q5 and/or Q10 can be checked routinely by monitoring the output current of the Trip Actuator Amplifier with the circuit tripped. If this value exceeds 0.1 ampere the output power transistor, Q5 (or Q10) may be defective.

Non-coincidence Logic Unit

Because of the redundancy features incorporated in the Non-coincidence Logic Unit, certain malfunctions will not affect its operation. To ensure that the redundancy features are maintained the unit will be inspected, and redundancy circuits checked every six months.

Check for Diodes CR1 through CR18

a) Connect 24 volts d-c operating power to the unit.
b) Disconnect all input signals from unit.
c) Apply 24 volts d-c to input terminal E1.
d) Place jumper across diode CR10.
e) Measure voltage across diode CR1. If diode is blocking correctly, voltage should be approximately 24 volts. If there is not a voltage or much less than 12 volts, replace the diode.
f) Remove the jumper from diode CR10 and place jumper across diode CR1.
g) Repeat Step c), reading voltage across diode CR10.
h) Remove jumper from diode CR1.
i) Disconnect 24 volts d-c from input terminal E1 and connect 24 vdc to input terminal E2.
j) Repeat Steps d) through i) to check diodes CR2 through CR18.

Check for Failed Redundant Transistors

a) Connect 24 volts d-c operating power to unit.
b) Connect 24 volts d-c signal to all inputs.
c) Measure the output voltage at E11. This voltage should be approximately 16 volts.
d) Apply 12 volts d-c to the cathode of diode CR19 in the base circuit of transistor Q1 to prevent Q1 and Q2 from turning off.

e) Remove all input signals. The voltage at E11 should drop to less than 1 volt. If the voltage does not drop, check transistors Q3 and Q4 for malfunction.

f) To check transistors Q1 and Q2, disconnect the 12 volts d-c from CR19 and connect the 12 volts d-c to the cathode of CR21. Check that the output voltage at E11 is less than 1 volt.

3.4 "Evaluate the possibility of: (a) a single short across one of the relay matrices nullifying certain process instrument scram functions; (b) loss of manual scram capability owing to failures in the safety system or manual scram circuitry; and (c) loss of certain safety functions if the scram reset relay were stuck closed or shorted."

This topic was reevaluated at 10 MW upgrade (see Hazards Summary Report, Addendum 4, Appendix A).

(a) A single short across one of the relay matrices will not nullify any process scram functions. It will be noted on drawing 212E760 that the input E6 to the non-coincidence logic units appears to include all flow scrams. However, it will be noted that the relay matrix through which the input E7 to the non-coincidence logic units is made up includes contacts on relay 1K13. Relay 1K13 de-energizes, opening the scram input to E7 whenever any of the following conditions exist:

1. Low reactor loop pressure
2. High reactor loop temperature
3. Low reactor loop flow

and the system is set up to operate above 100 KW. Therefore, the conditions of low pressure or low flow in the reactor loop will cause a scram input to both E6 and E7 through two independent relay matrices. Although pool loop low flow scram contacts do not appear in both input circuits (E6 and E7), this condition is annunciated at 90% of normal.

(b) As presently shown on drawing 212 E760, the manual scram contact is shown in the circuit to non-coincidence logic unit input E7. This contact has been moved to between terminals TB2-2 and EE-17 (drawing coordinates C-13) in the 115 volt supply to the trip actuator amplifiers. Therefore, a failure in the safety system will not result in a loss of manual scram capability. In the event of a failure in the manual scram circuitry, a scram may be easily and quickly initiated by actuating switch 1S1 or switch 1S14.

(c) If both the scram reset contacts and contacts 7 and 8 of relay 2K16 become shorted, the trip actuator amplifier will still respond to a scram signal.

When relay 2K16, contacts 7 and 8, are closed and the scram reset switch is closed, Q2 and Q7 of the trip actuator amplifier are forward biased. Then the +16 volts from the non-coincidence logic unit enters the trip actuator amplifier causing Q1 and Q6 to be forward biased, bringing the trip actuator amplifier into conduction and supplying magnet current to the control rods.

If, with contacts 2K16-7 and 8 plus the scram reset switch shorted, the input from the non-coincidence logic unit becomes zero (the scram condition), Q2 and Q7 are forward biased. However, Q1 and Q6 become reverse biased which, through normal operation of the remainder of the circuits, interrupts magnet current, allowing the control rods to drop.

3.5 "Describe the tests that have been performed and the in situ tests that will be performed to verify safety system response to variations in supply voltage and frequency and to the extremes of expected ambient temperature conditions. Describe the nature and frequency of tests that will be performed to detect possible component or module failures."

The nature and frequency of tests that will be performed to detect possible component or module failures was discussed in Section 3.3.

Tests to determine the response of the control system to variations in supply voltage and frequency and to extremes of ambient temperature have been performed by the manufacturer of the system, the General Electric Company, Nuclear Electronics Products Section of APED. Amplifier response time and calibration, bi-stable amplifier response, and trip values, of the complete system were tested and were shown to remain within tolerance over these conditions:

Supply Voltage	115 volts ± 10%
Supply Frequency	60 cycles ± 5%
Temperature	32°F to 120°F

Certified results of in-plant tests are available from General Electric for the system installed at Columbia, Missouri, or for similar systems installed elsewhere.

The response of a control system to variations in voltage, frequency, and temperature depends entirely on circuit design and the quality of components used in manufacture. Since these factors are not changed in transporting equipment from one point to another and since the equipment was thoroughly tested prior to shipment, it has been concluded that a repetition of environmental tests here would be of no value.

3.6 "Describe the method for insuring redundancy in detectors which initiate building isolation."

Four independent detectors are used to initiate containment building isolation. Two detectors are located in the exhaust plenum on the fifth level of the reactor containment building, and two are located near the reactor pool surface. If any one of these detectors senses a radiation level above the trip setpoint, a reactor scram and isolation is automatically initiated.

The air plenum detectors are isolated from each other in that the operation of one does not affect the operation of the other. The pool surface detectors are also isolated from each other, but have different trip setpoints. The detector with the lower trip setpoint can be temporarily cutout to avoid an inadvertent scram and isolation during controlled experimental transfers or minor maintenance in the reactor pool area. The pool surface detector with the higher setpoint always remains in continuous operation.

These detectors are powered by independent low voltage power supplies and are, therefore, not subject to a single failure event. The power supplies are monitored by an alarm circuit which alerts the reactor operator in the event power to a detector is lost. The resulting loss of signal from the detector will initiate a reactor isolation and scram.

Power to the area radiation monitoring system is routed from the uninterruptible power supply (UPS) and the emergency power distribution system, which will provide continuous power in the event of a loss of normal site power.

3.7 "Provide an evaluation justifying the acceptability of the 125% of power level scram setting."

The underlying reason for providing a high power scram function is to afford protection to the reactor core and associated equipment. It is therefore important to determine that the scram level settings intended for the MURR will indeed suffice to provide this necessary protection. It will be shown that a scram level setting of 125% of nominal power (5 MW) will be more than adequate to provide safe shutdown with no burn-out in the event of overpowered operation.

The method used in arriving at the conclusions contained herein will now be given. A literature search was made to find burn-out correlations applicable to the MURR. Using these correlations, steady state burn-out heat fluxes were then computed for the MURR conditions of operation at 5 MW. Since there was much variation between the various computed burn-out heat fluxes, ϕ_{DNB} the most conservative, i.e., the smallest ϕ_{DNB} was taken. The average heat flux for 5 MW operation was then multiplied by the appropriate hot channel and hot spot factors to obtain the peak heat flux for a nominal 5 MW power with ± 10% power uncertainty. Finally, the peak 5 MW heat flux was divided into the burn-out heat flux to give the maximum nominal operating power multiplier for no burn-out. It was found that the burn-out power level thus derived was much in excess of the 25% over-power scram trip setting. In this manner, the 125% scram trip setting was justified.

In obtaining a burn-out heat flux, ϕ_{DNB}, various correlations were tried using the MURR operating conditions at a nominal 5 MW power. A summary of these calculations is presented in Table 3.7.1. Details of the burn-out heat flux calculations are presented in Appendix II. Although it was felt that the Gunther correlation provided the most realistic ϕ_{DNB}, it was decided that the Bernath correlation should be used as a much more conservative estimate. Thus, for the remainder of this report, ϕ_{DNB} is to be evaluated as 1.975×10^6 BTU/ft^2-hr, the Bernath value.

Table 3.7.1 Burn-out Correlation for 5 MW Conditions

Correlation	ϕ_{DNB}(BTU/ft2-hr)
1. Lowdermilk et al.[1]	2.543×10^6
2. Gunther[2]	3.521×10^6
3. Schrock[3]	5.237×10^6
4. Bernath[5]	1.975×10^6

The calculations of the hot channel-hot spot heat flux proceeded as follows:

5 ?

$$\phi_{(SMW)} = R\, P_r\, P_a\, \bar{\phi}_{(10\ MW)}$$

where

$\phi_{(SMW)}$ = hot spot heat flux (BTU/ft2-hr),

$\bar{\phi}_{(10\ MW)}$ = average core heat flux at 10 MW nominal power, ← why

why 10 MW value? why not 5 MW?

R = reference multiplier for 5 MW, 10% uncertainty with worst case non-uniform loading,

P_r = average hot channel power density/average core power density

P_a = peak hot channel/average hot channel power density

These were values evaluated according to TM-WRP 62-10.[4]

$\bar{\phi}_{(10\ MW)}$ = 172,244 BTU/ft2-hr for 93% heat

R = 0.784, for 5.5 MW and worst case non-uniform loading

P_r = 2.263 ↑ (10% uncertainty)

P_a = 1.443

These parameters resulted in a peak hot spot heat flux, $\phi_{(5MW)}$, of 4.215×10^5 BTU/ft^2-hr for 5 MW worst case loading and 10% uncertainty in power measurement.

Worst case non-uniform loading refers to a disadvantageous mixing of lightly and heavily loaded fuel elements. Internuclear[6] found that the worst case occurs when one heavy (6.5/8 Kg-U) element is loaded with seven light (3.5/8 Kg-U) elements in which case the peak to average heat flux multiplier is 1.425.

The maximum steady state operating power for no burn-out was then computed using the following relationship:

$$\text{Power} = 5\,\frac{(\phi_{DNB})}{(\phi_{(5MW)})}$$

$$\text{Power} = 5\,\frac{(1.975 \times 10^{6})}{(4.215 \times 10^{5})} = 23.4\ \text{MW}$$

Hence it is to be anticipated that burn-out might occur when a steady state power of 23.4 MW is exceeded, but not for lower power levels.

The hot channel discussed above includes two fuel plates nearest the island and extending around the full circumference. The hot spot was calculated by Internuclear Corporation to be about 18 inches down from the top of the core.

As a result of these calculations, it appears that a power level of 6.89 MW (125% of 5.5 MW) will result in no damage to the core nor will it compromise reactor safety.

References

1. W. H. Lowdermilk, C. D. Lanzo, B. L. Siegel, "Investigation of Boiling Burn-out and Flow Stability for Water Flowing in Tubes," NACA TN-4382 (September 1958).
2. F. C. Gunther, "Photographic Study of Surface Boiling Heat Transfer with Forced Convection," Trans. ASME, Vol. 73 (1951), pp. 115-123.
3. H. A. Johnson, V. E. Schrock, S. Fabic, F. B. Selph, "Transient Boiling Heat Transfer and Void Volume Production in Channel Flow," SAN-1007 (March 1963).
4. W. R. Pearce, "Core Heat Transfer and Fluid Flow--Preliminary Working Curves," Internuclear TM-WRP 62-10 (June 1962).
5. H. Etherington, "Nuclear Engineering Handbook," (Bernath Correlation) (1958), pp. 9-78.
6. Shapiro, D. M., Kim Y. S., and Putnam, G. E., "Reactor Physics Analysis for the University of Missouri Research Reactor," Internuclear TM-DMS-62-5 (1962).

3.8 "List the loads on the emergency power supply. Discuss the type of periodic tests performed to assure reliable performance."

The emergency power system is driven by a water cooled Cummins, six cylinder, turbocharged diesel engine. It is provided with a 270 gallon skid mounted diesel fuel storage tank and a mechanically driven fuel injection system. It is capable of assuming full load from a cold start in seven seconds. A 24 volt, nickel-cadmium storage battery is used for the EG starting system.

Attached to the diesel engine is a four pole generator equipped with a brushless permanent magnet exciter. It produces 60 cycle, 277/480 volt, 3 phase power, and has a continuous standby capacity of 275 kW. The design of the exciter and regulator provides for voltage regulation of better than plus or minus 2%. Stable generator output voltage and frequency are established within two seconds after the transition between no load and full load conditions.

The automatic transfer switch (ATS) is equipped with an adjustable .5 to 3 second delay on starting, preventing plant operation on instantaneous line failures, and an adjustable 0 to 25 minute delay on retransfer to commercial power. Incorporated in the unit is a static type dual rate float/equalizer charger with automatic and manual charge control to maintain the startup battery fully charged.

The emergency bus is routed through the automatic transfer switch to an Emergency Distribution Panel (CTR-1) shown on Figure 3.8.1. This distribution panel feeds the following emergency electrical loads:

1) Exhaust Fan EF-13
2) Exhaust Fan EF-14
3) Diesel Room Distribution Panel which provides control power for the EG room ventilation system.

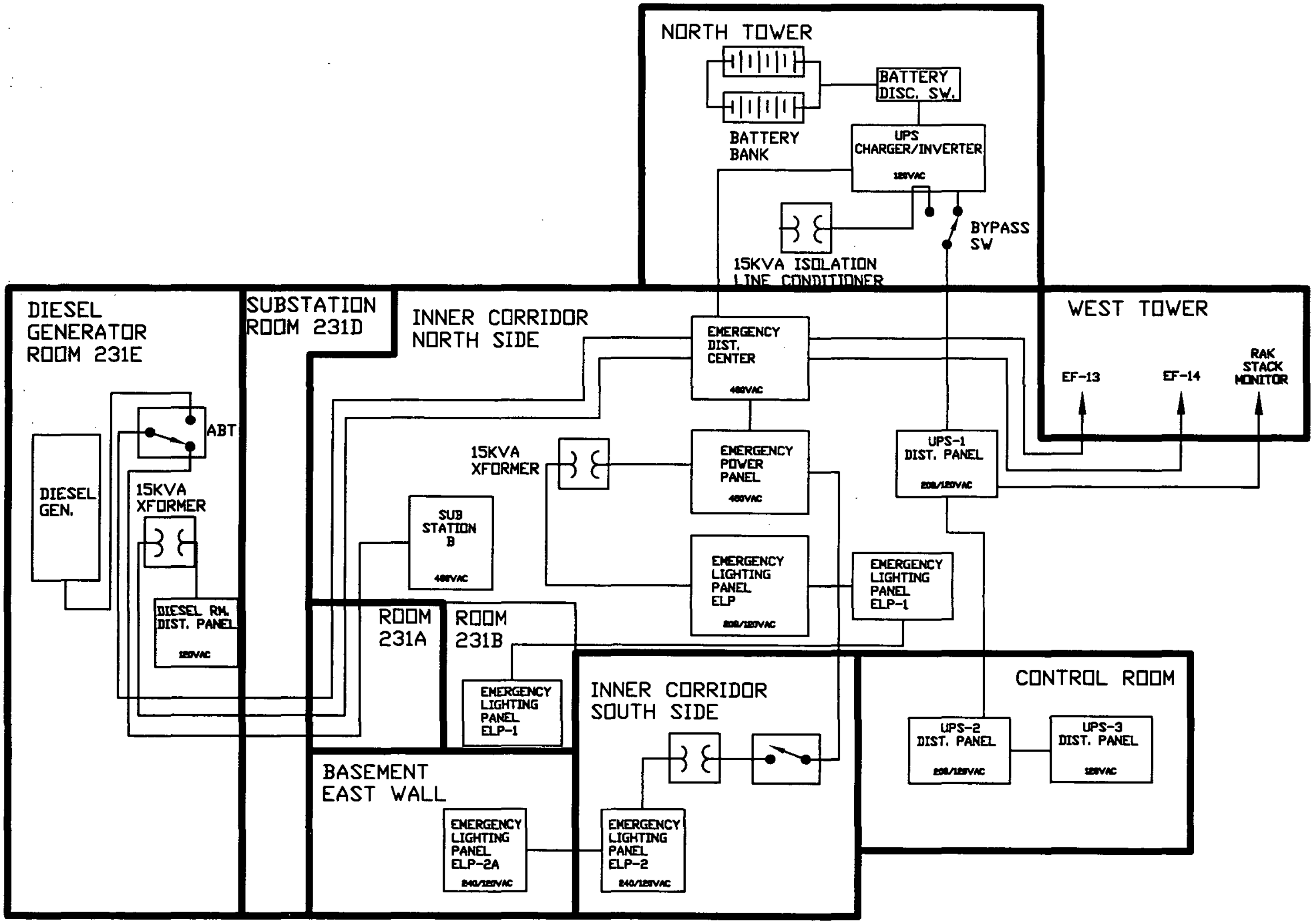
2002
2004

system, nitrogen station, fire protection system, and the evacuation alarms. 2004

4) Emergency Power Panel which feeds through a transformer and distribution panel to supply exit lights, stairway lights, fan failure alarm, intercommunications ~~system, nitrogen station, and the evacuation alarms.~~ This emergency power panel also feeds the emergency air compressor, the motorized isolation doors, the truck entry door and pedestrian entry doors. 2002

5) 120 volt distribution panel via either an uninterruptible power supply or a line conditioner. This distribution panel provides the following loads:
   a) Stack Off-gas Monitor
   b) Reactor Control Power (control rods, rod run-in, safety system, and Servo amplifier)
   c) Annunciator Panel
   d) Area Radiation Monitoring System
   e) Neutron and Process Monitoring Instruments

The generator will run for approximately 30 minutes weekly under no load conditions. The generator is load tested on at least a semi-annual interval.



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DATE 10/19/04
DRAWN BY: C. JONES
CHECKED BY: ENGINEER
CODE
REVISION NUMBER: 4
REVISION DATE 10/19/04

University of Missouri-Columbia
FACILITIES OPERATIONS
research reactor facility

FIGURE 3.8.1

MURR NUMBER: 2272
SHEET 1 OF 1

FIGURE 3.8.1

3.9 "Define each peak to average power ratio used and summarize the assumptions and methods of calculation used."

Please refer to the Hazards Summary Report, Section 5.5.2.3.

P_r = the ratio of the average power density in the hot channel to average core power density. This factor is based upon the power distribution being the same as the flux distribution for a 5 Kg core, uniformly loaded, with the rods out. This ratio is the r-θ component of the peaking in the core and takes into consideration both radial and circumferential peaking. The radial peaking was found to be the highest at the inner radius of the fueled region and near the side plates. The radial peaking factor calculations were based on the assumption that the fueled annulus of the reactor was homogeneous, therefore, a small correction factor was introduced to account for the inner fuel plate being slightly outside the inner radius of the fueled region.

P_a = axial peaking is the ratio of the maximum power density in the hot channel to the average power density in the hot channel with the 5 Kg core uniformly loaded and rods half inserted.

P_aP_r is now the ratio of maximum power density in the hot channel to the average core power density for the uniformly loaded 5 Kg core with the rods in the 50 percent inserted condition.

The overall value of the P_{max}/P_{ave} (P_aP_r) for the Missouri core under the conditions above is then 3.66 based upon two dimensional analysis in the R-θ and R-Z directions. The calculations were made using the PDQ computer code with four lethargy groups in the R-Z plane and the CURE code with three lethargy groups in the R-θ plane of reference.

The references used are Section 5 of TM-DMS-62-5 (Reactor Physics Analysis for the University of Missouri Research Reactor - Internuclear Company) and TM-WRP-62-10 (Core Heat Transfer and Fluid Flow - Internuclear Company).

The factor R_3 corrects for deviation in core loading from 5 Kg as well as for considering the possibility of loading any combination of light elements (350 U^{235}) and heavy elements. It should be noted that the calculations presented in the hazards analysis allows operation to 5 MW with as many as four light elements in the core. It is not anticipated that operation above 100 kW will be done with other than a uniform core and it may be definitely stated that operation above 5 MW will only be allowed with a uniform core loading (ref. Fig. 5.6 of Hazards Summary Report).

The corrections F_b and F_θ allow for a reasonable error of tolerance on the parameters as listed in Table 5.4. An overall error of 26.3% is considered on the bulk temperature factor and 33.2% on the film factor.

3.10 "Describe the supply and exhaust air doors and drives, and discuss the means of assuring that these doors will close and seal when needed."

The doors are 1/4" metal plate welded to a frame 5' 5/8" high by 5'2 1/8" wide and cover an opening 4'0" square. They are supported by two 1/2 ton trolleys riding on a 6" I-beam 12.5' long which is supported by attachment to the building. To insure proper alignment the doors travel in a guided slot at the bottom.

Each door is driven by a 3 phase, 440 volt, 1 hp, 1750 rpm motor through a Boston Ratiomotor (M126-20HU) lowering the output to 87.5 rpm. Attached to the shaft of the ratiomotor is a KSD-11-1 sprocket which, through a chain drive, drives another shaft which has two KSD-11-1 sprockets attached. A clutch permits disengaging the motor drive for manual door operation. Around the sprocket is a RC60 roller chain which passes around another KSD-11-1 sprocket located at the closing end of the door. The roller chain is attached to the door causing the door to move in response to the motor drive.

The doors are sealed by inflatable seals mounted in the door facing. The air for the seals is supplied from the main air system which is backed up by an emergency air compressor in case of a main power system failure.

The air is fed through a manual stop valve at 80 to 100 psi and reduced locally. The line contains a relief, three-way solenoid valve, for inflating and deflating the seals, and a pressure switch for sensing when the air is bled off. The pressure switch activates permitting the doors to open when the seals deflate.

The actuation of the isolation or evacuation switches located on the control console, or the evacuation switch located in Room #202 (MURR Lobby), or on detecting high radiation level above the set point in the building exhaust air plenum or above the pool, will energize two relays in parallel, with a contact from each in parallel, completing the circuit to the closing coil of the motor causing the motors to drive the doors to the closed position. The doors upon closing actuate their respective limit switch. This causes three-way solenoid valves to energize permitting the gaskets to inflate. After a time delay of approximately ten seconds the solenoid valves de-energize.

Redundancy is accomplished by the use of two detectors, two relays, two power supplies, and two compressed air supplies.

Further discussion of the isolation system reliability and assurances that the doors will close when needed is presented as part of Section 3.20.

3.11 "State the maximum allowable containment building leak rate at 2 psig. Describe in detail the method and frequency of testing the leak rate in the "as is" condition and of extrapolating to 2 psig. Describe the frequency and type of checks performed on the isolation system including tests for leakage in the inflatable gasket system. Provide the time limits for automatic closure of penetrations after radiation detection. Reevaluate the doses from the maximum credible accident using the maximum integrated leak fraction over the course of the accident. Justify all pressure and dose reduction factors. Discuss the adequacy of provisions for vacuum relief in the containment building."

A) Discussion

The results presented in the following paragraphs will revise data and information presented in University of Missouri Hazards Summary Report dated July 1, 1965, Sections 3.4 and 13.4.2 through 13.7.

The reactor containment was tested for leak rate on January 5, 1966. As a result of this test the containment building leak rate was determined to be a maximum of 11.5% per 24 hours at 2 psi over-pressure. The leak rate was determined by the reference vessel method as presented in the American Nuclear Society proposed standard Leakage Rate Testing of Containment Structures for Nuclear Reactors. The evaluation of doses from the maximum credible accident are based on the leak rate determined by the test and this peak leak rate of 11.5% per 24 hours shall be considered as the maximum allowable at 2 psig.

The above leak check was made at an initial 2 psi over-pressure, therefore, extrapolation to 2 psi was not necessary. Subsequent tests of the containment will be made annually to determine leak rate and containment will be pressurized to approximately 1.9 psi initial over-pressure. The leak rate at 2 psi will be obtained by extrapolating data observed from leak rates at 0.5, 1.0, 1.5 and 1.8 psi over-pressure as determined from the leak test.

It is to be noted that the average leak rate over a 24 hour period based on the peak leak rate stated will be less than 10 percent per day.

The reactor isolation system shall be checked for proper operation at least every 90 days. This check will be performed by a radiation trip of one of the two isolation detectors in the isolation system. The built-in source will be used on the detector. In order to assure that the system has isolated properly a visual check of all components will be made to see that they are in the isolated position.

Experience with the existing inflatable gasket system to-date has shown that a significant leak in the gasket can be detected audibly without difficulty. The gasket is not in danger of being deflated due to a small leak because it is under constant pressure from both the main and emergency air compressors.

The time limit for automatic closure of the isolation valves and doors is based on the measured time required for the ventilation doors on the reactor fifth level to close and seal. The required time is seven seconds from the initiation of the isolation signal until the doors are closed and sealed. The automatic valves on the 16 inch ventilation pipes are quick acting and close in about three seconds.

B) Evaluation of Doses from the Maximum Credible Accident:

1. There is no change in the postulations leading up to the cause of the accident.
2. The meltdown was increased from 10% to 20%. Investigation has resulted in no firm value for the percent meltdown. Opinions range anywhere from one percent to fifty% with fifty% admittedly very conservative. The value was increased to 20% due to observations in the SL-1 accident and SPERT reports indicating that 20% would be more in order.
3. The rate of release of the activity contained in the building is based on data obtained from the building leak test performed in January 1966. The calculated doses take into consideration not only the radioactive decay of the fission products but also the reduction of leak rate due to the over pressure decay in the building.

4. As in the original report, Sutton's equation was used to determine the down-wind concentration of cloud activity. However, the diffusion coefficient C_y and C_z were changed from 0.4 and 0.07 to 0.04 and 0.08 for C_y and C_z respectively. Again there is not total agreement among authorities as to particular values for these coefficients. The values chosen for the revised calculations are based on data reported in IDO-12005 (Workbook in Atmospheric Diffusion Calculations - G. A. DeMarrais) and the Ames Hazards Report. It is assumed that an inversion condition exists and the wind velocity is one meter per second. The stability parameter remained the same at 0.5.
5. The calculation of the thyroid dose from Iodine was based on original activities determined from Table II - Burnett, Reactors, Hazard vs. Power Level. This change was made due to the large contribution of beta activity to the thyroid dose, whereas Table 13.3 of the Hazards Summary Reports total gamma activity only.
6. The corrections applied to Sutton's equation were, for external dose, fallout and building effect. For internal or thyroid exposure, building effect only. Note that the correction for wind variability, as originally considered, has been omitted.

The thyroid dose, as calculated, is considered to be very conservative due to the following:

1. The decay of building pressure is based on leakage only and does not consider loss of pressure due to heat lost to the building containment and components. The heat loss will decrease the release rate as a function of time after the accident.
2. The calculations assume that there is no retention of iodine in the pool water and no iodine plate out on the building and components.
3. Sutton's equation does not consider terrain effect on diffusion. The sitting of the Missouri reactor is such that a significant amount of dispersion would be expected.
4. The calculation assumes that the persons exposed are on the cloud centerline throughout the duration of exposure. For long exposure times this is not considered likely.
5. The building leak test, upon which the calculations are based, was made with no paint on the interior of the containment walls. A subsequent leak test shows that

the leak rate is approximately 8 percent or a reduction of 3.5% due to painting. Because the reduction was due to the painting and the paint coat is subject to some deterioration with age, the calculations were based on the unpainted condition.

6. The total energy release as discussed in Section 13.4.4 of the Hazards Summary is much less than that required to raise the pressure in the building to two psi over pressure.

At the boundary of the reactor exclusion area (500 feet from the containment) the external dose is 2.8 Rem following a two hour exposure. The infinite dose to the thyroid after two hours of exposure is 304 Rem. (The reference for thyroid dose calculations is The Physics of Radiology by Johns - page 566.)

Calculations were performed on the IBM 1620 Model II computer.

Total results of dose calculations are presented on graphs, Figures 3.11.1, 3.11.2 and 3.11.3.

Vacuum relief in the containment building is provided by the water seal trench which will relieve at approximately 1.7 psi external pressure. Of main concern is the ability of the containment to withstand this negative pressure. The architect responsible for the design of the building has calculated that - 1.7 psi puts no limiting stresses on the building.

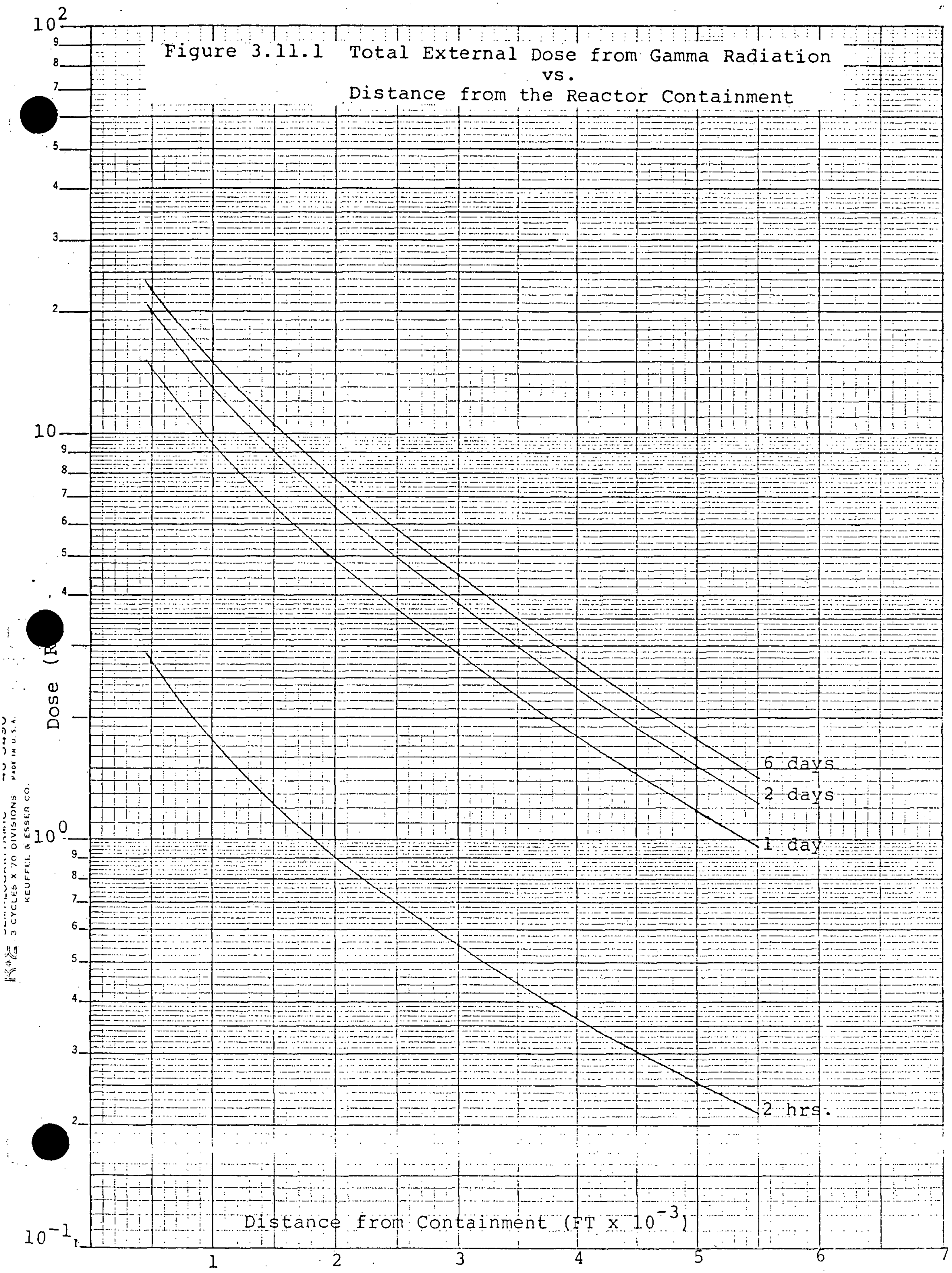


Figure 3.11.1 Total External Dose from Gamma Radiation vs. Distance from the Reactor Containment

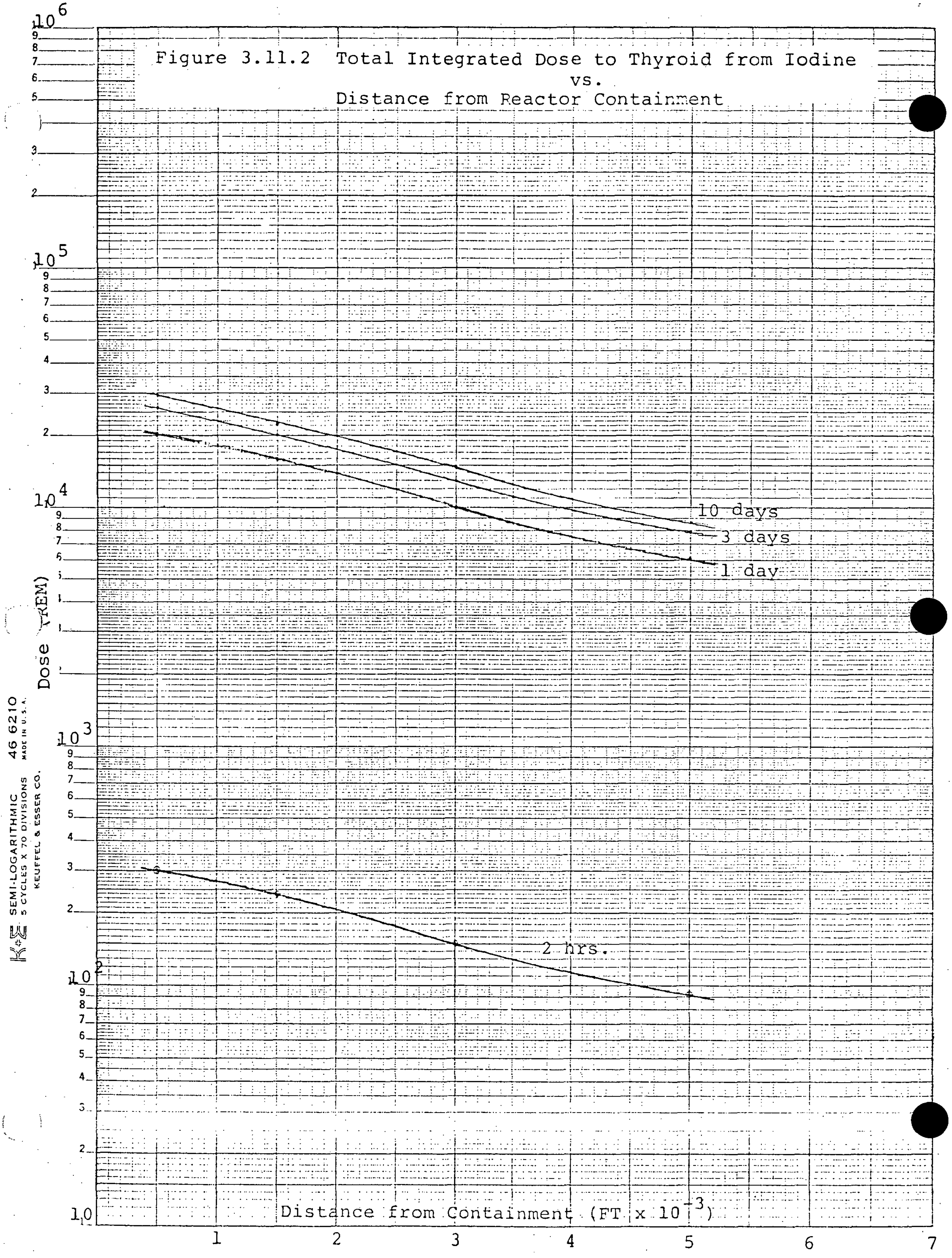
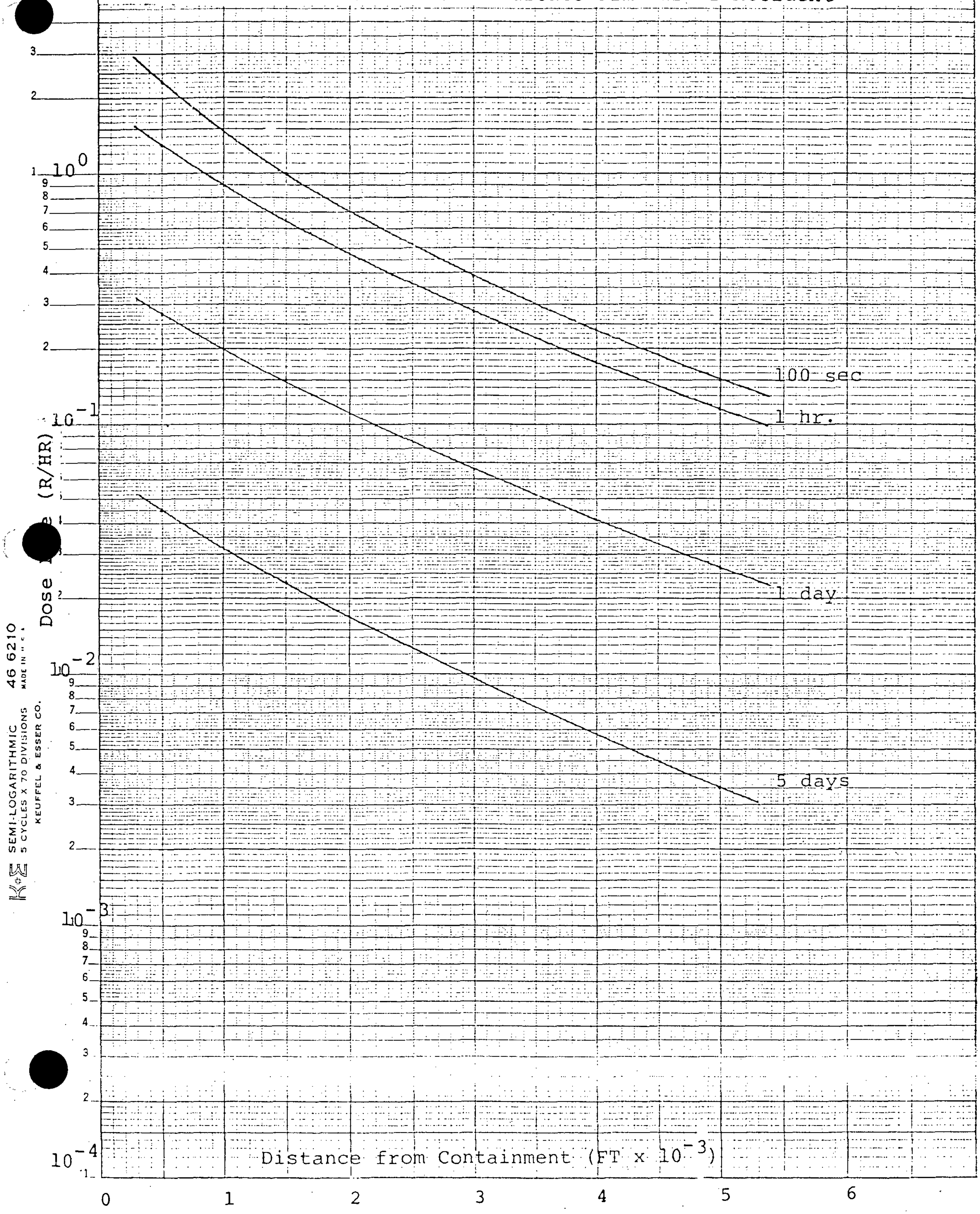


Figure 3.11.2 Total Integrated Dose to Thyroid from Iodine
vs.
Distance from Reactor Containment
10 days
3 days
1 day
2 hrs.
Dose (REM)
Distance from Containment (FT x 10^{-3})
10^6
10^5
10^4
10^3
10^2
10
1
2
3
4
5
6
7
SEMI-LOGARITHMIC 46 6210
5 CYCLES X 70 DIVISIONS MADE IN U.S.A.
KEUFFEL & ESSER CO.

(-9)

Figure 3.11.3 External Dose Rate to Whole Body from Fission Product Gamma Radiation vs. Distance from Containment Various Curves Indicate Time after Accident



3.12 "Evaluate the magnitude and consequences of reactivity insertion caused by partial melting of fuel elements in the maximum credible accident and in flow blockage accidents in which several fuel plates in the center of an element melt."

A full discussion of the question of the consequences of a partial loss of fuel plates cannot be included with this report. A detailed analysis requires considerable digital computer calculation. The calculations are being accomplished on the University of Missouri's IBM 7040. Unfortunately computer availability has been such that completion of the analysis could not be accomplished in time to be included here.

The computation, using codes designated AIM-6 and EXTERMINATOR, are expected to show that loss of centrally located fuel plates would result in a decrease in the effective neutron reproduction factor. This assumption is based upon that fact that the mass coefficient of reactivity, $(\Delta k/k) / (\Delta M/M)$, has the rather large value of 0.117 for a 5.2 kilogram loading. The mass coefficient, however, was calculated using an homogenized core rather than by perturbation techniques. A definite conclusion cannot be stated prior to the detailed analysis mentioned above. Such an analysis will be submitted later.

3.13 "Evaluate the consequences of the valves to the in-pool convective loop failing to open following a loss of primary coolant flow and a reactor scram."

It is postulated that after continuous operation for a 30 day period at 5 MW, the reactor experiences a scram due to loss of reactor coolant flow and/or system pressure, with the following events occurring.

a. P501 turns off.
b. V-507 A and 507 B closes, isolating the reactor loop.
c. V-543 A and 543 B open, venting the reactor loop to the atmosphere.
d. V-546 fails to open prohibiting reactor decay heat from being dissipated to the pool through the in-pool heat exchanger.

Under these circumstances it is shown below that the water in the reactor vessel may be raised to near saturation temperatures but there will be no net formation of steam. The reactor core remains covered with water and fuel plate temperatures do not reach melting conditions.

The fission product energy release rate from a reactor which has been operated for 30 days at 5 MW is equal to 6.9 percent of operating power. Therefore, the core will be producing decay heat at a rate of 0.069 x 5 MW = .345 MW, which converts to 327 BTU/second. Approximately 62.7 percent of the core decay heat is dissipated within the core region of the pressure vessel. Therefore, the initial rate at which energy is being delivered to the reactor coolant is 205 BTU/second.

The amount of water available to absorb the decay energy is assumed to be that in the pressure vessel from the bottom of the core to the top of the vessel, plus that in the piping up to the in-pool convective loop and valve. The total volume of water available is 9.17 cubic feet which, at a temperature of 155°F, is 650 pounds of water. The amount of heat required to raise this quantity of water to saturation temperature (assumed to be 212°F) is

$$506 \text{ lbs.} \times \frac{1 \text{ BTU}}{\text{lb } ^\circ\text{F}} \times (212-155) = 31{,}920 \text{ BTU}$$

The total time required to release this quantity of heat is about 420 seconds. The heat release rate 420 seconds after shutdown is 66 BTU/second.

Finally, we calculate the rate at which heat will be transferred from the 212°F water inside the pressure vessel to the 100°F water in the pool outside the pressure vessel.

The film coefficient for heat transfer from a round tube of water for the condition under discussion is very close to 233 BTU/hr•ft^2•°F. The total heat transfer area, including the pressure vessel and the piping to the in-pool convective loop valve, is 54 ft^2. One may now calculate the required piping outside wall temperature for a heat removal rate of 66 BTU/second from:

$$(T_W - 100) = \frac{66 \text{ x } 3600 \text{ BTU/hr}}{54 \text{ ft}^2 \text{ x } 233 \text{ BTU/hr} \bullet \text{ft}^2 \bullet {}^\circ\text{F}}$$

$T_W = 118.88°F$

Since the inside temperature of the pressure vessel is assumed to be much higher (212°F) than the calculated outside wall temperature required for heat removal, the conclusion is that there will be no net formation of steam in the pressure vessel following the assumed valve failure. There is, however, the likelihood of considerable nucleate boiling in the core region for the first several minutes following the scram.

The data on decay heat rates were taken from "Fission Product Decay Power," a report prepared by D. M. Shapiro of the Internuclear Company.

3.14 "Evaluate, with and without scram, the startup accident with all four shim rods withdrawn at maximum rates from interlock neutron level."

A) Summary

An analysis was made, using an analog computer, of the power history of an excursion resulting from the uncontrolled, continuous withdrawal of the four control rods used with the MURR. The initial power level was assumed to be that at shutdown with a one (1) curie startup source (i.e., 1.95×10^{-4} watts).[4] The rods were considered to be bottomed initially and then withdrawn, in bank, at maximum rate from the core.

Briefly, the analysis followed in this order: First, an approximate calculation was performed in order to determine the period of the reactor as the control rods passed through the critical position. Another calculation gave the power level at critical. These calculations allowed the evaluation of initial conditions for the analog computer simulation, which computed the power history of the excursion from the time the rods passed through the critical position. The ability of the scram process to effectively shutdown the reactor, was evaluated using the technique of Moore and Grossman.[5]

The conclusions drawn from this analysis are that an uncontrolled rod withdrawal at the source power level will result in damage to the reactor, unless immediate steps are taken to insert negative reactivity. The computer results imply that unless compensating action is taken, power will rise at rates as much as five megawatts per second. Under such circumstances, the power will rise, causing core voiding, burn-out, and possibly fuel plate meltdown. The excursion would eventually be reversed by inherent reactor shutdown mechanism such as temperature and void coefficients or core disassembly should scram and rod run back circuits fail. These mechanisms are more fully discussed in another report.[2]

The time scale on which a startup accident occurs warrants some attention. The bulk of the time is spent with the rods being pulled and the reactor in a subcritical condition. The first 14.7 minutes of rod withdrawal pass with the reactor in a subcritical state. During this time, the reactor power increases by a factor of ten over the startup interlock power. After the reactor achieves criticality, power builds up at increasing periods, limited only by the power feedback parameters and rate of δk insertion. Within thirty seconds after the rods have passed the critical rod position, power will have risen to burn-out conditions. Forty-five seconds after criticality, power is expected to have risen above 100 MW. Unlike the case of step induced prompt critical excursions, however, both manual and automatic scrams afford a high degree of protection to the reactor.

B) <u>Preliminary Calculations</u>

The method of analysis involved calculating the critical power level and the associated rate of change of power by analytic procedures. These would then serve as initial conditions for the analog simulation program.

The first step involved the estimation of the reactor period at criticality. Here, the method employed was that given by Schultz where subcritical and prompt supercritical asymptotic approximations were used.[6] The relationships used in computing periods for subcritical and super prompt reactivity conditions were respectively:

$$P > -\frac{\delta k}{\gamma} \qquad \text{for the subcritical case}$$

and

$$P < \frac{\ell^*}{\delta k - \bar{\beta}} \qquad \text{for the prompt critical case}$$

where:

$P =$ period (seconds)

$$\delta k = \frac{k_{eff} - 1}{k_{eff}} \text{ (reactivity)}$$

$\gamma =$ rate of change of reactivity (δk/sec.)

and where $\bar{\beta}$ and ℓ^* have their usual kinetics interpretations.

The reactivity insertion rate was evaluated on the basis of straight line segment approximations to the rod worth curve and rod drive rate.[5]

Using these asymptotic approximations, the reactor period was plotted against reactivity as is seen in Figure 3.14.1. It will be noted that the region $0 \leq \delta k \leq \bar{\beta}$ represents a region of discontinuity for these approximations. It was necessary to interpolate in order to obtain information in this region. On the basis of these calculations, it was estimated that the reactor period is between five and seven seconds at the time the control rods pass through their critical positions.

In order to compute the power level at critical, it was recalled that period is defined as:

$$P = \frac{n}{dn/dt}$$

Thus, a differential equation is obtained involving the reactor period:

$$\frac{dn}{dt} = \frac{n}{P}$$

Using previously computed values for reactor period, from rod bottomed to criticality, and, using the initial source interlock power level, it was possible to integrate the above equation to obtain an approximation to the critical power level.

$$n_{crit.} = 9.36\ n_{initial}$$

or

$$n_{crit.} \cong 2 \times 10^{-3} \text{ watts}$$

Having at hand the critical power level and the period, the rate of change of power (dn/dt) was obtained. The critical power level and the rate of change of power were considered to be initial conditions to be applied to the analog simulator. Actually, it was only necessary to use the rate of change of power since the power level was essentially zero.

C) Analog Simulation

The analog simulation was performed using two Electronics Associates Inc. TR-10 analog computers. Except for minor details, the physical model for the reactor kinetics and power feedback are as described in detail in another report submitted in answer to question 15.[2]

Basically, the computer solves the single, averaged delayed neutron group kinetics equations. Simultaneously the core and island temperatures and the core void fraction are generated as functions of power. These, through their respective coefficients, modify the reactivity as seen by the reactor, and provide the basis for inherent shutdown mechanisms.

The computer was scaled so that it would provide information for power ranging from zero to 100 megawatts. The simulator circuit is shown in Figure 3.14.2, where common analog symbology is used. Reactivity was inserted at a rate of 2.78×10^{-4} δk/sec., a value corresponding to the rate of insertion at the critical rod position.

D) Results

The results of the computer run are shown in Figure 3.14.3, where power and the reactivity of the reactor as shown as functions of time. Referring to the reactivity, δk increases as a ramp function until the power rises to about four (4) megawatts. At this point, power feedback begins to be felt, altering the "zero power" kinetics. The reactivity reaches a maximum and then reverses under the influence of the power feedback. The power increases slowly at first, but after about 25 seconds after the critical rod positions are passed, the power level changes rapidly, approaching a rate of about five megawatts per second.

As in a previous report,[3] the burn-out power level is considered to be 23.4 megawatts. In order to protect the reactor, the excursion should be terminated before the power level reaches this value. Using the method of Grossman and Moore,[1] and a curve of reactivity as a function of time during rod drop, it was determined that the startup accident can be safely terminated by scrams initiated at 5.5 MW and below. The period scram setting of 8 seconds will cause a shutdown even before the reactor becomes critical. In light of this, it appears that automatic scrams will adequately terminate a startup accident. It further appears that inherent shutdown mechanisms acting alone are insufficient to safely terminate a startup accident excursion. It appears that a startup accident would be detected with ample time remaining for a shutdown by manual scram, should automatic scrams and run backs fail.

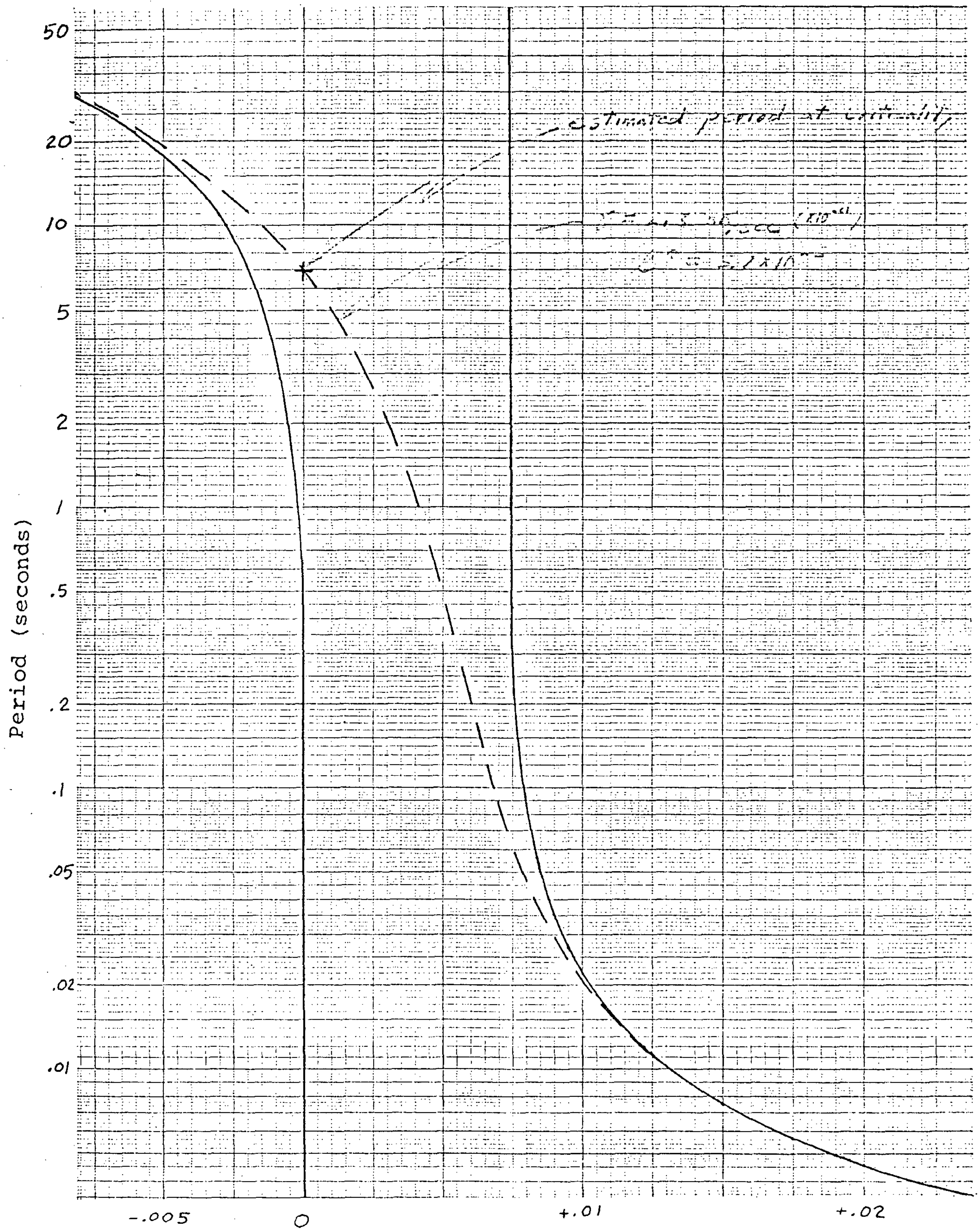


Figure 3.14.1 Reactivity (δk)
Period versus Time during Rod Withdrawal

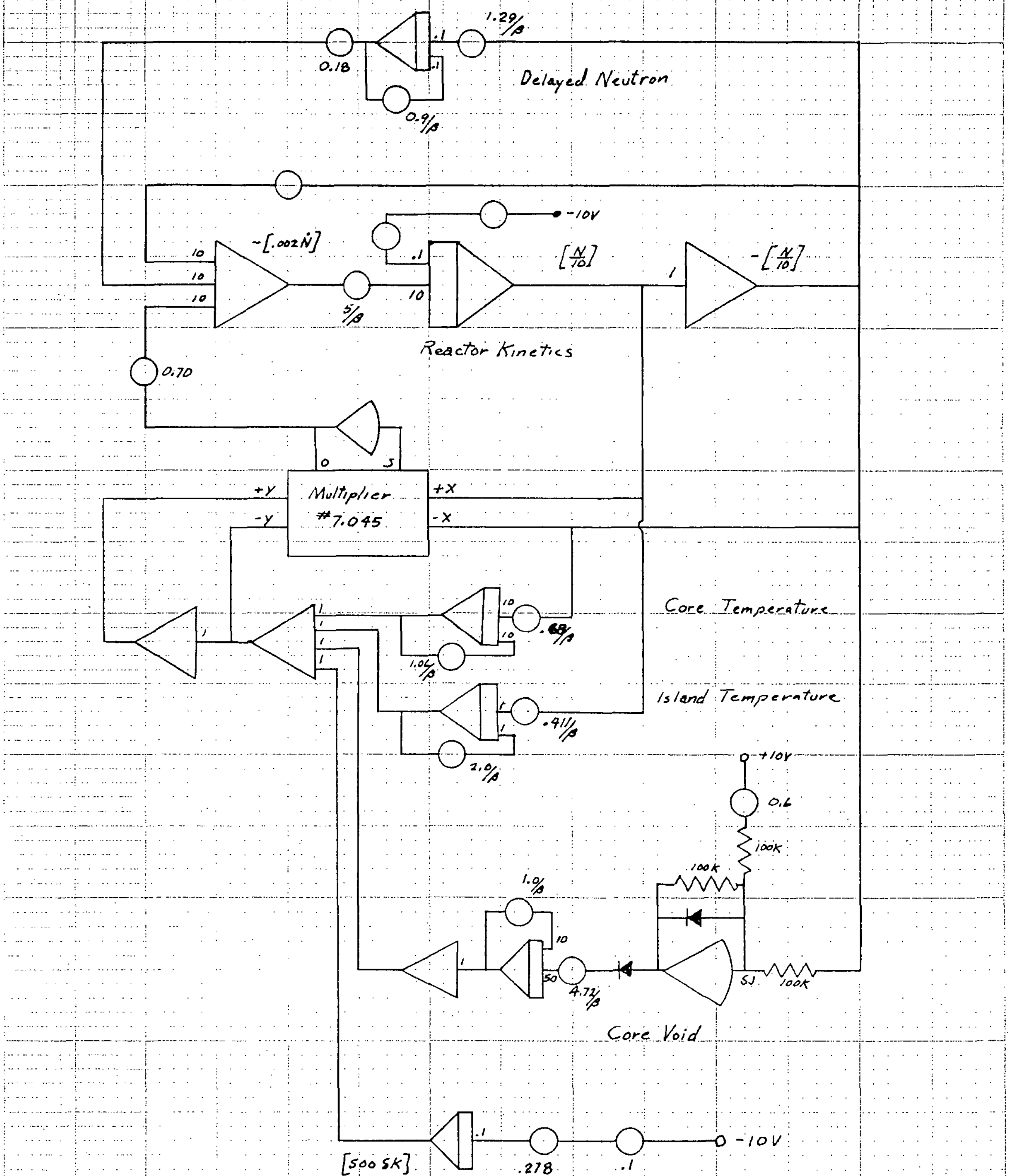


Figure 3.14.2 Analog Simulator Circuit

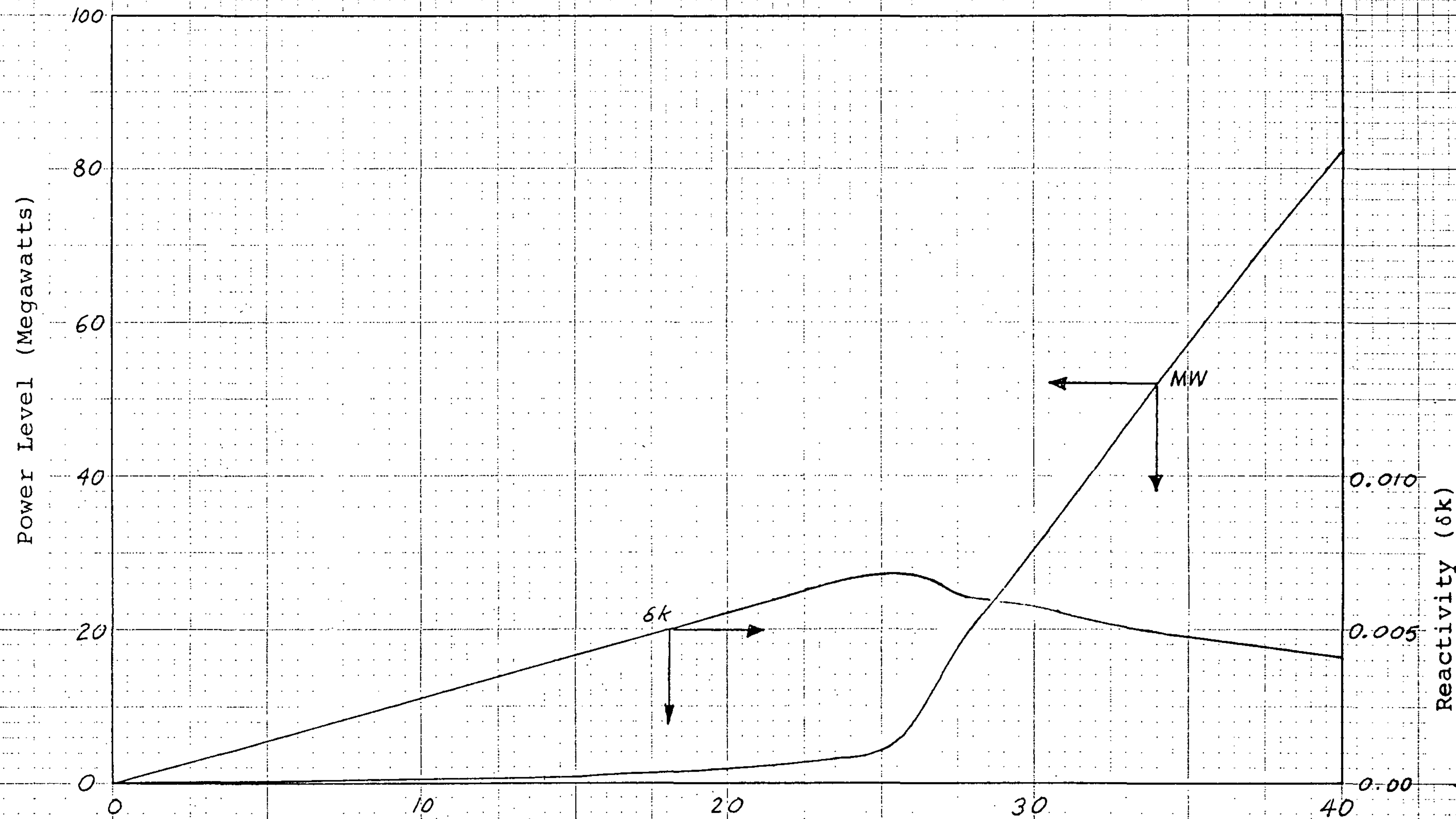


Figure 3.14.3 Time, Seconds after Rods Passed Critical Position

References

1. Grossman, S. R. and Moore, K. V., A Method of Estimating the Kinetic Effect of Scram Rods, IDO-16777, May 1962.
2. McCright, G. H. Jr., A Study of Burnout in the MURR as a Result of Step Induced Excursions, MURR internal communication, January 31, 1966.
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4. Shapiro, D. M., Missouri University Research Reactor Design Data, Vol. II, Internuclear Company, TM-DMS-62-5, September 28, 1962, pp. 129 ff.
5. Shapiro, D. M., Op.Cit., p. 86.
6. Schultz, M. A., Control of Nuclear Reactors and Power Plants, McGraw-Hill, New York, 1961, pp. 381 ff.

3.15 "Evaluate, with and without automatic scram, the maximum step reactivity insertion that can be tolerated without fuel hot spot melting under the various flow and pressure conditions under which you expect to operate. Relate this to the maximum reactivity worth of a single experiment and to voiding of the island."

3.15.1 Introduction

This section represents a summary of work done to establish a maximum reactivity step insertion which can be tolerated by the University of Missouri Research Reactor without suffering core damage. The result of this study is the establishment of a conservative reactivity value, which, if not exceeded, will not result in hot spot burn-out. The resulting value for reactivity is then related to the respective worths of experiments and to island voiding.

Both digital and analog computation was employed in arriving at the conclusions reported herein. In addition, heavy reliance was placed on the results of the SPERT test series and the SL-1 accident investigation.

As a result of the calculations, in conjunction with the SPERT and SL-1 data, carefully qualified conservative estimates were obtained regarding the magnitude and duration of power transients, approach to burn-out, and pressure excursions.

On the basis of these calculations, it is highly improbable that core damage would result from the failure of any single experiment or island voiding.

3.15.2 Shutdown Mechanisms

A reactor, when subjected to a step increase in reactivity, follows a reasonably predictable course. This course is determined by the magnitude of the reactivity insertion, the reactor kinetics and the power feedback parameters. Upon receiving a positive step in reactivity, a previously critical reactor will increase power at a rate determined by the magnitude of the step and the effects of changing power.

In order to avoid the possible consequences of a power excursion, the reactor must be promptly shut down. This may be accomplished by reactor scram functions or inherent shutdown mechanisms associated with the reactor design. In prompt critical excursions, the inherent shutdown mechanisms are very important since the transient will have progressed to dangerous levels before a scram can take effect. Time delays associated with power feedback are much shorter than scram delays, and so, the importance of inherent shutdown mechanisms for fast transients is greatly enhanced.

There are several important mechanisms which cause a reactor to shut itself down or to stabilize after an excursion. These mechanisms will be discussed in the next several paragraphs.

A useful mechanism for small excursions (less than prompt critical) is the moderator temperature-reactivity relationship. In general, reactors are designed such that the temperature coefficient is negative. In most cases a delayed critical transient can be successfully controlled by the temperature coefficient alone, should scram or rod run-back circuits fail simultaneously. Usually, such a condition will result in reactor power stabilizing at a power higher than normal, but not sufficiently high as to cause a departure from nucleate boiling and subsequent burn-out. Thus, even if safety trips fail to function, excursions due to reactivities less than the effective delayed neutron fraction would be of little major consequences.

As reactor periods become shorter, the peak power tends to increase. Another mechanism, the void coefficient, then becomes important in limiting excursions. In this, the reactivity changes as a function of moderator density which, in turn, is dependent on steam voids.

In a discussion of ultimate shutdown mechanisms, it is appropriate to insert a discussion of the results of the BORAX and SPERT tests. The results of these tests lead to several general conclusions which are applicable to water moderated, enriched uranium reactors such as the MURR. Some of these general conclusions are here summarized.

1) The nuclear excursions became less severe as the initial moderator temperature was increased toward the boiling point.
2) Changes in initial power level had no measurable effect on the severity of the transient.
3) Increasing the system pressure from 0 to 2500 psig caused only a slight increase in peak power and a broadening of the power burst. Total energy released by as much as a factor of two.
4) Increasing the moderator-coolant flow increased equilibrium power and broadened the post peak side of the power bursts. This is due primarily to the increased coolant effectiveness which partially defeats the inherent shutdown mechanisms.
5) Longer fuel plate thermal time constants tend to greatly increase the severity of the excursions by delaying the onset of the inherent shutdown mechanisms. Thus, the unusually thick (0.035") cladding of the SL-1 fuel contributed to the severity of that accident over what might be predicted from SPERT test results.
6) The destructive pressure excursions that were noted in SL-1, SPERT I and BORAX at periods shorter than above five milliseconds appear to be due to significant vaporization and/or melting in the fuel meat prior to clad melting. The fuel, in this weakened state, is then dispersed into the moderator, thereby vastly increasing the heat transfer area and causing flash boiling. This, in turn, results in the 6 to 10 kpsi pressure excursions and destruction of the reactor. It is believed that such effects could not arise from an unmelted core. Only by dispersing melted and/or vaporized fuel plate material could the necessary heat transfer area be obtained. This was confirmed in SL-1 and SPERT by noting the appearance of debris found in the reactor vicinity. Metal-water reactions apparently cannot be responsible for the pressure surges but might contribute--especially if significant fuel vaporization takes place.
7) Excursions caused by ramp reactivity changes are comparable to step induced transient if the comparison be made on the basis of minimum reactor period.

Considering the time scale in which prompt critical transient takes place, it appears immaterial whether or not a scram occurs. The eventual equilibrium power will be dictated by moderator voiding due to boiling. A reactor scram could only affect a shutdown from this equilibrium power; thus, in considering prompt critical excursions, reactor scrams would provide only post-excursion shutdown. Reactor scrams will, however, provide increased protection in slower delayed critical excursions resulting from step reactivity insertions. In this light, the calculations to be presented consider only self-shutdown of the reactor through inherent shutdown mechanisms.

3.15.3 Method of Analysis

Both analog and digital computers were used in the analysis presented in this report. In general, heaviest reliance was placed on the digital analysis, the analog simulation being used as a supplement. A description of the methods used in the analysis will follow in the succeeding paragraphs.

The major portion of the digital analysis information to be presented was obtained from calculations performed on the Argonne CDC-3600 digital computer at Argonne National Laboratory. The Chic-Kin code was used in performing this calculation. The Chic-Kin code was used in performing this calculation. The Chic-Kin code is a fortran computer program designed to combine transient heat transfer and flow relations with the reactor kinetics equations. This code originated at the Bettis Atomic Power Laboratory and was authored by J. A. Redfield of the Westinghouse Electric Corporation.

In making the transient analysis of the University of Missouri Research Reactor, studies were made of four general cases with respect to initial reactor period. These cases were chosen for comparison with SPERT data and were classified in the following manner:

a) Delayed critical, long periods,
b) Delayed critical, short periods,
c) Prompt critical, period > 10 m-seconds,
d) Prompt critical, period ≤ 10 m-seconds.

Preliminary investigations showed that the latter three cases would prove most interesting, since the automatic and/or manual rod run-backs and scrams should safely terminate long period excursions. In light of this, computer runs were made using the Chic-Kin code for reactivity steps of 0.005, 0.010 and 0.025 δk. These runs were made in two parts. First, an average channel run was made. Core average thermal and kinetic parameters allowed the use of a power coefficient with the kinetics solution. The object of this step was to obtain power with respect to time for each of the transients. A second run was then made in which the power versus time relations previously computed, were used with hot channel parameters. In this way it was possible to investigate hot channel temperatures and heat transfer. These calculations permitted an investigation of the self shutdown mechanisms within the MURR reactor based solely on coolant-moderator density changes.

The results of these computer runs were compared with experimental data from SPERT and BORAX tests, and SL-1 data. Using the experimental data as guidelines, it was possible to estimate the relative hazard to the MURR for each transient excursion and to identify probable shutdown mechanism. Table 3.15.1 lists pertinent parameters of the MURR, SPERT, BORAX and SL-1 reactors for purposes of comparison.

Details of the Chic-Kin code digital kinetics simulator and the analog simulation are presented in Appendix III.

Table 3.15.1

Reactor Parameters

	BORAX	SPERT	SL-1	MURR
Plate Thickness (in.)	.020	.060	.120	.050
Meat Thickness (in.)		.020	.050	.020
Clad Thickness (in.)		.020	.035	.015
Channel Width (in.)		.179	.310	.080
Meat Material	U-Al	U-Al	U-Al	U-Al
Clad Material	Al	Al	Al	Al
Convection	Nat'l	Nat'l	Nat'l	Forced
Pressure (psi)	~6	~6	~6	65
Void coefficient (%δk/cm^3)		-6×10^{-4}	-10^{-4}	-11.75×10^{-4}
β_{eff}	.007	.007	.007	.00738

Destruction Parameters

	BORAX	SPERT	SL-1
Maximum pressure (psi)	6K-10K	3K-4K	10K
Maximum power	2300 MW	$(1.9 \pm .4) \times 10^4$ MW	
Period (m-sec.)	2.6	3.2	4-5
Energy (MW-sec.)	135	31	133 ± 10
Chemical energy (MW-sec.)		3.5	24-33

3.15.4 Results

The Chic-Kin digital code and the analog simulation allowed certain conclusions concerning reactivity inputs which would result in no burn-out. In the digital analysis, burn-out conditions were noted directly on the output data. For the analog computer analysis, burn-out was determined to occur when the reactor power surpassed the burn-out power given in a previous report.[6] The results of these calculations and the conclusions drawn therefor will now be presented.

A) Long Period Delayed Critical Transients:
Experience with reactor control system analysis and calculations performed on the analog reactor simulator convinced the investigator that long period transients are not very interesting from the standpoint of hazards evaluations. For such transients, ordinary scram and rod run-back circuits provide adequate protection. In addition, for such small excursions as, say 0.001 or 0.002 δk, the automatic shimming circuit should successfully compensate for the added reactivity. Even in the event of a complete instrumentation failure, the temperature and void coefficients should cause the power to stabilize without burn-out. The analog computer simulation program was employed to confirm the above results.

In performing the calculations, analog computer runs were made for reactivity steps of from 0.1% to 0.5% δk with normal flow conditions. The results of these computer runs are shown in Figure 3.15.1. It was found that the positive island temperature coefficient caused a minor broadening of the transient peak. Island effects also resulted in a slightly higher final stabilized power level. The calculations showed that burn-out will not occur as a result of reactivity steps less than 0.005 δk in magnitude. All of these transients stabilize at power levels less than 23.4 MW, the minimum value for burn-out.

B) Short Period Delayed Critical Transients:
This case was treated by using an 0.005 δk initial step insertion in the Chic-Kin program. It was found d that power increased rapidly with practically no overshoot from the initial 5 MW to roughly 11 MW. At this point, the period became very long as the moderator density effect on reactivity began to be felt. The power history of this transient is plotted in Figure 3.15.2. A longer run-out would show that power continues to rise on a long period to an equilibrium power level.

A comparison of Figure 3.15.2 with the analog results for an 0.005 δk step from Figure 3.15.1 reveals a minor discrepancy in the initial prompt response. A definite power peak, absent in the digital analysis, was present in the analog run for the 0.005

δk step. This is believed to be due to the over-simplification afforded by Van Rennes' power feedback function which includes only a single averaged thermal delay. The inclusion of the island effects in the analog runs also had the effect of increasing the amount of overshoot. From a practical standpoint with respect to effects felt by the reactor, the two runs have essentially identical consequences. The analog simulation allowed the following of the power transient to its final equilibrium power.

The computer runs and similar data from SPERT IV tests indicated that no burn-out would be encountered during any delayed critical excursion so long as normal coolant flow was maintained in all channels.

C) Prompt Critical with Periods Longer Than 10 Milli-seconds:

A 0.01 δk reactivity step was used in making calculations for this case. This value of reactivity is greater than the maximum worth of a single experiment were it to suffer total "collapse." A reactivity step of this magnitude would result in a reactor period of roughly 16 milliseconds in the MURR. Chic-Kin code gave the following calculated results for the 0.01 δk step.

The average channel run showed that no burn-out would occur in the average channel. Thus, no burn-out was predicted for channels in which the hottest point was attended by heat fluxes less than the core average heat flux. More than 80% of the fuel falls into this category and is presumed secure from damage. The computer run for the hot channel, however, indicated that subcooled departure from nucleate boiling (DNB) might occur at about 116 milliseconds after initiation of the transient. This result is at variance with the results from the SPERT test series in which no burn-out or melting was indicated for periods longer than about 6 milliseconds. This apparent discrepancy is best explained by the conservative nature of the hot channel factors and burn-out correlation.

An analog computer run was made for the 0.01 δk step to determine the effects of the positive temperature coefficient in the island. The result of the analog run was that the island effect broadened the peak and increased post peak power by about 30%.

The analog results are shown in Figure 3.15.3.

The conclusions drawn from these calculations and the SPERT data is that a 0.01 δk step probably represents the conservatively maximum step reactivity insertion which would preclude core damage. Figure 3.15.4, 3.15.5 and 3.15.6 show the Chic-Kin code results for power history, radial fuel plate temperatures, and axial core temperature distribution for the hot channel.

D) Prompt Critical With Periods Less Than 10 Milliseconds:
This case is partially treated in the section entitled A Worst Case Transient, where a 0.025 δk reactivity induced transient is considered. Since this case, resulting in a 3.2 m-sec. period, represents the largest conceivable transient for the MURR, it is sufficient here to consider only cases where the reactor period lies between 3.2 and 10 milliseconds. No computer runs were made for reactivities between 0.01 and 0.025 δk, but the wealth of SPERT I and IV data for reactivities in this range allow a prediction of conditions in the MURR for similar circumstances.

On the basis of SPERT I and SPERT IV data, and calculations, it is conservatively predicted that hot spot burn-out occurs for periods around 16 milliseconds. As the period is made shorter, relatively small pressure surges and thermal expansion tend to cause bowing of the fuel plates and an increasing tendency toward melting. No destructive pressure surges are predicted for periods longer than about 3.2 milliseconds (the worst case).

Figure 3.15.7 illustrates the power history of the 0.025 δk induced transient to the point where subcooled departure from nucleate boiling occurred. The Chic-Kin program was unable to proceed and so, was terminated. The SPERT I-D 3.2 millisecond destructive test is also included for comparison. Figures 3.15.8 and 3.15.9 show the radial fuel plate temperatures through the meat and cladding, and the axial cladding surface temperature distribution for various times after transient initiation. Both Figures 3.15.8 and 3.15.9 refer to the hot channel run.

E) A "Worst Case" Transient:
While proceeding with this study, several conclusions were afforded which are actually outside the scope of this report but are pertinent to a hazards evaluation of the MURR. These interesting conclusions will presently be disposed of.

As was noted in Table 3.15.1, the SL-1, BORAX, and SPERT cores were destroyed by temperatures and pressures under the influence of reactor periods shorter than 5 milliseconds. In this light it will prove of interest to postulate a similar excursion for the MURR and deduce what sequence of events will follow.

In the BORAX destructive test, a large amount of cladding surface was molten and the central region of the m,eat was beginning to vaporize at the time of peak power. It is believed that the vaporized meat caused dispersion of the molten and vaporized fuel into the moderator, thus causing the pressure surge which destroyed the core. The period was about 2.6 milliseconds, and was short compared to the thermal time constant of the fuel plate. The pressure surge occurred just after the time when the power was at the peak.

A similar evaluation of data obtained from the SPERT I destructive test shows that there was no melting of the cladding until more than 5 milliseconds after the power peak, and then only a small amount of surface was affected. The SPERT fuel temperatures were well below the boiling point of aluminum and, in order to obtain dispersal of the molten fuel, some hydrodynamic effect, such as a water hammer, was necessary. This, in part, explains the 15 millisecond delay between the power peak and the destructive pressure transient. The reactor period which attended this test was about 3.2 milliseconds.

The SL-1 was similar to the BORAX test in that significant vaporization in the meat caused a violent dispersion of molten fuel material into the moderator. This again is the most probable cause of the destructive pressure surge. The fact that the meat was significantly vaporized while the reactor was on a period of 4 to 5 milliseconds is interesting since similar circumstances were found for BORAX on a 2.6 millisecond period. This can be partially explained in terms of the fuel plate thermal time constant. The thickness of the meat and cladding of the SL-1 fuel resulted in a thermal time constant much greater than that for BORAX and SPERT reactors. This delayed the moderator density reactivity effects until after the meat had reached the boiling point. Thus, for the SL-1, the destructive pressure excursion occurred for a slower period than was true for either BORAX or SPERT.

In the three cases cited above, the intent was to illustrate the importance of a short thermal time constant for fuel plates and that the violence with which reactor cores are "disassembled" is closely related to the dispersal of hot (molten or vaporized) fuel element material. The importance placed on short thermal time constants implies that fuel cladding should be minimized consistent with low fission product leakage. In this regard, the MURR fuel plate has better characteristics than any of the three examples, having a meat thickness of 0.015 inches. The coolant channel of the MURR is only 0.08 inches wide and is another factor which promotes an enhanced insurance against short period transients. This is due to a shorter plate to water thermal time constant which promotes nucleate boiling sooner in the transient sequence and a faster termination of the excursion.

A postulated reactivity insertion of about 0.025 δk would give a period of about 3.22 milliseconds, which is comparable to the SPERT I destructive test. Calculations using the Chic-Kin code indicated nucleate boiling and subsequent departure from nucleate boiling (DNB) after about 0.017 seconds, and at a power of 600 MW. At this point, the computer run was terminated due to an inability of the program to handle subcooled DNB conditions. It can be surmised that the power continues to rise to perhaps 2300 MW during which time perhaps 35% of the fuel plates will show melting. The peak power and prevalence of melting are based on SPERT I data. It is pertinent to note that both of these values would be conservative, since both the fuel plate and plate to water thermal time constants are shorter than was true for SPERT. It is also significant that the negative void coefficient of MURR is <u>twice</u> that for SPERT and <u>twelve</u> times that of SL-1. Thus, the peak power attained will be less than that for SL-1 or SPERT, and shutdown will proceed faster.

In spite of the enhanced safety features due to the larger void coefficient and shorter thermal time constants present in the MURR, it must be presumed that a pressure surge similar to that of the SPERT test will occur. Since the pressure vessel and coolant piping are of about the same diameter, piping to vessel loss coefficients will be significantly lower than those for SL-1 and other similar reactors. Thus, the piping will most likely dissipate a large amount of the surge energy. As a result of the destructive pressure surge, it may be expected that ruptures would occur in the pressure vessel and in the coolant piping in the vicinity of the reactor. It is expected that nearly all of the kinetic energy involved in the excursion would be dissipated in the pool.

F) Conclusion

As a result of the above calculations, it is expected that burn-out with minor fuel rupture will occur for reactivity step insertions greater than 0.01 δk. Also, it can be concluded that the failure of a single experiment worth up to 0.01 δk would not result in core damage. If the test hole were suddenly voided, the resultant reactivity insertion is 0.0104 δk. It is further concluded that serious damage from pressure surges induced by flash boiling may be anticipated for reactivity inputs of greater than 0.025 δk. Thus, the sudden and complete voiding of the island would probably result in heat core damage.

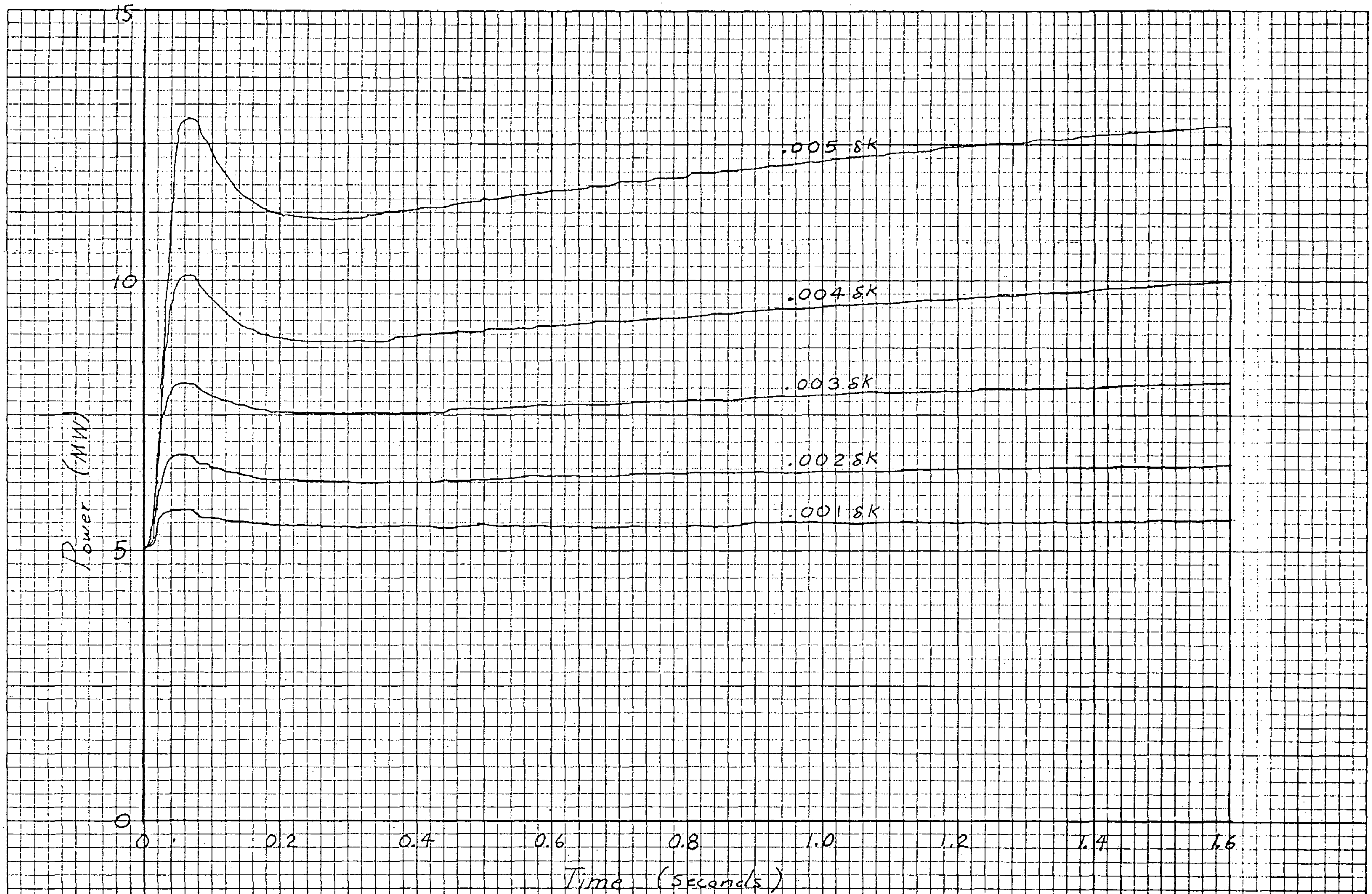

Figure 3.15.1 Analog Results for Sub-Prompt Excursions

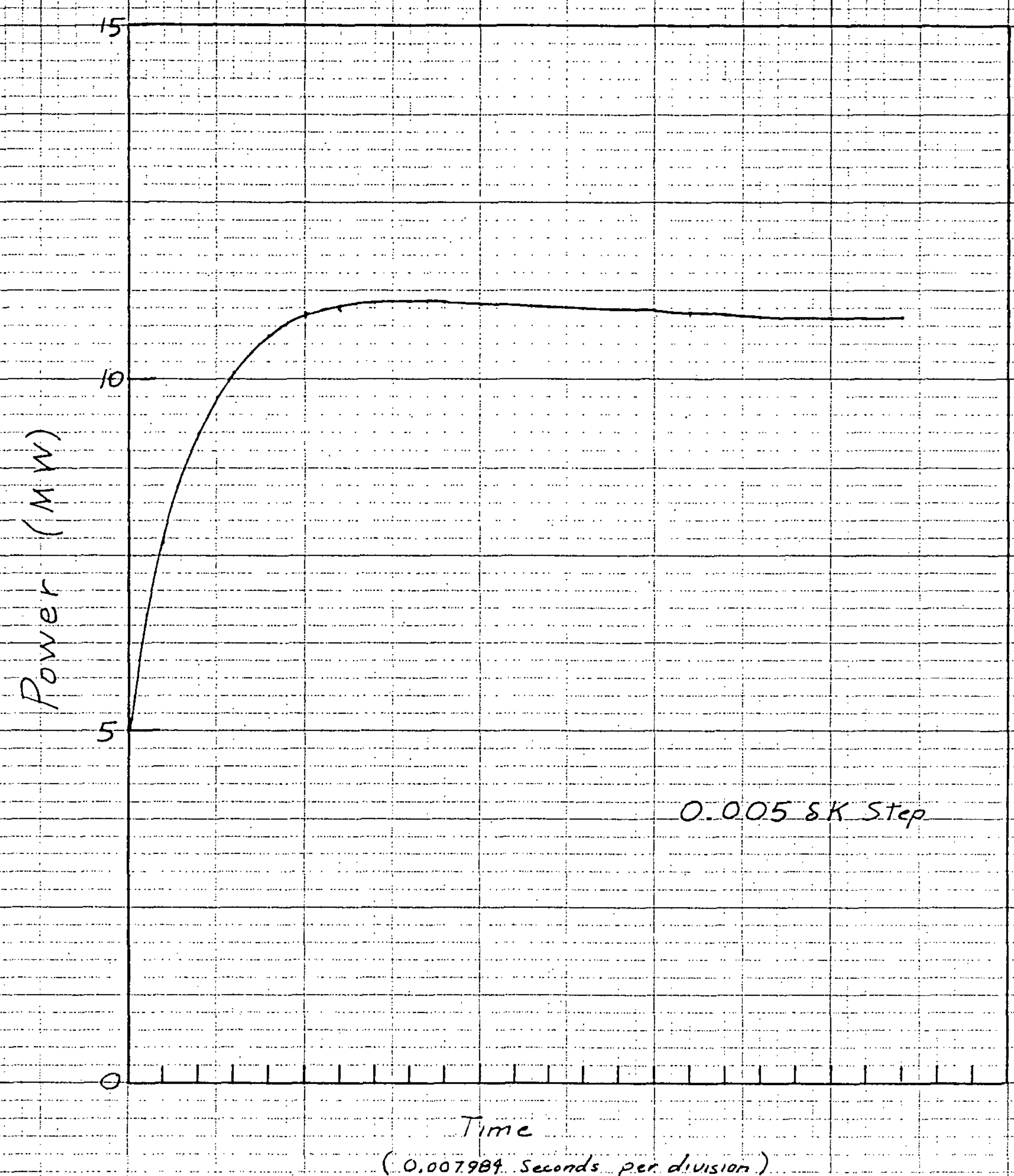

Figure 3.15.2 Chic-Kin Results for 0.005 δk Transient

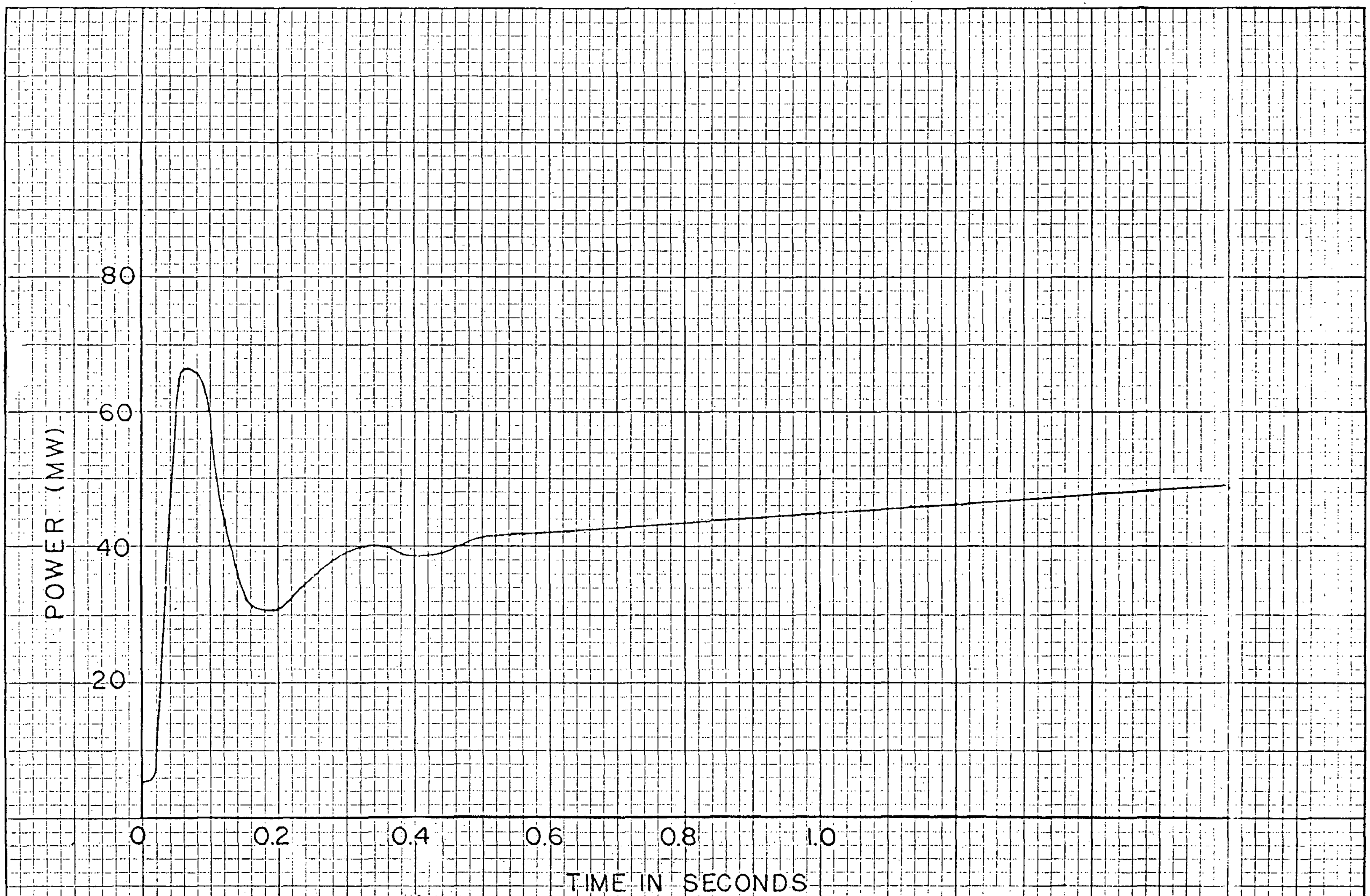

Figure 3.15.3 Analog Results for 0.010 δk Transient

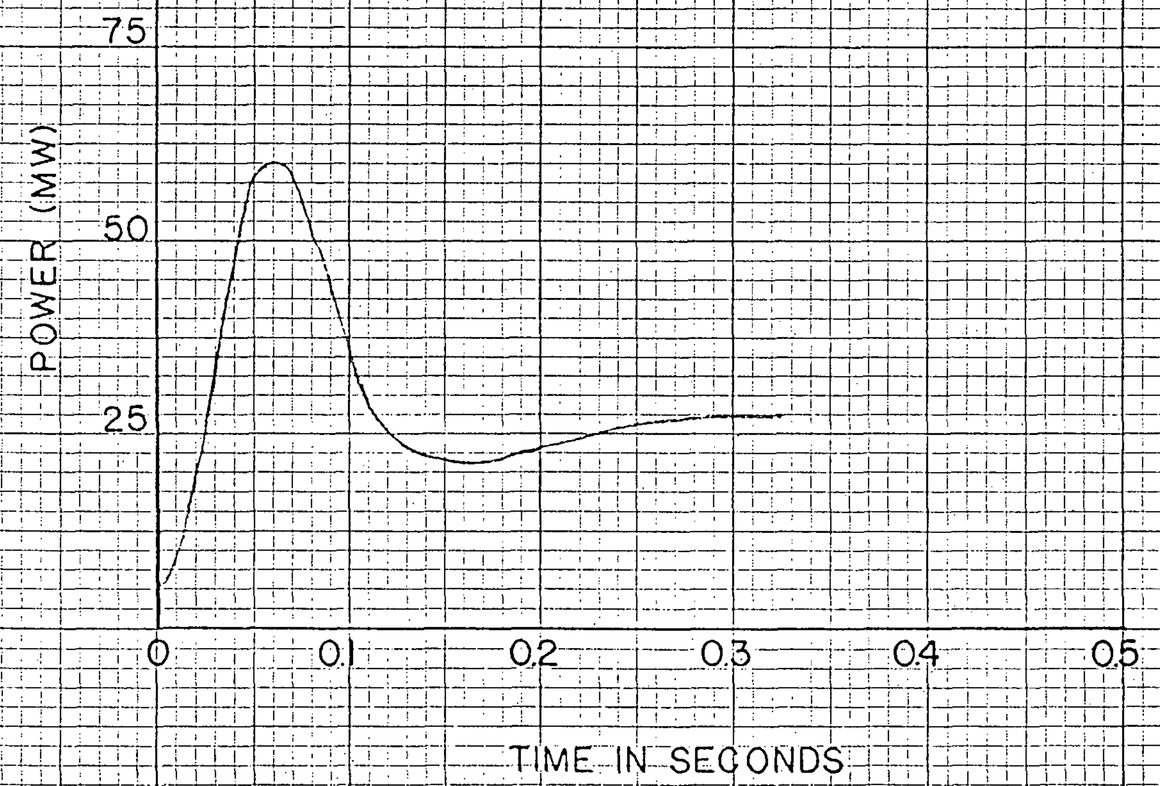

Figure 3.15.4 Chic-Kin Results for 0.010 δk Transient

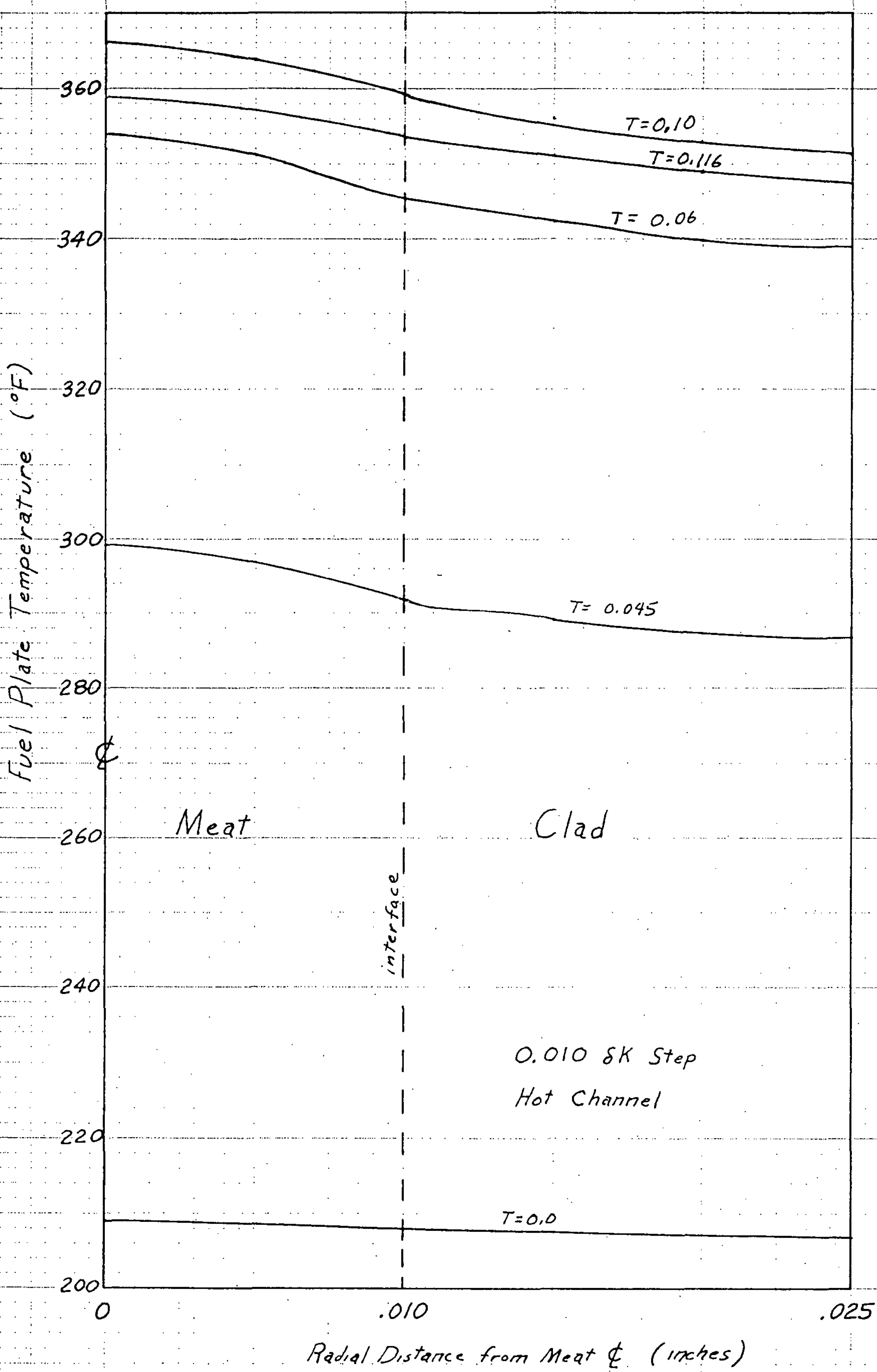

Figure 3.15.5 Radial Hot Spot Fuel Plate Temperature Distribution, 0.010 δk Transient

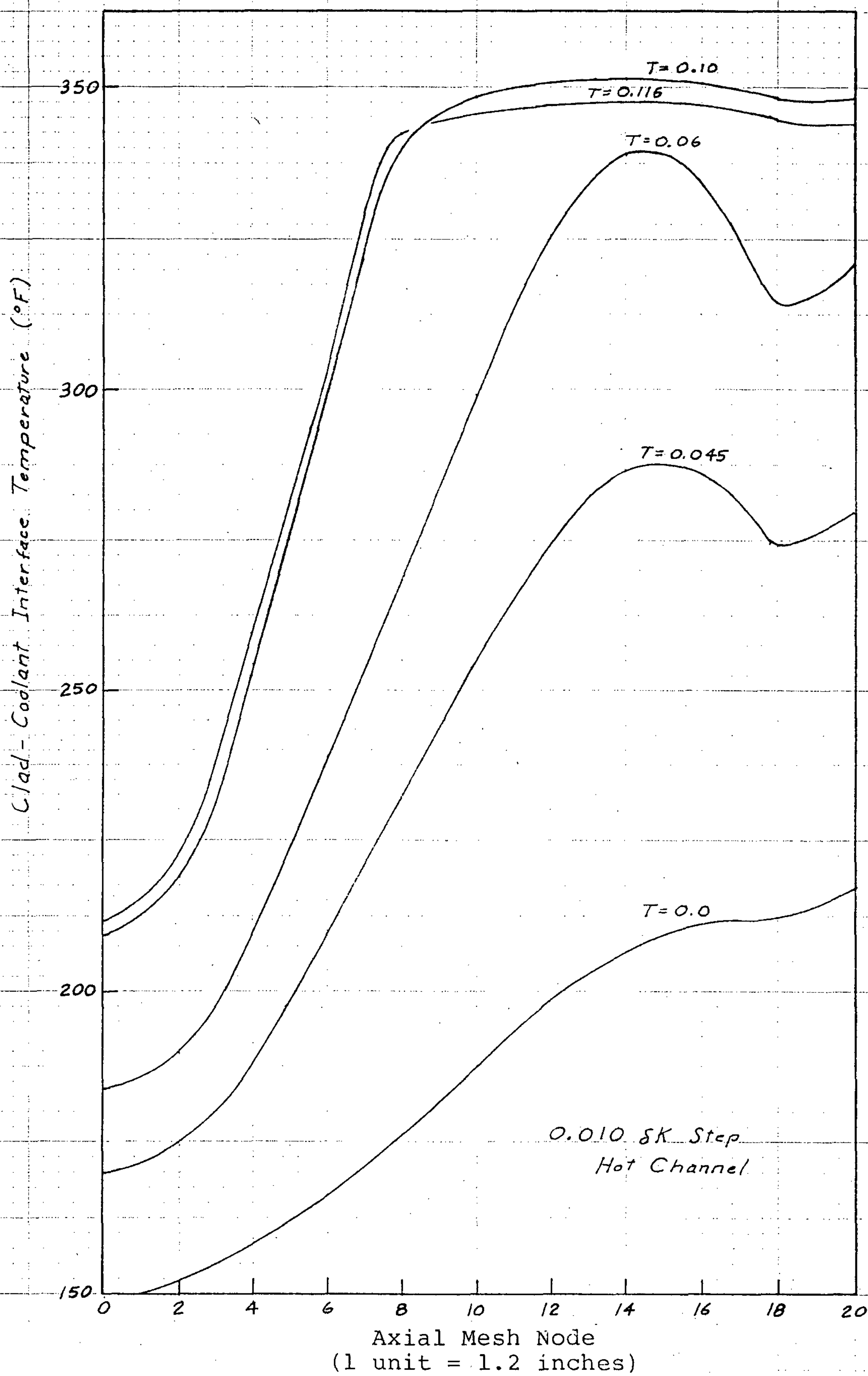

Figure 3.15.6 Axial Hot Channel Plate Temperature Distribution, 0.010 δk Transient

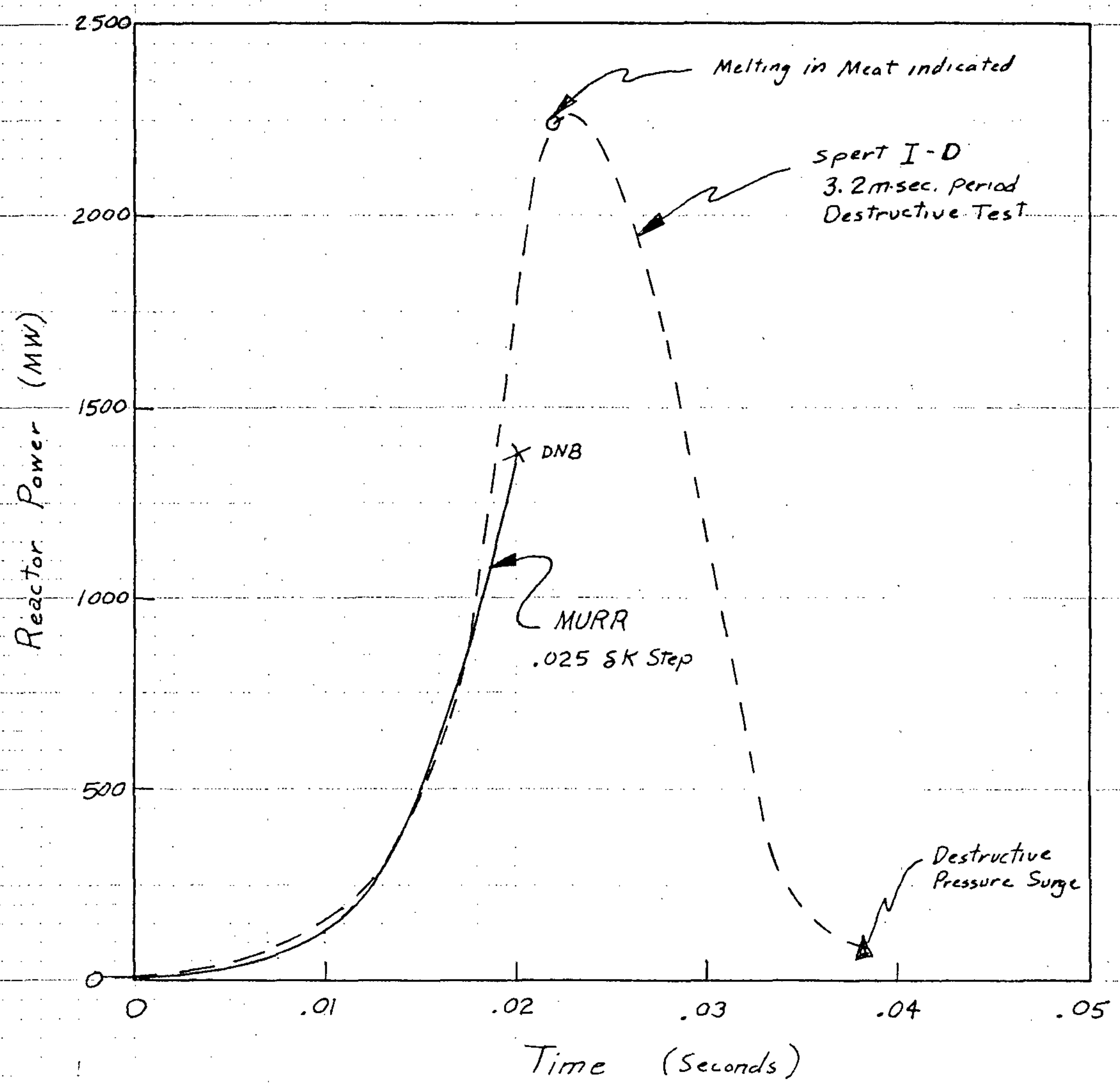

Figure 3.15.7 Chic-Kin Results for 0.025 δk Transient

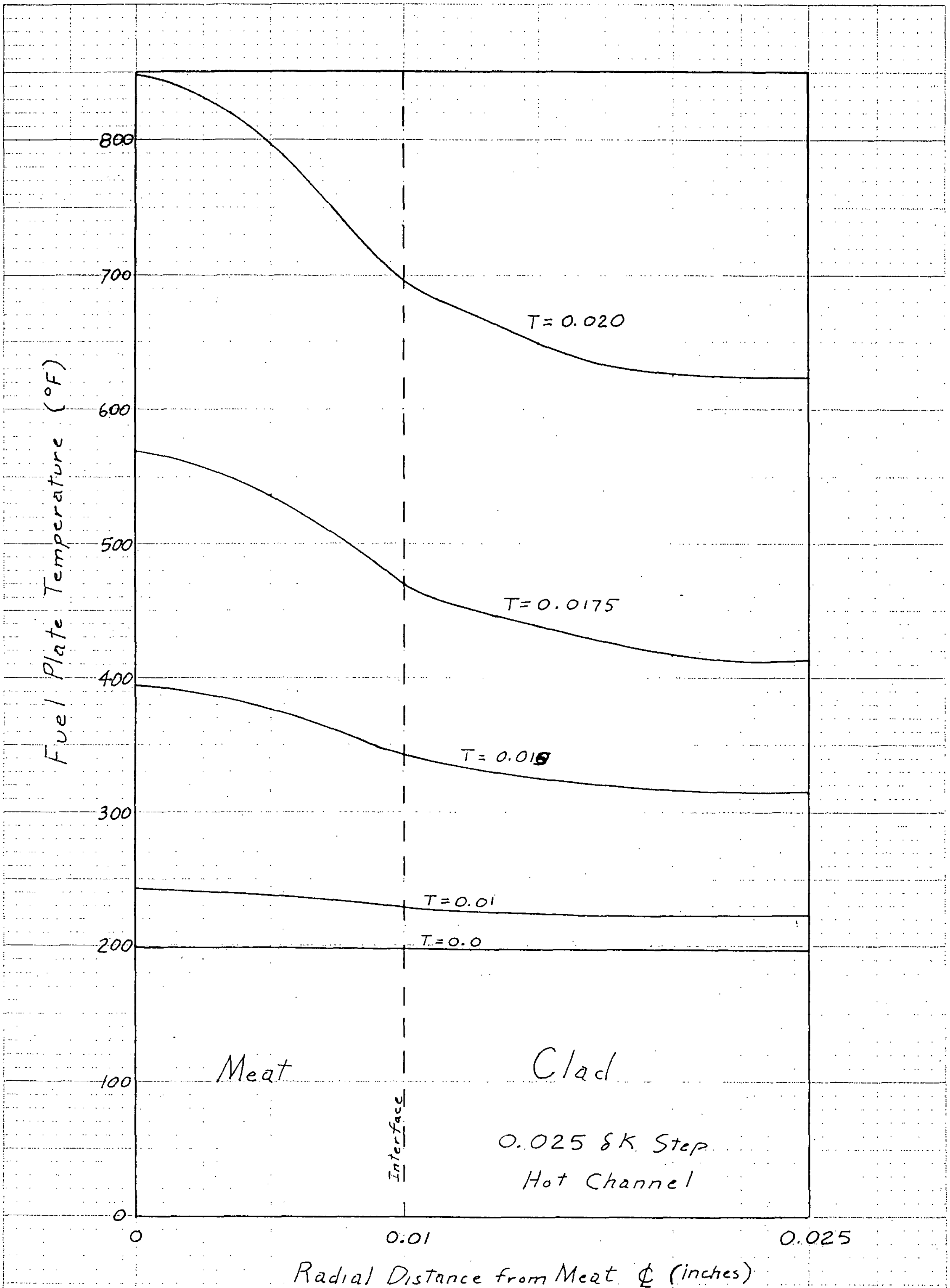

Figure 3.15.8 Radial Hot Spot Fuel Plate Temperature Distribution, 0.025 δk Transient

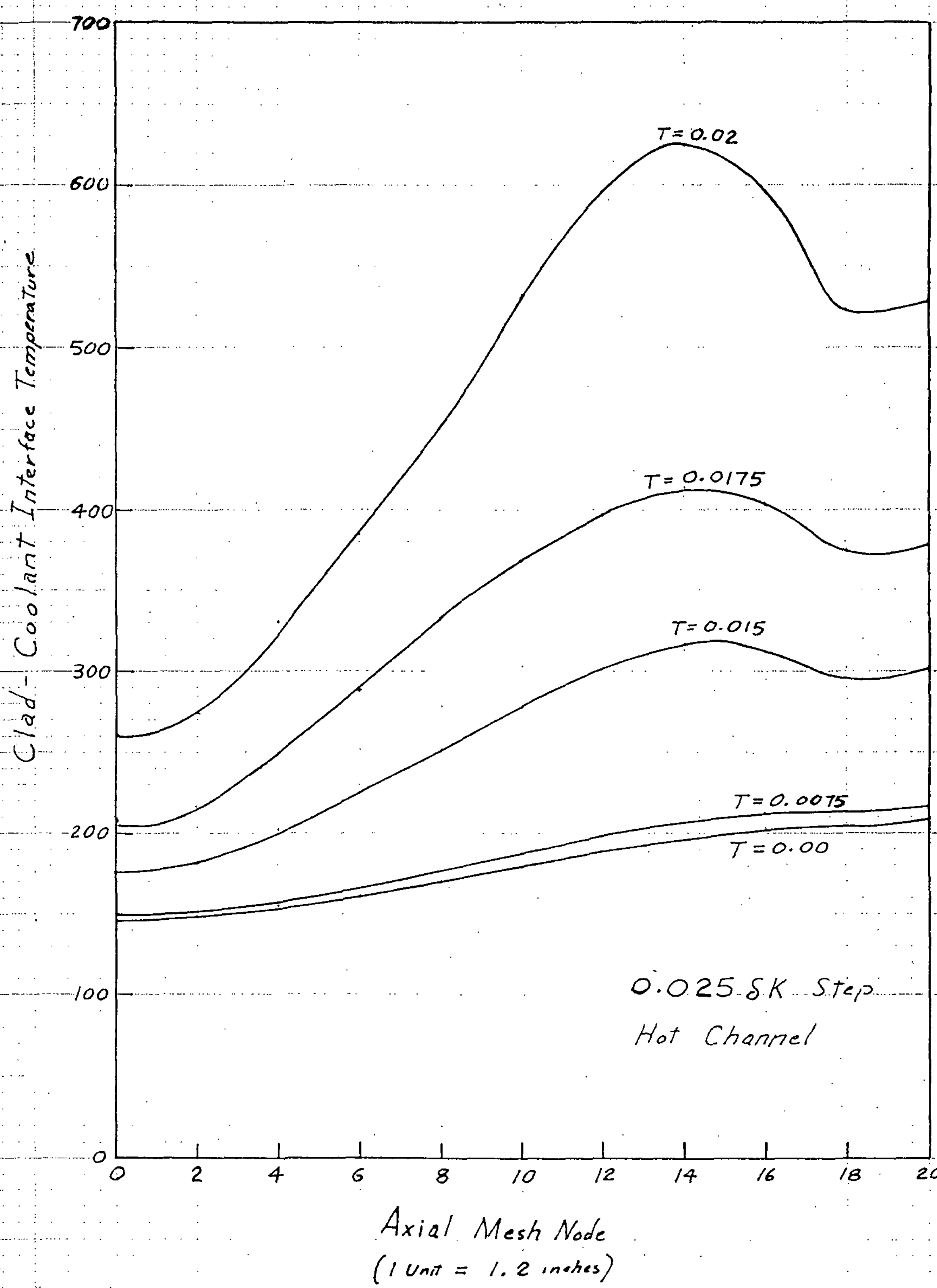

Figure 3.15.9 Axial Hot Channel Fuel Plate Temperature Distribution, 0.025 δk Excursion

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3.16 "Calculate the height to which the pool water will drain if the six inch drain pipe is sheared in the tunnel and the emergency water supply is turned on. Evaluate the reactivity effect of voiding to this height and of voiding the entire pool. Discuss the tests required to assure operability of the emergency raw water supply and the control of any valves in the emergency raw water cooling system external to the reactor which could shut off the emergency raw water supply."

In calculating the height to which the pool water will drain if either the pool six inch supply or return line is completely sheared and the emergency water supply is turned on, several factors must be considered. First, the break must occur between the valve 519A and the pool or valve 509 and the pool. The separation of the two pipe sections must be complete, otherwise there will be no significant loss of water. Second, it must be assumed that the pool convective valve, V547, is closed. If V547 were open it can be closed from the control room.

If the break were to occur in the six inch return line from the pool, drainage would be down through the reflector tank. The resistance to flow presented by the reflector region is such that 13 feet of water would remain over the top of the reflector at equilibrium. In this case the reactivity effect will be zero.

If the break occurs in the six inch line that supplies water to the pool, drainage is through the diffuser. The diffuser is a vertical section of pipe with 36 horizontal rows of seven 5/8" diameter holes. Considered alone, the final pool level would be approximately 10 rows below the top of the diffuser. However, the top of the diffuser is below the top of the reflector tank, but since, in this case, pool water is not draining through the reflector tank, it will remain full. Here again the reactivity effect is zero.

The worst case occurs if both six inch pool water circulation lines are severed, in which case the reflector may become completely drained. If the control rod are fully inserted, k_{eff} will go through a maximum of 0.93 with the reflector region approximately 50% voided, and decrease to a value of less than 0.88 when the reflector region is completely voided. If the reactor were just critical as the reflector region became voided, k_{eff} might reach a value of 1.05 with the reflector region approximately two-thirds voided, but would decrease to less than 1.0 as voiding becomes complete.

In the event of a break, it is important to consider how much time would be available to take corrective action. If pool water were draining out through the diffuser, the calculated time for the free surface to reach the level of the top of the reflector tank is six (6) minutes. With water being supplied to the pool at the rate of 1000 gpm the pool surface would reach the reflector tank in about ten (10) minutes. This is clear that ample time is available to secure the break.

The raw water supply has only one valve, which is the main control valve at the pool edge, inside the containment building. This valve is not locked and is under the control of the reactor operator. There are three post valves exterior to the building between it and the main fire water supply line. Provision is made on these valves enabling them to be padlocked in the open position. These three valves will remain in the locked open position at all times, except as authorized by the reactor supervisor, with the key under the control of the reactor supervisor.

Prior to filling the pool with demineralized water the emergency fill line will be tested to determine the flow rate available under the static head of pressure at the outlet to the pool. This value has been calculated to be 1300 gpm under normal conditions.

Subsequent tests will be to verify that the static head necessary to deliver at least 1000 gpm is available at the outlet to the pool. This will be done every 30 days.

3.17 "Describe the general classes of experiments for which authorization is sought under the initial operating license and discuss the safety aspects and criteria that will be considered during internal review and approval of specific experiments. Discuss the responsibilities of the Reactor Advisory Committee in review and approval of experiments and changes to the facility. Indicate the degree of independence and operating organization that will be maintained by the committee."

Let us consider the first portion of this question, namely: "Describe the general classes of experiments for which authorization is sought under the initial operating license and discuss the safety aspects and criteria that will be considered during internal review and approval of specific experiments."

A number of experiments are currently being prepared for insertion into the reactor. In general, these can be subdivided into two classes. The first class may be called neutron beam experiments. The second may be generally termed neutron irradiation and isotope production. Under the heading of neutron beam experiments can be gathered all those projects under development which will utilize one of the beam ports.

The criteria applied to any reactor experiment may be summarized as follows:

a) criticality considerations;
b) heat generation considerations;
c) shielding considerations; and
d) off-gasing and/or chemical reaction.

The second class of experiments subject to review in light of the criteria enumerated above might be generally classified as "neutron irradiation and isotope production." Under this general classification one finds the grouping of experimental facilities including the flux trap position, the graphite reflector positions, the pneumatic tubes, and any in-pool samples external to the reflector. All experiments of this type are subject to review by the Reactor Manager and the Reactor Health Physicist.

2000

The Reactor Manager critically reviews the proposed experiments to ascertain the reactivity effect, the problem of heat generation, the possibility of sample decomposition, and the general precedence for this type of irradiation through the review of a Reactor Utilization Request (RUR). The Health Physicist reviews the proposal to ascertain whether the experimenter possesses the experience and equipment to cope with the expected radiation level. Further, the Health Physicist determines whether the irradiated material may be safely used in the particular environment suggested by the experimenter. ~~It is pertinent to point out that samples positioned in~~ "Secured and unsecured experiment samples positioned in the flux trap will not be moved during reactor operation. Moveable experiments in the flux trap, graphite reflector, and pneumatic tube positions may be moved in and out while the reactor is operating." ~~those samples in the graphi~~ ~~and out while the reactor is~~ 2000

In any instances where the Reactor Manager feels that he is not qualified to make a judgement pertaining to the safety of a proposed experiment he may refer the experiment to the appropriate subcommittee of the Reactor Advisory Committee. This group will make a finding after subjecting the experiment to an extensive review. Their finding may very well require additional non-reactor investigations or possibly additional safety features not originally envisioned.

The second part of this question has to do with the Reactor Advisory Committee. The question segment referred to reads as follows: "Discuss the responsibilities of the Reactor Advisory Committee in review and approval of experiments and changes to the facility, indicate the degree of independence from the line operating organization that will be maintained by the Committee."

Responsibilities:

The Reactor Advisory Committee is the Committee of the University of Missouri appointed by the University of Missouri-Columbia (UMC) Chancellor to satisfy requirements imposed by the federal government. The University and the Nuclear Regulatory Commission expect this Committee to review and make recommendations concerning experimental and operational activities at the Reactor Facility.

Responsibilities of the Committee are partially set forth in the Technical Specifications portion of the reactor operating license as follows:

1. Review and make recommendations concerning proposed changes to reactor equipment or procedures when such changes have a safety significance, involve an amendment to the operating license including a change in the Technical Specifications, or create an unreviewed safety question as defined by 10 CFR 50.59.

2. Review and make recommendations concerning proposed tests or experiments significantly different from any previously reviewed or which involve an unreviewed safety question as defined by 10 CFR 50.59.

3. Review circumstances of all abnormal occurrences and violations of the Technical Specifications and the remedial measures taken or to be taken to prevent recurrence.

The Committee shall act in an advisory capacity to the Director of the Reactor Facility in matters pertaining to the safe operation of the reactor and with regard to planned research activities and use of the facility building and equipment. It may independently explore policies and procedures as they relate to interaction with other administrative elements of the University and with clients of the Reactor Facility that are not part of the University. It will respond to matters brought before it by the Director, researchers, or other University administrative officials.

The Committee, through its Chairman, may appoint subcommittees consisting of students, faculty, and staff of the University when it is deemed necessary to delegate a part of its responsibilities. Membership on subcommittees need not be limited to appointed members of the Committee. Subcommittees may be authorized to act in behalf of the Committee.

It will be noted from the above that the Reactor Advisory Committee has two very important roles to play. First, it is the jury for ascertaining the safety of any experiments which the Reactor Manager or the Reactor Health Physicist feel are subject to question. In particular, these two individuals will possess and continually develop a precedence of past experiments which they are confident will not present any troubles in the reactor. In those instances where there is any question as to the safety of an experiment they will refer this to the Reactor Advisory Committee for review. The second important function of the Reactor Advisory Committee is to make a periodic review of reactor operations to ascertain that these operations are being carried out in a safe and economical manner.

3.18 "Describe plans for evacuation drills within the exclusion area and the criteria for considering a drill successful."

Our "Exclusion Area Evacuation Plan is presented in paragraphs which follow. We expect to conduct an evacuation drill twice a year. The criteria for a successful drill are listed at the end of this section.

Upon hearing the evacuation alarm all personnel within the exclusion area without preassigned tasks will proceed to points beyond the exclusion area limit according to the following plan.

1. All personnel within the containment building will exit through the east door. They will proceed north via the access road to a point beyond the exclusion area limit.
2. All personnel within the electrical shop, the mechanical shop, the storage and office supply rooms and the health physics office will exit through the east door. They will proceed north via the access road to a point beyond the exclusion area limit.
3. All personnel with the main offices, the lobby, and lounge, the men's locker room, the women's locker room and the dark room will exit through the east door. They will proceed north via the access road to a point beyond the exclusion area limit.
4. All personnel within the meeting room, the library, and the laboratories and offices on the south corridor will exit through the south door. They will proceed east and south via the access road to a point beyond the exclusion area limit.
5. All personnel within the below grade area exterior to the containment building including the demineralizer area, the liquid waste storage area and the heat exchanger room will exit through the south door via the south stairway. They will proceed east and south via the access road to a point beyond the exclusion area limit.
6. All personnel within the laboratories and offices on the west corridor will exit through the west door. They will proceed south via the driveway and east and south via the access road to a point beyond the exclusion area limit.

7. All personnel within the conference room and the laboratories and offices on the north corridor will exit through the north door. They will proceed east and north via the access road to a point beyond the exclusion area limit.
8. All personnel within the mechanical equipment rooms and personnel the roof top or in any tower will exit through the north door. They will proceed east and north via the access road.
9. All personnel within the cooling tower and pipe tunnel will exit through the east door of the cooling tower. They will proceed east and south via the access road to a point beyond the exclusion area limit.
10. All personnel within the USDA facilities will exit through the nearest building exit. They will proceed east through the parking lots and north via the access road to a point beyond the exclusion area limit.

Preassigned Tasks

See the EOP for preassigned tasks of the facility director, laboratory supervisor, reactor supervisor, associate reactor supervisor, reactor operator, assistant reactor operator and reactor health physicist.

The mechanical technician will ascertain that the mechanical equipment rooms and below grade area exterior to the containment building are evacuated and secure in addition to his tasks as indicated in the EOP.

The laboratory supervisor's secretary will notify the USDA main laboratory building of the exclusion area evacuation by telephone. The electrical technician will ascertain that the USDA facilities are evacuated and secure in addition to his tasks as indicated in the EOP.

Criteria for Successful Drill

1. Evacuation alarm is heard by all personnel within the reactor facilities.
2. Total evacuation is completed in less than 15 minutes.
3. Preassigned tasks are completed in less than 15 minutes.

3.19 "Describe controls to prevent direct irradiation hazard or leakage of radioactivity from the pneumatic tube system both inside and outside the containment."

The pneumatic tube system, from send-station to in-pool terminal, is a complete closed tube. At no point in this system is there air leakage in or out of the tube other than at the point of sample insertion. The send-receive terminals in every case (there are seven such terminals) are located in air flow fume hoods within laboratories designed specifically for working with radioactivity. Every hood has a linear face velocity of 100 feet per minute or more. All air exhausted from these hoods is filtered through HEPA filters. The only exit point for contaminated air contained within this tube system is at the pneumatic blower located below grade at a position adjacent to the east wall of the containment building. During normal operation the exhaust air from the blower is exhausted through the 4-inch exhaust line back into the facility exhaust stack plenum.

The approximate effluent volume from these blowers is 200 cubic feet per minute.

It is to be expected that argon-41 will be generated in the pneumatic tube system by activation of the air used to propel irradiation containers through the system. However, it is to be noted that 100 cubic feet per minute effluent from the pneumatic tube system achieves a dilution factor of 200 prior to release of the air from the exhaust stack. Argon effluent levels and expected dilution were defined in earlier reports.

Direct radiation hazards from the operation of the pneumatic tube system is controlled by: (a) distance; and (b) the rapid rate of travel of the irradiation container through the system. In addition, the use of the pneumatic tube system by an experimenter is subject to administrative controls that can be initiated either by: (1) the principal experimenter; (2) the reactor operator; or (3) the health physicist.

With respect to the distance control, it should be noted that those parts of the pneumatic tube system located in the reactor pool enter the biological shield approximately four feet below the pool surface. At that point each tube penetrates the biological shield and drops immediately down to a point approximately 15 feet above the beamport floor, or about 9 feet above the head of a person standing on the beamport floor. From the containment building the tubes pass through a steel plate and exit into the outer laboratory structure. Immediately after passing through the steel plate the transport tubes curve upward to the ceiling of the mechanical equipment room located in the laboratory building. At this point they are approximately 13 feet off the floor and are located behind the inner walls of the laboratories. This space is not normally occupied.

From this location they divide into north and south routes (south no longer used, terminal blocked at basement panel and in inner corridor) that carry two tubes in each direction. These tubes then proceed to the west on the north side of the containment building. Each tube branches off from the main tube and penetrates the laboratory wall at a point above the drop ceiling, from whence it proceeds down into the receive terminal located within the air flow fume hood of the laboratory. Direct radiation hazards within the hood will be minimized by the erection of a lead brick barrier or by the use of a receiving pot having at least a 2-inch thick lead wall.

The "rate-of-travel" control also minimizes direct radiation hazards. The irradiation container will move through the tube at a speed of 30 to 45 feet per second. This speed is advantageous in that a localized direct radiation hazard will exist only temporarily at any point on the tube as the container is moved through the tube from the reactor to the laboratory station.

The administrative control involves the utilization of the pneumatic tube system by an experimenter. It is intended that samples will not be inserted into the system for long periods of time nor in a quantity able to produce large amounts of radioactivity. It is more generally the case that an experimenter seeks to produce measurable quantities of radioisotopes for the identification of a particular isotopic species. A case in point would be their utilization in activation analysis. In such technology, one cannot cope with curie quantities of gamma-emitting isotopes since it is desirable to produce only enough radio-activity so that the analysis can be completed by the use of a conventional counting system.

A further administrative control can be exerted if an unusual operation occurs, e.g., the sticking of a container in any portion of the transport tube. The experimenter is immediately cognizant of the fact that the container has not been received at the laboratory station because his control station on the wall of the fume hood will not register the flight of the container as he tries to discharge it into the hood. At present, the experimenter is to notify the reactor operator of this malfunction through the laboratory-reactor console communication system. At that time, the reactor operator can advise the health physicist or the principal experimenter of the condition and the necessary radiation safety controls put into operation. If the condition is considered to be of extreme hazard, the principal experimenter and/or health physicist can order a complete cessation of operations in that part of the laboratory.

In summary, these controls serve to limit the direct radiation hazard from the pneumatic tubes. The distance of the tubes from personnel working in the area, the speed at which the sample container is transported through the system, and administrative controls over the operation of the system by an experimenter will minimize the problems that could be experienced in operating the system without such controls.

3.20 "Provide sufficient information, including schematics as applicable, to verify that the containment isolation system possesses adequate redundancy and reliability characteristics to assure that no single component or circuit failure could render any portion of the system inoperative."

The devices and components which operate to accomplish containment building isolation are arranged as shown in Figure 3.20.1. Those factors which assure adequate redundancy and reliability characteristics are discussed in the following paragraphs.

(1) All relays which perform isolation functions by energizing have been paralleled by a second relay in such a way that if one or the other fails to energize (assuming that they do not fail simultaneously), the unaffected relay will perform as necessary to accomplish isolation.
(2) All relays which de-energize to perform a function are assumed to be fail safe.
(3) The solenoids which operate the quick closing 16" and 4" exhaust valves are de-energized to shut valve, therefore, assumed fail safe.
(4) The containment building exhaust line contains two 16" butterfly valves. The 16" A valve is air/open--spring/close (assumed fail safe) while the 16" B valve is air/open--air/close. Air to the valve operators is supplied from the main facility compressor and the emergency compressor. The 16" B valve has an additional dedicated emergency air compressor. Adequate redundancy is attained by the fact that there are two separate isolation valves, and the 16" B valve has redundant air sources.
(5) Power to operate the supply and return fan damper doors comes from both normal electrical supply and emergency motor-generator supply, therefore, they will operate in case of a simultaneous isolation and loss of electrical power. Additionally, there are air-piston operated backup doors in the air supply and return plenums which are air/open--gravity/close. These backup doors close automatically with any automatically initiated containment building isolation. They will fail closed upon loss of air or electrical power.

In asking how much reliance can be placed on the reactor containment isolation system, two important factors must be born in mind.

First, the isolation system will be regularly tested. The operation of the complete system is tested prior to every reactor startup. The operation of the motorized isolation doors will be tested at least once weekly if the reactor is operating continuously. A "fault" record will be maintained and immediate corrective action taken upon detection of any malfunction.

Second, at least two major failures must occur before the building isolation system cannot perform its intended function. Containment building isolation is needed only in the event that fission products or other large sources of radioactivity have been released into the containment building. This circumstance requires a major system failure; an event which itself is highly unlikely. A failure of the isolation system must have occurred <u>at the same time</u> before it can be said that the system has failed to perform its functions.

Regular testing of the containment isolation system, together with a sound program of preventative maintenance, will assure that the likelihood of a failure to isolate at any given time is extremely small.

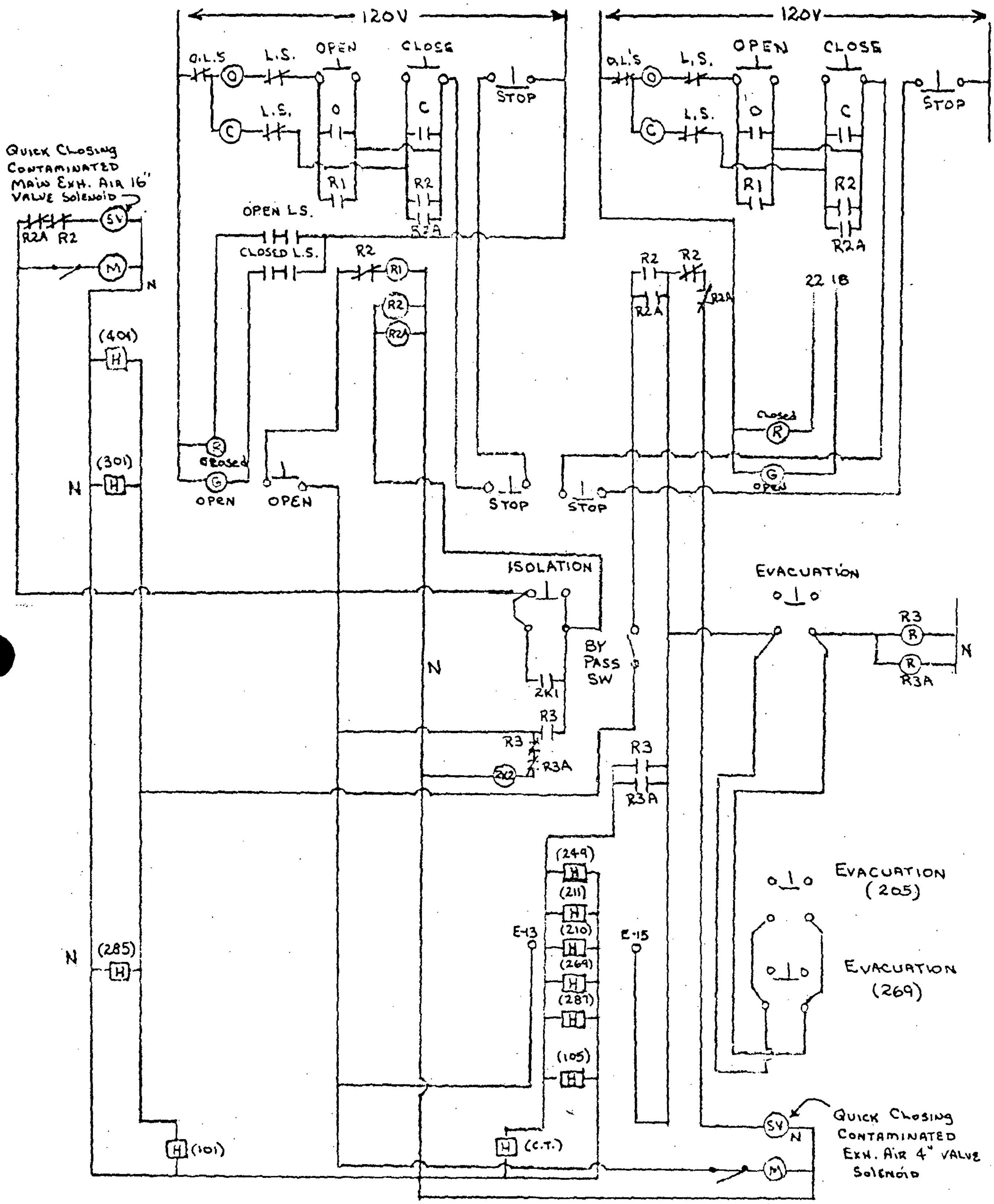

Fig. 3.20.1 Reactor Isolation and Evacuation System

3.21 "Describe the design used to assure that: (1) the master trips for reactor isolation on high radiation level cannot be adjusted from the instrument panel, or (2) the meter setting cannot be turned off scale to make the trips inoperable."

The system which provides automatic reactor isolation on high radiation level is an area radiation monitor of standard design and frequent nuclear industry application. An isolation trip occurs when a count rate comparator which monitors the output of a particular detector senses a preset trip level. The preset trip level is set by adjusting a potentiometer located behind the front cover plate of each channel.

Reactor operating procedures require that the reactor isolation trip settings be checked routinely as part of the startup checklist. In no case may a trip setting be deliberately changed, if the reactor is in operation, unless the change is authorized by the Reactor Manager.

The Eberline Radiation Monitoring System (RMS II) is on the vertical console immediately in front of the reactor control console and in full view of the reactor operators.

If a reactor isolation trip setting was changed, the change would be observed and would have to be deliberate, since access to the trip setting potentiometer requires the use of screwdriver to reach through the front cover plate of each channel.

3.22 "Submit drawings showing the general arrangement of the ventilation systems and associated dampers and controls for the containment and laboratory areas."

All fresh air for both the laboratory and the containment building enters through dampers on the north and south faces of the east tower and through the dampers of each RTAH unit. Fresh air entering through the north and south dampers passes into receiving plenums and through steam preheat coils.

The fresh air then passes through a dust filter, moving on through supply fan No. 1 (SF-1) heating and cooling coils, and finally into the double duct air distribution system.

Fresh air from the north and south RTAH's passes through chill water coils for air conditioning and secondary system reactor waste heat coils for heating, and is distributed via ceiling grills in the north and south corridors. Fresh air for the containment building passes up from the receiving plenum and is mixed with containment building return air. Containment building return air, driven by return fan RF-2, enters the east tower through the motorized isolation door No. 505. The mixed return plus fresh air passes through a dust filter, cooling coils, heating coils, and motorized isolation door No. 504 to supply fan SF-2. From supply fan (SF-2) the air is distributed throughout the containment building.

Laboratory building air and containment building air are never mixed. The supply and return fan pairs are interlocked so that if SF-2 is off, so is RF-2, however, RF-2 may be off with SF-2 on. If either of the motorized isolation doors (504 or 505) are closed, SF-2 and RF-2 are both off.

Exhaust air from both the containment and the laboratory building enters the atmosphere through either exhaust fan EF-13 or 14 and then through the exhaust stack located in the west tower. Either one or the other (EF-13 or EF-14) exhaust fan is on at all times. The other fan is a standby. Failure of the on-line fan automatically activates the standby and also activates a warning light in: (a) the reactor control room; and (b) in the facility lobby. Failure of both fans activates an alarm buzzer only in the reactor control room.

Laboratory exhaust air is picked up at the fume hoods, passed through absolute filters, and delivered to EF-13 or 14.

Containment building exhaust air enters EF-13 or 14 through a 16 inch line with two (2) quick-closing isolation valves. This exhaust air is picked up from beamport experiments storage ports, beamports, the thermal column, the Nuclepore film shield box, and the pool surface air sweep.

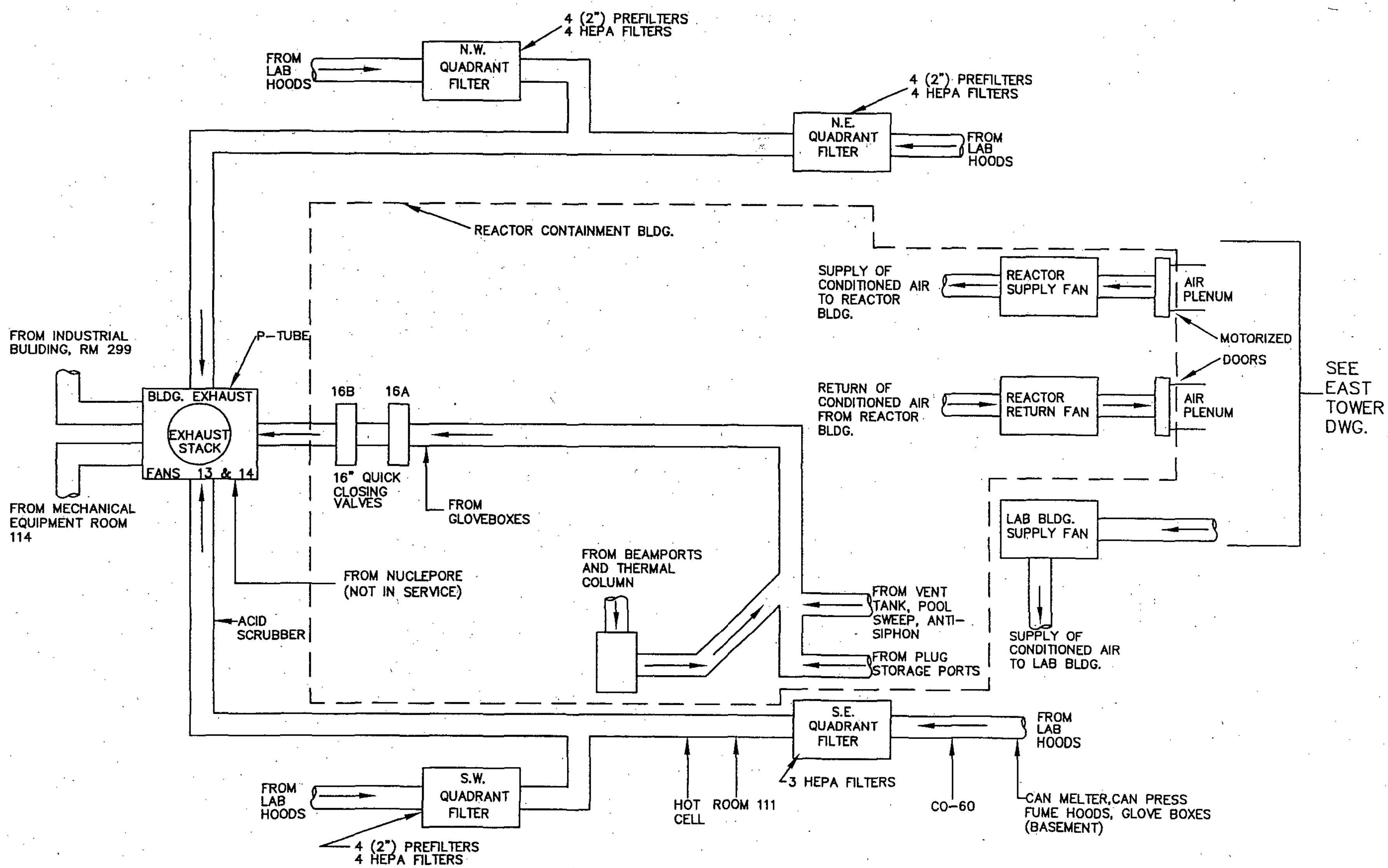

FIG. 3.22.1

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REVISION DATE: 12/19/02

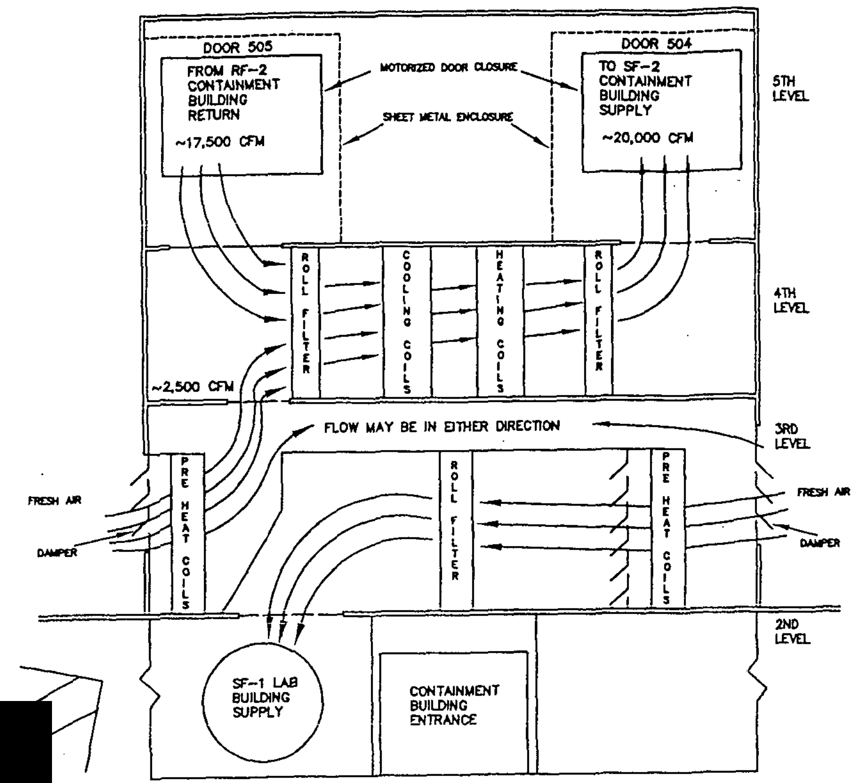
UNIVERSITY OF MISSOURI-COLUMBIA
FACILITIES OPERATION
research reactor facility

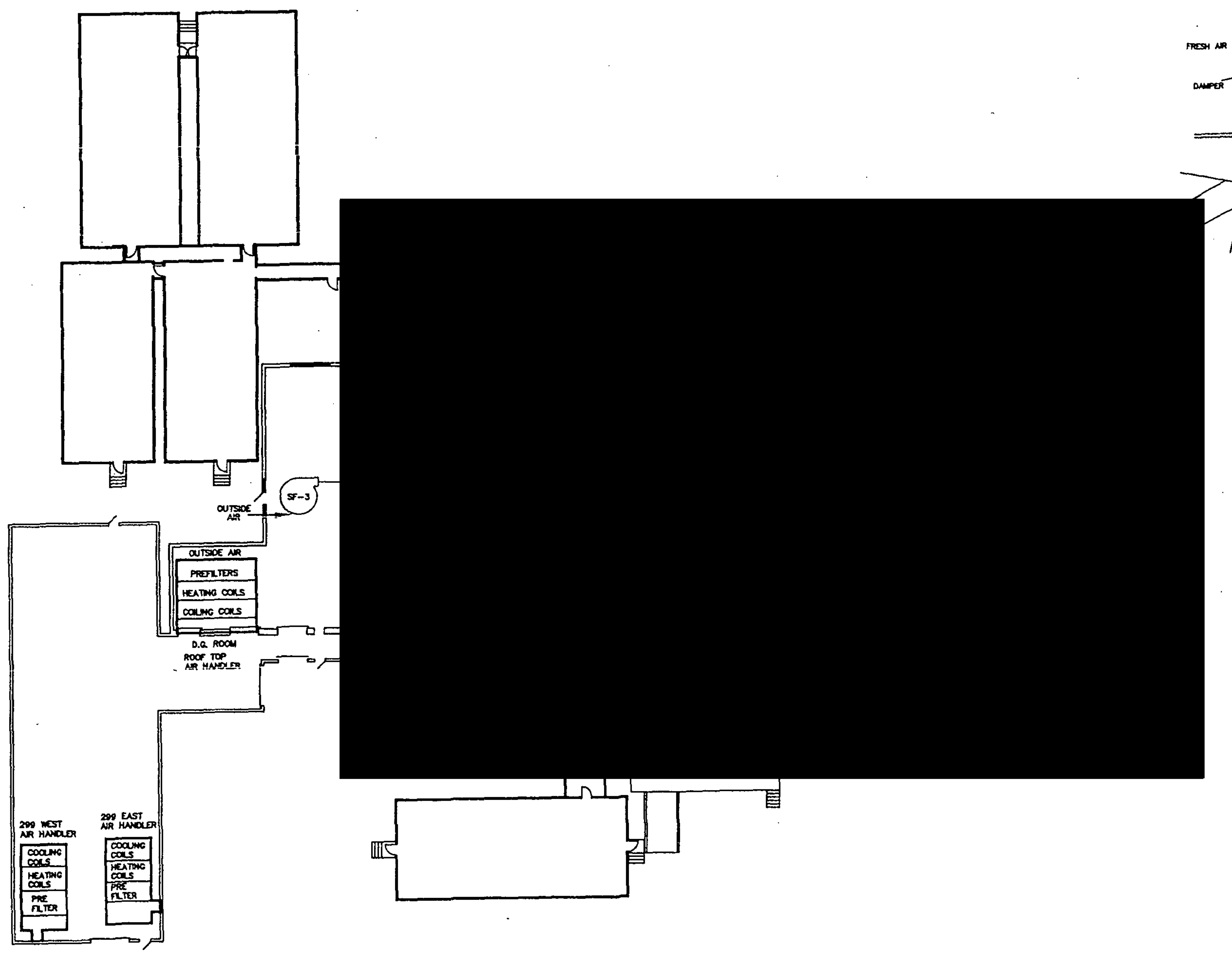
DIAGRAM OF LABORATORY AND CONTAINMENT BUILDING VENTILATION SYSTEM

MURR NUMBER: 2420
SHEET 1 of 1

2002

FIGURE 3.22.2





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REVISION DATE:

UNIVERSITY OF MISSOURI-COLUMBIA
FACILITIES OPERATIONS
research reactor facility

MURR SUPPLY AIR SCHEMATIC

MURR NUMBER: 1125

SHEET 2 OF 3

3.23 "Describe the means used to assure that the reactor can be shutdown if all safety rods fail to drop into the core region."

There is no installed emergency shutdown system which is capable of accomplishing a shutdown with all rods stuck out. A number of possible systems are being considered including (a) a poison sleeve positioned around the flux trap above the core region; and (b) a boric acid-borax solution injector system. Either of these methods would suffice but engineering design and cost out has not been done.

It is felt that start-up plus calibration efforts will more clearly define the need for a back-up shutdown capability. To illustrate, the control rods will be exercised through fifty rod drops prior to critical loading. These drops should demonstrate the capability of the system to stay within tolerance. There should be no scratching or binding of the rods or the rod gaps.

At intervals after start-up rod drop times and rod tolerances will be measured. These data will be carefully scrutinized in an effort to forestall the development of any rod drop problem.

The staff of the reactor facility will investigate, design and procure prices for the installation of a suitable emergency shutdown system. Upon completion of this task the design will be submitted to the AEC.

Appendix II

Heat Flux Correlations
Applicable to Section 3.7 of this Report

The following is a listing of the four burn-out correlations originally considered for use in the steady-state burn-out calculations. All are usable for pressures of 65 psia and 100 to 150° of subcooling and channel flow velocities of 11.55 feet per second. The Bernath correlation was finally selected as one which is both reliable and conservative.

1. Lowdermilk Correlation (1)*

$$\phi_{DNB} = \frac{270\ G^{.85}}{D^{.12}\ (L/D)^{.85}}$$

G = 2.545×10^6 #/hr-ft^2
(11.55 f/sec @ 150°F)

L/D = 150

D = Hydraulic Diameter = .16/12 feet

ϕ_{DNB} = 3.521×10^6 BTU/hr-ft^2

2. Gunther's Correlation(2)

ϕ_{DNB} = $0.135\ V^{0.5}\ \Delta T_{sub}$ BTU/in^2 sec

V = 11.55 f/sec

ΔT_{sub} = $T_{sat}\ T_{liq} = 148$

ϕ_{DNB} = 3.521×10^6 BTU/hr-ft^2

3. Schrock Correlation(3)

$$\phi_{DNB} = \left[12300 + \frac{67V}{D^{.6}}\right] \left[102.5 \ln P - 97 \left(\frac{P}{P+15}\right) + 32 - Tw\right]$$

V = 11.55 f/sec

Tw = 150°F

P = 65 psia

D = .16/12 feet

ϕ_{DNB} = 5.237×10^6 BTU/hr-ft^2

4. Bernath Correlation (5)

$$\phi_{DNB} = \left[5710 \left(\frac{D'_e}{D'_h}\right)^{0.6} + 48 \frac{\overline{U}_L}{(D'_e)^{0.6}}\right] \bullet \left[102.6 \ln P' - 97.1 \frac{P'}{P'+15} - \frac{\overline{U}_L}{2.22}\left[\frac{D'_h}{D'_e}\right]^{0.6} + \bar{t}_b\right]$$

D'_e = hydraulic diameter = 0.16/12 ft

D'_h = heated perimeter

$\overline{U}_L$ = 11.55 f/sec

P' = 65 psia

ϕ_{DNB} = 1.975×10^6 BTU/ft^2-hr

*References are in the main text, Section 3.7.

Appendix III

Step Reactivity Insertion Analysis: Digital and Analog Simulation Details

A) Digital Analysis

Input data for the Chic-Kin code will be discussed here. With the exception of the initial reactivity step and certain transient heat transfer parameters, average channel input data was the same for each run. Input data for each of the hot channel runs were the same, except for tabulated transient power history for each run was obtained from the average channel runs. Hot channel and hot spot factors used were those used by Internuclear in the design of the MURR.

In making up the input data it was necessary to give the average volumetric thermal source strength. This calculation was performed in the following manner:

At 5 MW, the average heat flux is given as:

$$\overline{\phi} = \frac{1}{2}(172{,}244) = 86{,}122\ \mathrm{BTU/hr-ft^2}$$

The meat thickness is 0.020 inches or 0.001667 feet. Thus, the volumetric thermal source strength is:

$$q^* = \frac{\overline{\phi}}{.001667} = 5.16 \times 10^7\ \mathrm{BTU/hr-ft^3}$$

It was assumed that 93% of the heat was generated in the core region. It was further assumed that 5% of the core region heat was generated directly in the coolant-moderator by neutron collision and gamma absorption. Applying the appropriate factors gave the (meat) volumetric thermal source strength, q*, as:

$$q^* = 4.92 \times 10^7\ \mathrm{BTU/hr\text{-}ft^3}$$

It has been assumed that q (r,z,t) is separable as:

$$q(r,z,t) = q^* q_1(r)\, q_2(z)\, q_3(t)$$

such that

$$q_{(t)} = \frac{q^* q_3(t)}{(1+r_w)\int_{fuel} q_1(r) q_2(z)\, dVol}$$

Where q_1 (r) and q_2 (z) are radial and axial distribution functions and are defined such that the integral in the denominator of the above equation is unity. The fraction of heat delivered directly to the coolant moderator, r_w, is 0.05. q_3 (t) is the instantaneous power relative to the initial reference power. The distribution function, q_2 (z), was taken from Internuclear design data; q_1 (r) was assumed unity in the meat region and zero in the cladding.

Data for the reactor kinetics portion of the program was evaluated using Keepin and Wimett parameters for a uranium-235 reactor. Each β_i (the i^{th} group delayed neutron fraction) was multiplied by the ratio of the effective delayed neutron fraction to the number fraction (0.00738/0.0064). The prompt neutron lifetime was entered as 5.7×10^{-5} seconds, which is the new value computed by General Electric.

Power feedback effects considered in this study are limited to coolant-moderator density effects and reduced flow area due to plate thermal expansion. The moderator density coefficient of reactivity was computed from the Internuclear value for void coefficient. The density coefficient was subsequently checked by comparison with the temperature coefficient and moderator density change over a 20°F change in temperature.

$$\Delta k/k = -0.233 + 0.00233 \text{ (\% Density)}$$

$\Delta k/k$ was assumed to change linearly with percent density over the small changes in density encountered.

No account was taken of the Doppler fuel temperature coefficient. This would be an additional negative coefficient. This would be an additional negative coefficient of reactivity of small significance for the present problem. Reactivity effects due to the island temperature coefficient were neglected in the digital runs. While the is.and is a region of positive temperature coefficient, the relatively small fraction of heat delivered to the island renders the overall reactor power coefficient negative. Further, the thermal time constant involved in heating the island water is an order of magnitude longer than that of the core region and so, for the fast transients considered here, should indeed render the island effects negligible.

Hydraulic parameters were evaluated for an inlet water temperature of 140°F and an initial flow rate of 11.55 ft./sec. Table III-1 gives pertinent hydraulic dimensions of the typical flow channel. Channel roughness was taken as midway between a smooth channel and commercial cast iron pipe, in agreement with the design value reported by Internuclear.

Table III-1 Hydraulic Parameters

Channel length	24.00 inches
Plate half thickness	0.025 inch
Channel half thickness	0.040 inch
Channel hydraulic diameter	0.160 inch

Inlet and exit plena were redefined for the purpose of the Chic-Kin code to be the inactive portions of the fuel plates at either end of the fuel elements. This gave an area ratio of unity and inlet and exit loss coefficients of zero. The pressure drop across the active core region thus defined, was computed to be 2.463 psi, corresponding to a 3.7 psi drop across the overall core region at normal flow conditions. When contraction and expansion losses were considered for the ends of the fuel elements, substantial agreement with the 3.7 psi design value is obtained. The core pressure drop was held constant throughout the problem, as was recommended by the author of the Chic-Kin code.

The Chic-Kin code was provided with the system pressure, the burn-out heat flux (ϕ_{DNB}), the fraction of heat to voids (f_r), bubble collapse time (τ), and a minimum heat transfer coefficient. For this problem, the system pressure was taken as 65 psia, and was held constant at this value for all runs. In computing the burn-out heat flux, several prominent correlations were studied. It was felt that the Gunther correlation provided the most conservative burn-out heat flux value consistent with engineering practice. The burn-out heat flux was evaluated as 3.521×10^6 BTU/hr-ft^2.

The bubble collapse time (τ), was a quantity which could not be readily computed. A value for τ of 0.1 seconds was taken as a reasonable value. This value was based on precedent values as noted on page 41 of the Chic-Kin code user's manual, where a comparison is made with published SPERT data. A parameter study by Dr. R. P. Morgan at Argonne National Laboratory indicated that a τ of 0.1 seconds results in more rigorous conditions for the reactor than is true for smaller values of τ. It appears quite improbable that τ would ever exceed 0.1 seconds and that the use of this value is quite conservative.

In computing the fraction of heat flux to voids (f_r), use was made of Figure 8 of the Chic-Kin user's manual. This figure was derived from the work of Schrock, et al., and was originally reported in SAN 1007. An estimate for f_r was made at atmospheric pressure and for a period determined in each run by the stable period associated with the reactivity step. These estimates were made for 140° of subcooling, based on a moderator-coolant average temperature of 150°F. The value thus computed was the corrected for pressure by multiplying by the ratio of specific volumes of vapor:

$$f_r(65) = f_r(14.7)\,\frac{V_g(14.7)}{V_g(65)} = 4.09\, f_r(14.7)$$

For a 0.025 δk step reactivity input, f_r was found to be 0.0192. It was necessary to recalculate f_r for each transient run.

The minimum heat transfer coefficient, HMIN, was computed using the relationship given in the user's manual:

$$\text{HMIN} = \frac{\phi s}{\Delta t} = \frac{\sqrt{K\rho c}}{\sqrt{p}}$$

where:

p = stable period (e.g., 3.22 m-sec for 0.025 δk step)

K = 0.386 BTU/hr-ft-°F

ρ = 62.1 lb/ft^3

c = 1.0004 BTU/lb-°F

Upon substitution of the above values, HMIN was found to be 5180 for a 0.025 δk step input. It was necessary to recompute HMIN for each transient.

The axial reactivity weighting function was supplied to the computer for use in computing spatial dependence of the power feedback effects. This function is determined by the importance function or adjoin flux, which, for a one lethargy group approximation, is proportional to the square of the flux. The axial reactivity weighting factor was obtained for each of the axial mesh points, Z(j), for the case of the rods half inserted. The axial power distribution P(j) for the above case was taken for each mesh node and squared. To find the axial reactivity weighting factor, ALFA(j), it was necessary to observe that:

$$\sum_{J} \mathrm{ALFA(j)} \equiv 1.0$$

Thus if:

$$K \sum_{J} P^2_{(j)} = 1.0 = \sum_{J} KP^2(j)$$

then:

$$KP^2_{(j)} = \mathrm{ALFA(j)}$$

It was then necessary to sum up all the values of power squared, $[P(j)]^2$, corresponding to the several mesh nodes, and divide this sum into unity. Using the values obtained from the local to average power distribution curve, the weighting factor proportionality constant was found to be 0.04277. Thus ALFA(j) was given by:

$$\mathrm{ALFA(j)} = 0.04277\ [P(j)]^2$$

Values for ALFA(j) were computed for each axial mesh interval and were presented to the computer as tabulated data.

The above data was punched and compiled as an input card deck and was run with the Chic-Kin program. The results were tabulations showing instantaneous power distribution, fuel plate temperatures and reactivity compensated.

B) <u>Analog Simulation</u>

Due principally to the very short thermal time constant of the MURR fuel plate, it was necessary to use extremely small time increments for computer stability. This meant that excessively long computer runs were necessary to obtain information for more than the first second of the transient. Thus, the short thermal time constant, while a fortunate design feature, made digital analysis very expensive for post-peak study. In this light, the analog computer appeared as an attractive alternate to expand knowledge of after-burst phenomena.

In order to make the analog simulation, it was necessary to delete from the analysis all spatial dependence of power feedback effects. This meant that core averaged parameters had to be employed in order to develop power feedback mechanisms. This also precluded an accurate determination of spatial dependent variables such as temperatures and heat fluxes. It was, however, a simple matter to include the island in the analysis.

A basic description of the analog simulator begins with an analog computer solution of the "point form" of the reactor kinetics equations. To provide for power feedback, computer circuits were devised to simulate core and island temperature changes and core voiding. These simulated physical effects were translated into reactivity effects through their appropriate coefficients. Two Electronics Associates Inc. type TR-10 analog computers were used simultaneously in order to provide sufficient capacity for the simulation. A more detailed description of the computer-simulator will follow.

The reactor kinetics simulator was developed from the point form kinetics equations using a single averaged delayed neutron group:

$$\frac{dn}{dt} = \frac{\delta k}{1} n - \frac{\beta}{1} n + \lambda c$$

$$\frac{dc}{dt} = \frac{\beta}{1} n - \lambda c$$

where:

$\beta = 0.00738$ (delayed neutron fraction), 1 = 5.7 x 10^{-5} seconds (prompt neutron lifetime),

λ = 0.090 (average delayed neutron precursor decay constant),

n = neutron density (proportional to power),

c = delayed neutron precursor concentration.

The value of 0.090 given for λ was derived by a curve fitting technique. For this, the Bode plot of a one group reactor transfer function was fitted as closely as possible to the transfer function of a six delay group reactor.

Reactivity effects due to core and island temperature variation were treated by means of the Van Rennes technique. Basically, this is a linearized transfer function relating temperature change to power change in a system where there is coolant flow. Using Laplace transform terminology, the Van Rennes transfer function is:

$$\frac{\overline{T}(s)}{P(s)} = \frac{1/mc}{s + {}^{2Fc/m}}$$

where:

$\overline{T}$ is the average moderator-coolant temperature, (F°),

m is the total moderator mass, (lb.),

c is the specific heat of the moderator material, (BTU/lb),

F_c is the coolant-moderator mass flow rate, (lb/sec),

P is the reactor power applied to the coolant, (BTU/sec.).

Both the core and the island temperature effects were represented by separate applications of the Van Rennes formula. Table III-2 lists core and island parameters used in computing the temperature feedback transfer functions. When the core and island temperature coefficients were included the following transfer functions were obtained:

$$\frac{\frac{\delta k_{(s)}}{1}}{MW_{(s)}} = \frac{-23.9}{S + 10.6} \qquad \text{(core)}$$

$$\frac{\frac{\delta k_{(s)}}{1}}{MW_{(s)}} = \frac{+1.435}{S + 2} \qquad \text{(island)}$$

Where $MW_{(s)}$ is the Laplace transform of the reactor power expressed in megawatts and the signs represent the sense of the feedback.

Table III-2 Temperature Feedback Parameters

Core temperature coefficient	- 7.30 x 10 Δk/F
Island temperature coefficient	+ 7.43 x 10 Δk/F
Core coolant velocity	11.55 ft/sec
Island coolant velocity	2.00 ft/sec
Core volume	33.00 liters
Metal to water ratio	0.77
Core coolant flow	1800 gpm

Feedback consideration due to core voiding from steam formation was based upon the methods used in the Chic-Kin code. The relationship between the void fraction (R_g) and the thermal flux (ϕ) is given by the equation:

$$\frac{dRg}{dt} = f(p)\,\phi - Rg/r$$

where:

$$f(p) = \frac{fr\,Vg}{1_1 h_{fg}} = 2.19\,fr \quad \text{if } \phi > \phi nB$$

or

$$f(p) = 0.0 \text{ if } \phi < \phi\, nB$$

l_1 = channel half thickness,

fr = heat fraction to voids,

r = bubble collapse time,

Vg = specific volume of steam at 65 psia,

h_{fg} = enthalpy for fluid to steam change of state at 65 psia,

ϕ nB = thermal (heat) flux at which nucleate boiling commences.

This relationship rigorously applies where the thermal heat flux is less than the departure from nucleate boiling. For this reason, results are less accurate for transients in which burnout is predicted. Applying the void coefficient for the MURR core and appropriate unit conversion factors, and taking the Laplace transforms, the core void transfer function becomes:

$$\frac{\delta k_{(s)}}{MW(s)} = \frac{2.443\ fr}{s + 1/r} \qquad \text{for } \phi > nB$$

$$= 0 \qquad \text{for } \phi < \phi\ nB$$

Since spatial dependence was not accounted for, the effective power at which nucleate boiling takes place had to be determined by comparison with the digital computer results. It was found that excellent correspondence was afforded for a "power at boiling" of 60 MW for the 0.010 δk step transient. No attempt was made to include boiling in the island since this appears quite improbable. Figure III-1 shows the basic power feedback loops used in the analog simulation.

The analog computer diagram is presented in Figure III-2, where the symbology is standard for analog computer programming. The analog computer gave information about the instantaneous reactor power throughout the duration of the transient. Through its use, the equilibrium after-burst power was obtained.

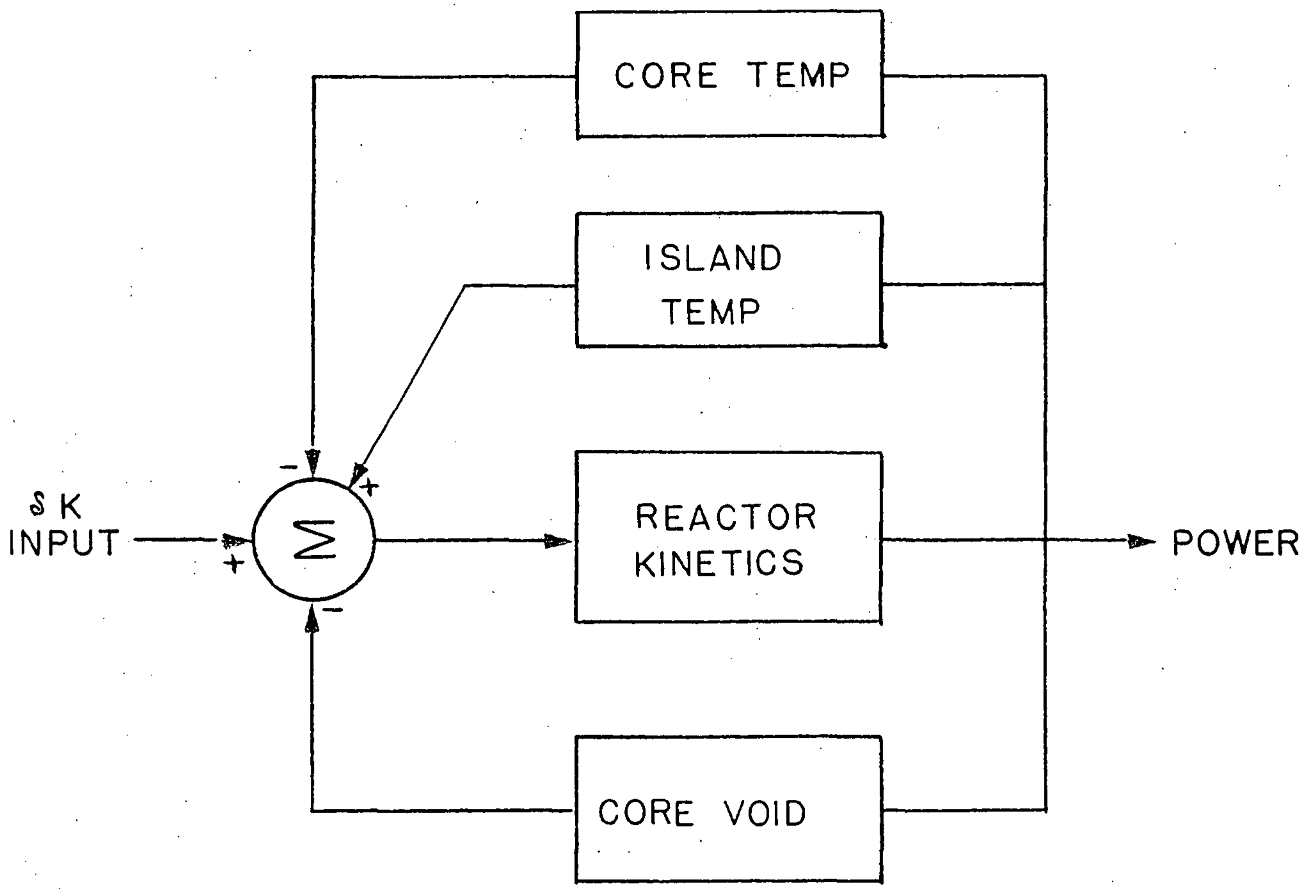


Figure III-1 Reactivity Feedback Paths

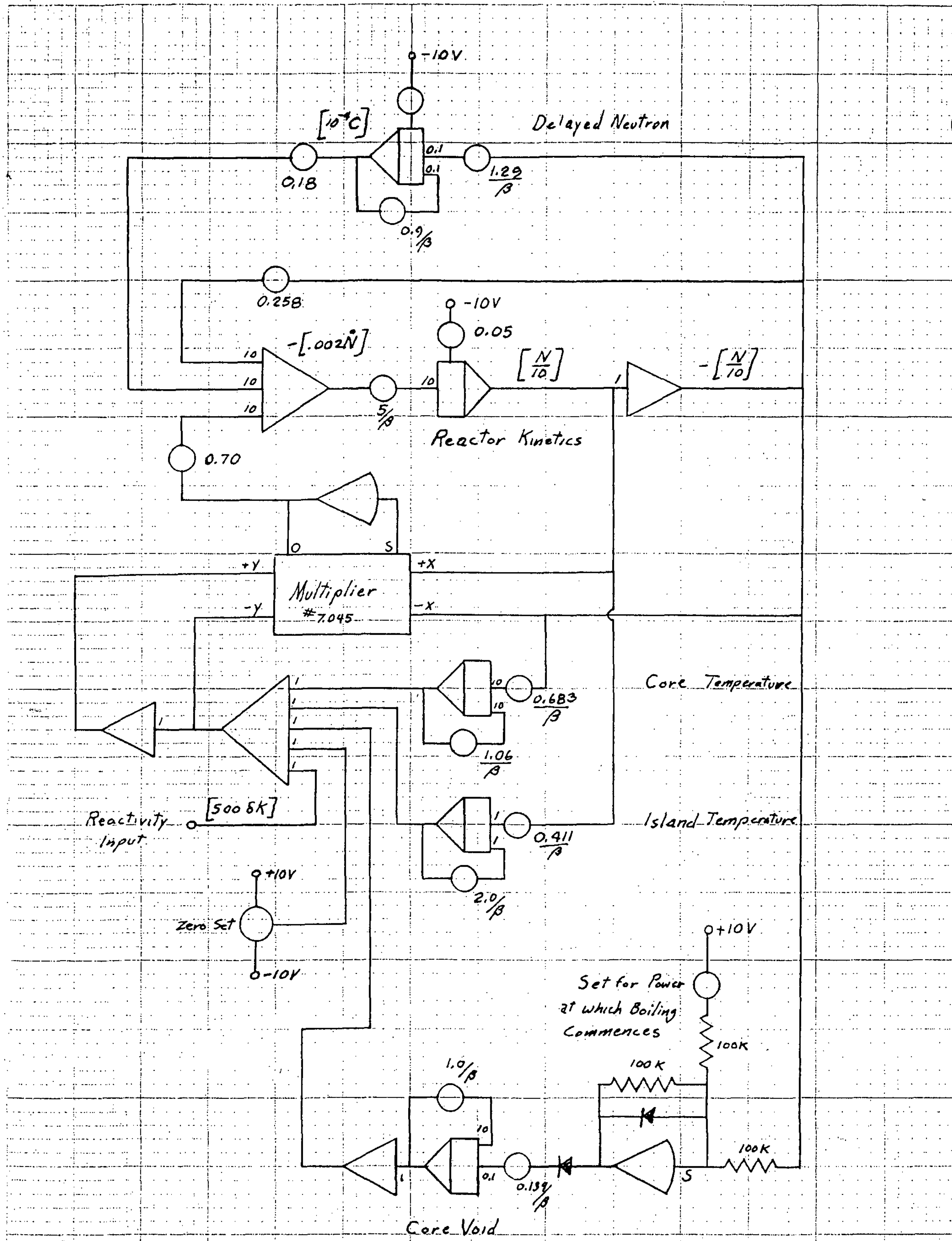


Figure III-2 Analog Computer Circuit

ADDENDUM NO. 2
HAZARDS SUMMARY REPORT
University of Missouri
Research Reactor Facility

Compiled and Edited by
The Staff
Research Reactor Facility

Submitted by
The University of Missouri
Columbia, Missouri

May 1966

PREFACE

The following material has been prepared in response to a request for clarification of 17 items which were sent to the University of Missouri by Dr. R. L. Doan with a letter dated April 1, 1966.

The seventeen items are discussed in the order in which they appeared on the list as presented to us.

Preceding the discussion of the seventeen items mentioned above is a short discussion of the reactivity effects resulting partial core melting and the loss of fuel plates central to a fuel assembly. This discussion is adjunct to that given in Section 3.12, page 39, of Addendum One to the Hazards Summary Report.

Also included with this submittal is O. G. Kelly and Company, Incorporated Drawing No. 65-9663-4 of the boral control blades for the reactor.

CONTENTS

DISCUSSION

"Continuation of the Discussion Presented in Section 3.12 of Addendum One to the Hazards Summary Report"

In an effort to determine the change in the core reactivity that would result from the loss of single fuel plates from a fuel assembly we have investigated one typical case. To investigate this case we enlisted the aid of the Oak Ridge National Laboratory where two computer runs using the EXTERMINATOR multi-group diffusion code on an IBM 7090 computer were made. First, a run was made to obtain k_{eff} for the normal MURR core. A second calculation was then made to determine k_{eff} in a similar core, but with two diagonally opposite fuel plates missing. The two fuel plates were located in the central region of their respective fuel elements and were replaced by water in the second run.

The calculations were run using the four lethargy group and r-theta geometry options and the 5 kg reference core loading as described in the Internuclear reactor design data. Macroscopic cross sections and neutron lethargy intervals were taken from the Internuclear data for use in the present calculations.

The results of the two runs are as follows:

1. The normal core (all fuel plates intact) was used as a control, and yielded a calculated k_{eff} of 1.10117.
2. The second core with two fuel plates missing, yielded a calculated k_{eff} of 1.10221.

Thus, the replacement by water of two diametrically opposite fuel plates, centrally located in their respective fuel elements, results in an increase k_{eff} of 0.00104, or a reactivity increase of approximately 0.1% δk.

REFERENCES

1. Tobias ML, Fowler TB, Vondy DR, "Exterminator-A Multigroup Code for Solving Neutron Diffusion Equations in One and Two Dimensions, "ORNL-TM-842.
2. Shapiro DM, "Missouri University Research Reactor Design Data," Vol. II, Internuclear Company, TM-DMS-62-5, Sept 1962.

1. "In reference to 3.5 of Addendum No. 1, describe the in-situ test to be performed to verify safety system response to extremes of expected ambient temperature conditions."

The in situ tests to be performed to verify safety system response to extremes of expected ambient temperature conditions are as follows:

Close all doors to the Control Room and set the room thermostat to 90°. After room temperature has stabilized read and record room temperature. Complete the operations described in I, II, and III below.

I. Channel 2 and 3

1. Place RAMP switch (S3) to VARIABLE.
2. Adjust Z14R3 to obtain an indication on the PERIOD meter, which corresponds to period scram trip, eight (8) seconds.
3. Verify trip lamp lights and scram annunciation occurs.
4. Adjust Z14R3 to obtain an indication on the PERIOD meter which corresponds to short period rod run-in, ten (10) seconds.
5. Verify trip and that magnet current has reduced to zero.
6. If the response expected after completion of items 2 through 5 is not obtained, determine and record the input level required to achieve safety system action.

II. Channel 4

1. Set S1 to TRIP TEST and set range switch to the 0-125% full power range.
2. Set trip adjust potentiometer (R1) to obtain an indication on the front panel meter for high level scram trip, (125%).
3. Verify trip.
4. Set trip adjust (R1) for power level rod run-in trip, (115%).
5. Verify trip and that magnet current has reduced to zero.
6. If the response expected after completion of items 2 through 5 is not obtained, determine and record the input level required to achieve safety system action.

III. Channel 5 and 6

1. Place selector switch S1 in CAL position.
2. Connect current source to J8 and apply input current, equivalent to desired trip point of the Power Level Scram Trip, (125%).
3. Verify trip.
4. Apply input current equivalent to desired trip point of the rod run-in trip, (115%).
5. Verify trip and that magnet current has reduced to zero.
6. If the response expected after completion of items 2 through 5 is not obtained, determine and record the input level required to achieve safety system action.

2. "In reference to 3.17 of Addendum No. 1, provide additional discussion on your criteria for determining which experiments require approval of the Reactor Advisory Committee."

It is the intent of the reactor operating staff to submit all new experiments to the Reactor Advisory Committee for review. Additionally, any experiment which falls into one or more of the following three categories will be submitted to the committee for review.

(1) Any experiment which is to be installed in a beam tube for long term usage.

(2) Any experiment which will require substantial modification of the "as built" reactor experimental facilities.

(3) Any experiment which involves high temperatures, low temperatures, high pressure, vacuum, highly corrosive material, or explosive materials, in any combination.

In reviewing an experiment brought before it the Reactor Advisory Committee will evaluate the following as well as other safety related factors: reactivity, corrosion, temperature, pressure, explosive effects, experiment containment, and heat removal.

Each experimenter making a request for reactor time will complete a form, a copy of which is included in this report. This form will be submitted to the Reactor Advisory Committee for their finding. In those instances where a second or third experiment is identical to the first one (or varying to a limited extent) there will be no resubmittal for approval.

All experiments conducted external to the reactor will be performed by the experimenter. It will be the function of the reactor staff, the principal experimenter, and the reactor health physicist to supervise this experimental work. It is not their function to perform the experiments for the faculty member or graduate student.

(This is a historical representation of a Reactor Utilization Request (RUR). RURs currently in use include all information asked for in this form, but may include more detailed information and analysis.)

Date of Request: ____________________

REACTOR UTILIZATION REQUEST
Research Reactor Facility
University of Missouri

1. Applicant: Department:
   Office Address: Telephone number:

2. Proposed experiment (if other than service irradiation). Use attached sheets as necessary.

3. Experimental facility requested.

4. Material to be irradiated.
   Chemical form:
   Weight (grams):

5. Estimated maximum activity of irradiated material:

6. Description of encapsulation.

7. Irradiation instructions.

8. Disposition of irradiated material.

9. Authorizations

Laboratory Supervisor - Date

Reactor Health Physicist - Date

Reactor Advisory Committee Action - Date

Reactor Supervisor Approval - Date

Safety Evaluation File Number

Instructions for Completion of Reactor Utilization Request Form

Item 1. Self explanatory.

Item 2. Provide sufficient description of the proposed experiment to permit a full evaluation of the hazards associated with the conduct of this particular experiment. Also indicate if this experiment is one of a kind or is one of a series of like experiments.

Item 3. State the specific experimental facility which is to be utilized, i.e., pneumatic facility, beam tube, thermal column, reflector irradiation position, flux trap, or in-pool.

Item 4. Describe the material to be irradiated in sufficient detail to permit an estimation of its physical, chemical and nuclear characteristics.

Item 5. Provide an estimate of the maximum activity expected in the irradiated material.

Item 6. Provide a description of how the material being irradiated is to be encapsulated. If the material to be irradiated contains special nuclear material, explosive materials, or highly corrosive material, special encapsulation requirements will be imposed. Please consult with the Research Reactor staff in determining encapsulation requirements.

Item 7. State any special instructions for the irradiation, such as, length of the irradiation, integrated flux desired, and flux desired.

Item 8. State how the irradiated material is to be disposed of. If it is to be delivered to another laboratory or another building, provide shipping instructions. If the irradiated material is to be discarded, so state. State Radiological Safety Office authorization if applicable.

Item 9. Each reactor utilization request must be approved by the laboratory supervisor, the reactor health physicist, and finally by the reactor supervisor. Some experiments, as determined by the reactor supervisor, and all experiments being run for the first time will be submitted to the Reactor Advisory Committee for review. Each authorized Reactor Utilization Form will include reference to a safety evaluation report. This report will be on file in the Research Reactor offices and will provide all the analysis and technical details upon which the safety evaluation is based. Many experiments, such as routine neutron activation analysis experiments, may refer to a single safety evaluation analysis. The safety evaluation is prepared by the reactor staff.

3. "In reference to 3.21 of Addendum No. 1, please confirm our understanding that plastic dowels will be mounted on meter faces to prevent the contact arms from being positioned beyond the maximum trip level values."

The Eberline RMS II has external access to the alarm and trip setting potentiometers but this must be done using a screwdriver to reach through a front cover plate of each channel. No inadvertent or unmonitored change of set points is likely. The Eberline system uses voltage comparators to change the state of alarm relays for trip functions, instead of alarm contact arms used in the tracerlab ARMS.

4. "It is our understanding that the initial loading procedure presented in your application has been changed. Provide a discussion of your new loading procedure."

The initial reactor fuel loading procedure has been modified from that as described in Section 11.2 of the Hazards Summary Report. In contrast to that described in Section 11.2, the fuel will be loaded with moderator present. The prime reason for this change is due to the necessity of water being in the pool to a level such that the dash pot assembly in the control rod mechanism is covered with water. The reactor core will be taken critical on a partial loading when it becomes possible to do so. Critical, as far as the fuel loading is concerned, is inferred to be cold critical or a power of essentially just a few watts. In order to keep the pool as calm as possible it is planned to keep the pool circulating system turned off, this will mean that for the fuel loading operation the safety trips associated with pool circulation will be bypassed. Specifically, the pool flow, natural convection valve and pool system isolation valve trips will be defeated for loading.

To prevent accidentally dropping a fuel element during loading, a safety wire will be attached to the fuel element in addition to the positive lock on the fuel handling tool. The safety wire will be removed after the element is in place in the core.

Only one element will be loaded into the core at a time and multiplication data will be taken and analyzed following the loading of each element. The loading procedure first loads the core with four kilograms of U-235 in the form of 4-650 g elements and 4-350 g elements. The 350 gram elements will then be replaced by four 650 gram elements to bring the core loading to 5.2 Kg of U-235 in eight elements.

The instrumentation for startup is as described in Section 11.2 of the Hazards Summary Report.

The position of the control rods will be two rods partially withdrawn to provide shutdown capability and two rods fully inserted. The degree to which the partially cocked rods are withdrawn will depend upon analysis of the data.

In order to support the loading sequence the procedure has been simulated to as close a degree as possible with the University's subcritical assembly and discussed at length with at least three authorities on reactor loading; they are R. Cochran, Texas A&M, S. MacKay, General Electric Co., and R. B. Hamilton, General Electric Co.

5. "Describe the availability of qualified consultants for critical core loading and the responsibilities of these consultants."

The two nuclear groups involved in the design and construction of this reactor are Internuclear Company of St. Louis, Missouri, and General Electric Company, San Jose, California. Each of these groups will provide consultant services to the startup crew. Additionally, it is the intention of the University of Missouri to procure the services of a "non-involved" outside consultant.

Internuclear Company of St. Louis performed the preliminary and final design work on the reactor facility. A member of their staff, Dr. Donald Shapiro, performed the core physics computations. Dr. Shapiro will be on hand during reactor startup as a consultant and advisor to the startup crew. His functions, comments and criticisms will be delivered to Dr. A. H. Emmons the Director of the total facility. A resume on Dr. Shapiro is included in this report.

Another person who will act as a consultant and advisor and who has been intimately involved in the construction supervision of the reactor facility is Mr. Fred Flint of Internuclear Company, St. Louis, Missouri. Mr. Flint has been the job supervisor representative from Internuclear during the past three years. He will be on-site and available as a consultant during the reactor startup. His resume is included.

The General Electric representative who will be on-site during the reactor loading and low power operation is Mr. S. David MacKay. Mr. MacKay has been on-site during the latter few phases of reactor construction and installation. He has been responsible for conducting the startup and precritical operations of all reactor systems and for conducting acceptance testing. A resume on Mr. MacKay is included with this report.

Dr. Robert Cochran, Head of the Texas A&M Nuclear Science Center, has been retained by the University of Missouri to act as a consultant during fuel loading and initial criticality runs. Dr. Cochran is familiar with the system in that he served as the license examiner for the AEC during operators licensing procedures of some weeks ago. He will come on-site and work through the Director of the facility during the period of fuel loading and reactor startup. Dr. Cochran's qualifications are well known to the Commission for whom he has served as a consultant for several years. His resume is attached.

6. "It is our understanding that several component designs which have been modified since your Construction Permit Application have not been fully described in your current application. Please describe and evaluate, where necessary, the above design changes. In particular, check those items which were referred to in our letter to you dated May 26, 1961, and in the AEC Staff Evaluation dated October 31, 1961."

Referring first to the AEC letter dated May 26, 1961, the preface to Amendment No. 1 to Exhibit A of Class 104 License Application, Docket No. 50-186 states, "In a letter dated May 26, 1961, from Martin B. Biles, Chief, Test and Power Reactor Safety Branch, Division of Licensing and Regulation, additional information was requested and certain conditions were stipulated with regard to: (1) reactor site; (2) Argon-41 effluent control; (3) reactor containment; and (4) reactor design. This amendment to the University's original Preliminary Hazards Report presents information on each of these subject categories."

The reactor site has been considered in detail in prior submittals to the commission.

Argon-41 effluent control was discussed in detail in Amendment No. 1 referenced above, and is given further discussion under Item No. 11 of this submittal.

The reactor containment building construction is now complete and leak rate measurements have been made. The total leakage in a 24 hour period from the containment building with the interior walls uncoated was 8.6% of the contained building volume when the interior of the building was raised to an initial overpressure of 2 pounds per square inch. After the application of a sealant on the interior walls the total leakage in 24 hours was reduced to approximately 6.0% of the contained building volume. Because building leakage exceeds that value which was stated in the Construction Permit Application, new calculations of a conservative maximum dose which might be delivered to points outside the containment building under assumed maximum fission product release conditions have been made which use the measured building leak rate data. The revised calculations were discussed in Addendum No. 1 to the Final Hazards Summary Report and are discussed again under Item No. 9 of this report.

With reference to Sections 3.3 and 3.4 of Addendum No. 1 to the Preliminary Hazards Report, the final design of the pedestrian entry and the equipment entry deviates somewhat from that discussed. The pedestrian entry doors and the equipment entry door, as presently installed, operates as illustrated by Figure 3.2 of Amendment No. 1 to the Preliminary Hazards Report. These doors, however, are not fitted with auxiliary latching. When the door is in the closed position the inflatable gaskets press the doors which are held to within 3/8 inch of the door jam by adjustable guide wheels and provide an air tight seal. The two building ventilation system isolation doors are designed in the same way as the pedestrian and equipment entry doors.

The main features of the reactor design were discussed in Amendment No. 1 to the Preliminary Hazards Report and in the Final Hazards Summary Report. The reactor fuel to be used, the test program, and the acceptance testing program have been discussed in the above referenced documents. The systems for core support, reflector support, and experimental facilities support were discussed in Section 4.0 of Amendment No. 1 to the Preliminary Hazards Report.

A complete reactor pressure vessel specification is included with this submittal. This specification provides a complete description of all design conditions, strengths, safety factors, codes, and other fabrication conditions.

The reactor's reflector, experimental facilities, control system, and waste treatment system have been discussed in prior submittals.

Additional information not previously submitted concerning radiation damage, stresses, and heating is appended to this report in the form of technical memorandums prepared by the Internuclear Company during the course of reactor design and analysis.

7. "In reference to 3.4 of Addendum No. 1, provide required schematics to justify that relay 1K13 provides the necessary safety system redundancy for the E6 scram matrix. Also provide justification that either a scram is not required on "Pool Loop Low Flow" or that redundancy requirements are met on this circuitry."

NOTE: This topic was reevaluated at 10 MW upgrade (see Hazards Summary Report, Addendum 4, Appendix A).

Figure 7-1 shows the circuitry which operates relay 1K13. When relay 1K13 is de-energized the safety system receives a scram signal through logic input E7. Relay 1K13 is energized if the wide range monitor is set for a full scale power level of 50 KW, the power selector switch is in the 0.1 MW position, valves 507A and 507B are closed, valves 543A and 543B are open, and valve 546 is closed. If the power selector switch is in the 5 MW position, all ranges of the wide range monitor may be used, but valves 507A and 507B must be open, valve 546 must be closed, and reactor pressure, temperature and flow must be normal.

To provide redundancy in Pool Loop Low Flow scram protection, the circuitry which monitors pressure drop across the reflector tank, which now provides only an alarm, will be modified to provide a scram signal. This circuit will be set so that if the reflector differential pressure deviates to either side of normal, a scram will be initiated. The precise pressure settings for scram will be determined by measurement during pre-operational testing.

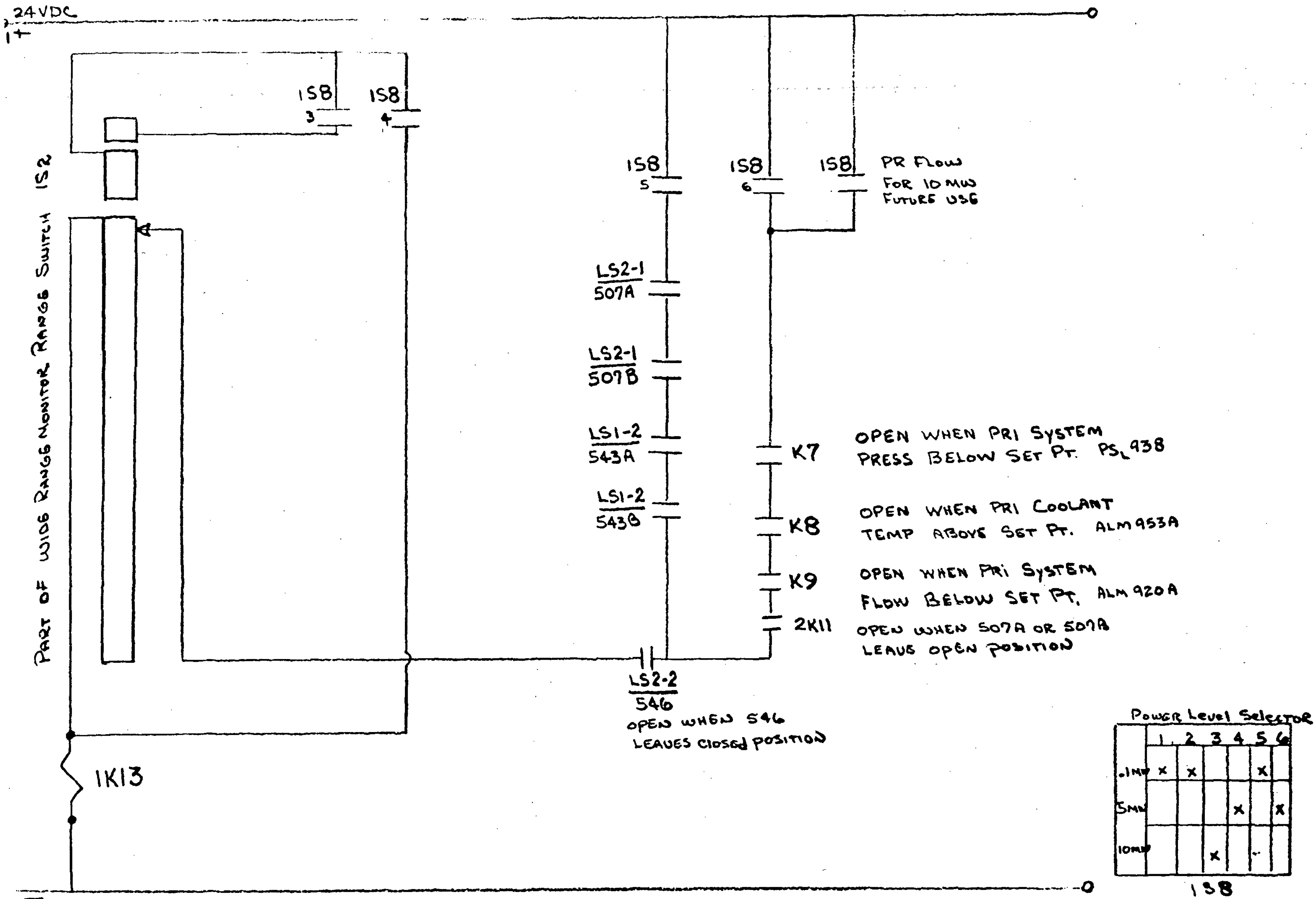


Figure 7-1. Relay 1K13 Energizing Circuits

8. "In reference to 3.10 and 3.20 of Addendum No. 1, describe the redesign of the isolation system which will provide sufficient functional redundancy such that no single failure can prevent isolation. In particular, describe the redesign of the circuitry and the additional drives for the ventilation system doors. Also describe the manner in which the elevator door type interlock on these doors will be defeated during non-maintenance periods."

Redundancy in containment building isolation is being provided by the addition of a second set of isolation doors. Figure 8-1 shows a sectional view of the added isolation doors. The doors are enclosed from above by a steel plenum chamber which becomes part of the building containment.

The doors are operated by a double acting pneumatic cylinder and are held open against gravity by air pressure. The air which drives the pneumatic cylinder is delivered through two normally energized solenoid valves. When the valve is energized the door is held open. If the solenoid or supply power fails, the door is driven closed by gravity.

These "backup" isolation doors operate in the event of radiation existing in the ventilation chambers of the east building tower or high radiation levels at the reactor bridge. A remote indication and alarm is mounted in one channel of the Eberline RMS II radiation monitoring system in the control room. The radiation detection system used employs a G-M tube that has a dynamic range of 0.1 to 10,000 mR/hour. In the event of instrument failure, loss of signal to the instrument will initiate a reactor isolation.

The independence of the backup isolation detector trip is achieved by having its channel mounted in a separate rack unit from the primary building exhaust plenum trip unit. The design of the circuitry that initiates a Reactor Isolation (2K1A and 2K1B) will further provide independence between detectors.

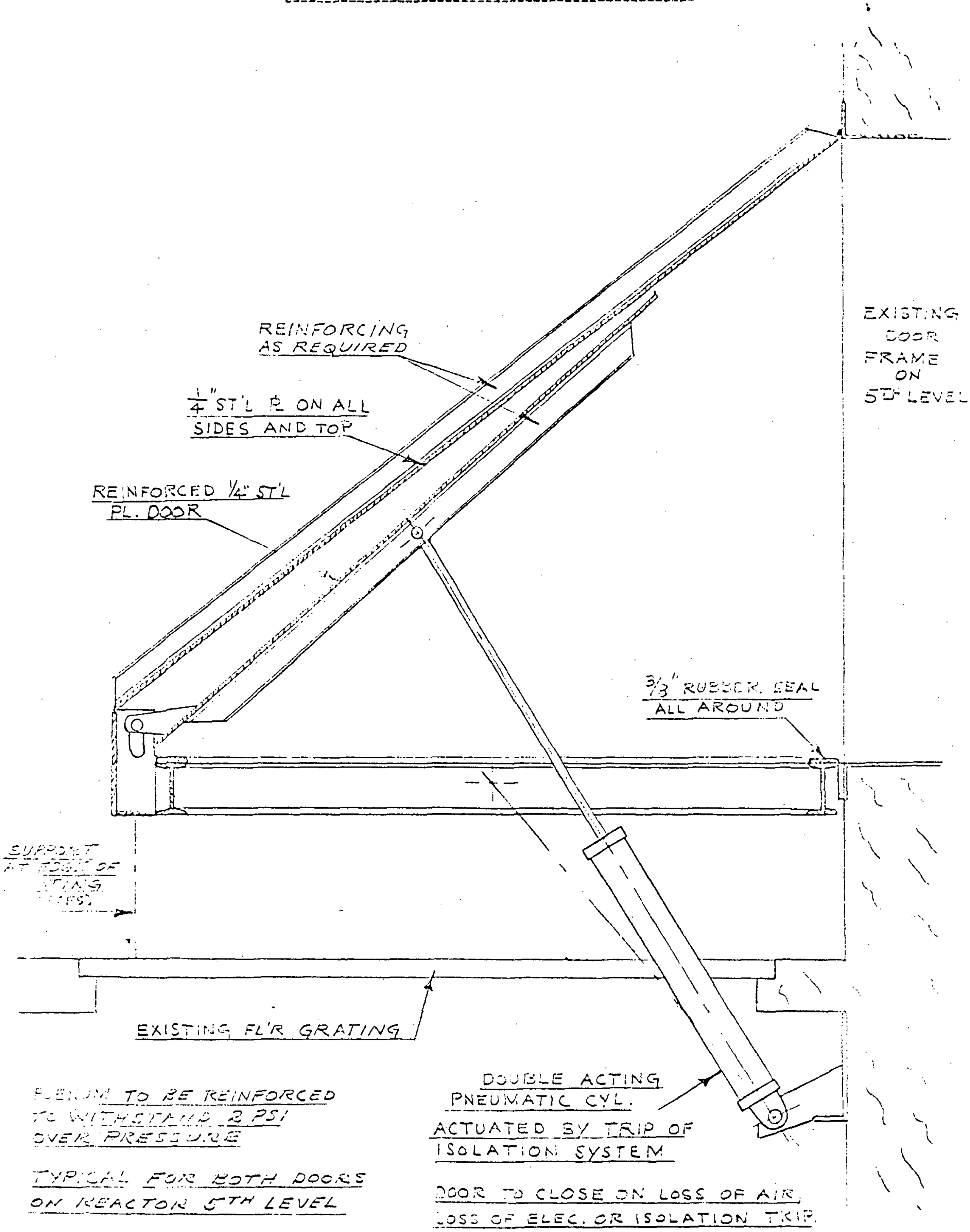
Solenoid power, which is taken from the commercial electrical supply, is required to keep the backup isolation doors open. In the event of the failure of commercial power the doors will close and remain closed until power is restored. In the event of the loss of the compressed air supply the doors will be closed by gravity.

With regard to the safety edge (referred to as an elevator door type interlock in Dr. Doan's communication of April 1, 1966) it is our belief that this circuit should be in operation at all times. Firstly, we are concerned about the potential hazard to personnel if one were to neglect to energize the safety edge circuit while accomplishing routine maintenance or inspection. Secondly, if an object were inadvertently left in the doorway and the door attempted to close without safety edge protection, serious mechanical damage to the operating mechanism could result when the door encountered the object. We believe that it is preferable to permit the safety edge to return the door to the open position. The door will only close if the operator pushes the button to close the door. Control room indications will alert the operator to this abnormal condition and immediate corrective action can be taken. In the meanwhile, of course, the backup isolation doors will have operated, completing the containment building isolation.

BY [illegible] DATE 6-11-66 SUBJECT CROSS SECTION OF BACKUP CONTAINMENT ISOLATION PLENUM

SHEET NO. 1 OF 1

Figure 8-1



9. "The long-term thyroid doses presented in Figure 3.11.2 of Addendum No. 1 appear to be inconsistent with the two-hour doses presented. Reanalyze the thyroid doses taking into account containment pressure decay."

The long-term thyroid doses presented in Figure 3.11.2 of Addendum No. 1 are in error by a factor of 10. A plotting error resulted in the one, three, and ten day iodine dose to the thyroid being a factor of ten too high.

A more detailed analysis of the thyroid doses is presently being prepared and is being sent under separate cover.

10. "Section 3.15 of Addendum No. 1 analyzed transients starting at 5 megawatts. Provide an evaluation of the consequence of transients starting from an initial power of 10 megawatts."

An analog computer study was made of step reactivity induced transients for the University of Missouri Research Reactor when operated under "10 Megawatt" conditions (e.g., 3600 gpm coolant flow and 10 MW initial power level). The purpose of this study was to obtain an estimate of the maximum step reactivity insertion which can be tolerated by the Missouri Reactor without suffering hot spot burn-out. A previous study, made using 5 megawatt conditions resulted in a maximum tolerable step of about 0.01 δk reactivity (1). The present findings for the 10 megawatt conditions differ, as will be discussed.

The results of this study were that a step reactivity insertion of up to about 0.008 δk could be tolerated with assurance of safety from burn-out. The results of the present analog study indicate that a scram assisted shutdown should be initiated soon after the initiation of the 0.008 δk transient. This conclusion is occasioned more by the increased coolant flow than by considerations of the higher initial power level.

To provide a guideline for evaluating reactor transients with respect to safety from boiling burnout, the Bernath correlation (2) was applied to the Missouri Reactor design, where the pertinent parameters are:

Coolant flow rate	3600 gpm
Coolant pressure	65 psia
Inlet coolant temperature	140°F
Coolant velocity	23.1 fps

The hot spot and hot channel factors used were those of previous reports, i.e. P_a - 1.443 and P_r = 2.263 (3), (4). Using these parameters and the 10 megawatt core average heat flux of 172,244 BTU/ft^2-hr, the power at which boiling burn-out should occur was computed to be about 47 megawatts.

The burn-out power of 47 megawatts, being a steady state value, was considered applicable only to the post-burst part of the transients, where power was slowly changing. Transient burn-out heat fluxes for the peak power burst part of the transient would be much greater than the steady state value, owing mainly to the fact that prompt critical reactor periods are comparable to the bubble formation times. For the transients considered here, it is therefore quite improbable that burn-out would occur as a result of the peak power bursts, even though it should exceed the 47 MW steady state burn-out value.

On the basis of the SPERT test series, it was expected that the initial power would have little effect on the character of the transients. The increased coolant flow of the 10 MW case as compared to the 5 MW conditions, however, would tend to broaden the burst slightly and cause a rather large post peak affect. The latter affects are due to the increased flow tending to defeat the inherent negative temperature and void coefficient shutdown mechanisms. The analog results confirmed these considerations and demonstrated the desirability of complete reactor shutdown after the post-peak region of transient.

Figure 10-1 shows the results of the analog computation for step induced transients. The reactivity insertions considered ranged from .0025 δk to 0.010 δk and, with the exception of transient A, all transients were for 3600 gpm coolant flow rates. Transient A was computed using an 1800 gpm (5 MW) coolant flow rate, and was included for purposes of comparison. These transients are very conservative since no boiling induced void feedback was included. Any core voiding due to local boiling would tend to reduce the power levels from those shown, especially in the post burst region.

Referring to Figure 10-1, the 0.008 δk transient appears to be the worst tolerable transient if a scram assisted shutdown can be initiated with 400 milliseconds of the step insertion. Using the method described by Grossman and Moore (5), the scram shutdown characteristics were computed for the Missouri Reactor and applied to the 0.008 δk transient as shown. Under these conditions, then, it would appear quite impossible that burn-out would occur for the 0.008 δk transient.

The question, then, is whether or not a reactor scram can be initiated automatically within 400 milliseconds. The electronic circuits associated with the safety system can certainly respond to an over-power or short period condition in a negligible amount of time. That the rod magnet and rod drop dynamic characteristics are sufficient is illustrated in Figure 10-2, in which the fraction of rod worth is plotted against the time after loss of magnet current (6), (7). Figure 10-2, based upon tests performed by General Electric, was used as the basis for computing the effects of a scram initiated 0.4 seconds after a 0.008 δk transient. The subsequent power trace is shown by the dashed line of Figure 10-1.

Based on conservative calculations and the results just described, there is reasonable assurance that a 0.008 δk step increase in reactivity can be safely tolerated by the University of Missouri Research Reactor. Spert III and IV data (8), (9) indicate that much larger step reactivity insertions can be tolerated. Another, and more complete analysis will be presented regarding 10 MW flow condition transients when application is made for an amendment to the Missouri Reactor operation license to permit 10 megawatt operation.

<u>References</u>

1. "Hazards Summary Report," Addendum I, University of Missouri, p. 53ff.
2. Etherington H, "Nuclear Engineering Handbook" (1958), p. 9-78.
3. "Hazards Summary Report," Addendum I, University of Missouri, p. 18ff.
4. Shapiro DM , Kim VS and Putnam GE, "Reactor Physics Analysis for the University of Missouri Research Reactor," Internuclear TM-DMS-62-5 (1962).
5. Grossman SR and Moore KV, "A Method of Estimating the Kinetic Effects of Scram Rods," IDO-16777.
6. General Electric Co., A.P.E.D., Correspondence with University of Missouri.
7. Shapiro DM, Design Data, Vol. II, Internuclear TM-DM-62-5 (1962), Figure 7-2.
8. Toole CR, "The Effects of Pressure and Flow on Room Temperature Power Excursions in SPERT III," IDO-16918 (1964).
9. Crocker JG and Stephen LA, "Reactor Power Excursion Tests in the SPERT IV Facility, IDO-1700 (1964).

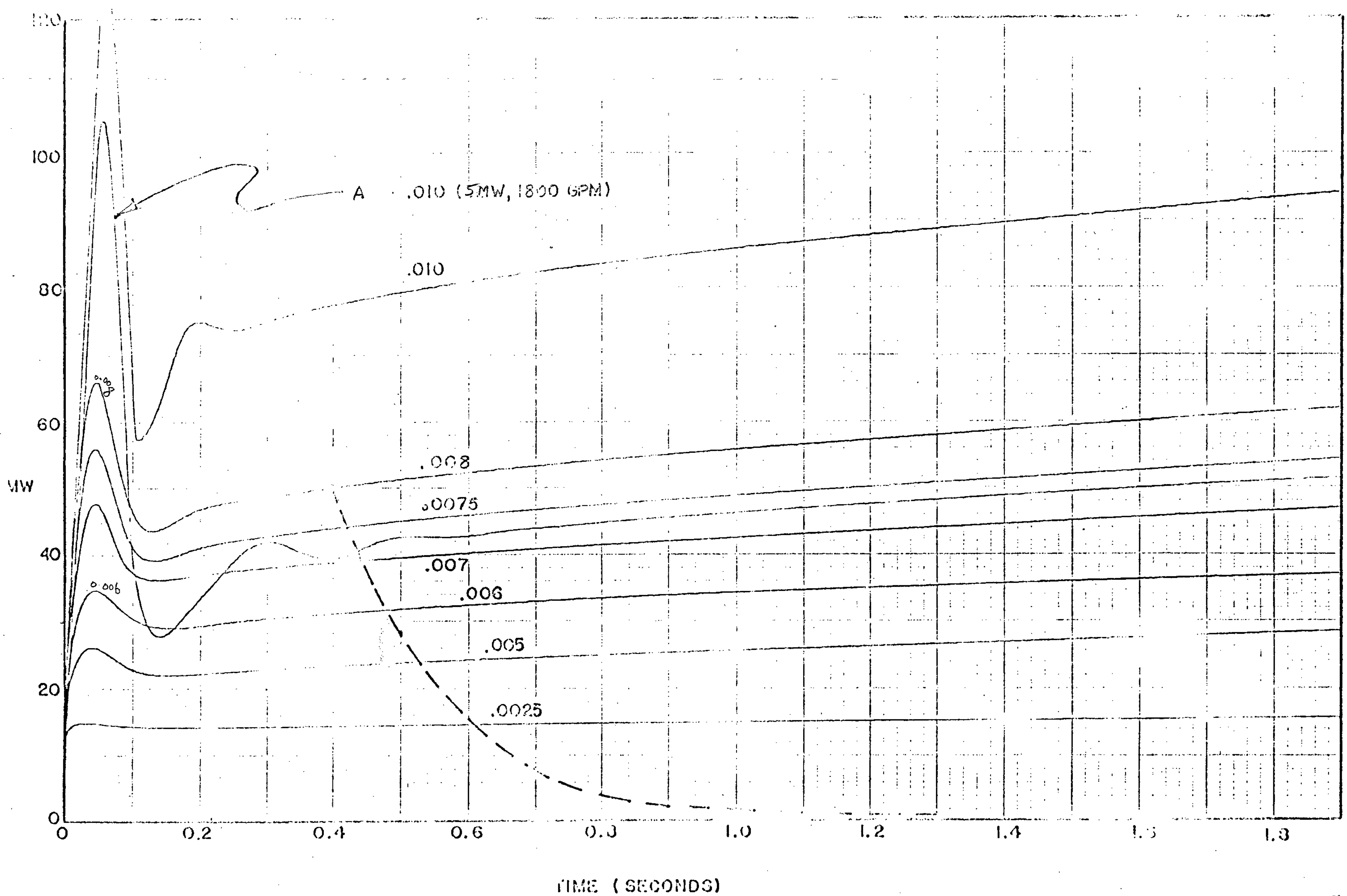


Figure 10-1. 10 Megawatt Transients

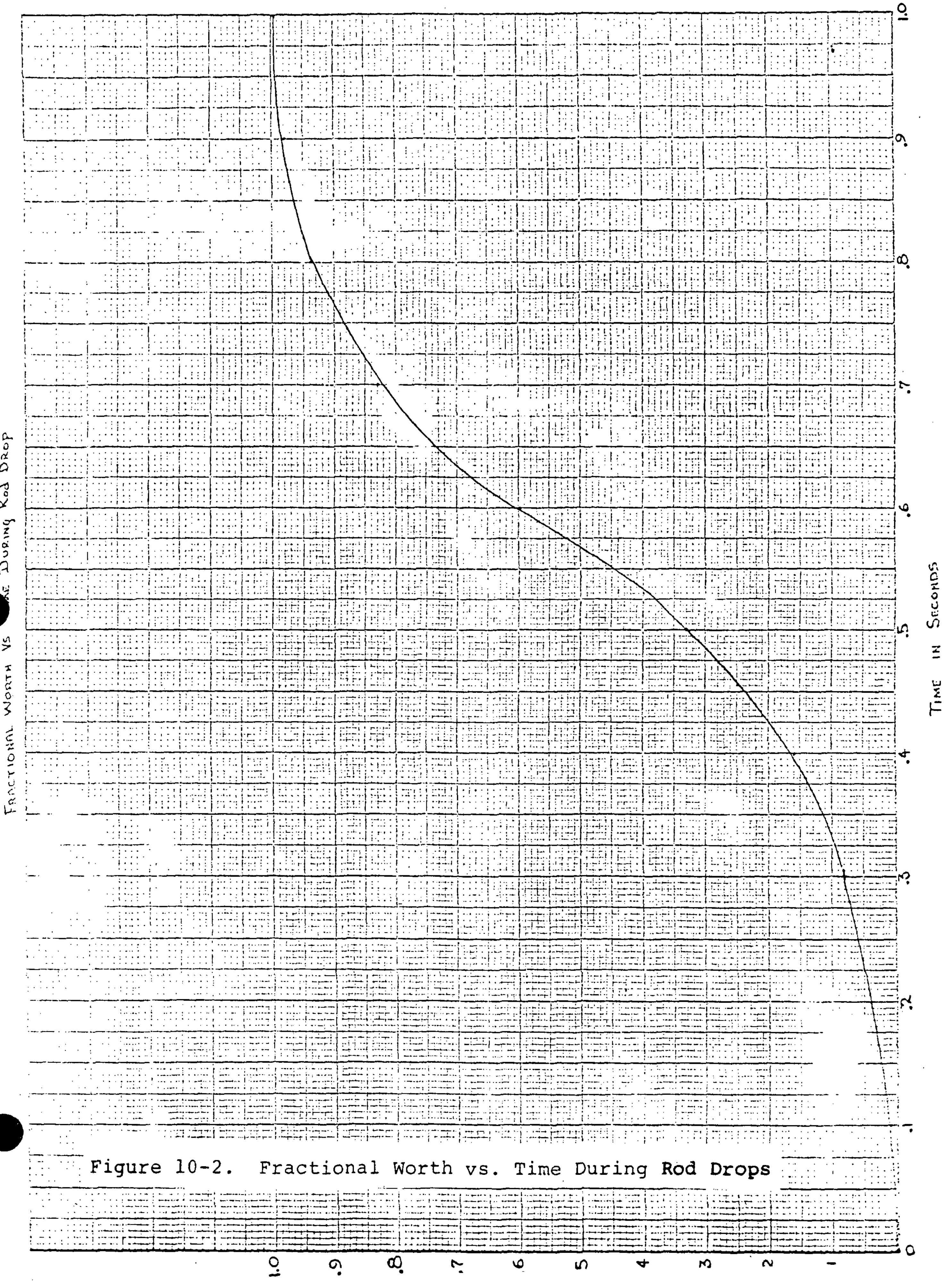


Figure 10-2. Fractional Worth vs. Time During **Rod Drops**

11. "If you require an atmospheric dilution factor to meet 10 CFR Part 20 limits on Argon-41 concentrations, state the factor you are requesting and provide a justification for this number."

Note: This question is no longer relevant due to Facility Modification package 88-7. See 1989-1990 Annual Report for updated information.

Section 4.2.7 of the Preliminary Hazard Report presents information of the then proposed techniques of disposal of Argon-41 to the atmosphere. In that report it is indicated that the facility will be serviced by two stacks; one for non-irradiated air disposal, the other for reactor off-gases including irradiated air. Amendment No. 1 to the Preliminary Hazard Report, pages 14 through 19, describes the single stack system as designed and installed. A schematic of the off-gas exhaust system is presented on page 103 of Addendum One to the Hazards Summary Report. This latter reference also defines measured exhaust volumes in the system.

The measured rate of air discharged from the stack is 20,411 cubic feet per minute. Of this volume, 1544 cfm is exhausted from the reactor building and of that only 255 cfm is potentially irradiated air (Ar-41 contaminated). The 255 cfm is exhausted from six beamports, one thermal column, and the four terminal pneumatic system. If it is assumed that there are 2% air voids in the thermal column, that two 6 inch and one 4 inch beam tubes are empty, and that all four pneumatic terminals are in use the generation rate for Argon-41 at 10 MW power level is 520 μc per second. then the computed stack discharge activity level at full power (10 MW) is 5.4 x 10^{-5} μc /cm^3.

Then the maximum ground level concentration using all of the most conservative assumptions would be:

$$x_{max} = \frac{C_z}{C_y} \frac{2Q_c}{e\pi \bar{u} h^2}$$

$$x_{max} = 1 \times 10^{-7} \ \mu C/cm^3$$

where:

C_z = C_y

Q_c = 520 μc/sec.

$\overline{u}$ = 4 meters/sec.

h = 17.4 meters

This result comes from the assumption that (a) the stack off-gas has no vertical velocity (b) Cz is equal to Cy, and (c) continuous operation at 10 MW power level with no time averaging of discharge levels.

In actual fact the stack data is as follows:

Area of stack exit	=	7.07 ft^2
Exit volume	=	20,411 ft^3/min
Exit velocity (V_s)	=	2890 ft/min
	=	48 ft/sec
	=	14.6 m/sec

which provides an added "effective" stack height which may be computed from any one of a number of formulas (1). The most conservative of these is that one by Rupp, Beal, Bornwasser and Johnson (5) which does not take any buoyancy credit.

$$\Delta h = 1.5 \frac{V_s}{\overline{u}} d$$

where:

Δh = computed rise of plume center line above stacks,

V_s = mean stack effluent velocity (meters/sec),

$\overline{u}$ = wind velocity (meters/sec), and

d = inside diameter of stack in meters.

The stack inside diameter is 38 inches than d is unity.

The cooling equipment installed with this reactor facility will limit (for at least two years) the power level to 5 MW or less. It is planned that the facility will be operated only on one shift, however we would like to place such a restriction on operation (i.e., an 8 hour per day limit). Then consider the maximum ground concentrations at 5 MW power level, continuous operation, under an inversion condition.

$$X_{max} = \frac{C_z}{C_y} \frac{2\,Q_c}{e\pi \bar{u}\, h^2}$$

$$= 6.95 \times 10^{-9}\ \mu C/cc$$

where:

C_z = 0.07

C_y = 0.4

$\bar{u}$ = 1 meter/sec

h = $h + \Delta h = 17.4 + 21.9 = 39.3$ meters

Q_c = 260 μC/sec

And under lapse conditions:

$X_{max} = 2.32 \times 10\text{-}8\ \mu C/cc$

where:

C_z = 0.25

C_y = 0.25

$\bar{u}$ = 6 meters/sec

h = $17.4 + 3.6 = 21.0$ meters

Q_c = 260 μC/sec

The validity of these calculations is subject to more stringent review as operational data is taken. At the time of this writing the applicant institution requests that a "dilution factor" of 1000 be allowed where:

$$D.F. = \frac{X_s}{X}$$

and

D.F. = dilution factor,

Xs = source concentration at stack in μC/cc

X = concentration downwind in μC/cc

This dilution factor would be applied to measure values of stack effluent activity levels. The suggestion that a D.F. of 1000 be allowed follows from this "averaged" calculation:

$$x_{max} = \frac{C_z}{C_y} \cdot \frac{2 \times 520\ \mu C/sec}{e \pi \bar{u} h^2}$$

$$= 5.48 \times 10^{-8}\ \mu C/cc$$

when

C_z = C_y

$\bar{u}$ = 4 m/sec

h_{eff} = 23.5 meters

Q_c = 520 μC/sec

Then

$$D.F = \frac{X_s}{X} = \frac{5.4 \times 10^{-5}}{5.4 \times 10^{-8}}$$

A review of the reactor site characteristics will reveal to the reader that the reactor is situated in a valley (see Figure 1.1, page 2, Amendment No. 1, Preliminary Hazard Report). The top of the reactor stack is at the elevation 672 feet. To the north, east, and south, the edge of the valley is at least 1000 feet distance from the reactor, but to the west the land rises steeply to a maximum of 700 feet elevation. This maximum elevation occurs 800 feet to the west.

In the consideration of stack dilution one must include the consideration of people. Attention is called to Table 1.2 and 1.3 of the above cited reference. It will be noted that from 0 to 5000 feet in the three 45° sectors (Sectors 5, 6 and 7) bounded by a line to the south and a line to the northwest there are only eight private residences and these at more than 4500 feet to the northwest. The major portion of the area to the west and within 3000 feet of the reactor is the University golf course. This comprises a tract of 157 acres of extremely low population density. This golf course, together with many acres of additional land to the west of the site, is the property of, and under the sole control of, the University of Missouri. It is not intended that this land will ever be used for dormitories, hospitals, apartments or any other residence type of dwelling.

In summary, a stack dilution factor of one-thousand is requested for computation of Argon-41 release rates. The effluent Argon-41 levels will be measured and, prior to any increase in power over 5 MW, additional data and computations will be submitted to the A.E.C.

References:

1. Moses H., Strom G.H., and Carson J.E. "Effects of Meteorological and Engineering Factors on Stack Plume Rise," Nuclear Safety, Vol. 6, No. 1, p. 1 (1964).
2. Hosler C.R. "Climatological Estimates of Diffusion Conditions in the United States," Nuclear Safety, Vol. 5, No. 2, p. 184 (1963-1964).
3. Soman, S. D. and Abraham, P., "Study of Airborne Radioactivities in the Operation of a High Flux Research Reactor," Health Physics, Vol. 11, No. 6, p. 497 (June 1965).

4. Martin, J.E., "The Correlation of Wind Tunnel and Field Measurements of Gas Diffusion Using Krypton-85 as a Tracer," Ph.D. Thesis, University of Michigan (1965).
5. Rupp, A.F., Beall, S.E., Bornwasser, L.P., and Johnson, D.H., "Dilution of Stack Gases in Cross Winds," USAEC Report AECD-1811, p. 1-15 (1948).
6. Gifford, F.A., Jr., "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," Nuclear Safety, Vol. 2, No. 4, p. 47 (June 1961).
7. Gifford, F.A., Jr., "Atmospheric Dispersion Calculations Using the Generalized Gaussian Plume Model," Nuclear Safety, Vol. 2, No. 2, p. 56 (December 1960).

12. "Discuss the means to assure rapid evaluation of the area between 500 feet and 1500 feet from the reactor in case of emergency and estimate the time required to evacuate this area assuming other laboratory and office buildings are built and are fully occupied."

Upon notification of the emergency all personnel in laboratory and office spaces within the area from 500 to 1500 feet from the reactor building will evacuate the area according to the following plan.

1. All personnel within the Golf Course Greenskeeper building, the Sanitary Engineering Research Laboratory, the Animal Shelter, the Animal Psychology Laboratory, and the Animal Nutrition Laboratory and offices will proceed east along the access road and north on Route K to a point beyond the 1500 foot limit.
2. All personnel with the Space Sciences Research Center will proceed north and east along the access road and north on Route K to a point beyond the 1500 foot limit.
3. All personnel who have evacuated the reactor building to the north will proceed as indicated in item 2 above.
4. All personnel who have evacuated the reactor building to the south will proceed south and east along the access road and south on Route K to a point beyond the 1500 foot limit.

The maximum time necessary for the execution of this extension the Exclusion Area Evacuation Plan will be thirty minutes.

13. "Discuss the magnitude of possible water hammer effect due to normal closing of the primary loop isolation valves, and accidental closing of either the inlet or outlet valve due to a false signal or valve linkage failure. In this connection consider the consequences of cyclic stresses and vibration of the primary system. Discuss means of limiting the magnitude of the effect. Describe the pre-op tests that will be performed to check this effect."

An analysis of the water hammer induced pressure pulse which might result from closure of valves in the reactor primary loop, using methods of conservation of momentum, was accomplished. This analysis revealed that valve closure times of 0.5 seconds or greater would result in a net reaction force on the piping system, acting at the center of gravity, of less than 500 pounds. This compares to a steady flow reaction force of 250 pounds. The maximum pressure buildup accompanying the flow deceleration is 38.0 psi. The same analysis indicated that a valve closure time of as low as 0.02 seconds would result in a loop pressure buildup of only about 800 psi.

As presently installed and adjusted the butterfly type valves for reactor core isolation have a closing time on the order of 4 to 6 seconds. At this closure time the net loop reaction force and the loop pressure buildup are insignificant. At the design minimum valve closure time of 1 second the resulting net reaction force and loop pressure buildup are still well below any limiting stresses.

In the event of valve linkage failure such that the valve and its actuating mechanism are completely decoupled, no water hammer is likely to occur. The reason for this is that the valve will not seat without being driven closed. The water flow through the butterfly valve has as much tendency to force the valve open as to force it closed.

During preoperational testing the reactor isolation valves will be closed from full flow conditions. The effects of water hammer will be noted by noting any audible indications and by observing any movement of the primary coolant piping. Also, valve closure times with full flow and with no flow will be measured to determine whether or not flow influences closure time.

14. "In reference to 3.23 of Addendum No. 1, assessment of the possible need for a backup shutdown system will, as you suggest, depend on demonstration of control rod reliability under normal conditions, and analysis of those conditions which could cause binding of the rods (e.g., distortion of the pressure vessel due to pressure surges originated by nuclear excursions of sudden valve closure, collection of debris in guides, etc.). Your assessment of such matters and your views are desired as to whether a backup shutdown system would be necessary to mitigate the possible consequences of accidents that could occur under the presently planned scope of operations or, if not, whether limited provisions should be made for installation of such a system to cover any expanded future programs."

The need for a backup shutdown system on the University of Missouri Research Reactor is not at all clear. The reactor design is such that with the core in the cold-clean condition (maximum k excess) it will be subcritical with one rod fully withdrawn. However, rod operating position changes from approximately one-half withdrawn at the beginning of core life to fully withdrawn at the end of core life. At the end of core life, of course, the k excess is at a minimum. The significance of this is that, under normal operating conditions, two and possible three rods could stick in the operating position, and the reactor would still be shutdown by an adequate margin.

It must be conceded that a buildup of debris, rod dimensional changes or deterioration of off-set mechanism bearing surfaces may reduce rod drop speed, and under extreme conditions may even result in complete binding. These conditions would proceed very slowing, would not affect all rods uniformly, and, because of the requent operation of the rods, would not go undetected.

Binding as a result of pressure vessel distortion due to pressure surges of any kind is highly unlikely. The system is equipped with pressure relief valves which, because this is a completely filled system, will very effectively limit any pressure surge.

In summary, the likelihood of conditions developing, undetected, that would result in rod binding is remote. Additionally, it would be necessary for all four rods to simultaneously completely stick before shutdown capability is lost.

Nevertheless, plans are being prepared for the future installation of a boric acid solution shutdown system. This system, which is presently being designed, consists of a 250 gallon stainless steel tank to be located on the 5th level of the reactor containment building, an air pressure supply, and stainless steel piping to the reactor primary coolant system. The 250 gallon tank is maintained at a pressure of 85 psig. In the event auxiliary shutdown should be required, up to 250 gallons of a 4% solution of boric acid may be injected under pressure into the rector primary coolant loop. This quantity of solution is sufficient to poison out the reactor even when further diluted by the entire volume of the loop. If at any time future operating experience should indicate the necessity, the system will be installed.

15. "Page 3 of Addendum No. 1 states that pressure relief valves in the primary system will be used rather than a rupture diaphragm. Provide a description of these valves and their location and discuss the design criteria for the valves in relation to relief of pressure from boiling and water hammer. Will the valves allow pool water flow into the primary system after relieving pressure? If not, discuss this design change in regard to the need for flooding the core after a steam water expulsion as postulated in the maximum credible accident (p. 13-24 of the Hazards Summary Report)."

Two relief valves are installed in the primary system. One relief valve is located in the piping tunnel on the reactor primary inlet line. Its purpose is to protect the piping in the equipment room from possible overpressure when pressurizer valve 527C and isolation valves 507A and 507B are closed. The second relief valve is located as shown on the attached sketch. The valves are 2 inch Mipco series 1, O.W. valves.

In the event of reactor scram and failure of both the forced and natural convection cooling, the heat from 5 MW operation will be dissipated without operation of the pressure relief valve. From 10 MW operation about two cubic feet of water will be expelled from the primary system through the relief valve in order to relieve the pressure (1).

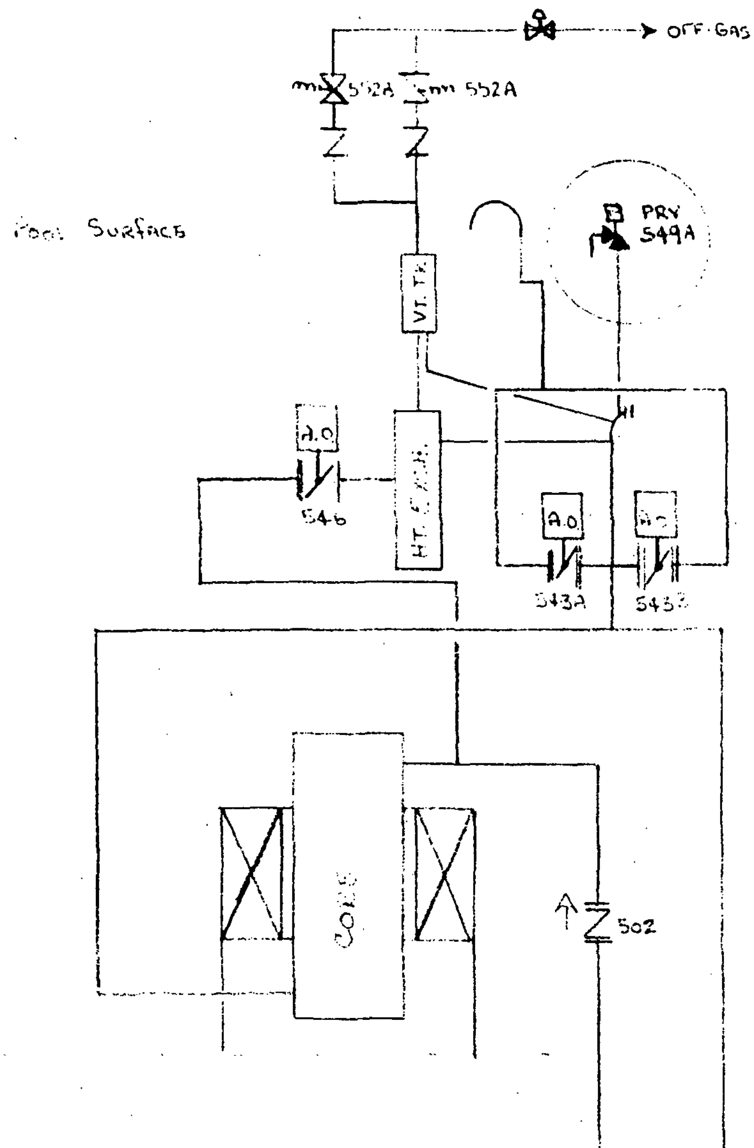
Two factors prevent water hammer from the operation of these valves. The small pipe opening into the large system will tend to suppress any hammer, also the overpressurization of the system due to steam formation alleviates the solid water system and will act to suppress any pressure surges.

These valves will not allow pool water to enter the core. Provision for this is by two check valves located in the pool on the primary system as part of the original design (ref. valves 550C and 550D Figure 5.1 of Hazards Summary Report).

One advantage of the relief valves is that after the pressure has been relieved the valves close, preventing further release of gases or water in the primary system, keeping the potential radioactivity release to the containment to a minimum.

---

[1] Verbal communication with General Electric Company.



[redacted]

Figure 15-1. Location of Reactor Coolant Loop Pressure Relief Valves 549A and 549B.

16. "In reference to our discussion on March 14 concerning the siphon break line, your views are desired as to whether or not modifications should be made to reduce the possibility that activity would be discharged directly from the primary system into the containment under certain accident conditions."

It is our considered opinion that the siphon break system should not be modified from the present design at this time. The most significant factor in arriving at this conclusion arises from a consideration of transient analysis. The reactor characteristics which cause it to recover from a reactivity transient are the negative temperature and void coefficients. The more important of these in this discussion is the void coefficient. The opening of the anti-siphon valves will very rapidly and very effectively reduce the pressure in the primary system. This pressure reduction will permit boiling thus introducing void and a concomitant sharp decrease in reactivity which will limit the power excursion. This situation was discussed in Section 13.5.3 of the Hazards Summary Report, but in that discussion the importance of the shutdown action of partially voiding the core was not emphasized.

It seems to us far more important to limit the magnitude of an excursion and total fission product release than to attempt to minimize the introduction of fission products into the containment building at the expense of possible enhancement of core damage.

The anti-siphon system as presently designed is sufficiently simple to permit reliable routine operation. At the same time it permits us to maintain the primary system pressurized so that any small fission product leak, such as might occur through a single leaking fuel plate, can be isolated by the primary coolant system.

17. "The design described in the construction permit application provided a water leg seal for coolant lines between the reactor room and the heat exchanger room. This seal afforded considerable advantage in assuring that a positive seal would be maintained between the two rooms and in assuring that process system failures outside the reactor room would be unlikely to uncover the core or provide an open pathway to a region outside containment. Explain the basis for and justify elimination of this water leg seal from the design."

The water leg seal for the piping between the reactor pool and the heat exchanger room which was described in Section 4.1.2 of the Preliminary Hazards Report was eliminated from the final design because a more detailed analysis showed that the advantages in inclusion of the water leg seal were insignificant compared to the additional complexity in piping construction required by its use.

At the time the Preliminary Hazards Report was prepared the building design was such that the portion of the reactor coolant piping tunnel between the reactor biological shield and the containment building was open to containment building atmosphere. The trench was shielded by removable sections of concrete that provided no sealing. With this design a water seal was necessary to complete the containment seal.

As presently designed, reactor containment begins at the outer end of the coolant piping penetrations into the reactor pool. The coolant piping tunnel is simply outside containment. The only gain in including a water leg is the extension of containment by a few feet. This minimal gain is more than offset by the problems associated with the introduction of complex piping arrangements and by the possible corrosion problems that might result from immersion of the aluminum pipes in untreated water.

APPENDIX I

Specification for Reactor Pressure Vessel

GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPT.
P.O. BOX 254
175 CURTNER AVENUE
SAN JOSE, CALIFORNIA

TRANSMITTAL

PROJECT: UNIV. OF MISSOURI

REQUISITION: 408-16950

Date: AUGUST 11, 1964

TO: DISTRIBUTION

FROM: E. H. SEVERANCE
ENGINEERING PLANNING

SUBJECT: REACTOR PRESSURE VESSEL

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RESEARCH REACTOR FACILITY
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UNIVERSITY OF MISSOURI

R1-207-2

ATTACHED	THEY ARE TO BE USED FOR	THESE ARE	PLEASE NOTE	PLEASE
___ REQUISITIONS	___ DISTRIBUTION	___ PRELIMINARY	X REVISION	___ COMMENT
___ PRINTS	___ ESTIMATING	___ UNCHECKED	___ HOLDS	___ APPROVE
___ PHOTOSTATS	___ ORDERING MAT'L	___ CHECKED		___ DISTRIBUTE
X SPECIFICATIONS	___ SECURING QUOT.	X FINAL		
___ TRACINGS	___ CONSTRUCTION			
___ REPRODUCIBLE	___ INFORMATION			
	X PROCUREMENT			

DRAWING NUMBERS, TITLES & COMMENTS

21A5125, Revision 1 - SPECIFICATION FOR REACTOR PRESSURE VESSEL

RECEIVED
NOV 12 1964
C. L. T. GABLER A.I.A. & ASSOCIATES
ARCHITECTS & ENGINEERS
3300 BOOK TOWER DETROIT, MICHIGAN 48226

CORNELIUS L. T. GABLER A.I.A.
AND ASSOCIATES
ARCHITECTS AND ENGINEERS
DETROIT MICHIGAN

ARCHITECTURAL DEPT.
BY Dettloff DATE 12-16-64
STRUCTURAL DEPT.
BY .... DATE ....
MECHANICAL DEPT.
BY .... DATE ....
ELECTRICAL DEPT.
BY .... DATE ....

IN CHECKING OVER CONTRACTORS DRWGS THIS OFFICE ASSUMES NO RESPONSIBILITY FOR THEIR ACCURACY SEE SPECIFICATIONS

APPROVED X
APPROVED PROCEED
AS NOTED

ATPE-139 (2-64)

This sheet is not part of the equipment specification. It is to be included with the Material Request, indicating in general the things that Vendor should include in the quotation and pointing out where performance guarantees, etc., should be added to the contract.

1.0 PERFORMANCE GUARANTEES

In addition to the warranty of material and workmanship, the Seller shall guarantee that the completed equipment will meet the specific performance requirements at design conditions as specified in Section 5.0 of Specification 21A5125.

The Seller shall guarantee that the equipment furnished will be suitable for the service described herein and will safely meet all the conditions of operation specified in this specification.

2.0 INFRINGEMENT PROTECTION

2.1 All royalties or other charges for patents to be used in the specified equipment shall be considered as included in the contract price.

2.2 The Seller agrees to indemnify and hold harmless the Buyer against any and all judgements, damages, cost and expenses which may be awarded against the Buyer in any suit, action or proceeding brought against the Buyer for infringement or alleged infringement of a patent, by a court of competent jurisdiction, arising out of the use of the specified vessel by the Buyer in the ordinary course of their operation for the purposes herein stated. The Seller further agrees that if any suit or suits for infringement of a patent or patents is instituted against the Buyer as specified above because of the use of said vessel, the Seller will assume defense thereof if promptly notified of any such suits. It is expressly understood that in assuming the defense of such suit or suits, the Seller shall have control of same, but the Buyer shall be kept fully informed of the progress thereof, and shall have the right to confer about and give advice and assistance regarding same.

3.0 PROPOSAL DATA

The following information shall be submitted with the Seller's proposal:

3.1 Deviation from this specification shall be noted.

3.2 Outline drawings showing principal dimensions and the location of all nozzles and key locating dimensions. (This may take the form of Drawing 237E501 with all exceptions noted).

3.3 Descriptive literature, if available, shall be provided describing facilities, manufacturinging capability (especially pertaining to aluminum), etc.

3.4 A complete description (including chemical and physical properties) of alternate materials proposed for all pressure-containing parts.

3.5 <u>Spare Parts</u>

A list of spare parts recommended for five (5) years of operation shall be submitted as a separate part of the proposal. Prices shall be quoted separately.

1.0 SCOPE

1.1 This specification defines the engineering requirements of the equipment specified herein.

1.2 The work done by the Seller in accordance with this specification shall include all necessary design, development, analyses, drawings, evaluation of materials and fabrication methods, fabrication, shop testing, inspection and preparation for shipment and installation requirements and procedures.

1.3 Seller shall have representative at site during the installation and handling process.

2.0 RESPONSIBILITY

The Seller shall accept full responsibility for his work and for compliance with this specification. Review or approval of drawings, procedures, data or specifications by the Buyer with regard to general design and controlling dimensions does not constitute acceptance of any designs, materials or equipment which will not fulfill the functional or performance requirements established by the purchase contract.

3.0 GENERAL DESCRIPTION

3.1 The aluminum vessel assembly will be used as a pressure container supporting the nuclear research reactor core and reflector.

3.2 The equipment to be furnished in accordance with this specification shall be the reactor vessel and removable heads with attachments, reflector tank, nozzles, etc., arranged as shown on Drawing 237E501 and complete with attachments for vessel support, support of reactor core, lifting and handling of the vessel head, and support of reflector and reflector tank.

3.3 Suitable lifting arrangement shall be provided to allow installation of assembly at the site. Final location is in an 8-foot diameter by 30-foot deep tank.

3.4 With entire assembly in water, the normal surface depth of water to pool floor is 30 feet.

3.5 The plant site is the Research Reactor Facilities, located at Research Park, University of Missouri, Columbia, Missouri. User will be curators of the University of Missouri.

4.0 CODES

4.1 The reactor vessel design, fabrication and testing shall be done in accordance with the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section VIII, latest edition and applicable latest addenda, and Case Interpretations pertaining to nuclear pressure vessels and the laws, rules, and regulations of the State of Missouri. The intent of paragraph P-96 of Section I of the Power Boilers shall be met, although the vessel is non-ferrous. The Seller shall propose the details to meet the intent of P-96 for the approval and concurrence of the Buyer.

4.2 The completed vessel shall be stamped with the applicable ASME Boiler and Pressure Vessel Code stamp and by any marks or identification required by the State of Missouri. Location of stamp shall be on part 24 of Drawing 237E501.

4.3 The Seller shall be responsible for obtaining approval of deviations from the Code after approval of the deviations by the Buyer.

4.4 The intent of this specification is to supplement the requirements of the codes specified above and to encompass the means whereby the design objective is satisfied.

4.5 ASTM material specifications shall be per latest revision.

5.0 DESIGN REQUIREMENTS

5.1 Operating Conditions

5.1.1 Internal - Reactor

5.1.1.1 Power Generation - Fuel 10 MWt

TITLE REACTOR PRESSURE VESSEL

5.1.1.2	Fluid:	Demineralized water, 1 μ mho conductivity
5.1.1.3	Flow:	3600 gpm
5.1.1.4	Fluid Temperature In:	140°F
5.1.1.5	Fluid Temperature Out:	159°F
5.1.1.6	Pressure In At Reactor Inlet:	60 psig

5.1.2 External - Pool

5.1.2.1	Flow:	1200 gpm
5.1.2.2	Fluid:	Demineralized water, 1 μ mho
5.1.2.3	Fluid Temperature:	
5.1.2.3.1	Bulk Pool:	93°F
5.1.2.3.2	Pool Outlet:	100°F
5.1.2.4	Pressure:	Static head of pool water per Figure 1, attached

5.2 Design Considerations

5.2.1 Internal Pressure and Temperatures

5.2.1.1	Design Pressure (Reactor Inlet):	100 psig
5.2.1.2	Design Maximum Metal Temperature:	250°F
5.2.1.3	Hydrostatic Test Pressure (per code):	150 psig

5.2.2 Dead and Live Loads

5.2.2.1	Total Core (Fuel) Weight:	160 pounds
5.2.2.2	Reflector Weight:	

TITLE REACTOR PRESSURE VESSEL

5.2.2.2.1	Total Beryllium	350 pounds
5.2.2.2.2	Total Graphite	1800 pounds
5.2.2.3	ΔP Across Core at 3600 gpm:	13 psig
5.2.2.4	ΔP Across Reflector at 1200 gpm:	2 psig
5.2.2.5	ΔP Across Island Tube:	2 psig
5.2.2.6	Vessel Support Reactions:	Per figure #3
5.2.2.7	Control Rod Bracket Impact Load	1200 inch-lbs damped in 3 inches
5.2.2.8	Control Rod Bracket Dead Load (each)	300 pounds
5.2.3	Pipe Reactions	As shown on figures #1 and #2 attached.

5.2.4 Cyclic Loading

Reactor operating cycle is 8 hours on - 16 hours off - weekend shutdown or continuous at 10 MW for 40 days.

5.2.5 Design Objective

The objective shall be to design and fabricate this reactor pressure vessel to have a useful life of twenty years under operating conditions specified by the Buyer.

6.0 DESIGN ANALYSIS

6.1 Requirements

The design analysis is divided into three divisions, each of which is to meet specific requirements. The divisions and the requirements for each division are as follows:

6.1.1 ASME Boiler and Pressure Vessel Code Design

6.1.1.1 This design is required to obtain the ASME Boiler and Pressure Vessel Code stamp. The Code formula and the rational design required to fulfill the intent of the Code will be used to size most parts of the main shell to enable material orders to be placed.

6.1.1.2 This design analysis will be used to define the parts of the reactor vessel that may require experimental stress analysis. This analysis shall account for all combinations of peak loads in conjunction with maximum metal temperatures that may be coincident on the reactor vessel for their influence on membrane stresses at any steady operating conditions.

6.1.2 Steady-State Analysis

6.1.2.1 This design is the investigation of all principal stresses and shears and their combinations at all critical sections of the reactor pressure vessel that result from the combinations of steady-state loadings. The results of the above design will be used to augment the design under 6.1.1 encompassing those designs not covered by Code formulas. Parts of the design under 6.1.1 will partially fulfill these requirements.

6.1.2.2 The results of this design will be used in conjunction with the analysis of cyclic operations under 6.1.3.

6.1.2.3 This type of design will be used to analyze data from any experimental stress analysis work where possible.

6.1.2.4 The results of this design will be used to judge the adequacy of the design of the reactor pressure vessel and structures using the allowable stress limits as set forth in Appendix I.

6.1.3 Transient Analysis

The transient analysis is the analytical testing of the designed reactor vessel placing all operating loads and cycles of operation on each critical part or parts and determining the alternating stresses for the various loadings. The cyclic loadings, number of cycles or operation and method of determining the alternating stresses and fatigue life of the reactor vessel given in Appendix I will be used as a basis of judging whether the design objective is fulfilled. The analysis will also be used to check adequacy of any required thermal baffling used to control or limit thermal stresses.

TITLE REACTOR PRESSURE VESSEL

⚠ 6.1.4 The Buyer will perform the steady-state analysis paragraph 6.1.2 and the transient analysis paragraph 6.1.3 after receipt of detail drawings and Code Design data paragraph 6.1.1 from the Seller. These data will then be transmitted to the Seller to obtain the Code stamps.

6.2 Calculation of Stresses

6.2.1 The detailed structural analysis required to meet the requirements of 6.1 shall be made for the stresses resulting from internal pressure, external and internal loadings and the effects of steady and fluctuating temperatures and loads for regions given in 6.3 which involve changes of shape, structural discontinuities and points of concentrated loadings (control rod brackets, reflection support and fuel support.)

6.2.2 Where dimensions and loading conditions permit, the adequacy of structural elements will be verified by comparison with completely analyzed elements. The calculations shall include a complete analysis of stresses under steady state and transient conditions to determine suitability of the design with respect to the allowable stress given in Appendix I and to determine the operational limitations with respect to fatigue of the reactor vessel materials over the life of the reactor vessel (Design Objective) using the loading conditions supplied by the Buyer.

6.3 Parts of the Reactor Pressure Vessel to be analyzed shall include all nozzles of a nominal size of six inches and over and those regions of the vessel shown in dark lines on Figures 1, 2 and 3 attached.

6.4 Calculations

The calculations shall be clear and in sufficient detail to permit independent checking. Specific references shall be given for all formulas and methods used or the formulas and methods shall be derived independently.

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6.5 Descriptions of Computer Programs

If computer programs are used to obtain solutions to design problems, the Seller shall furnish the Buyer the description of each different computer program used. These descriptions shall be furnished with the first issue of the design calculations incorporating such programs. The computer program description shall include computer type, program capabilities, assumptions, limitations and statement of availability.

1 6.6 (Deleted)

6.7 Summary Report

After completion of the reactor vessel design, the Seller shall furnish the Buyer additional copies of all calculations plus a summary report of results of all computations. Each copy shall be bound in a suitable paper binding and indexed.

7.0 CONSTRUCTION

7.1 Weld Joints

7.1.1 Weld joints shall be located so that openings in the shell do not intersect any longitudinal or circumferential welds in the vessel where possible.

7.1.2 Weld joints shall be designed to facilitate a maximum of radiographic examination to ASME Boiler and Pressure Vessel Code Standards UW-51.

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7.2 Vessel Supports

Vessel supports, internal supports, their attachments and adjacent shell shall be designed to take maximum combined loads as given in Section 5.2 and in Figure 1, 2 and 3 attached. There shall be no gross yielding of the pressure vessel supports causing permanent displacement.

7.3 Head Closure

7.3.1 The head closure shall be designed for removal and reassenbly remotely under 12 feet of water. Seller shall specify and submit bolt torque requirements.

7.3.2 The head seal shall be a flexatallic gasket with a packing gland seal at the island tube as shown on Drawing 237E501.

7.3.3 Stud, nut and washer arrangement shall be basically as shown on Drawing 237E501.

7.3.4 The vessel head is to be removed over the island tube.

7.3.5 Zero leakage is allowed using water as the pressurizing fluid at design pressure, having used operating bolt-up loads. This is required after all ASME Boiler and Pressure Vessel Code-required hydrotesting.

7.4 Flanges

7.4.1 Flange Joints and Facings

Unless otherwise stipulated, flange joint types shall be of ASA form for steel flanges but rated in accordance with Appendix II, Section VIII of Boiler and Pressure Vessel Code.

7.4.2 Standard Flange Sizes

Standard flange sizes shall be limited to those sizes, having proper pressure and temperature ratings, that require bolt sizes from 3/4" to 2" in diameter.

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7.4.3 Standard Flange Types

Flanges requiring welding to attach to the reactor vessel or piping shall be limited to those types that attach by full penetration welds such as welding neck. Threaded or socket welding type are not permitted.

7.4.4 Studs and Nuts Thread for Standard Flanges

The threads on studs and nuts 3/4" through 2" in diameter shall be in accordance with Paragraph UCS-11 of Section VIII of the ASME Boiler and Pressure Vessel Code.

7.5 Stress Concentration

Care shall be taken in design and fabrication to minimize stress concentrations at changes in sections or penetrations. Fillet radii shall be equal to at least half the thickness of the thinner of the two sections being joined. If reinforcement for openings requires local increased vessel shell thickness, such reinforcement shall extend at least one and one-half times the diameter of the opening from the center of the opening. These requirements are not to be construed as a waiver for evaluating the stresses for use in the analysis for cyclic operation.

8.0 MATERIALS

8.1 Alternate Materials

The Seller shall be free to suggest alternate materials during preparation of detailed drawings and shall bring such alternates to the attention of the Buyer but shall not make substitutions without approval of the Buyer. All materials to be used shall be indicated on the Seller's detailed drawings.

8.2 Records

The Seller shall maintain complete records showing use of all materials so that it will be possible to relate every component of the finished reactor vessel to the original certification of the material and the fabrication history of the component.

TITLE REACTOR PRESSURE VESSEL

8.3 Base Material

8.3.1 All aluminum alloy 6061-T6 in the pressure vessel shall be heat treated to the T-6 condition after welding and before final machining.

8.3.2 Forgings

Forgings shall be aluminum, ASTM B247, Alloy 6061-T6.

8.3.3 Plate

Plate shall be aluminum, ASTM B209, Alloy 6061-T6.

8.3.4 Base material shall be selected and worked to produce as fine a grain size as practical. Conditions that border on critical grain growth shall be avoided.

8.3.5 Forged material ingots shall be produced by "vacuum degassed pouring."

8.4 Attachments

Aluminum material used for internal attachments shall be Alloy 5052, 5086 or 6061-T6.

8.5 Pipes and Tubes

Pipes and tubes shall be ASTM B241, Alloy 6061-T6, or ASTM B210, Alloys 5154 or 6061-T6.

8.6 Studs, Nuts, Bushings and Washers

All austenitic stainless steel studs, nuts, bushings and washers shall be solution heat treated. The studs and the nuts shall be new malcomized as per G. E. Specification P11BYP1-S3.

8.6.1 Studs and bolts shall conform to ASTM A193, Grade B8.

8.6.2 Nuts shall conform to ASTM A194, Grade 8.

8.6.3 Bushings and washers shall be Type 304 austenitic stainless steel.

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8.7 Weld Electrodes and Rods

Material for weld electrodes and rods shall be selected from ASTM B285 for aluminum or equivalent for other processes and reported to the Buyer for approval.

9.0 FABRICATION

9.1 General

9.1.1 Procedures

The Seller shall submit for the Buyer's approval all weld and weldor qualification procedures, all heat treating procedures including those for preheat and postheat treating, all repair procedures, all cleaning and rinsing procedures, all preserving procedures and a list of all cleaning agents and preservatives together with their chemical composition. Results of exploratory trials and tests to establish procedures are desirable evidence and may be included.

9.1.2 Definitions

Major defects in base material and weld metal, as referred to in this specification are defined as defects which are equal to or greater in depth than 10% of the thickness of the section, or 3/8" whichever is less.

9.1.3 No tool shall be used during fabrication which may deposit copper on any portion of the assembly.

9.2 Fabrication Procedure Qualification

The procedures used for the fabrication of the pressure vessel shall be qualified. The qualification of procedures shall include the following:

9.2.1 The Seller shall submit proposed procedures to the Buyer in advance of performing the work of qualifying.

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9.5 Surface Condition

All surfaces in contact with primary coolant water shall have a finish of 250 micro-inch or better, except as indicated on Drawing 237E501. In the case of weld joints, local grinding required for X ray shall be considered as meeting this requirement.

10.0 INSPECTION AND TEST

10.1 Procedures

The Seller shall submit for the Buyer's approval all test procedures inspection procedures, and certified copy of test results.

10.2 Ultrasonic Inspection - Base Material

10.2.1 Plate

10.2.1.1 Method - All plates for the reactor vessel shall be ultrasonically examined by the longitudinal beam technique in accordance with ASTM A 435. Ultrasonic transducers may be any convenient area up to one square inch.

10.2.1.2 Reference Specimen - A reference specimen shall be used to calibrate the equipment. The reference specimen shall be of the same nominal thickness and composition as the material being tested and it shall have a flat-bottom hole with a depth of ten per cent of the thickness for thicknesses over two inches, and a depth of 25 per cent of the material thickness for thicknesses two inches and less. Calibration hole diameter shall be one-half inch for material four inches and less in thickness and shall be 5/8 inch for material more than four inches in thickness.

10.2.1.3 Acceptance Criteria - A defect from which one or more continuous ultrasonic indications cause a loss of back reflection greater than the reference defect which is monitored during movement of the transducer two inches in any direction shall be unacceptable. In addition, any continuous echo indication, regardless of back reflection

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indications, covering an area in which the shortest dimension exceeds two transducer widths and the longest dimension exceeds four inches shall also be unacceptable.

10.2.2 Forgings

10.2.2.1 Method - All forgings for the reactor vessel shall be examined by the ultrasonic method in accordance with ASTM A388, Recommended Practice for Ultrasonic Testing and Inspection of Heavy Steel Forgings, and paragraph 10.2.1.1, previous page. Ring forgings and other hollow forgings shall, in addition, be tested using the shear wave technique.

10.2.2.2 Reference Specimen - The reference specimen shall have the same nominal thickness and composition as the forging. Calibration of equipment shall be in accordance with paragraph 10.2.1.2 for longitudinal wave inspection. For the shear wave technique, a groove one inch in length, no greater than 1/8-inch wide and three per cent of the nominal thickness in depth shall be used.

10.2.2.3 Acceptance Criteria - Acceptance for longitudinal wave inspection shall be as described in paragraph 10.2.1.3. For shear wave inspection, a screen indication in excess of that associated with the standard groove defect (refer to paragraph 10.2.2.1) is unacceptable unless the defect is removed and the forging repaired.

10.3 Welds

All welds shall be examined by radiographic, ultrasonic and/or liquid penetrant inspection.

10.3.1 Radiographs

10.3.1.1 Film and technique for radiographing and standards of acceptance shall be in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code, paragraph UW-51. The double film tedhnique shall be used. Gamma rays shall not be used unless approved by the Buyer.

10.3.1.2 The Seller shall retain both sets of radiographs for the period of time required by the ASME Code. After this period, they will be sent to the Buyer upon request.

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10.3.1.3 Films shall be suitably marked to identify the weld. Film identification markings shall coincide with the detail drawing markings for each weld.

10.3.2 Liquid Penetrant Inspection

All welds shall be liquid penetrant inspected. Liquid penetrant indications in the welds and in the base material adjacent to the weld shall be considered unacceptable if one of the following conditions exist:

10.3.2.1 Crack-like indications or incomplete fusion

10.3.2.2 Linearly-disposed spot indications of four or more spots spaced 1/4 inch or less from edge-to-edge of the indication.

10.3.3 Ultrasonic Inspection

In addition to the above inspections, ultrasonic inspection of welds as required by ASME Code Case 1273N shall be performed in accordance with Code Case 1275N.

10.4 Repairs

10.4.1 All defects shall be removed to sound metal and verified by liquid penetrant inspection. The defective area shall be repaired by welding, using approved procedures. All major weld-repaired areas shall be radiographed. All major and minor defect repaired areas shall be liquid penetrant inspected, using approved procedures.

10.5 Studs, Nuts, Bushings and Washers

Each finished part shall be liquid penetrant inspected.

10.5.1 Any linear indications oriented in a general circumferential plane are unacceptable.

10.5.2 Linear indications oriented in a general axial plane shall be investigated to determine their nature. Cracks or other sharply-defined linear indications are unacceptable.

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10.6 Test Plate Evaluation

The Seller shall prepare a welded "test plate", or plates of the reactor vessel shell material. The Seller shall prepare test specimens of the base metal, weld metal and weld-heat-affected zone metal in accordance with Appendix II.

Tests required by ASME Code Section I, paragraph P96 shall also be prepared from this "test plate".

10.7 Hydrostatic Tests

10.7.1 Code Test

1 After completion of fabrication but prior to shipment, the reactor vessel shall be pressure tested in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII. Tests shall be conducted at an internal pressure of 120 psia at a water temperature of 160°F for a period of 72 hours. Measurement of pressure axial and radial growth at 24-hour intervals shall be made. The pressure vessel shall exhibit zero leakage during this test. A certified report of all hydrostatic test data shall be submitted.

10.7.2 Sealing Test

To meet requirements of 7.3.5.

10.8 Dimensional Control

Throughout fabrication, dimensional control shall be mandatory, and prior to shipment, a complete recheck shall be made of controlling dimensions as mutually agreed to prior to the start of fabrication.

11.0 PREPARATION FOR SHIPMENT

11.1 Cleaning Procedures

The Seller shall submit for the Buyer's approval, cleaning procedures, preserving procedures and a list of cleaning agents and preservatives together with their chemical content. In lieu of a complete chemical analysis, the Buyer shall accept a report which states the chlorides, fluorides, sulfur and alkaline content. Other harmful elements should also be reported.

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11.2 Interior and Exterior Surfaces

All surfaces of the reactor vessel shall be thoroughly cleaned to be free of iron, lubricant, weld spatter, chips, and other foreign materials. After the interior is cleaned and dried, the reactor vessel body and top head shall be sealed to prevent entry of water, dirt or other foreign material. The seals used on the reactor vessel nozzles shall not affect the weld preparation or flange faces of the nozzles.

11.3 Small Parts

Small, loose pieces, including bolting, tools, gaskets, etc., shall be adequately crated or boxed, for protection during shipment. All pieces shall be marked with the equipment piece number, 301.

11.4 Shipping Weight and Dimensions

Estimated shipping weights and over-all clearance dimensions shall be supplied to the Buyer at least four months in advance of shipping.

11.5 Shipping Frame

The shipping frame shall be designed to support the vessel adequately and securely during the intended modes of shipment and to permit movement by crane handling at the site.

12.0 SUBMITTALS

12.1 Drawings

12.1.1 Outline Drawings - A drawing depicting the outline of the reactor vessel indicating over-all dimensions, location and size of nozzles, location of supports and shipping weights.

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12.1.2 Assembly Drawings - A section drawing depicting the arrangement of the functional parts, parts list and material designatioms. Identify each piece of material contained in the pressure vessel with the mill heat number.

12.1.3 Detail Drawings - Drawings for details of construction such as weld preparations, surface finishes, finished dimensions, lifting attachments, insulation attachments, thermocouple pads, flanges and supports.

12.1.4 Drawings for Approval - Outline, assembly and detail drawings shall be submitted for approval. The detail drawings submitted shall be for design details enumerated in 12.1.3 which are required for coordination with piping and structure and design details which are at variance with the Code or the requirements of this specification.

12.1.5 Controlling Location Arrangement Drawings

One or more drawings shall be devoted exclusively to outline dimensions such that mating components designed and supplied by others, such as piping, anchor bolts, instruments, etc., may be procured for an exact fit with the reactor vessel assembly. These drawings shall show reference to the controlling detail drawings and show over-all dimensions and locations on reactor vessel.

12.1.6 Drawings to be Certified - Outline, Assembly and Detail drawings for design coordination shall, upon completion of the design, be certified to be correct with no further changes required. No alterations may be made to the design after certification without the approval of the Buyer.

12.1.7 As-Built Drawings - Prior to shipment of the reactor vessel, the Seller shall provide an Outline drawing with the actual measured significant dimensions which have been designated prior to start of fabrication. If the final construction differs from the previously submitted Assembly and Detail drawings, corrected drawings shall be provided by the Seller.

12.2 Design Items Requiring Submittal to the Buyer for Approval

12.2.1 Deviations from this specification

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12.2.2 Design analysis calculations

12.2.3 Material selections, deviations or substitutions

12.3 Fabrication Procedures, Qualification Procedures and Processes Requiring Submittal to the Buyer for Approval

12.3.1 Heat treatment procedures. Included would be preheat for welding or cladding, postheat for welding or cladding, heat for forming and heating for stress relief.

12.3.2 Welding and weld repair procedures

12.3.3 Weld and weldor qualification procedures

12.3.4 Cleaning and preserving procedures with chemical composition of solutions or agents.

12.4 Inspection and Test Procedures Requiring Submittal to the Buyer for Approval

12.4.1 Ultrasonic test procedures and results

12.4.2 Liquid penetrant test procedures and results

12.4.3 Radiographic examination procedures and results

12.4.4 Hydrostatic test procedures and results

12.4.5 Leak check procedure and results

12.5 Three copies of material specifications shall be furnished the Buyer for information during procurement stage.

12.6 Records

The Seller shall maintain records of all material qualifications, all weld and weldor qualifications and all process qualifications required by this specification and the material specifications. In addition, the Seller shall maintain records of all tests, such as ultrasonic radiography and hydrostatic. A list of the

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records shall be submitted to the Buyer on completion of the job. The Buyer shall be able to obtain certified copies of such records for a five-year period.

12.7 Photographs

The Seller shall provide the Buyer with progress photographs of the vessel during each significant stage of manufacture. The photographs shall be glossy prints, 8" x 10"; one set consisting of one negative and three prints and the maximum number of sets not to exceed ten.

INSTRUCTION MANUAL, DRAWING & DATA REQUIREMENTS

REVIEW OR APPROVAL OF DRAWINGS, PROCEDURES, DATA, OR SPECIFICATIONS BY THE BUYER WITH REGARD TO GENERAL DESIGN AND CONTROLLING DIMENSIONS DOES NOT CONSTITUTE ACCEPTANCE OF ANY DESIGNS, MATERIALS, OR EQUIPMENT WHICH WILL NOT FULFILL THE FUNCTIONAL OR PERFORMANCE REQUIREMENTS ESTABLISHED BY THIS SPECIFICATION AND THE PURCHASE CONTRACT.

CERTIFIED DRAWINGS SHALL SHOW THE PURCHASE ORDER AND THE EQUIPMENT NUMBER.

TYPE OF DRAWING, MANUAL OR DOCUMENT		NO. REQUIRED & TYPE OF COPY	DUE
1. OUTLINE DIMENSIONS AND FOUNDATION REQUIREMENTS	APPROVAL CERTIFIED	2 PRINTS 1 REPRODUCIBLE	* **
2. ASSEMBLY AND CROSS SECTION DRAWINGS WITH PARTS LIST WITH MATERIAL DESIGNATIONS	APPROVAL CERTIFIED	6 PRINTS 1 REPRODUCIBLE	* **
3. SHOP DETAIL DRAWINGS	APPROVAL CERTIFIED	___ PRINTS ___ REPRODUCIBLE	* **
4. WIRING DIAGRAMS	APPROVAL CERTIFIED	___ PRINTS ___ REPRODUCIBLE	* **
5. ENGINEERING SCHEDULE TO INCLUDE DATES FOR START AND FINISH FOR DESIGN CALCULATIONS, DATA, MATERIAL SELECTIONS, APPROVAL DRAWINGS & DOCUMENTS	APPROVAL	2 REPRODUCIBLE	WITHIN 15 DAYS AFTER AWARD OF ORDER.
6. FABRICATION SCHEDULE WHICH DETAILS THE SEQUENCE OF FABRICATION & INDICATES START & FINISH OF EACH PHASE.	APPROVAL	2 REPRODUCIBLE	WITHIN 30 DAYS AFTER AWARD OF ORDER.
7. DESIGN CALCULATIONS	APPROVAL	6 PRINTS	PRIOR TO FABRICATION
8. PROCEDURES		6 PRINTS	30 DAYS AFTER ORDER
9. INSTRUCTIONS FOR ERECTION OR INSTALLATION, OPERATION AND MAINTENENCE		20 MANUALS	20 DAYS PRIOR TO SCHEDULE SHIPPING DATE.
10. CODE CERTIFICATES		3 ORIGINAL COPIES	10 DAYS AFTER SHIPMENT
1. CERTIFIED PERFORMANCE DATA	CERTIFIED	___ PRINTS	
C F RECOMMENDED SPARE PARTS R one YEAR'S OPERATION TH PRICES		2 LISTS	WITH QUOTATION
13. AS-BUILT DRAWINGS TO INCLUDE ALL PERTINENT DIMENSIONS, INCLUDING LOCATION & LENGTH OF NOZZLES & SUPPORTS. OUTLINE DIMENSION WITHIN ± 1/16" ACCURACY.		2 REPRODUCIBLES	PRIOR TO SHIPMENT

* WITHIN 60 DAYS AFTER AWARD OF PURCHASE CONTRACT AND PRIOR TO FABRICATION

** WITHIN 60 DAYS AFTER RECEIPT OF APPROVAL DRAWINGS OR WITHIN ______ DAYS OF RECEIPT OF PURCHASE CONTRACT IF APPROVAL DRAWINGS ARE NOT REQUIRED.

PREPARED BY	ENGINEERED BY		GENERAL ELECTRIC	DATA & DWG. REQUIREMENTS FOR:	
E. H. Severance	W. A. Pitt		ATOMIC POWER EQUIPMENT DEPT. P. O. BOX 254 2151 SOUTH FIRST STREET SAN JOSE · CALIFORNIA	REACTOR COOLING & POOL LOOP HEAT EXCH.	
APPROVED BY		DATE		ATTACHMENT A TO SPEC: 21A5125	REV. NO.: 1
[signature]				PROJECT: UNIV. OF MISSOURI REQ. 408-16950	
			ISSUED BY ENGINEERING: REACTOR PLANT & SYSTEMS	DATE: 8-11-64	SHT. 1 CONT. ON FINAL

ATPE-350-A

1.0 BASIS FOR DETERMINING STRESSES

1.1 The determination of stresses under this specificeation is based on the maximum shear theory. Stresses are expressed in terms of the "stress intensity" which is defined as twice the maximum shear stress, or as the largest algebraic difference between any two of the three principal stresses at a given point.

1.2 Terms relating to stress determination which are used in this specification are defined as follows:

1.2.1 Membrane Stress: The component of normal or direct stress which is uniformly distributed and equal to the average value of stress across the section under consideration away from structural discontinuities.

1.2.2 Local Membrane Stress: The component of normal or direct stress which is uniformly distributed and equal to the average value of stress across the section under consideration at a structural discontinuity.

1.2.3 Bending Stress: The component of normal or direct stress which varies with the distance from the centroid of the section under consideration.

1.2.4 Primary Stress: A direct stress of shear stress developed by the imposed loading which is necessary to satisfy the simple laws of equilibrium of external and internal forces and moments. The basic characteristic of a primary stress is that it is not self-limiting. In the absence of strain hardening, a primary stress which considerably exceeds the yield strength will result in failure, or at least gross distortion. A thermal stress is never classified as a primary stress. Examples of primary stresses are:

1. Membrane stress in a circular cylindrical of spherical shell due to internal pressure.
2. Bending stress in the central portion of a flat head due to pressure.

1.2.5 Secondary Stress: A direct stress or shear stress developed by the constraint of adjacent parts of by self-constraint of a structure. The basic characteristic of a secondary stress is that it is self-limiting. Local yielding and minor distortions can satisfy the conditions which cause the stress to occur, and failure from one application of the stress is not to be expected. Examples of secondary stresses are:

1. Thermal stress
2. Bending stress at a gross structural discontinuity

1.2.6 Gross Structural Discontinuity: A source of strain or stress intensification which affects a relatively large portion of a structure and has a significant effect on the over-all stress or strain pattern in the structure as a whole. Examples are head-to-shell and flange-to-shell junctions a and junctions between shells of different diameters or thicknesses.

1.2.7 Local Structural Discontinuity: A source of stress or strain intensification which affects a relatively small volume of material and does not have a significant effect on the overall stress or strain pattern nor on the structure as a whole.

Examples are opening and nozzle connections, small fillet radii, small attachments, and incomplete weld penetrations.

1.2.8 Peak Stress: The highest stress in the region under consideration. The basic characteristic of a peak stress is that it does not cause any noticeable minor distortion and is objectionable only as a possible source of fatigue crack or a brittle fracture. It is the increment added to the primary or secondary stress by a stress concentration such as a notch or fillet. Examples are:

1. Thermal stresses in cladding materials and in vessel walls due to rapid temperature changes in the contained fluid
2. Stresses at local structural discontinuities

1.2.9 Operation Cycle: An operational cycle is defined as the initiation and establishment of new operating conditions followed by a return to the operating conditions that prevailed at the begining of the cycle. Intermediate operating conditions may be developed before return to the initial conditions.

2.0 ALLOWABLE STRESSES

The calculated stress intensities shall fall within the allowable values permitted for the combinations described below of the following catagories of stress intensity:

2.1 Primary membrane stress intensity, P_m (See definitions in para. 1.2)

2.2 Primary local membrane stress intensity, P_l (See definitions in para. 1.2)

2.3 Primary bending stress intensity, P_b (See definitions in para. 1.2)

2.4 Secondary Stress intensity, Q (See definitions in para. 1.2)

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2.5 Peak stress intensity, L (See definitions in para. 1.2)

2.6 Primary Membrane Stress Intensity (P_m): Computed as the average value across the thickness of a section of the primary stresses produced by design pressure and other mechanical loads, but excluding all secondary, peak and local membrane stresses. The allowable value of this stress intensity is S_m, and is equal to the allowable stress values permitted by the ASME Boiler and Pressure Vessel Code.

2.7 Primary Local Membrane Stress Intensity (P_l): Computed as the average value across the thickness of a section of the primary stresses produced by design pressure and other mechanical loads including the effect of structural discontinuities, but excluding secondary and peak stresses. The allowable value of this stress intensity is 1.5 S_m.

2.8 Primary Membrane and Primary Bending ($P_m + P_b$), or Primary Local Membrane and Primary Bending ($P_l + P_b$) Stress Intensity: The highest stress intensity of any location across the thickness of a section calculated from the combination of primary membrane plus primary bending stresses, or primary local membrane plus primary bending stresses (whichever produces the higher stress) produced by design pressure and other mechanical loads, but excluding all secondary and peak stresses. The allowable value at this stress intensity is 1.5 S_m.

2.9 Primary and Secondary Stress Intensity ($P_m + P_b + Q$) or ($P_l + P_b + Q$): The highest stress intensity at any location across the thickness of a section calculated from the combination of primary and secondary stresses due to any loading, but excluding peak stresses. The allowable value at this stress intensity is 3 S_m.

2.10 Peak Stress ($P_m + P_b + Q + L$) or ($P_l + P_b + Q + L$): The highest stress intensity at any point across the thickness of a section calculated from the combinations of primary, secondary and peak stresses produced by pressure and mechanical and thermal loads in combination. The allowable value of this stress intensity is dependent on the number of times the loads are to be applied, and is obtained from the fatigue curve of Figure 4 for the material by means of the methods of analysis for cyclic operation described in Section 4.0.

2.11 Triaxial Stresses: When all principal stresses are high and one signed such that the stress differences and the stress intensity are small these principal stresses shall be limited to 3 S_m where this condition occurs.

2.12 Stud Stresses. Stud primary membrane stresses shall be limited to 30,000 psi at steady state design pressure and temperature for bolting material having a minimum yield point at room temperature of 100,000 psi.

2.13 Pure Shear: The average pure shear stress across a section loaded in pure shear shall be limited to 0.6 S_m at temperature.

2.14 Bearing Stress: The average bearing stress under maximum loads shall be limited to 1.5 S_m at temperature.

3.0 CYCLIC OPERATION

The suitability of a vessel component for specified operating conditions involving cyclic applications of leads and thermal conditions shall be evaluated by comparing the resulting stresses with strain-cycling fatigue data which are represented as follows:

3.1 Fatigue Curve: The allowable amplitude of the alternating stress component (on-half of the alternating stress range) is plotted in Figure 4 against the number of cycles allowed. This stress amplitude is calculated on the assumption of elastic behavior and, therefore, has the dimensions of stress but does not represent a real stress when the limit of elastic behavior is exceeded. The fatigue curve is obtained from uniaxial strain-cycling data in which the imposed strains have been multiplied by the elastic modulus and a suitable safety factor has been applied, so as to make the calculated stress intensity amplitude and the allowable stress amplitude directly comparable.

3.2 Fatigue Diagram: The mean stress component during a cycle is plotted as the abscissa, and the alternating stress component (one-half the alternating stress range) is plotted as the ordinate in the modified Goodman diagram in Figure 5. Since thermal stresses and concentrated stresses may safely be allowed to exceed the yield stress for limited numbers of cycles, the proposed method takes account of the shift in the mean stress component when yielding occurs.

3.3 Cumulative Damage Hypothesis: To evaluate the effect of alternating stresses of varying equivalent stress amplitude, a modified Miner's criterion is used. This criterion assumes a linear damage relation and can be expressed as follows:

$$\sum \frac{n_i}{N_i} = 0.8$$

where:

ni = anticipated number for cycles of a given stress amplitude
Ni = maximum allowable number of cycles at the same stress amplitude as obtained from Figure 5, following the procedure in procedures in Section 4.0.

4.0 DESIGN FOR CYCLIC LOADING

The following steps are involved in determining the suitability of a vessel for a specified cycles of operation on the basis of the stresses at a given point.

4.1 Principal Stresses: Consider the value of the three principal stresses at the point versus time for the complete cycle, taking into account both gross and local structural discontinuities and thermal effects. These are designated as σ_1, σ_2, σ_3, for later identification.

4.2 Stress Differences: Determine the stress difference $S_{12} = \sigma_1 - \sigma_2$; $S_{23} = \sigma_2 - \sigma_3$; $S_{31} = \sigma_3 - \sigma_1$ versus time for the complete cycle. In what follows, the symbol S_{ij} is used to represent any one of these three stress differences.

4.3 Alternating Component: Determine the extremes of the range through which each stress difference (S_{ij}) fluctuates and find the absolute magnitude of this range. Call this magnitude $S_{r\ ij}$ and let

$$S_{alt\ ij} = 1/2\ S_{r\ ij}$$

4.4 Mean Component: For each stress range $S_{r\ ij}$, determine the mean value of the range and call its absolute value $S'_{mean\ ij}$. This is the basic value of the mean component. The actual value, $S_{mean\ ij}$ for use in comparison with the Fatigue Diagram depends on whether or not the limit of elastic behavior, S_b, has been exceeding during the cycle, and is determined as follows:

1. For any S_{ij} for which $S_{alt\ ij} + S'_{mean\ ij} \leq S_b$, the mean component is equal to its basic value and

$$S_{mean\ ij} = S'_{mean\ ij}$$

2. For any S_{ij} for which $S_{alt\ ij} + S'_{mean\ ij} > S_b$ and

$$S_{alt\ ij} < S_b, \quad S_{mean\ ij} = S_b - S_{alt\ ij}$$

3. For any S_{ij} for which $S_{alt\ ij} \geq S_b$,

$$S_{mean\ ij} = 0$$

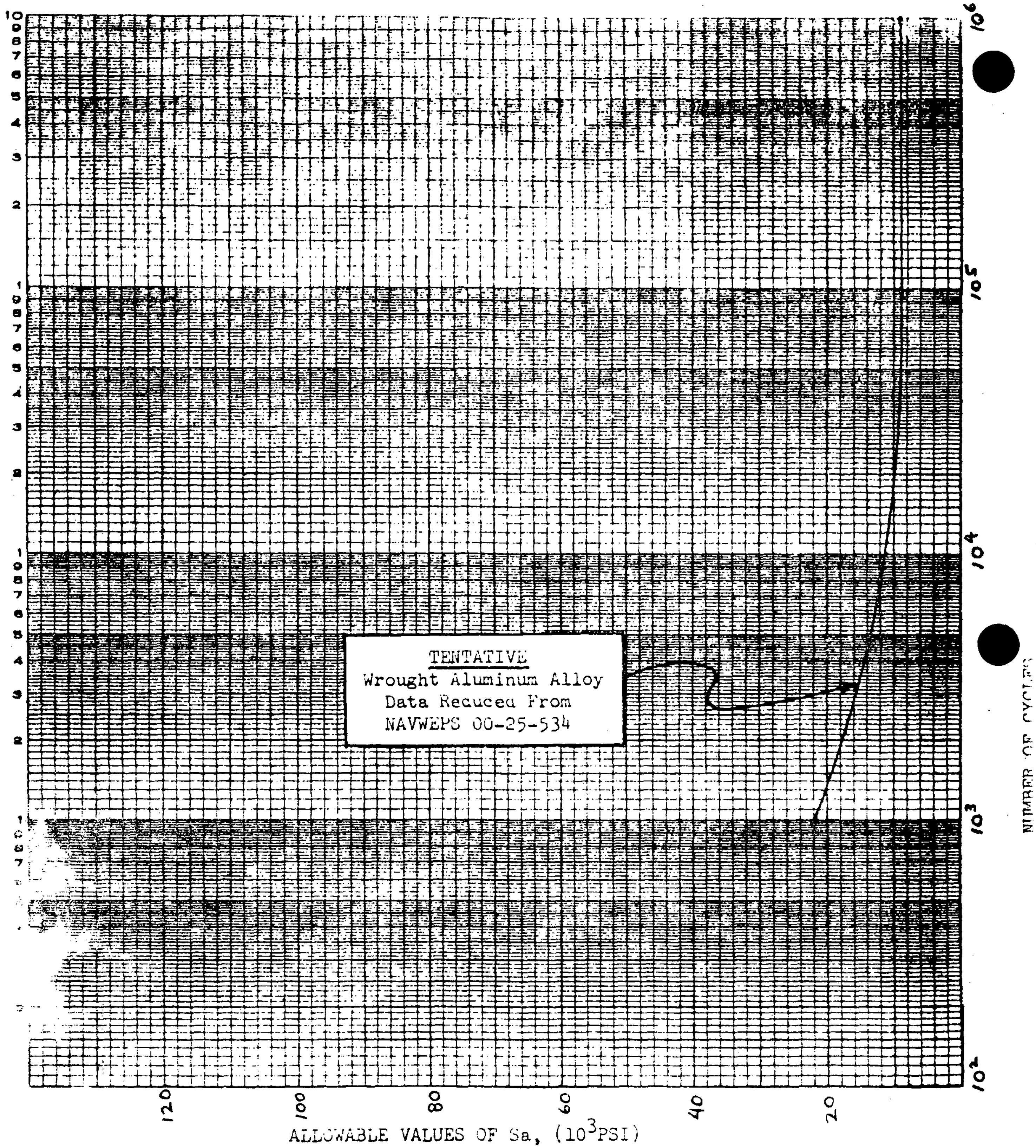
4.5 Use of Fatigue Diagram: Construct a fatigue diagram as shown in Figure 5 for each cyclic loading condition. The value of S_a to be used in the fatigue diagram is the allowable stress amplitude obtained from the Fatigue Curve (Figure 4) for the specified number of cycles. Plot the points (S_{mean} 12, S_{alt} 12), (S_{mean} 23, S_{alt} 23) and (S_{mean} 31, S_{alt} 31) on the fatigue diagram. Draw straight lines through these points from S_u on the abscissa (mean stress) axis to the intercept on the ordinate (alternating stress) axis). This is the equivalent alternating stress intensity for zero mean stress. Consider only the largest value of the intercept for each cyclic loading condition.

Determine from the Fatigue Curve (Figure 4) the allowable number of operating cycles for each of the cyclic loading conditions. Call these values N_1, N_2, N_3, etc., and call the corresponding expected number of cycles n_1, n_2, n_3, etc.

Calculate the usage factors, U_1, U_2, U_3, etc., for the operational cycles from the relations

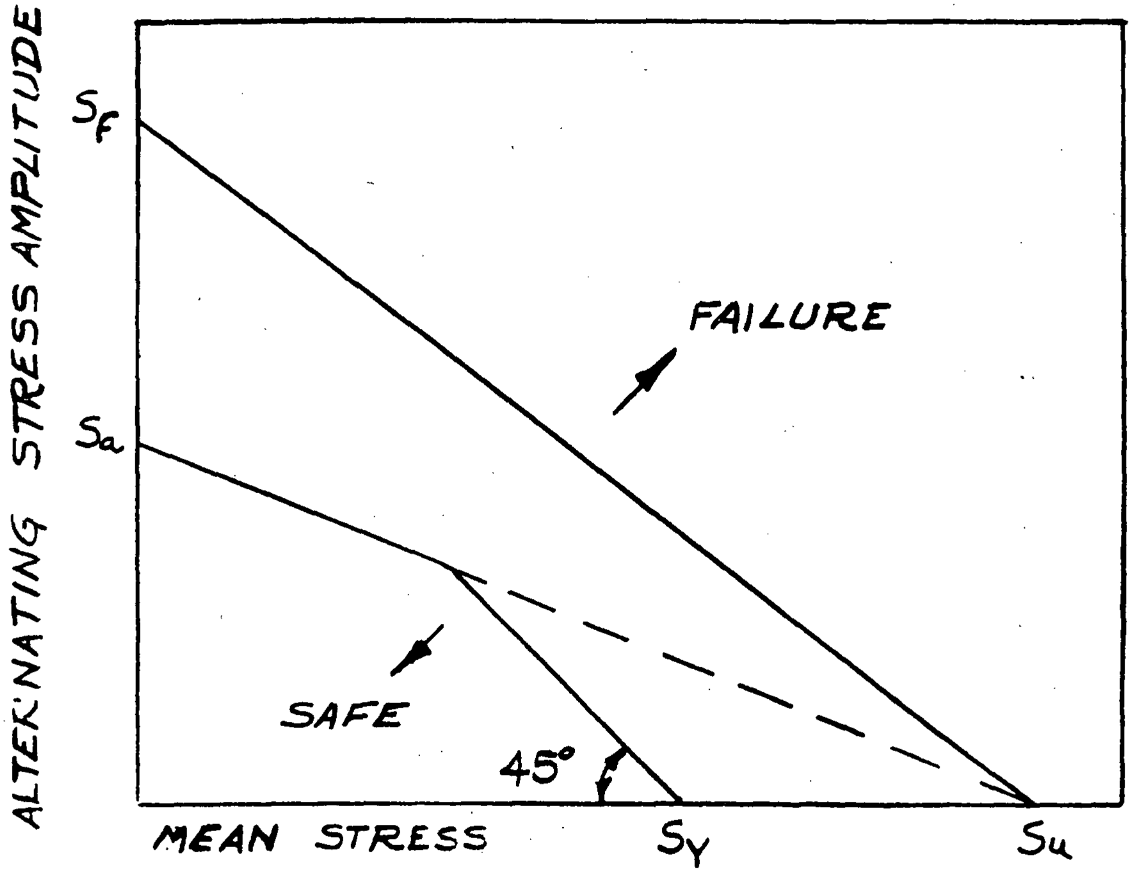
$$U_1 = \frac{n_1}{N_1}, \; U_2 = \frac{n_2}{N_2}, \text{ etc.}$$

The cumulative usage factor, U, calculated as the sum of U_1, U_2, U_3, etc., shall not exceed the value of 0.8.



DESIGN FATIGUE CURVE
FIGURE 4

TITLE REACTOR PRESSURE VESSEL - ANALYSIS



S_u = MINIMUM ULTIMATE STRENGTH.

S_Y = MINIMUM YIELD STRENGTH.

S_a = ALLOWABLE ALTERNATING STRESS AMPLITUDE.

S_f = FATIGUE STRENGTH AT SOME FINITE NUMBER OF CYCLES.

FATIGUE DIAGRAM
FIGURE 5

APPENDIX II

1.0 SURVEILLANCE TEST PROGRAM

1.1 Base Metal - Figure 6

1.1.1 The Seller shall furnish two plates, as shown in Figure 6 from the plate used to make the reactor vessel in the reactor core region or from a similar plate from the same heat.

1.1.2 The Seller shall heat treat these plates with the reactor vessel, or in similar fashion, to insure that they represent the metallurgical condition of the reactor vessel steel, as fabricated, in the reactor core region.

1.2 Welded Plate - Figure 7

1.2.1 The Seller shall furnish a welded plate representative of a reactor vessel longitudinal weld from the plate used to make the reactor vessel in the reactor core region or from a similar plate from the same heat.

1.2.2 The Seller shall heat treat the plate with the reactor vessel, or in similar fashion, to insure that it and the weld represent the metallurgical condition of a vessel weld, as fabricated, in the reactor core region.

1.2.3 The Seller shall furnish documents to the General Electric Company, Atomic Power Equipment Department detailing all metallurgical data and demonstrating that the weld was made in a manner similar to a reactor vessel weld. X-rays of the weld shall be furnished.

PREPARED BY H Severance	ENGINEERED BY WA Pitt	GENERAL ELECTRIC	TITLE REACTOR PRESSURE VESSEL - TEST PLATE EVALUATION	
APPROVED BY	DATE	ATOMIC POWER EQUIPMENT DEPT. P.O. BOX 254 175 CURTNER AVENUE SAN JOSE, CALIFORNIA	SPEC. NO. APPENDIX II to 21A5125	REV. NO. 0
L L Blankenship	4/24/64		PROJECT UNIVERSITY OF MISSOURI	
		ISSUED BY ENGINEERING REACTOR PLANT & SYSTEMS	DATE 4-7-64	SHT. 1 CONT. ON 2

TITLE REACTOR PRESSURE VESSEL - TEST PLATE EVALUATION

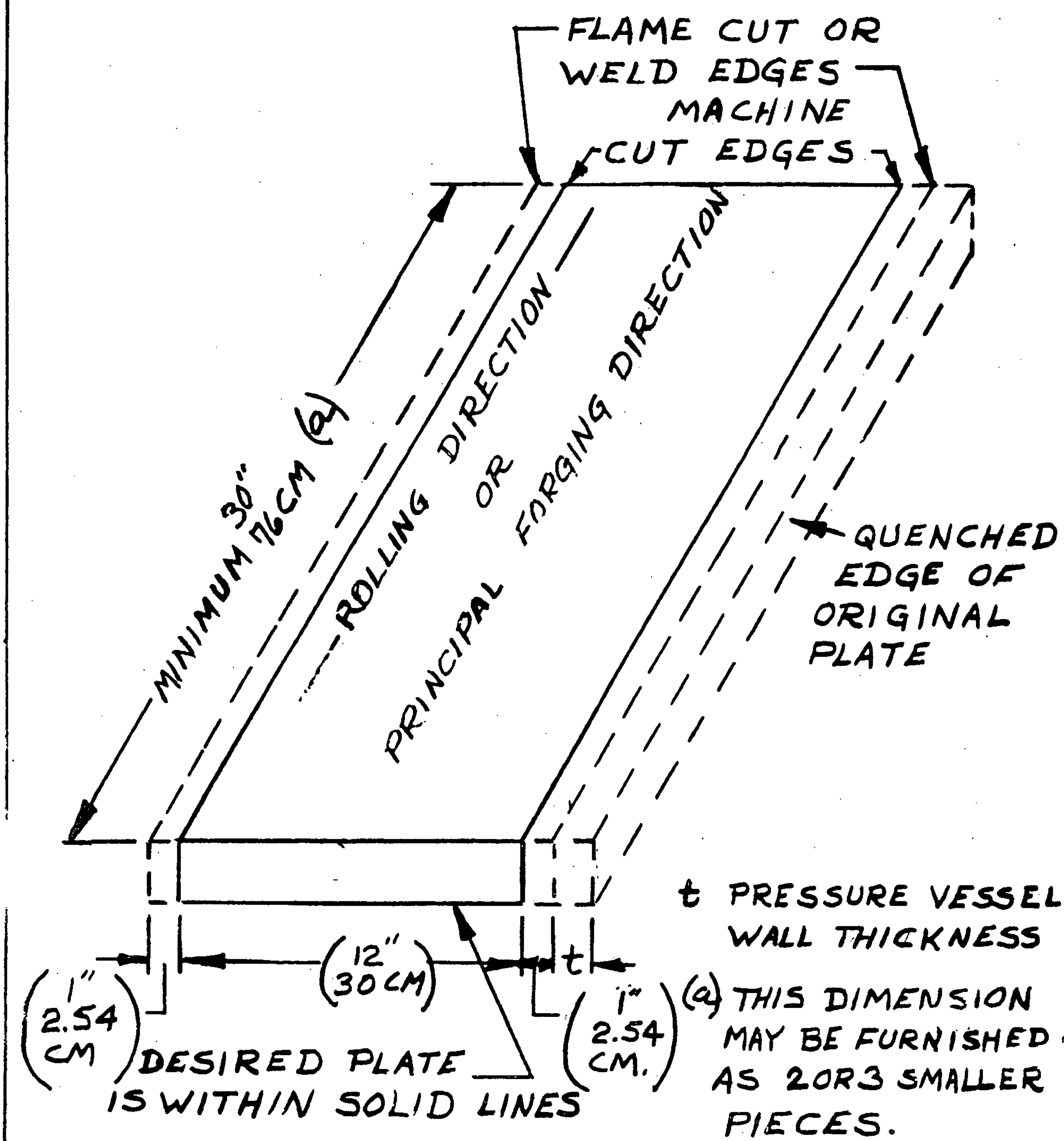
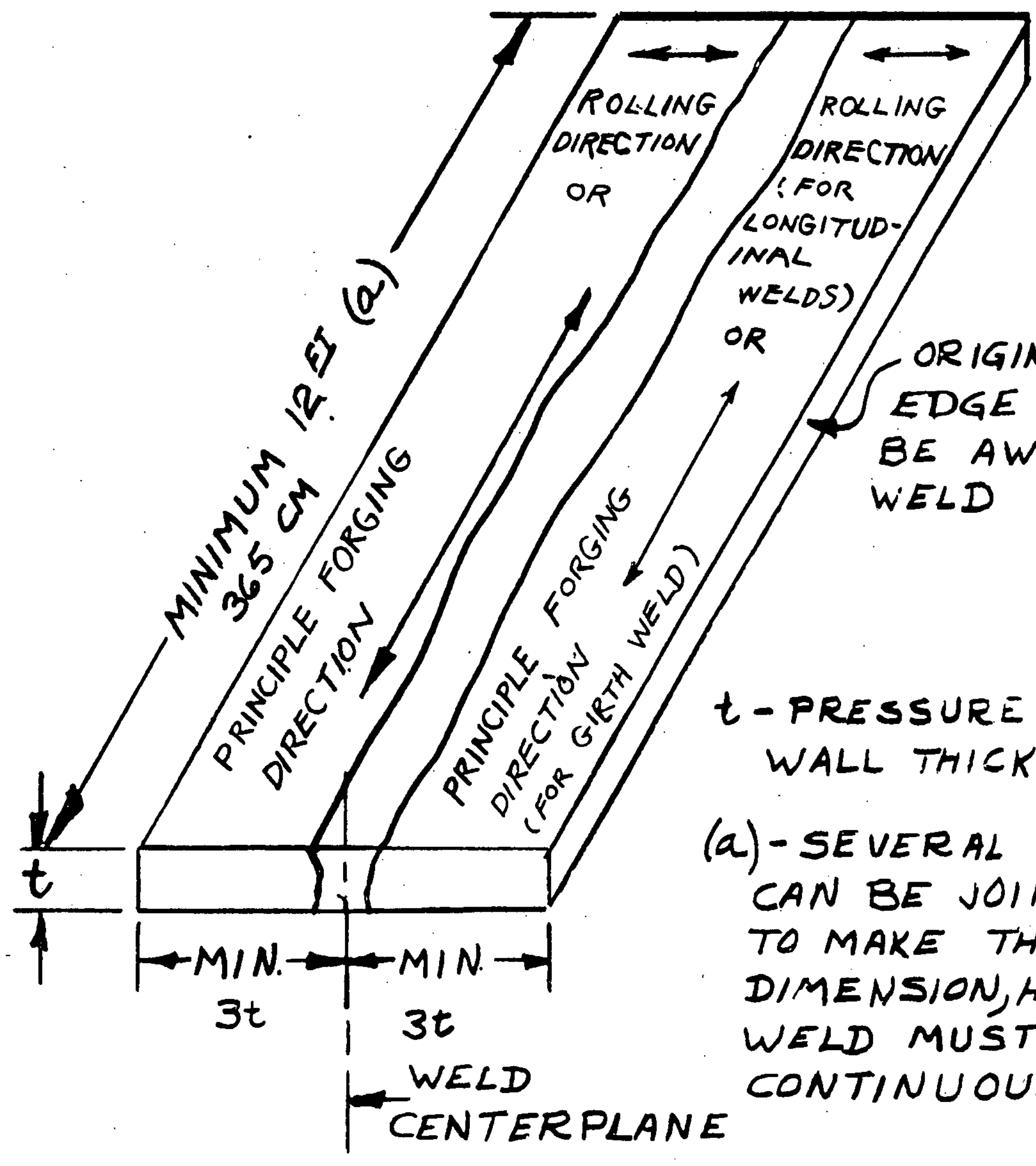


FIG 6 TEST PLATE
BASE METAL SPECIMENS

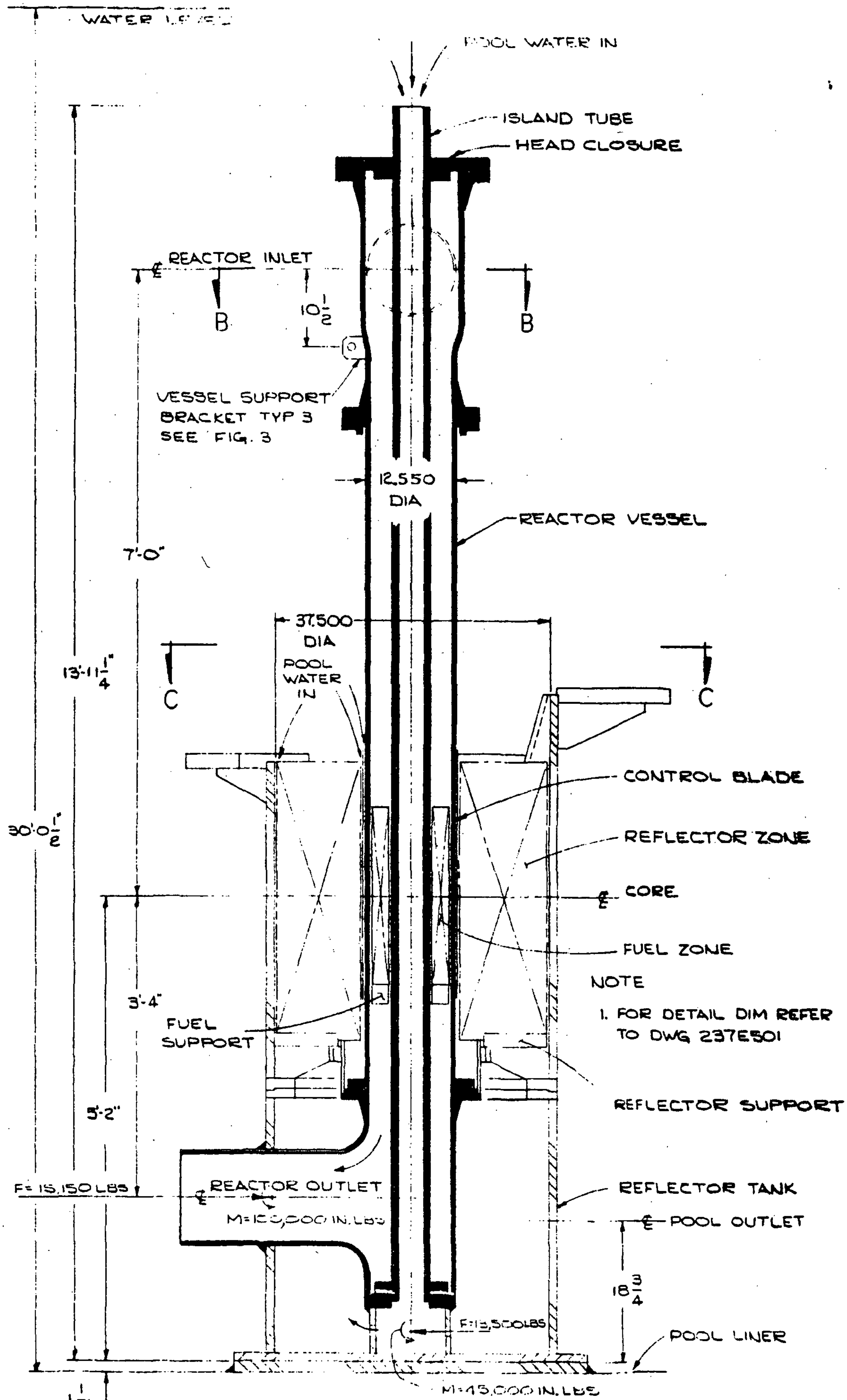
TITLE REACTOR PRESSURE VESSEL - TEST PLATE EVALUATION



t - PRESSURE VESSEL WALL THICKNESS.

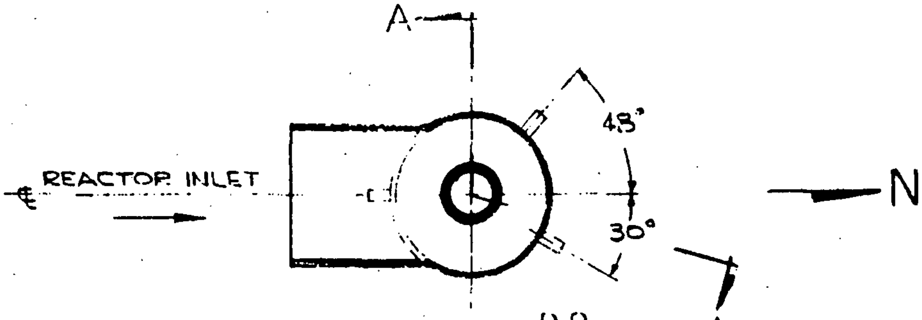
(a) - SEVERAL PIECES CAN BE JOINED TO MAKE THIS DIMENSION, HOWEVER, WELD MUST BE CONTINUOUS.

FIG 7 TEST WELD FOR WELD ℄ HEAT AFFECTED ZONE SPECIMEN

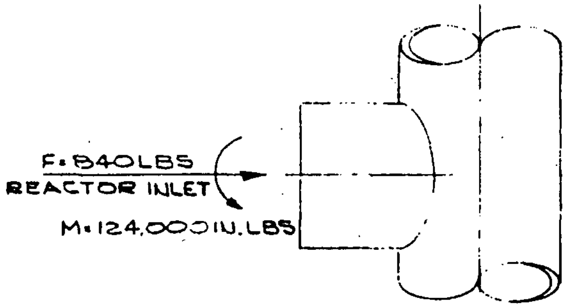


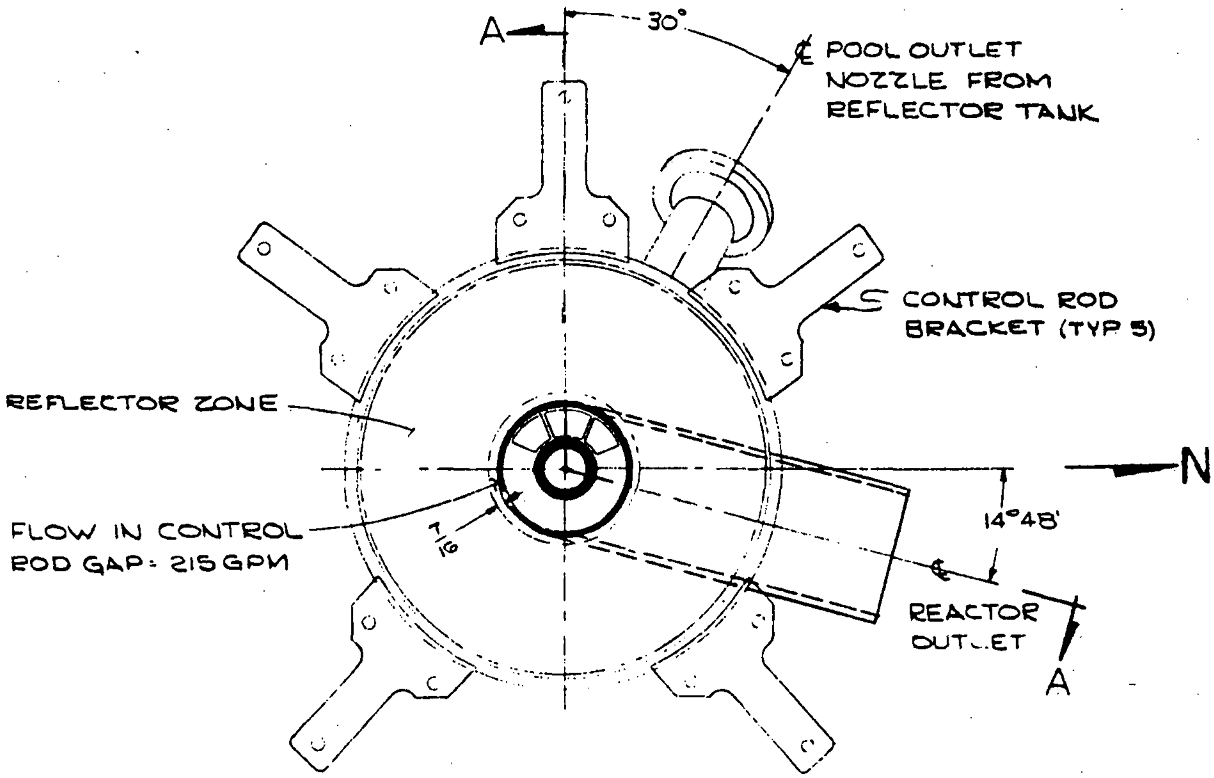
SECTION A-A

FIG. 1 — PRESSURE VESSEL AND REFLECTOR TANK DIAGRAM



SECTION B-B

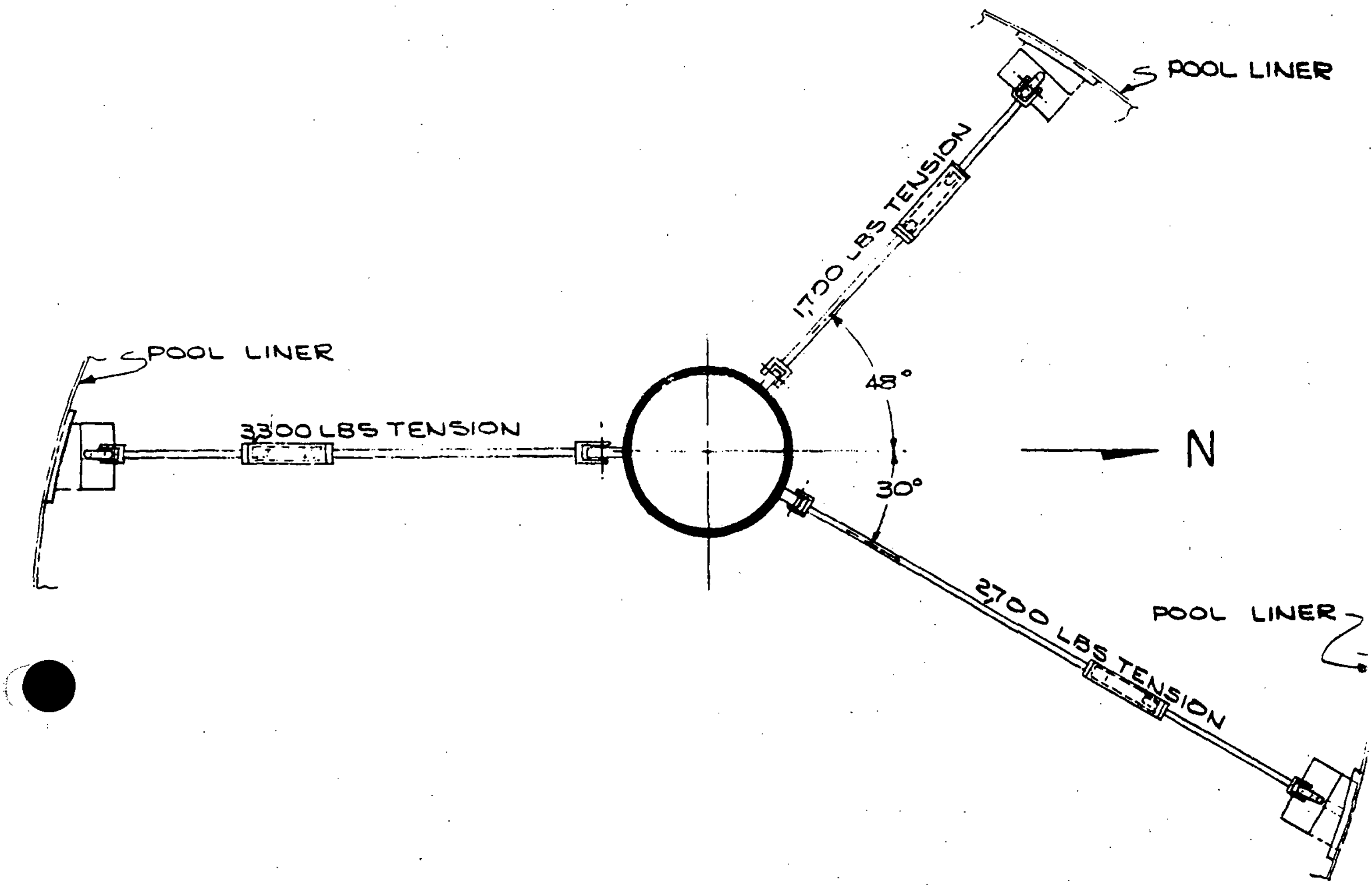




SECTION C-C

NOTE

1. FOR DETAIL DIM. REFER TO DWG. 237E501



NOTE
1. FOR DETAIL DIM. REFER TO DWG. 237E501

FIG. 3 — VESSEL SUPPORTS

APPENDIX III

Resumes of Qualified Consultants

February 1, 1966

CURRICULUM VITAE

Donald M. Shapiro, D.Sc.

Director of Computer Center
Missouri Institute of Psychiatry
5400 Arsenal Street
St. Louis, Missouri 63139

Home: [redacted]

Date of Birth: November 15, 1935
Place of Birth: Pittsburgh, Pennsylvania
Nationality: U. S.

Education	Field	Degree	Year
University of Pittsburgh, 1953-1956	Physics	B.S.	1956
Washington University, 1958-1966	Applied Mathematics and Computer Sciences	D.Sc.	1966

Academic Appointments

Research Associate, Washington University Computer Center	10/63-
Research Associate, Missouri Institute of Psychiatry	10/64-1/65
Research Associate, Department of Psychiatry, Missouri Institute of Psychiatry, University of Missouri School of Medicine	1/65-
Assistant Professor, Department of Psychiatry, Missouri Institute of Psychiatry, University of Missouri School of Medicine	2/66-

Appointments

Director, Computer Center, Department of Psychiatry, Missouri Institute of Psychiatry, University of Missouri School of Medicine	1/65-
Independent Consultant, Reactor Physics and Mathematics, Computer Methods	2/63-
Reactor Physicist, Internuclear Company	10/57-1/63
Reactor Physicist, United Aircraft Corporation (Connecticut Aircraft Nuclear Engine Laboratory)	6/56-9/57

February 1, 1966

PUBLICATIONS

Donald M. Shapiro, D.Sc.

1. Advanced Engineering Test Reaction Safety Studies. INTERNUC-49, Oct., 1959 (with P. C. Bertelson, O. J. Elgert, T. L. Francis, M. J. Kornfeld and G. E. Putnam).

2. MISPHT (Multigroup Internuclear Spherical Transport). INTERNUC-68, Oct., 1961 (with G. E. Putnam).

3. MIST (Multigroup Internuclear Slab Transport). INTERNUC-67, Aug., 196' (IDO-16856, May, 1963) (with G. E. Putnam).

4. The Application of the IBM Control System to EEG Analysis. PROCEEDINGS JOINT EASTERN-MIDWESTERN 1620 USERS GROUP MEETING, Pittsburgh, 1963: 402-417.

5. EEG Analysis by Digital Computer. I: Development of IBM 1710 System. ELECTROENCEPH. CLIN. NEUROPHYSIOL., 1965, 18:520 (with D. Bridger, T. Itil and M. Fink).

6. Comparison of Various Digital Computer Methods for EEG Analysis. PROCEEDINGS VI INTERNATIONAL CONGRESS OF EEG AND CLINICAL NEUROPHYSIOLOGY, Elsevier, 1965. Also, Psychiatric Research Foundation of Missouri, Publication 65-7, 1965 (with M. Fink).

7. Quantitative Analysis of the Electroencephalogram by Digital Computer Methods. III: Applications to Psychopharmacology. VII IBM MEDICAL SYMPOSIUM, Poughkeepsie, Oct., 1965 (with M. Fink).

8. Die Anwendung Von Digital - Computer - Methoden in Der Psychopharmakologie. ARZNEIMITTEL-FORSCH, 1966 (in press) (Presented at the German Neuropsychopharmacology Society, Nurnberg, 1965) (with T. Itil and M. Fink).

9. Discrimination of Amobarbital Effects by Quantitative Electroencephalography. Psychiatric Research Foundation of Missouri, Publication 65-5, 1965 (with T. Itil, C. Hickman and M. Fink).

10. Computer Analytic Classification of EEG Sleep Stages. Read at Society for Psychophysiological Research, Houston, 1965 (in preparation) (with M. Fink).

11. Quantitative Analysis of the Electroencephalogram by Digital Computer Methods (to be published) (with M. Fink).

February 1, 1966

EXPERIENCE RECORD

Donald M. Shapiro, D.Sc.

1965-1966

Director of Computer Center (IBM-1710) at the Missouri Institute of Psychiatry whose primary purpose is to implement and investigate the use of digital and hybrid computing systems for the automated analysis of complex biological wave forms in a real time environment. The facility also performs the statistical, data processing, and information retrieval functions necessary for other types of psychiatric research.

1963-1966

Independent consultant to Internuclear Company, Petrolite Corporation, and Westinghouse Electric Corporation, Astronuclear Laboratory in the fields of reactor physics and shielding, statistical methods, and digital computer techniques.

1963-1964

Research Associate in the Washington University Computer Center with project direction responsibilities in the areas of biomedical computations and intra-center research activities.

1957-1963

Reactor Physicist with Internuclear Company.

1962-1963

Project Physicist responsible for the reactor physics analyses for the detailed design of a 10 Mw flux trap research reactor for the University of Missouri and a 5 Mw research reactor for Kyoto University.

1961

Project Engineer for programming, coding, and development of an advanced multigroup slab transport program (MIST) for the IBM-7090 for Phillips Petroleum Company. Programmed and developed the analogous problem in spherical geometry as the MISPHT code. Project Engineer in charge of nuclear engineering services to General Electric Company, Aircraft Nuclear Propulsion Department.

1960

Performed consulting and analytical services relative to the Enrico Fermi Fast Breeder Reactor for Atomic Power Development Associates in the areas of neutron and reactor physics, shielding and computer programming.

1957-1959

Performed various analytical services in the areas of reactor physics and shielding, computer programming, hazards analysis and kinetics, on projects such as the Advanced Engineering Test Reactor, Fast Reactor analyses and critical experiment checkout for the Aircraft Nuclear Propulsion Department of General Electric Company and various Internuclear Company proposals.

1956-1957

Reactor Physicist for Pratt & Whitney Aircraft Corporation, CANEL Division.

Performed various analytical services in the areas of reactor physics, kinetics, shielding, and computer programming.

Frederick A. Flint
Mechanical Engineer

EXPERIENCE RESUME

1960 to Present

INTERNUCLEAR COMPANY
St. Louis, Missouri

Engineer in Charge (1963 to Present) - In charge of completing Internuclear Company's contractual commitments to the University of Missouri, following a decision by Petrolite Corporation, sole owner of Internuclear Company, to discontinue nuclear engineering consulting and design work. Responsibilities involved in conjunction with a $1,000,000, 5-10 megawatt, high flux, research reactor (the conceptual, preliminary and partial detail designs and specifications for which were provided by Internuclear Company) include the following:

1. Assist University of Missouri in making technical and economic evaluation of reactor bids.

2. Review and approval of reactor supplier's drawings and information for compliance with specifications.

3. Inspection of reactor components at fabricators' plants and after installation for compliance with specifications.

4. Inspection of reactor and systems during cold operational, start-up and check-out phases for compliance with specifications.

   Protection of Internuclear Company's nuclear and thermal performance and shielding warranties.

   The reactor, presently in the final stage of construction, will go critical late this year and be completely checked out early in 1966.

Resident Engineer (1962 to 1963) - Assisted with preparation of design criteria, preliminary design and specifications for a $900,000, 1-5 megawatt, research reactor for Kyoto University in Japan. Subsequently spent ten months in Japan in charge of the detail design phase. Responsibilities included:

1. Review and approval of Japanese subcontractors' drawings and information for compliance with specifications.

2. Served as consultant to Kyoto University and Japanese subcontractors on design problems.

3. Protect Internuclear Company's nuclear and thermal performance and shielding warranties.

Upon return to the United States purchased, inspected and shipped fuel elements for the Kyoto University Reactor.

Note: Reactor subsequently started up and checked out with only a few minor shielding problems, has been in operation over a year.

Test Engineer (1960 to 1962) - Worked at Oak Ridge National Laboratory under a sub-contract between Internuclear Company and Union Carbide Nuclear Company. Responsible for proof-testing certain major components for GCR-ORR Loop II (an in-pile facility designed to permit study of some of the major problems associated with radioactively contaminated, gas-cooled reactor systems). Also responsible for the design, construction and operation of the out-of-pile gas (helium) test loop wherein the proof-testing was accomplished. The components were given stringent performance tests under both design and emergency conditions at gas temperatures ranging to 1000°F, pressures ranging to 400 psig and flow rates ranging to 700 lbs/hr.

Components tested included:

Heater I, 169 kw, electrical cartridge, one pass, axial flow (unit failed)
Heater II, 282 kw, electrical cartridge, four pass, cross flow
Regenerator, core and shell, gas to gas, counterflow heat exchanger
Evaporator, tube and shell, gas to water, cross flow heat exchanger
Condenser, tube and shell, steam to water, cross flow heat exchanger

Authored report number CF62-9-35 entitled, "Performance Tests for GCR-ORR Loop II Main Heat Exchangers," setting forth analysis of proof-test results and performance data.

1957 to 1960

..ITY OF UTAH
La..e City, Utah

Student

1954 to 1957

GENERAL ELECTRIC COMPANY, AIRCRAFT NUCLEAR PROPULSION DEPARTMENT
Idaho Falls, Idaho

Designer - Assisted with the preparation of conceptual design and design criteria, coordinated detailed design with sub-contractor and reviewed sub-contractor drawings for compliance with design criteria for three in-pile test loops. The three test loops, designed for installation in test holes 33, 66 and 99, respectively, of the Engineering Test Reactor at the National Reactor Testing Station in Idaho, were similar but varied in size. All were designed for the purpose of performing research and development work on nuclear fuel elements in a gas coolant at temperatures ranging to 1500°F, pressures ranging to 300 psig and flow rates ranging to 24 lbs/min.

Designed majority of loop components for GEANP Loop II for installation in the HT-1 test hole of the Materials Testing Reactor at the National Reactor Testing Station in Idaho. GEANP Loop II was designed for the purpose of performing research and development work on nuclear fuel elements in a gas coolant at temperatures ranging to 1500°F, pressures ranging to 300 psig and flow rates ranging to 6 lbs/min. Subsequently followed fabrication of spare loop at the fabricator's plant.

1953 to 1954

BLAW KNOX COMPANY
Idaho Falls, Idaho

Draftsman - Assisted with the design and modification of the Chemical Processing Plant (for processing spent nuclear fuel elements) at the National Reactor Testing Station in Idaho, making such drawings as process flowsheets, equipment layouts, piping layouts and details, vessel details, mechanical equipment details, structural details and other drawings.

1951 to 1953

AMERICAN CYANAMID COMPANY
Idaho Falls, Idaho

Draftsman - Assisted with the design and modification of the Chemical Processing (for processing spent nuclear fuel elements) at the National Reactor Station in Idaho, making such drawings as process flowsheets, equipment [illegible]ts, piping layouts and details, vessel details, mechanical equipment details, structural details and other drawings.

19[illegible] to 1951

CITY ENGINEERING DEPARTMENT
Idaho Falls, Idaho

Assisted with the accomplishment of paving, sidewalk, sewer and airport improvement projects in the following capacities:

Draftsman - Made topographic, plan and profile and detail drawings, prepared engineering cost estimates and calculated assessment costs to owner of abutting property.

Instrumentman - Made city lot surveys, topographic surveys and line and grade surveys.

Rodman-Chainman - Assisted in making surveys.

1941 to 1946

UNITED STATES AIR FORCE

Private-1st Lieutenant - Successfully completed courses in airplane mechanics, Norden Bombsight maintenance and Minneapolis Honeywell Automatic Flight Control Equipment maintenance.

Served with the Eighth Air Force in England for 27 months.

Directly commissioned a 2nd Lieutenant while overseas.

PERSONAL DATA

Education: B.S. Mechanical Engineering, University of Utah, 1960 graduated with honors.

Courses in Advanced Calculus, Engineering Analysis and Vector Analysis, University of Tennessee, 1961.

Course in Fortran Programming for Engineers, Washington University in St. Louis, Missouri, 1964.

nal: Registered Professional Engineer in the State of Missouri; Registration Number E-11423.

Member American Nuclear Society.

Miscellaneous: [redacted]

[redacted] [redacted]

S. DAVID MAC KAY

Education

Bachelor of Science, Physics, Siena College, 1950

Master of Science course work completed at Union College, 1957

Employment History

1965 to Date	General Electric Company, Atomic Power Equipment Department; Plant Test Engineer
1962 - 1965	Allis-Chalmers Manufacturing Company, Atomic Energy Division; Project Manager, Elk River Reactor Project
1960 - 1962	Allis-Chalmers Manufacturing Company, Atomic Energy Division; Operations Supervisor, Elk River Reactor Project
1959 - 1960	Allis-Chalmers Manufacturing Company, Atomic Energy Division; Nuclear Engineer
1958 - 1959	Alco Products, Incorporated, Schenectady, New York; Project Engineer, Reactor Core Evaluation
1956 - 1958	Alco Products, Incorporated, Schenectady, New York; Nuclear Engineer, Criticality Facility
1955 - 1956	Knolls Atomic Power Laboratory, Schenectady, New York; Reactor Supervisor, Thermal Test Reactor
1952 - 1956	Knolls Atomic Power Laboratory, Schenectady, New York; Laboratory Assistant and Critical Assemblies Operator

Experience

Mr. MacKay's prime responsibility during his present assignment is the safe startup of a nuclear reactor. His duties include: preparation of startup and power escalation program and procedures, preoperational testing, and supervision of fuel loading, critical testing and power escalation.

Mr. MacKay was the Project Manager of the Elk River Reactor Plant, responsible for all project functions. These responsibilities included the coordination of design efforts; approval of design changes, procedural changes, and procurement of components; fulfilling the requirements of the reactor operating authorization and obtaining necessary changes; schedule developments; and administration of the contract to satisfy the requirements of various branches of the AEC and the electric utility company.

Prior to that assignment, Mr. MacKay was the Operations Supervisor of the Elk River Reactor, where his prime responsibility was the safe startup of the reactor. His duties included: preparing the startup and power escalation program and procedures; preoperational testing; and supervising fuel loading, initial criticality, and power escalation.

Previously, Mr. MacKay was employed by Alco Products, Incorporated, where he performed zero power tests for the Army Package Power Reactor at the Alco Critical Facility and served as instructor to Army personnel in reactor operation and basic reactor theory. Subsequently, Mr. MacKay transferred to the reactor analysis unit at Alco and was appointed Project Engineer for Core Measurement. In this capacity, he was responsible for the design, execution, and interpretation of nuclear experiments on the Army Package Power Reactor.

Before his employment with Alco, Mr. MacKay worked at the Knolls Atomic Power Laboratory. There he participated in the startup and operation of various critical assemblies including the Preliminary Pile Assembly, which contained fully enriched uranium, beryllium, metallic sodium, stainless steel, etc., and was used for preliminary design of the Submarine Intermediate (spectrum) Reactor core; the Proof Test Reactor, a full-scale mockup of the S3G core (Seawolf); the Advanced Test Reactor, which contained fully enriched uranium, zirconium, light water, etc., and various exponential pile assemblies. He was later given the responsibility for the safe operation and maintenance of their 10-kw Thermal Test Reactor. With this facility, he provided a service to the laboratory and industry in the form of neutron flux activations and reactivity coefficient and cross section measurements.

Professional Societies

Mr. MacKay is a member of the American Nuclear Society and is a former treasurer of the Northeastern New York Section.

Publications

Mr. MacKay's work has been documented in several publications available through the Office of Technical Information Extension at Oak Ridge National Laboratory. He has also presented a paper at the 1958 meeting of the American Nuclear Society dealing with core physics measurements on SM-1 through 9.1 Mw-years of operation.

BIOGRAPHICAL RESUME'

ROBERT G. COCHRAN

Born, July 12, 1919, Indianapolis, Indiana;
Professor and Head, Department of Nuclear Engineering;
Head, Nuclear Science Center,
Texas A&M University

EDUCATION:

B. A. Degree, Physics, Indiana University, 1948
M. S. Degree, Nuclear Physics, Indiana University, 1950
Ph.D. Degree, Nuclear Physics, Pennsylvania State University, 1957

EXPERIENCE:

Teaching

Associate Professor of Nuclear Engineering, Pennsylvania State University, 1954-1959
Professor and Head, Department of Nuclear Engineering, Texas A&M University, 1959 - present

Research

Research Assistant, Basic Nuclear Physics, Indiana University, 1948-1950
Development Engineer, Sarkes Tarzain Consulting Engineers, Bloomington, Indiana (part-time), 1947-1950
Nuclear Physicist and Group Leader, Nuclear and Neutron Physics, Reactor Physics and Reactor Shielding Research, Reactor Design, Oak Ridge National Laboratory, 1950-1954
Director of Research Reactor, Basic Nuclear Physics, Neutron and Reactor Physics, Pennsylvania State University, 1954-1959
Head, Nuclear Science Center, Reactor and Neutron Physics, Texas A&M University, 1959 - present

Consulting

University of Michigan, Ann Arbor, Michigan, Reactor Engineering, 1955
B&W, Lynchburg, Va., Reactor Engineering, 1956-57
Curtiss Wright Corp., Quahanna, Pa., & New Jersey, Reactor Engineering and Physics, 1956-58
Cook Electric Co., Chicago, Ill, Nuclear Physics, Radiation Damage, Reactor and Neutron Physics, 1956-58
AMF Inc., Greenwich, Conn., Reactor Physics and Engineering, 1956-61
USAEC, Washington, D. C., Reactor Physics, Nuclear Engineering and Reactor Operation, 1957 - present

Bendix Corp., Detroit, Mich., Reactor Engineering, 1960-61
U. S. Army, Watertown, Mass., Reactor and Nuclear Engineering, 1960 - present
U. S. Air Force, Washington, D.C., and Kirtland, New Mexico, Nuclear Engineering, 1960 - present
Sandia Corp., Albuquerque, New Mexico, Nuclear Engineering, 1961 - present

IN THE PROFESSION

Member, American Nuclear Society, 1955 - present
Member, American Physical Society, 1948 - present
Member, Phi Kappa Phi, 1961
Member, Sigma Xi, 1961 - present
Chairman, Sub-Committee on Research Reactors, National Academy of Sciences, National Research Council
American Society Engineering Education, 1962 - present
Member, N-6 Committee (Reactor Standards) American Society of Mechanical Engineers

Listed in

American Men of Science
Who's Who in Atoms
Who's Who in Engineering
Leaders in American Science

PUBLICATIONS

Cochran, R.G., D. E. Feltz, J. D. Randall, and J. V. Walker, "An Absolute Thermal-Neutron-Flux Standard," submitted to Nuclear Science and Engineering, September 1963.

Cochran, R.G., "Low-Cost Research Programs for Research Reactors", presented at the Symposium on the Programming and Utilization of Research Reactors held in Vienna, Austria, October, 1961.

Pratt, W.W. and R. G. Cochran, "Beta-and Gamma-Ray Spectra of Pd^{111}," Physical Review, 118, 1313-1315, June 1, 1960.

Pratt, W. W. and R. G. Cochran, "Search for Low-Energy Gamma Rays in O^{16}," Physical Review, 116, 1578-80, December 15, 1959.

Cochran, R. G., "Safety and Site Selection," Journal of Engineering Education, 49, 422-425, February 1959.

Cochran, R. G. and W. W. Pratt, "The 25 Minute Isomer of Se^{83}," Physical Review, 113, 852-856, February 1959.

Cochran, R.G., A. Jacobs, F. J. Remick, J. Maxey and G. Robinson, "Temperature Coefficient at Low Temperatures in a Heterogeneous Light-Water Moderated Reactor," Trans ANS, June 1958.

Cochran, R. G. and F. J. Remick, "Research and Teaching with a University-Owned Research Reactor," Paper presented at Second United Nations International Conference on the Peaceful Uses of Atomic Energy. Geneva, 1958.

Cochran, R. G. and W. W. Pratt, "Radioactive Decay of Se^{83}," Physical Review, 109, 878-83 (1958).

Breaseale, W. M., R. G. Cochran and K. O. Donelian, "The Swimming Pool Reactor and its Modifications," Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955, Vol. 2, pp. 420-427.

Cochran, R. G. and K. M. Henry, "Proton Recoil Fast-Neutron Spectrometer Part II Experimental," The Review of Scientific Instruments, 26, 757-62, August 1955.

Various classified and unclassified publications from 1948 to present.

Chandler, Ekillis M., M. A. Quddus and R. G. Cochran, "Updating Educational Experiments for a Low-Powered Reactor," Submitted to Twelfth Annual ANS Meeting in Denver, Colorado, June 20-23, 1966.

SEE ATTACHED SHEETS FOR EARLIER PUBLICATIONS AND CONSULTING WORK.

Consulting involving reactor startup and testing:

1. Served on startup crew for BSR at Oak Ridge National Laboratory.

2. Supervised startup and test program of the Pennsylvania State University research reactor.

3. Consulted for the University of Michigan during final design and startup of their research reactor.

4. Consulted for Hietichi (Japan) on the design of a research and test reactor.

5. Supervised the critical tests and test program for the IRL research reactor. (Also served as a reactor consultant for AMF Inc. during the construction of several foreign reactors).

6. Supervised the startup and test program of the University of Kansas research reactor. (Bendix Corporation).

7. Consulted for Watertown Arsenal during startup and testing of the ARMA research reactor.

8. Designed, supervised startup and test program of the Texas A&M University research reactor.

9. Consulted for Curtis Wright Corporation during startup and test of their research reactor. (Also contributed to the design and analysis of this reactor.)

10. Have also contributed to the design and analysis of several other reactors.

PUBLICATIONS
Dr. Robert G. Cochran

"Angular Distribution of the Al^{27} (d, OC) Mg^{25} Reaction and Energy Levels in Mg^{25}." A. D. Shelberg, M. B. Sampson and R. G. Cochran: Physical Review, 80:574-9 November 15, 1950.
"Fast Neutron Dosimeter Measurements for Experiment 1 in the Bulk Shielding Facility." R. G. Cochran and H. E. Hungerford; Oak Ridge National Lab., 1951, 7 p.
"Fast Neutron Dosimeter Measurements for Experiment 2." R. G. Cochran and H. E. Hungerford; Oak Ridge National Lab., 1951, 5 p.
"A Proton Recoil Type Fast-Neutron Spectrometer." R. G. Cochran and K. M. Henry; Oak Ridge National Lab. (ORNL-1479), 1953, 36 p.
"Fast Neutron Spectra of the BSF Reactor." R. G. Cochran and K. M. Henry; Oak Ridge National Lab., 1953, 11 p.
"Fast-Neutron Spectrum of the LITR." R. G. Cochran and K. M. Henry; Oak Ridge National Lab., 1953, 8 p.
"A Proto . Recoil Type Fast-Neutron Spectrometer." R. G. Cochran and K. M. Henry; Oak Ridge National Lab. (ORNL-1479), 1953, 36 p.
"Oak Ridge Research Reactor Shadow Shield Experiment." R. G. Cochran, K. M. Henry and J. D. Flynn; Oak Ridge National Lab., 1954, 11 p.
"Reactor Radiations Through Slabs of Graphite." R. G. Cochran, J. D. Flynn, K. M. Henry and G. Estabrook; Oak Ridge National Lab., 1954, 22 p.
"Reactivity Measurements With the Bulk Shielding Reactor." R. G. Cochran, J. L. Meem, T. E. Cole and E. B. Johnson; Oak Ridge National Lab. (ORNL-1682), 1954, 79 p.
"Reactivity Measurements with the Bulk Shielding Reactor; Control Rod Calibration; Beam Hole Coefficients." E. B. Johnson, F. C. Maienschein, K. M. Henry R. G. Cochran and J. D. Flynn; Oak Ridge National Lab., 1955, 34 p.
"Proton Recoil Fast-Neutron Spectrometer Part II. Experimental." R. G. Cochran and K. M. Henry; The Review of Scientific Instruments, 26:757-62, August, 1955.
Swimming Pool Reactor and its Modifications." W. M. Breaseale, R. G. Cochran and K. O. Donelian; Proceedings of the International Converence on the Peaceful Uses of Atomic Energy, Geneva, 1955, Vol. 2, p.420-427.
...ioactive Decay of Se^{83}." R. G. Cochran and W. W. Pratt; Physical Review, 109:878-83 (1958).
...rch and Teaching with a University-Owned Research Reactor." R. G. Cochran and F. J. Remick; Paper presented at Second United Nations International Conference on the Peaceful Uses of Atomic Energy; Geneva, 1958.
"The 23 Minute Isomer of Se^{83}." R. G. Cochran and W. W. Pratt; Physical Review, 113:852-856, February, 1959.
"Safety and Site Selection," Journal of Engineering Education, 49:422-425, February, 1959, R. G. Cochran.
"Search for Low-Energy Gamma Rays in O^{16}." W. W. Pratt and R. G. Cochran; Physical Review, 116:1578-80, December 15, 1959.
"Beta- and Gamma-Ray Spectra of Pd^{111}." W. W. Pratt and R. G. Cochran; Physical Review, 118:1313-1315, June 1, 1960.

TITLE REACTOR PRESSURE VESSEL

9.4.3.3 No chemical cleaning shall be used once the configuration is such that it cannot be thoroughly rinsed. Any cracks or crevices which trap chemical reagents and cannot be rinsed shall be avoided.

9.4.4 Any cracks, blow holes or other defects which appear on the surface of weld beads shall be removed by machining, chipping or grinding.

9.4.5 Temporary welds on the base material of the reactor vessel shall be located, where possible, on edges and areas that will be trimmed off.

9.4.6 Wide welds to overcome poor fit are not permissible. Poor fits shall be remedied by suitable means such as regrooving and approved by the Buyer's representative. Except for small cavities, the Seller shall not correct a plate edge deficiency unless approved by the Buyer's representative and the Buyer's representative may require that welding to correct a plate edge deficiency be subject to radiographic or other approved methods of examination.

9.4.7 Weld joints shall be designed and prepared, wherever possible, including nozzle attachments and special joints, to permit radiographic inspection to ASME Code standards in accordance with Code Case 1273N and shall be in accordance with methods acceptable to the Buyer.

9.4.8 The reactor vessel shall be free of retention pockets and crevices. Welds shall be uniform and blend smoothly into the adjacent metal.

9.4.9 All unacceptable defects visually observed or revealed by radiographic, ultrasonic. or liquid penetrant shall be repaired by procedures approved by the Buyer, and in the case of major repairs of base material, in the presence of the Buyer's representative.

UNIVERSITY OF MISSOURI
COLUMBIA

COPY

RESEARCH REACTOR FACILITY
OFFICE OF THE DIRECTOR

June 3, 1966

Dr. R. L. Doan
Director
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington 25, D. C.

Reference: Docket No. 50-186

Dear Sir:

On behalf of The Curators of the University of Missouri we wish to submit the enclosed Supplement to Addendum No. Two to the Final Hazard Report on the Research Reactor Facility at the University of Missouri. This supplement is submitted in accordance with your verbal request by telephone on May 23, 1966.

The enclosures include twenty-two copies of the supplement, three of which carry a copy of this letter and affirmation.

Very truly yours,

A. H. Emmons

A. H. Emmons
Director

AHE/kjb
enc.

THE CURATORS OF THE UNIVERSITY OF MISSOURI

ATTEST:

Mary Robnett

Mary Robnett, Secretary

By R. H. Bezoni

R. H. Bezoni, Comptroller

STATE OF MISSOURI)
) ss.
COUNTY OF BOONE)

On this 6th day of June, 1966 personally appeared before me R. H. Bezoni, who, being duly sworn, on his oath stated that he is the Comptroller of The Curators of the University of Missouri, a public corporation, and as such Comptroller was duly authorized to execute the foregoing application on behalf of The Curators of the University of Missouri.

Done at my office in Columbia, Boone County, Missouri

Notary Public

My term expires My Commission Expires April 28, 1968.

SUPPLEMENT TO
ADDENDUM NO. TWO
HAZARDS SUMMARY REPORT

This supplement discusses in further detail some subjects which have been treated in earlier submittals.

I. With reference to the discussion of the effect of losing single fuel plates from an element, which appears on page 1 of Addendum No. Two, some new information has been obtained which is pertinent.

The Advanced Test Reactor (ATR) at the National Reactor Testing station has a core with many similarities to the University of Missouri Research Reactor. The fuel is enriched uranium and the moderator is ordinary water. In cross section, the major dimensions of the ATR fuel assembly are the same as those of the UMRR assembly. The major differences are that the ATR assemblies are twice as long as the UMRR assemblies and they contain twice as much fuel. The ATR assembly contains 19 curved fuel plates vs. 24 in the UMRR units. The metal to water ratio in the two reactors is almost identical. In view of these basic similarities one might expect that the results of removing a fuel plate from either of the two assemblies would be comparable.

At ATR both calculational and experimental studies were made. Calculations indicated that the removal of one central fuel plate from each of 40 fuel assemblies would yield a total Δk of +0.00449. The calculated Δk resulting from the removal of 2 central plates from each of 40 assemblies was +0.00698.

For the UMRR the calculated Δk resulting from the removal of one fuel plate from 2 out of a total of 8 fuel assemblies was +0.00104. If this result is linearly extrapolated to the removal of one fuel plate from each of the 8 assemblies the result is Δk = +0.00416. This result compares very closely to the ATR calculation of +0.00449.

It is also interesting to note that the Δk measured in the ATR critical facility upon removal of the central fuel plate from each of 40 assemblies was +0.00209. This value is less than half the calculated value.

Another very important point that deserves mention concerns which of the fuel plates in an assembly would most likely melt during a transient. In both the ATR and the UMRR the fuel plates which experience the highest power density are those on the inner and outer radius of the core. The Δk resulting from the removal of fuel plates from the inner or outer radius, according to ATR critical facility measurements, is negative.

II. On page 19 of Addendum No. Two to the Hazards Summary Report a containment building backup isolation system is described. Upon completion of the installation of this backup system, the containment building will be pressurized with the primary isolation doors open and only the backup doors closed. With the containment building pressurized the backup door assemblies and the added plenum chambers will be soap bubble tested to assure that the system is adequately leak tight.

III. A feature of the reactor primary coolant loop not previously described is a flow restricting orifice within the in-pool piping.

A 1/8 inch stainless steel orifice plate, drilled to provide a pressure drop of 16 psi at a flow of 3600 gallons per minute, is installed in a flange in the reactor primary piping. This flange is at coordinates D-16 on Figure 4.5 of the Hazards Summary Report.

The purpose of this flange is to augment the deceleration of flow through the reactor in the event that a pipe break should occur such that the reactor loop isolation valves are not effective and the anti-siphon system must perform its intended function.

In the extremely unlikely event of a pipe break in the reactor inlet line at the reactor end of the pipe tunnel, this orifice will serve to halt flow before the liquid-air interface reaches the core.

IV. The following is a further discussion of the calculated doses to the thyroid resulting from the maximum credible accident. This report examines in more detail the fission product release to the containment and the pressure decay in the building.

Previous discussions of the building leakage were based on the leakage resulting from the test pressure of two psi overpressure. The pressure rise actually occurring from the maximum credible accident is much less.

Three factors contribute to the overpressure in the building. They are:

1. The fission energy release from operating power to inherent shutdown of the reactor.
2. The energy resulting from a possible metal-water reaction.
3. The energy release from a possible hydrogen gas recombination.

The first two sources contribute a calculated 270 MW-sec of energy and this energy is considered as being released to the building in the form of saturated steam. The third factor contributes an additional 120 MW-sec of heat energy added directly to the containment atmosphere. From the assumed initial conditions of $70^{\circ}F$, 50 percent relative humidity at standard atmospheric conditions, a total of 370,000 BTU and 264 pounds of water are added to the building containment. From psychrometric data this results in a new temperature of $88^{\circ}F$ and 78 percent relative humidity. The net increase in pressure in the building is then 0.81 psi.

The iodine inventory in the reactor core is obtained from data by Burnett, Table II. (40 days irradiation at 10 MW)

$I^{131} = 25.5 \times 10^4$ curies
$I^{132} = 40.0 \times 10^4$ curies
$I^{133} = 59.0 \times 10^4$ curies
$I^{134} = 69.0 \times 10^4$ curies
$I^{135} = 53.6 \times 10^4$ curies

The iodine in the core will be distributed according to the flux distribution such that a twenty percent meltdown will result in a greater than twenty percent release of iodine. The axial thermal flux profile for the reactor is shown in Figure 6-5 of Volume II of the Reactor Design Data prepared by Internuclear Corporation. From this data a 20 percent meltdown will result in an approximate release of 31 percent of the fission products. Considering 60 percent of the iodine to be washed and plated out the total inventory of iodines available for leakage is,

$I^{131} = 3.16 \times 10^4$ curies
$I^{132} = 4.96 \times 10^4$ curies
$I^{133} = 7.31 \times 10^4$ curies
$I^{134} = 8.55 \times 10^4$ curies
$I^{135} = 6.64 \times 10^4$ curies.

With the iodine evenly dispersed within the building each cubic foot of air in the building contains the following iodine concentration,

$I^{131} = 0.131$ curies/ft.^{3} air
$I^{132} = 0.206$ curies/ft.^{3} air
$I^{133} = 0.305$ curies/ft.^{3} air
$I^{134} = 0.356$ curies/ft.^{3} air
$I^{135} = 0.277$ curies/ft.^{3} air

The pressure in the building will decay due to the heat transferred to the containment walls and due to leakage. The inside walls will be at very nearly the same temperature as the inside air temperature and the concrete will have a temperature gradient to correspond to the outside temperature. The outside columns and aluminum siding will absorb most of the direct radiation from the sun so that the outside walls will be close to the outside air temperature. The inside heat transfer coefficient was calculated to be 0.5 BTU/HR-FT^2-^{o}F and the area of the inside of the building is approximately 22,000 ft^2. The heat transfer and corresponding building air temperature was calculated using iteration methods assuming that the massive concrete structure will absorb the heat without significant increase in temperature. (All the heat released in the accident, if absorbed by the concrete, would raise the temperature less than one degree Fahrenheit). The pressure decay in the building versus time is shown in Figure 4.1.

Using data obtained from the building leak check the volume of air leaking from the building is calculated and used to calculate the radiation dose to the thyroid.

It is to be noted that in approximately ten hours the over-pressure in the building has decayed to essentially zero. Following this point the building leakage becomes a function of the daily variation in temperature and the building essentially pulsates from night to day.

Doses to the thyroid at 2 hours and 10 hours are shown in Figures 4.2 and 4.3.

PRESSURE LOSS IN BLDG. DUE TO HEAT TRANSFER

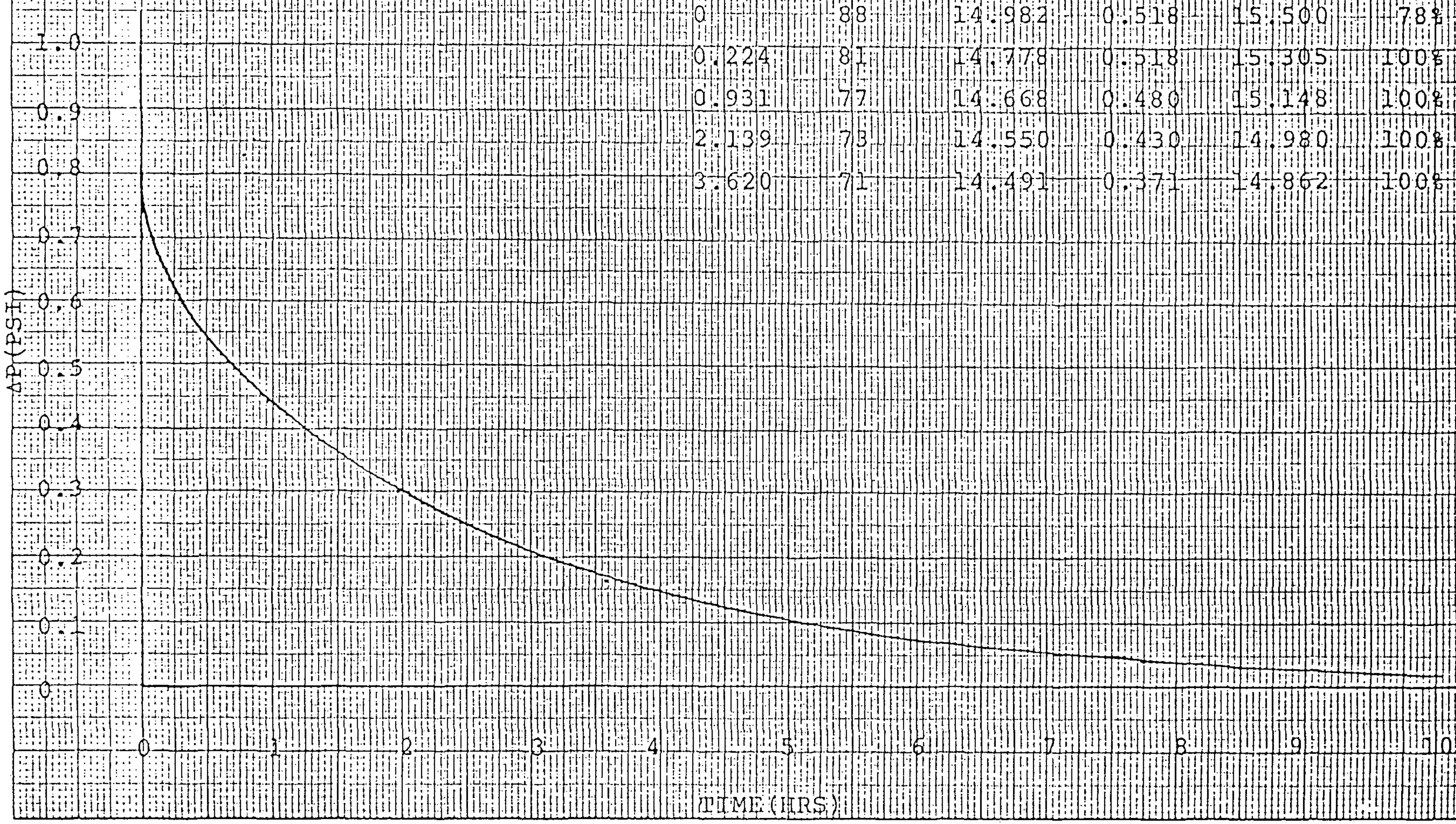
Fig. 4.1

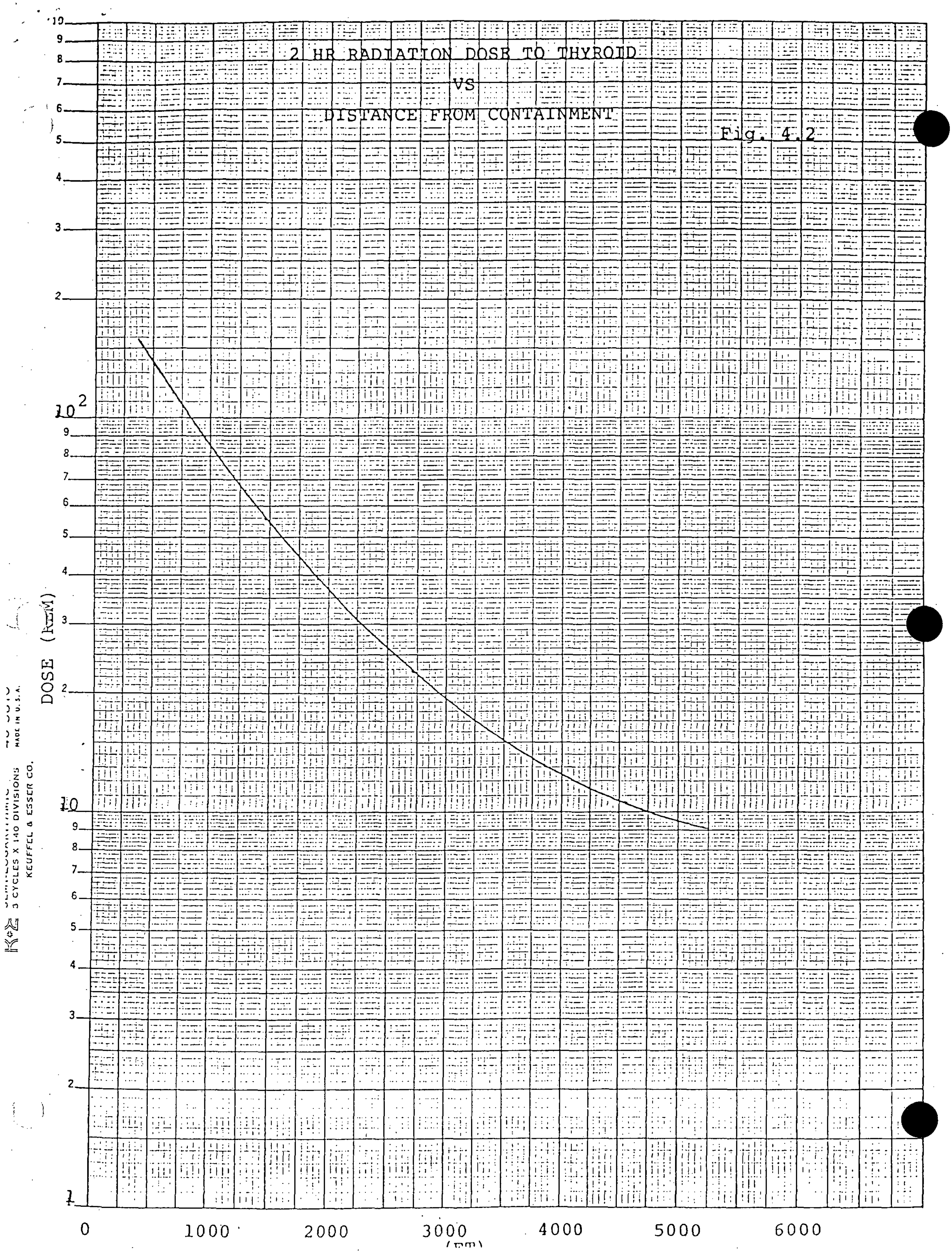
$g = hA\Delta T \quad h = 0.5\ \text{BTU/HR-FT.}^2\text{-}^\circ F$

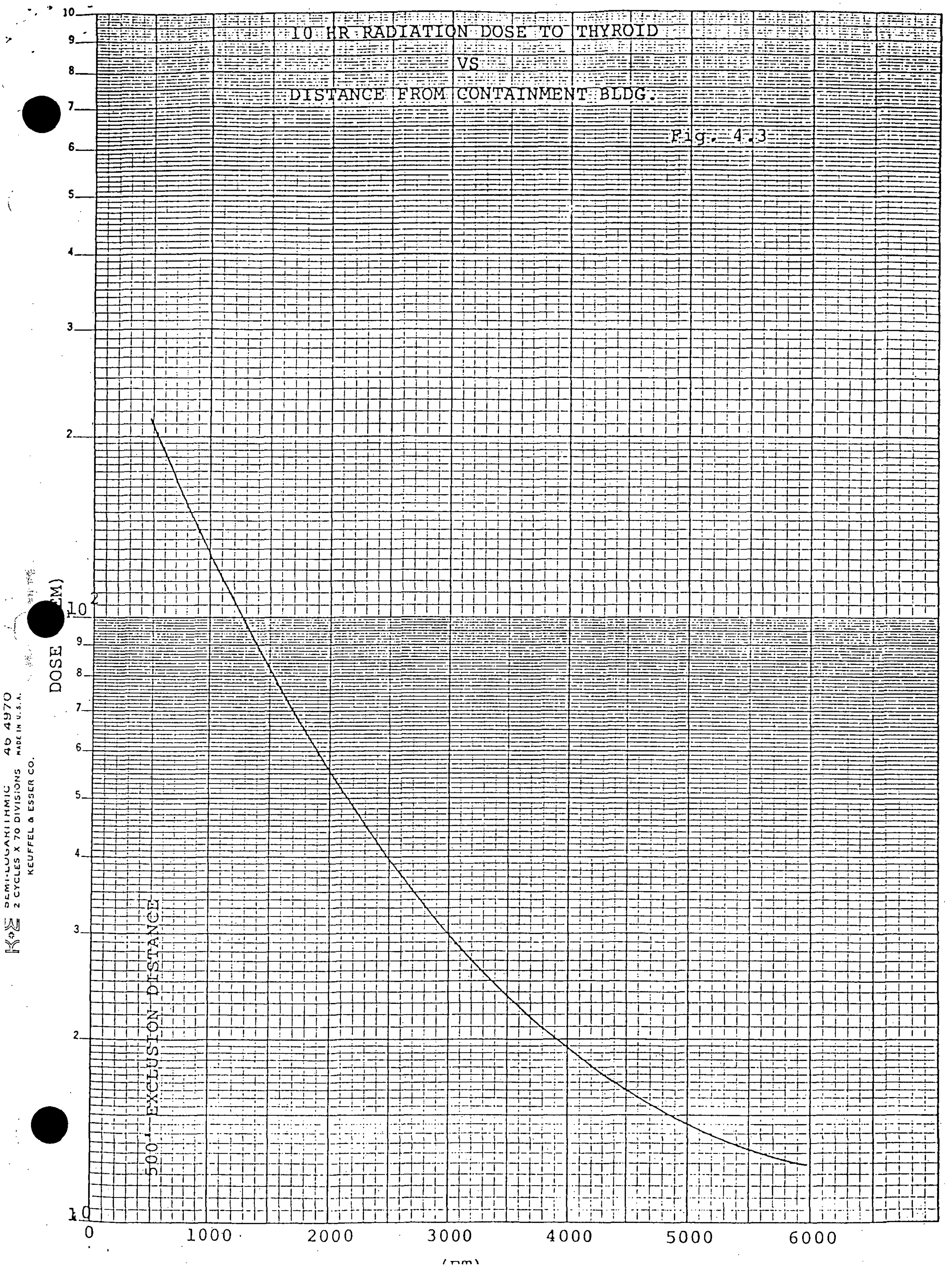
$A = 21{,}600\ \text{FT.}^2$

$\Delta T = {}^\circ F$

Time	Temp.	Pa	Pv	Pm	R.H.	ΔP
0	88	14.982	0.518	15.500	78%	0.800
0.224	81	14.778	0.518	15.305	100%	0.605
0.931	77	14.668	0.480	15.148	100%	0.448
2.139	73	14.550	0.430	14.980	100%	0.280
3.620	71	14.491	0.371	14.862	100%	0.162




2 HR RADIATION DOSE TO THYROID
VS
DISTANCE FROM CONTAINMENT
Fig. 4.2
DOSE (REM)
10^2
10
1
0
1000
2000
3000
4000
5000
6000
(FT)
K&E 3 CYCLES X 140 DIVISIONS
KEUFFEL & ESSER CO.
MADE IN U.S.A.


10 HR RADIATION DOSE TO THYROID
VS
DISTANCE FROM CONTAINMENT BLDG.
Fig. 4.3
DOSE
500 - EXCLUSION DISTANCE
0
1000
2000
3000
4000
5000
6000
K&E SEMI-LOGARITHMIC 46 4970
2 CYCLES X 70 DIVISIONS MADE IN U.S.A.
KEUFFEL & ESSER CO.

ADDENDUM NO. 3

University of Missouri
Research Reactor Facility

Compiled and Edited by
The Staff
Research Reactor Facility

Submitted by
The University of Missouri
Columbia, Missouri

August 1972

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1.0 INTRODUCTION

In July of 1965, the University of Missouri Research Reactor Facility submitted a Hazards Summary Report in support of its application for a reactor license to allow operation of the MURR reactor at a maximum power level of 10 MW. Addendum One to the Hazards Summary Report was submitted in February 1966 to make certain revisions to the original application and answer questions posed to the MURR by the Atomic Energy Commission. Additional questions were answered in May 1966 by Addendum Two to the Hazards Summary Report. Because of cooling equipment limitations at that time, the reactor was licensed for full power operation at 5 MW. Therefore, some of the questions answered in Addenda One and Two used 5 MW operation as a basis.

The necessary cooling equipment for higher power operation is now available to the University. The purpose of this document, Addendum Three to the MURR Hazards Summary Report, is to support the University of Missouri's application for an amendment to reactor license R-103 to allow operation of the reactor at the full design power of 10 MW. This report summarizes past operating experience of the MURR and describes the proposed modifications to allow 10 MW operation. Several questions addressed in Addenda One and Two required further safety analysis for 10 MW conditions and the appropriate study results are included in this report. Complete procedures for preoperational testing of the reactor systems and approach to 10 MW power level are included. Finally, an evaluation of the environmental impact of the proposed MURR power escalation is presented.

Under separate cover and also a part of this license amendment request are:

Technical Specifications for 10 MW Operation
Standard Operating Procedures for 10 MW Operation
MURR Annual Report

1.1 Brief History of Operation

The Missouri University Research Reactor first achieved criticality on October 13, 1966. An extensive low power testing and calibration program was carried out to verify that the critical physics and plant parameters were in compliance with reactor license R-103. The reactor was first operated at the presently licensed maximum power of 5 MW thermal on June 30, 1967.

Experimental reactor utilization has steadily increased and in turn MURR has evolved from an "operate on demand" schedule to the present schedule of 90 to 100 hours continuous operation per week at full power. Figure 1.1 shows the accumulation of total megawatt-days on the reactor. The reactor is presently operating at greater than 99% of the scheduled operating hours.

Fifty-seven fuel elements (seven cores) have been cycled through the reactor without a release of fission products. A meticulous fuel inspection program has detected anomalies in four fuel elements resulting in the elements being retired from service before their design burnup limit.

In the summer of 1971, the standard MURR fuel was upgraded. This new fuel was manufactured by powder metallurgy techniques and used uranium aluminide as the fuel meat. The ^{235}U loading was increased from 5.2 to 6.2 Kg per core. Low power physics testing proved the new fuel characteristics to be very close to the predicted values, and the new fuel was first put into full power operation on August 14, 1971.

Throughout the last five years of operation, the MURR has amassed an enviable record of reliable operation. The reactor has proven its inherent stability and safety and its close adherence to design predictions. The proposed power upgrade to 10 MW operation is expected to double the service capabilities of this valuable research tool.

A more complete report on the operation is included in the 71-72 Annual Report and is summarized as follows.

Total operating hours	21,030
Total hours at full power (5 MW)	19,010
Total experimental hours	138,105
Total megawatt days operated	4,026

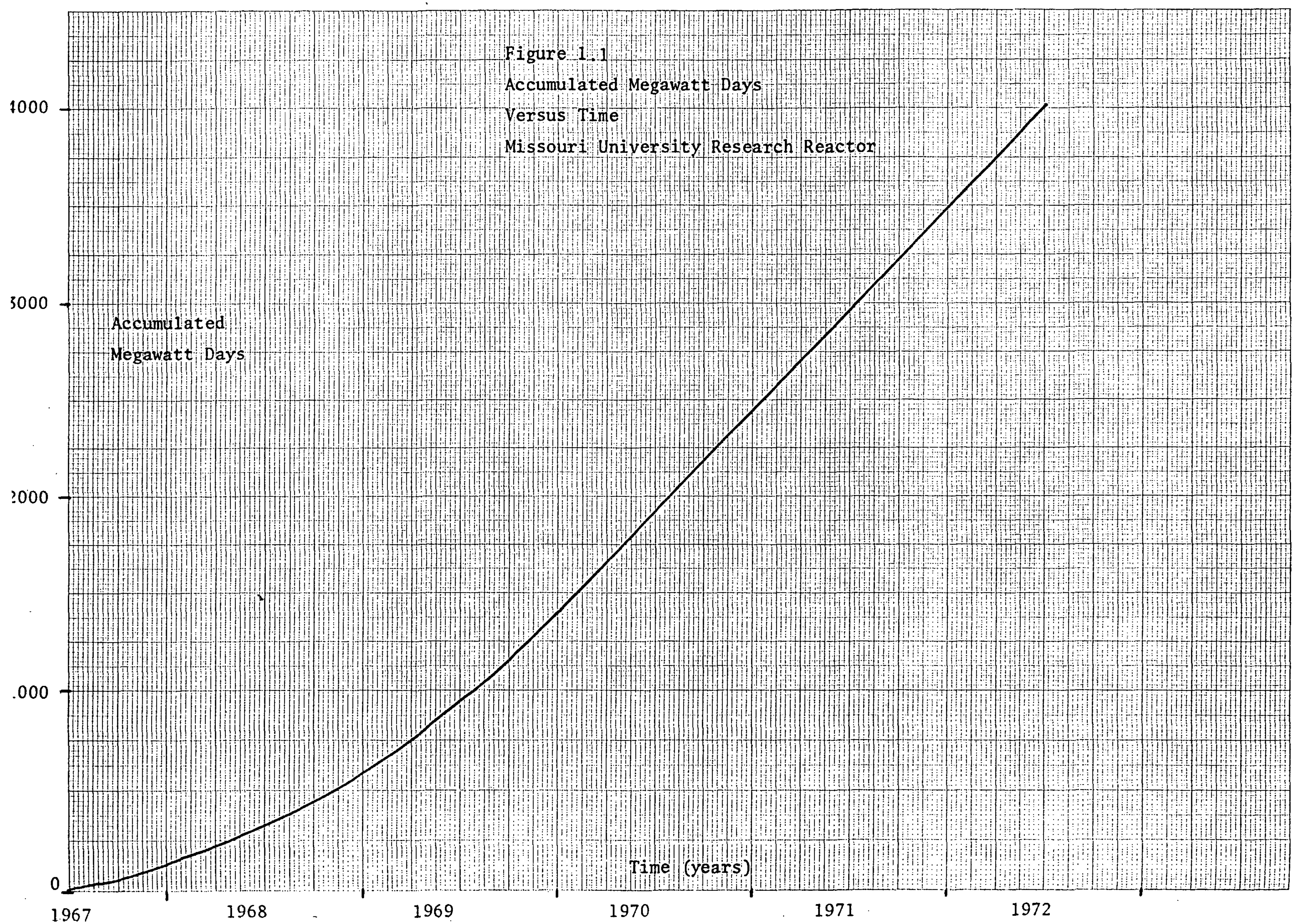


Figure 1.1
Accumulated Megawatt Days
Versus Time
Missouri University Research Reactor

2.0 DESCRIPTION OF 10 MW SYSTEM

2.1 Reactor Loop Cooling System

2.1.1 Introduction

The purpose of this section is to describe those component additions and modifications necessary to upgrade the reactor loop cooling (primary) system for 10 MW operation. This section will only cover new additions or changes to the system, but may describe in some cases other system components for clarification. Should questions arise about description of other components or their operation, further information can be found in the original Hazards Summary Report and its two addenda.

Where reference is made to installation of a component in this section, the materials and installation of the component will be in accordance with contractor specifications for upgrading the mechanical and electrical equipment. (1,2)

All components described in this section are shown on the Piping and Instrument Diagram (Figure 2.1).

2.1.2 Additional Primary Pump and Heat Exchanger

The major change in the primary system for 10 MW operation is the installation of an additional pump and heat exchanger in parallel with the existing pump and heat exchanger. They will be installed in a piping leg which will connect to flanges built into the existing primary loop. Each leg will contain the valves, fittings and instrumentation necessary to permit parallel operation with the existing components for 10 MW operation, and permit 5 MW operation with any one pump and any one heat exchanger. All new pipe and fittings will be schedule 40 aluminum 6061-T6. All connections which are not shown as flanged on Figure 2.1 will be welded by the inert gas shielded arc process. Consumable insert rings will be used on all piping butt welds. All welds will be examined radiographically. The new components will conform to the following specifications.

2.1.2.1 The new circulating pump (designated 501B) is a Worthington 6HN-173 centrifugal pump with 316 stainless steel components in contact with primary system water. The unit is driven by a GE 125 hp drip-proof motor. This unit exactly duplicates the existing pump assembly with the following exceptions.

a. The motor and pump are coupled by a Dodge Para-Flex coupling, type PS-110X in lieu of the gear type coupling on the existing unit.

b. The motor frame size is 405T on the new unit in lieu of the 445US on the existing unit.

2.1.2.2 The new heat exchanger (designated 503B) is a water-to-water shell and tube type which is a duplicate of the existing heat exchanger with the following exceptions:

a. The baffle plates and tie rods are made of type 304 stainless steel in the new unit.

b. The end plate flange is a lap joint flange instead of the slip-on type used in the existing unit. In December 2006, shell and tube-type heat exchanger 503B was replaced with a Graham Model GPE-60 plate-type heat exchanger with 201 plates and 1200 square feet of surface area. This heat exchanger is a duplicate of heat exchanger 503A, which was also replaced in December 2006. 2006

2.1.2.3 The "Y" type strainer (designated 597C) is an 8" aluminum body, flanged ends with stainless steel screen having .062-inch diameter perforations. The strainer will have a pressure drop of less than 3.5 psi at 1800 gpm. A 1/2-inch drain valve will be installed in the "Y". This valve will be of the same type as drain and vent valves in the existing system.

2.1.2.4 Gauges for pump 501B suction pressure (PI-916F) and discharge (PI-916E) pressure indication will be 4 1/2" dial size, type 1811T, U.S. Gauge Co. which is identical to the existing gauges on 501A.

2.1.2.5 The flow orifice (designated 913B) will be made of type 304 stainless steel and sized for 1800 gpm normal flow. This orifice is identical to orifice 913A in the existing system. In December 2006, the flow orifice flanges for 913B were replaced with dual-tap flanges to allow for additional flow monitoring sensing points. These flow orifice flanges are identical to those used for 913A, which were also replaced in December 2006. 2006

2.1.2.6 Valve number 540B is the throttle valve located downstream of heat exchanger 503B. This valve is an 8" diaphragm valve with 316 stainless steel flanged body and an ethylene propylene diaphragm. It is identical to valve 540A in the existing piping with the exception of the body material. Valve 540A has an aluminum body.

2.1.2.7 Valve number 517B (pump 501B discharge check valve) is an 8" check valve, aluminum flanged body with stainless steel disc and heavy walled, nonporous aluminum or stainless steel cap. This valve differs from existing valve 517A only in the materials used in the disc and cap. The stainless steel disc and heavy walled cap will provide additional strength.

2.1.2.8 The valves numbered 510D, 510E, and 510F will be 8" gate valves with aluminum flanged body and Johns-Manville 443V Buna-N or BNF V-Ring packing. These valves are identical to the existing 510 series valves with the exception of the packing material.

2.1.2.9 The valve 538B is a 3" diaphragm valve with aluminum flanged body and ethylene propylene diaphragm.

2.1.2.10 The valves 518AB, 518AH and 518AI will be 1/2" diaphragm valves with threaded aluminum bodies and ethylene propylene diaphragms. These valves are identical to the existing 518 series valves.

2.1.2.11 Valve 515Y will be a 2" diaphragm valve with aluminum flanged body and ethylene propylene diaphragm.

2.1.2.12 The valves numbered 595 are the cutout valves for the ΔP switches (DPS) and pressure switches (PS). They will be 1/4" Grinnell R-P&C 1070A, 18-8-Mo stainless steel globe valves. As of December 2006, the valves numbered 595C, 595D, 595E, and 595F are the isolation valves for normally installed but isolated pressure gauges used for long-term trending of heat exchanger differential pressure. 2006

2.1.2.13 The valves numbered 599V and 599W are isolation valves for drain lines from FT 912E. They will be 1/4" gate valves with aluminum threaded bodies.

2.1.2.14 Valve manifold number 568B is the isolation valve manifold for FE 913B. It will be an Anderson-Greenwood Model MI-VS4, assembly 2-8155-1, pressure rated at 6000 psi, 200°F, with body and trim of 316 stainless steel. "As of December 2006, FE 913B is instrumented with valve manifolds 568I and 568J." 2006

2.1.2.15 A calibration well will be installed adjacent to TE-980B on the outlet of heat exchanger 503B. This well will be identical to the RTD well for TE-980B and will be used for calibration of TE-980B.

2.1.3 <u>System Changes</u>

The purpose of this section is to discuss changes to be made to the existing primary system. These changes will be done in conjunction with the installation of the new equipment discussed in the previous section. The following changes will be made to the primary system.

2.1.3.1 The existing paddle switches are being removed because of unreliable operation and reactor safety. Operating experience has now shown that past scrams for no apparent reason were due to fluctuations on these switches. While not having occurred, inspections of these switches has shown that the paddle may break free and enter the system as a foreign object. The differential pressure sensors will be the new back up loss of flow protection. The ΔP method is being used on the pool system flow and has proven to be safe and reliable.

The differential pressure sensors will be designated DPS 929 and 928A and will be located across the reactor inlet and outlet piping and across the inlet and outlet of the existing heat exchanger 503A, respectively. These sensors will be identical to the differential pressure sensor (DPS 928B) which is to be installed across the new heat exchanger 503B. The installation of DPS-929 will be done in conjunction with another change involving the addition of isolation valves for pressure sensors PS-944A and PS-944B. Both pressure sensors are located on the reactor outlet piping. PS-944A will have isolation valve 595M and PS-944B will have valve

595N. The low pressure connection for DPS 929 will be between valve 595N and PS-944B. The high pressure connection for DPS 929 will be made to the reactor inlet piping and will use valve 595L as an isolation valve. The 595 series valve used in this modification will be identical to those described in paragraph 2.1.2.12.

In December 2006, DPS 928A and DPS 928B were removed as part of the primary coolant heat exchanger replacement project which consisted of replacing the shell and tube-type heat exchangers with plate-type heat exchangers. New dual tap flanges were installed for flow orifices 913A and 913B, which allowed an additional flow transmitter to be connected to each heat exchanger leg, thus eliminating the need for DPS 928A and DPS 928B. 2006

2.1.3.2 A 1/2-inch diaphragm valve, 518AA will be installed in the existing heat exchanger loop between FE-913A and valve 540A. This valve will be utilized as a vent to facilitate draining the heat exchanger 503A. This valve will be identical to the 518 series valves described in paragraph 2.1.2.10.

2.1.3.3 A new strainer will be installed in the existing heat exchanger loop. The old strainer will be removed and the new strainer installed in the piping on the reactor side of valve 540A. This strainer will be identical to the strainer described in paragraph 2.1.2.3.

2.1.3.4 TE-901A will be relocated to gain safer and easier access when doing RTD calibrations. The location now is in the overhead piping directly over the opening for the pipe tunnel which gives limited access and presents a hazard to personnel performing the calibration.

2.1.3.5 TE-980A (RTD) will be an addition to the existing heat exchanger loop and will be located on the outlet side of heat exchanger 503A. TE-980A will be utilized for determining the efficiency of heat exchanger 503A.

2.1.3.6 Calibration wells will be installed in the primary piping immediately downstream of RTD's TE-901A, TE-901B and TE-980A. The wells will be utilized for calibration of the RTD and will be identical to the RTD wells.

2.2 Pool Coolant System

2.2.1 Introduction

To facilitate 10 MW operation of the reactor, it was necessary to double the heat removal capability of the 5 MW pool water cooling system. At 10 MW upgrade an additional pump was installed in parallel with the existing unit. Subsequent analysis revealed the pool system will have adequate cooling with a single pump. Modifications have been made to the existing pool system to improve its reliability and to facilitate parallel operation of the pumps. All components described in this section are shown in Figure 2.1 (see Addendum 4 of HSR for revised Figure A.2--Piping & Instrument Diagram).

2.2.2 Additional Pool System Pump and Heat Exchanger

The major change in the pool system for 10 MW operation was the installation of an additional pump, doubling heat exchange capability, plus valve fittings and instrumentation associated with these components. The new pump has been installed in piping leg which parallels the existing pump of the existing system. Flanges have been provided in the existing system for connection of additional components. All pipes and fittings in the new legs will be schedule 40 aluminum 6061-T6. All connections which are not shown as flanged on Figure 2.1 will be welded by the inert gas shielded arc process. Consumable insert rings will be used on all pipe butt welds. All welds will be examined radiographically.

2.2.2.1 The circulating pump designated 508B is a Worthington Corporation centrifugal pump type 6CNG-104 and is driven by a 60 hp, 1750 rpm, drip-proof motor. This unit is identical to the existing pump with the exception of the motor frame size. The frame size of the new motor is 364T as compared to a frame size of 404US for the existing pump motor.

2.2.2.2 The heat exchanger designated 521 is a plate type heat exchanger constructed of stainless steel plates with EPDM gaskets. The heat exchanger will have spray shields installed on it.

2.2.2.3 The valve 531B is a manually operated 3" diaphragm valve with aluminum flanged body and ethylene propylene diaphragm. This valve is identical to valve 531A in the existing system.

2.2.2.4 Isolation valves 539D and F are 4" gate valves with aluminum flanged body and Johns-Manville BNF-V-Ring packing. This valve is identical to the existing 539 series valves with the exception of the packing material.

2.2.2.5 The check valve 535B is a 4" swing check with aluminum flanged body and stainless steel disc. This valve is identical to the existing valve 535A except its disc is made of stainless steel and its cap is strengthened.

2.2.2.6 Pump suction and pressure indicators PI 927C and D are type 1811T, U.S. Gauge Co. pressure gauges which are identical to the existing pressure gauges.

2.2.2.7 The Y-strainer designated 597B is a 6" aluminum flanged end strainer with stainless steel screen having .062 inch diameter perforation. This strainer will replace the presently installed cone type strainer.

2.2.2.8 The valves designated 522D and E are 4" diaphragm valves with aluminum flanged bodies and ethylene propylene diaphragms. These valves are identical to the existing 522 series valves.

2.2.2.9 A single flow orifice designated FE-921 is made of stainless steel and has four sets of flange taps, two sets attached to flow transmitters.

2.2.2.10 All 518 series valves are 1/2" aluminum screwed body diaphragm with ethylene propylene diaphragms. These valves are identical to the existing 518 series valves.

2.2.2.11 The 515Z valve is a 2" aluminum, flanged body, diaphragm valve with ethylene propylene diaphragm. This is identical to the existing 515 series valves.

2.2.2.12 The manifold valve (568D) for the ΔP transmitter is an Anderson-Greenwood Model MI-VS4 with 316 stainless steel body and trim. This valve manifold is identical to the existing manifold 568C.

2.2.2.13 The calibration wells will be RTD wells made of 304 stainless steel and having a 3" immersion. These wells will be used to calibrate the RTDs immediately upstream of them.

2.2.3 System Changes

Changes have been made to the existing pool system to facilitate operation of a single heat exchanger and to improve the capabilities of calibrating the RTDs and venting the system.

2.2.3.1 The outlet isolation valve on heat exchanger 521 has been changed from a 4" diaphragm valve of 522 series type to a 6" diaphragm type. This valve is used to control the flow through heat exchanger 521.

2.2.3.2 Valves 518U and 518AJ were added to the existing piping to provide a vent for this section of pipe. These valves are identical to the other 518 series valve described in paragraph 2.2.2.10.

2.2.3.3 The RTD designated TE-901D was moved from its existing location in the maze of pipes above the pipe tunnel to a location which is more accessible for repairs and maintenance.

2.2.3.4 Calibration wells were installed downstream of RTDs TE 901 C and D. These wells are identical to the wells for the RTDs and are used for calibration of the RTDs.

2.2.3.5 The RTD designated TE-980C has been installed to the inlet side of heat exchanger 521. This RTD along with TE-990C on the secondary water side will enable the operator to calculate the heat being removed by the heat exchanger and permits determination of the heat transfer coefficient for the heat exchanger which provides early warning of possible fouling of the heat exchanger.

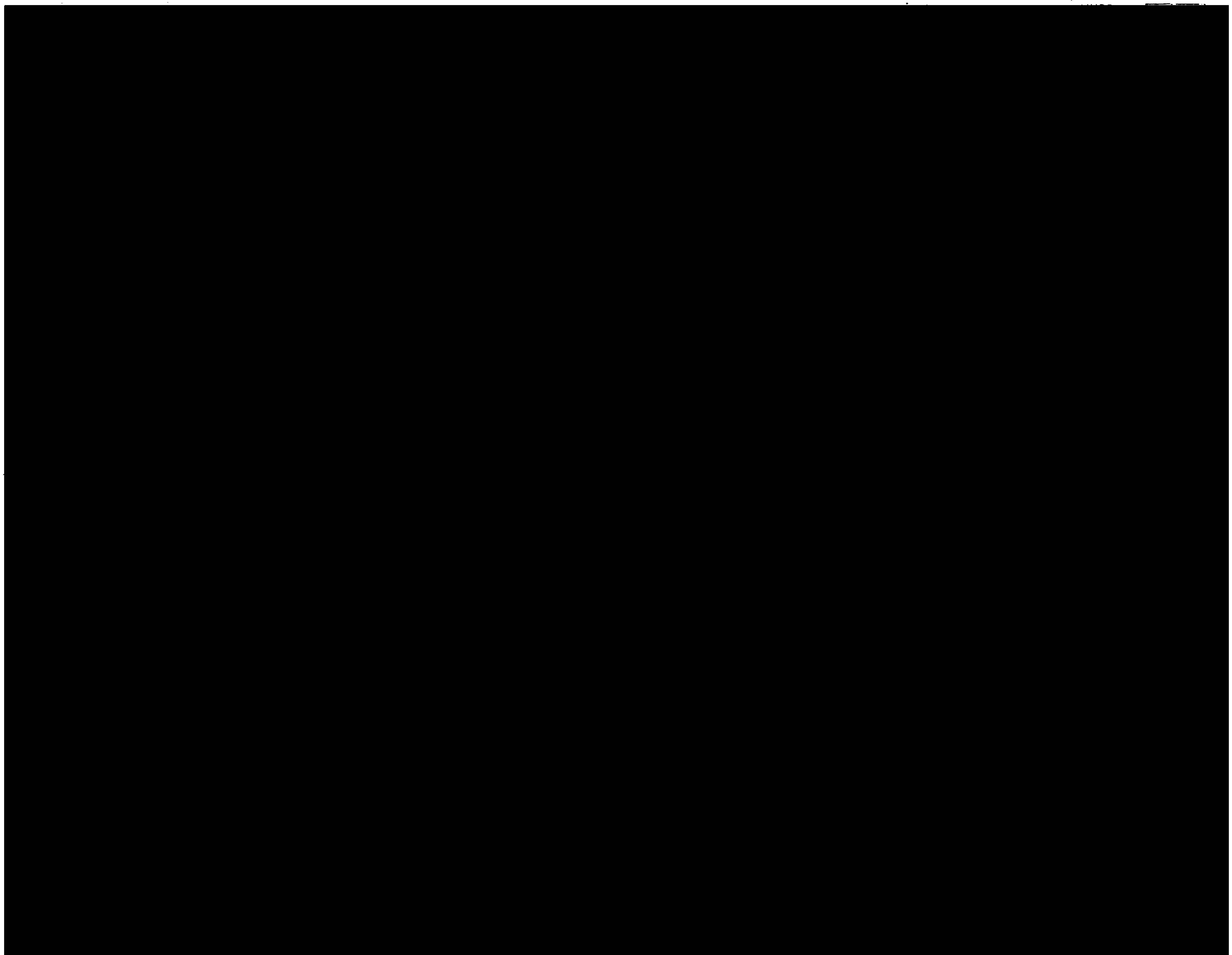


Figure 2.1

2006

2.3 Secondary Coolant System

2.3.1 Introduction

The secondary system will readily accept the addition of the new cooling equipment needed for 10 MW operation. The system was originally designed for 3 pumps and the piping has the necessary tie-in points which are presently capped. Refer to secondary cooling system drawing (Figure 2.2).

2.3.2 Equipment necessary to upgrade system to 10 MW capacity.

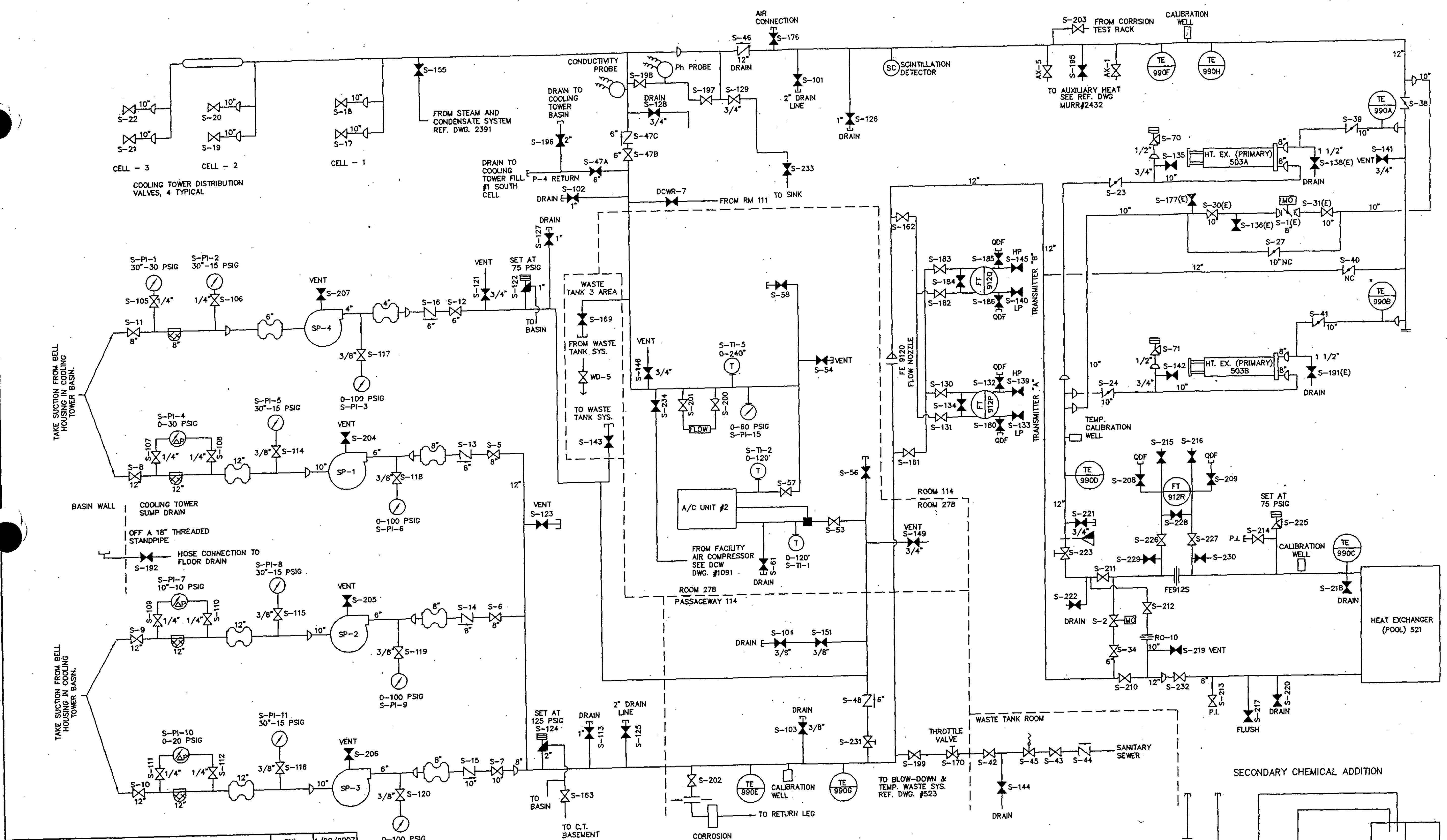
2.3.2.1 Cooling pumps (SP-3) is an Ingersoll-Rand pump, powered by 150 HP Toshiba motors ~~General Electric motors~~ and delivers 2200 gpm. 2006

2.3.2.2 Cooling tower cell #3 is a double flow, induced draft, Marley Model 452-102, series 14 with attached dual speed, Allis Chalmers powered, 40/10 hp, 1760/870 rpm, adjustable pitch cooling fan. This unit has already been installed, tested, and is presently in operation.

2.3.2.3 All valves and piping will be compatible with existing system components. The discharge check valve (S-15) and isolation valve (S-7) will be 10" instead of the presently installed 8" valves on the other 2 existing pumps to better meet design criteria.

2.3.2.4 Additional instrumentation will be covered in the electronics upgrading section. The only change in the present system will be the relocation of the scintillation detector from its present location of the combined pump discharge line to a point in the return piping downstream of the pool and primary heat exchangers to insure a faster response in the event of a leak from either pool or primary heat exchangers.

FIGURE 2.2



REV. NO.	DESCRIPTION	DR. BY	DATE
32	NEW HEAT EXCHANGERS AND COMPONENTS	RNL	1/22/2007
31	ADD VENT, DRAIN, LP AND HP LABLES	DGN	8/1/2006
30	REMOVED LiBr CHILLER	BAS	10/24/05
29	UPDATED PER CONTROL ROOM	BJN	2/12/04
28	MODIFICATION TO BLOWDOWN PIPING	BJN	7/3/03
27	RELOCATION OF BLOWDOWN PIPING	BJN	4/24/03
26	ADDED VALVE NOMENCLATURE	JJL	8/6/02
[illegible]	ADDED SECONDARY CHEMICAL ADDITION	JCA	8/9/01
[illegible]	MOD 01-5	JCA	6/21/01
[illegible]	UPDATED PER CONDITION ASSESSMENT	DGN	1/28/99
22	UPDATED PER CONTROL ROOM	JDR	5/5/98
21	CHANGED VALVE S-35 TO VALVE S-2	JDR	8/7/96
20	REMOVED FUTURE STRAINER & S-2 PER CONTROL RM.	JDR	11/2/95
19	UPDATED AND REVISED ON CAD, MOD. PACKAGE 94-5	JDR	8/10/95

ABBREVIATION REFERENCE

FCV = FLOW CONTROL VALVE
QDF = QUICK DISCONNECT FITTING
DCWR = DOMESTIC COLD WATER RETURN
S = SECONDARY SYSTEM
FT = FLOW TRANSMITTER
TE = TEMP. ELEMENT
AX = AUXILIARY HEATING AND COOLING SYSTEM
FE = FLOW ELEMENT
* TE REMOVED FROM WELLS

LEGEND

GATE VALVE
GLOBE VALVE
DIAPHRAGM VALVE
CHECK VALVE
RELIEF VALVE
NORMALLY CLOSED
AO AIR OPERATOR
SOLENOID OPERATOR
BALL VALVE
A/C CONTROL VALVE

PAGE 15

seal
2006

DATE: 4/26/95
DRAWN BY: JOEY RILEY
CHECKED BY:
ENGINEER: CPF
CODE:
REVISION NUMBER: 32
REVISION DATE: 01/22/2007

FIGURE 2.2

UNIVERSITY OF MISSOURI-COLUMBIA
FACILITIES OPERATION
research reactor facility

SECONDARY COOLING SYSTEM

MURR NUMBER: 502

SHEET 1 OF 1

2006

2.4 Electrical Distribution System

2.4.1 Introduction

The facility electrical distribution system will be modified as shown on Figures 2.3, 2.4 and 2.5. The system changes include the addition of motors and controllers for the new pumps (SP-3, P-501B and P-508B), splitting motor control center #2 into #2A and #2B, adding a welding breaker in motor control center #2B, increasing the current capacity of the wiring from substation B to motor control center #5, and adding six electrical penetration connectors in the containment building wall.

2.4.2 Description of Major Components

2.4.2.1 Pump SP-3 Motor

~~GE Model 5K405AK230B1~~
~~125 hp - 1780 rpm FL~~
~~460 volts - 60 cycle - 3 w~~
~~FL Amps 151~~
~~Type K - Frame 405TS~~
~~Nema Class Design B - Code G~~

Toshiba Model B1504ULF4USW
150 HP - 1780 rpm FL
460 volts - 60 cycle - 3 wire - 3 phase (variable speed rating)
FL Amps 170
Type TIKK - Frame 444T
NEMA Class Design B - Code G

2006

2.4.2.2 Pump SP-3, 501-B and 508-B Controllers

~~General Electric c~~
~~operation for the~~

"Pumps 501B and 508B have General Electric controllers sized to fit 7700 line controllers and provide operation for the associated motors. Pump SP-3 has a General Electric Evolution Series controller sized to provide operation for its motor."

2006

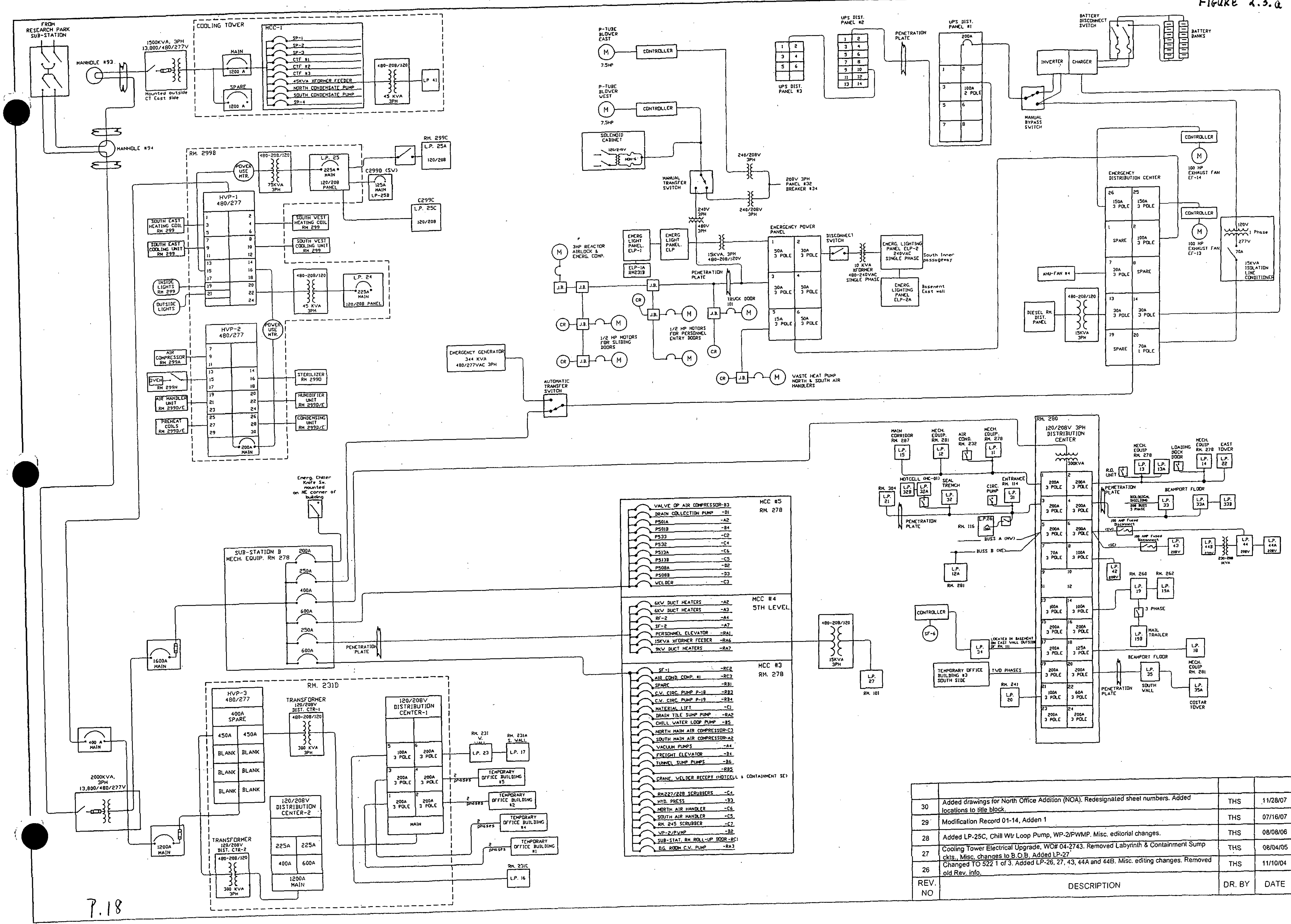
2.4.2.3 Pump 501-B Motor

GE Model 5K405AK205B1
hp 125 - 1780 rpm
460 volts - 60 cycle - 3 wire - 3 phase
FL Amps 151
Type K - Frame 405T
Nema Class Design B - Code G

2.4.2.4 <u>Pump 508-B Motor</u>
GE Model 5K364AK205
hp 60 - FL rpm 1770
FL Amp 149 74.5
Volts 230-460
3 phase - 60 cycle
Type K - Frame 364T
Nema Class Design B - Code G

2.4.2.5 Six pressure proof connectors will be placed in the penetration plate of east wall with connectors to contain 19 pins for #12 wire. Pyle National No. ZRELT-416-12S and No. ZPELT-416-12P.

FIGURE 2.3.a



P.18

REV. NO	DESCRIPTION	DR. BY	DATE
30	Added drawings for North Office Addition (NOA). Redesignated sheet numbers. Added locations to title block.	THS	11/28/07
29	Modification Record 01-14, Adden 1	THS	07/16/07
28	Added LP-25C, Chill Wtr Loop Pump, WP-2/PWMP. Misc. editorial changes.	THS	08/08/06
27	Cooling Tower Electrical Upgrade, WO# 04-2743. Removed Labyrinth & Containment Sump ckts., Misc. changes to B.O.B. Added LP-27	THS	08/04/05
26	Changed TO 522 1 of 3. Added LP-26, 27, 43, 44A and 44B. Misc. editing changes. Removed old Rev. info.	THS	11/10/04

seal 2007

DATE: 12/15/00
DRAWN BY: TOM SEEGER
CHECKED BY: ENGINEER: LAF
CODE:
REVISION NUMBER: 30
REVISION DATE: 11/28/07

FIGURE 2.3.a

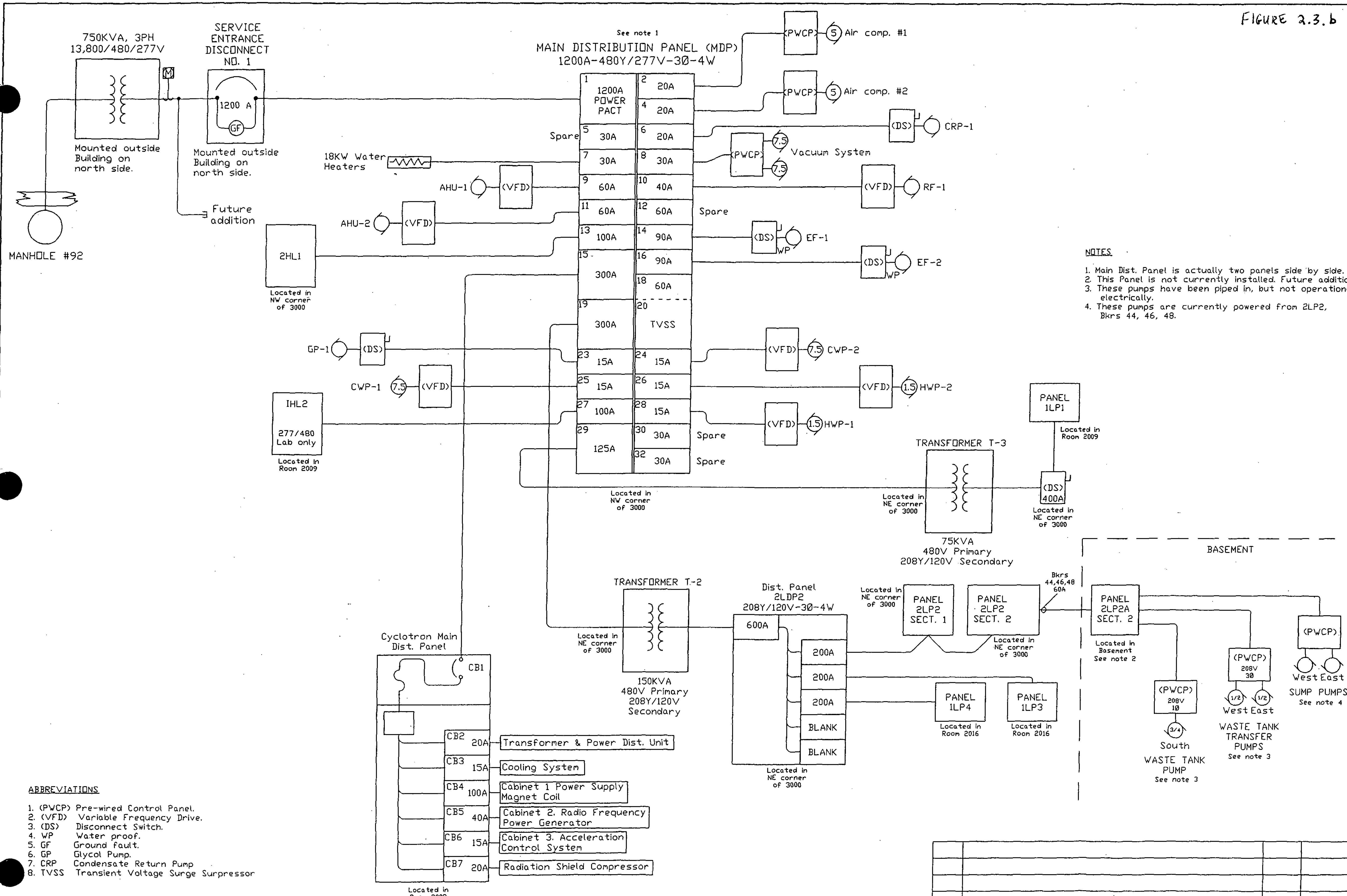
UNIVERSITY OF MISSOURI-COLUMBIA
FACILITIES OPERATIONS
research reactor facility

ELECTRICAL DISTRIBUTION
REACTOR/LABORATORY

MURR NUMBER: 522

SHEET 1 of 5

FIGURE 2.3.b



NOTES

1. Main Dist. Panel is actually two panels side by side.
2. This Panel is not currently installed. Future addition.
3. These pumps have been piped in, but not operational electrically.
4. These pumps are currently powered from 2LP2, Bkrs 44, 46, 48.

ABBREVIATIONS

1. (PWCP) Pre-wired Control Panel.
2. (VFD) Variable Frequency Drive.
3. (DS) Disconnect Switch.
4. WP Water proof.
5. GF Ground fault.
6. GP Glycol Pump.
7. CRP Condensate Return Pump
8. TVSS Transient Voltage Surge Suppressor

REV. NO	DESCRIPTION	DR. BY	DATE
0	Added drawings for North Office Addition (NOA). Redesignated sheet numbers. Added locations to title block.	THS	11/28/07

seal 2007

DATE: 11/28/07
DRAWN BY: TOM SEEGER
CHECKED BY: ENGINEER: CAF
CODE:
REVISION NUMBER: 0
REVISION DATE: 11/28/07

FIGURE 2.3.b

UNIVERSITY OF MISSOURI-COLUMBIA
FACILITIES OPERATIONS
research reactor facility

ELECTRICAL DISTRIBUTION
NORTH OFFICE ADDITION

MURR NUMBER: 522

SHEET 2 of 5

LP-11 MECH EQUIP RM 280

1	LTS RM 232, 232A
2	LTS RM 227
3	LTS RM 228, FAN RM 294
4	LTS RM 225
5	LTS RM 229, 294, WEST CORR (N)
6	LTS RM 223, 226, 280 ALLEYWAY, 220
7	RECEPT RM 226, 227, 228, WEST CORR (N)
8	LTS NORTH CORR (W), EAST DRIVE (N) NORTH PARKING SOUTH BANK
9	RECEPT RM 229, 231, 232
10	LTS RM 224
11	RECEPT W. CORR N, OUTSIDE BUILDING
12	LTS RM 222
13	LTS MACH RM 280
14	RECEPT RM 222, 223, 224, 225, 226
15	LTS MACH RM 280
16	RM 231 WASTE HEAT PUMP
17	LTS W. TOWER, EXIT LTS ABOVE N. EXIT
18	RM 231 WASTE HEAT PUMP
19	LTS & RECEPT NORTH TOWER
20	RM 231 WASTE HEAT PUMP
21	LTS & RECEPT WEST TOWER
22	RECEPT N. CORR W, OUTSIDE BUILDING
23	LTS & RECEPT NORTH TOWER
24	PARKING PHOTOCELL CKT, RECEPT RM 278 N
25	LTS & RECEPT NORTH TOWER
26	LTS NORTH PARKING LOT, NORTH BANK
27	RECEPT RM 232 (CLOTHES DRYER)
28	RECEPT RM 232B
29	SAME AS 27
30	LTS & FAN RM 232B
31	AIR COMPRESSOR (C.A.) JOHNSON
32	LABYRINTH SUMP PUMP CONTROL & LTS.
33	SAME AS 31
34	SAME AS 32
35	SAME AS 31
36	SAME AS 32

LP-12 MECH EQUIP RM 281

1	LTS RM 283, 288, 289, RECEPT 283
2	LTS RM 218
3	LTS RM 292
4	LTS RM 216
5	EXHAUST FAN RM 292
6	LTS RM 215, 215A
7	LTS RM 282
8	LTS RM 213
9	LTS RM 282
10	LTS NORTH CORR. EAST
11	LTS CORR. 281 NE, LTS RM 284
12	LTS RM 214, 217
13	RECEPT RM 282 EAST
14	LTS RM 212
15	
16	RECEPT RM 289
17	WATER FOUNTAIN EAST CORR, RECEPT RM 289
18	
19	RECEPT RM 282
20	RECEPT RM 216, 217
21	RECEPT RM 284
22	RECEPT RM 213, 214, L214A, NORTH CORRIDOR
23	RECEPT RM 282 WEST
24	RECEPT CORRIDOR 281
25	LTS IN PIPE CHASE
26	RECEPT IN EAST TOWER
27	AIR HANDLING UNIT RM 212
28	FILTER EAST TOWER, LOWER
29	SPARE
30	LTS EAST TOWER
31	RECEPT RM 214, 214A
32	FILTER EAST TOWER, TOP
33	
34	LTS RM 288A, RECEPT RM 288, 288A
35	SPARE
36	RECEPT RM 288

LP-12A N. INNER PASSAGE

1	SPARE
2	SPARE
3	SPARE
4	SPARE
5	SPARE
6	RECEPT STRIP NE WALL RM 218
7	SPARE
8	RECEPT CORNER OF EAST WALL RM 218
9	
10	
11	

LP-13 MECH EQUIP RM 278

1	LTS LOADING DOCK, RM 250, 251, 252 RECEPT 250, DOCK DOOR SOLENOID
2	SPARE
3	LTS RM 247
4	EXHAUST FAN
5	LTS RM 245
6	LTS RM 233, 234, 235, 236, 239, 240, 295, FAN RM 295, RECEPT WEST OUTER HALL
7	LTS RM 243, 246, 278 ALLEY
8	LTS RM 238
9	LTS RM 242, 242A, 242B, 242C
10	LTS SOUTH CORRIDOR WEST
11	LTS RM 244
12	LTS WEST CORRIDOR
13	RECEPT RM 235, 238
14	LTS WEST CORRIDOR
15	RECEPT RM 233, 235, WEST CORRIDOR
16	SPARE
17	RECEPT RM 240, 241, WEST CORRIDOR
18	RECEPT RM 242, 243, 244
19	RECEPT RM 246, SOUTH CORRIDOR
20	RECEPT SOUTH CORRIDOR
21	RECEPT RM 245, 246, 247
22	LTS CORRIDOR 278
23	RECEPT CORRIDOR 278
24	LTS CORRIDOR 276
25	SPARE
26	FAN CORRIDOR 278
27	RECEPT 238 WEST WALL (N), SOUTHWALL (W)
28	RECEPT 238 SOUTH WALL (E), WEST WALL (S)
29	RECEPT 238 EAST WALL
30	RECEPT 238 NORTH WALL, EAST WALL (N)
31	RECEPT RM 238 EAST WALL
32	SPARE
33	SPARE
34	SPARE
35	SPARE
36	SPARE

LP-13A MECH EQUIP RM 278

1	238 AIR HANDLER
2	SPARE
3	SAME AS 1
4	SPARE
5	SAME AS 1
6	SPARE
7	SPARE
8	TWIST-LOCK RECEPT RM 242C, 208VAC
9	SPARE
10	SAME AS 8
11	RECEPT RM 242C (EAST WALL)
12	RECEPT RM 242C (EAST WALL)

LP-14 MECH EQUIP RM 273

1	LTS RM 256,258, SOUTHEAST CORRIDOR
2	LTS RM 257
3	LTS RM 255
4	LTS RM 259
5	LTS CORRIDOR 273, RECEPT 273
6	LTS RM 260
7	RECEPT RM 271B
8	EMERGENCY LT. RM 273
9	LTS RM 271B
10	LTS RM 273 PIPE CHASE
11	LTS RM 271B
12	RECEPT EAST CORRIDOR, RM 269 (COKE MACHINE)
13	LTS CENTER CORRIDOR
14	LTS RM 269, RECEPT RM 267A
15	FAN RM 271B
16	LTS RM 267,271A
17	WATER FOUNTAIN EAST CORR, SOUTH SIDE
18	FAN RM 266, 267A, LTS RM 267A
19	RECEPT RM 251
20	RECEPT RM 273
21	RECEPT RM 267, 271A, CENTER CORRIDOR
22	RECEPT RM 271A, 271B
23	RECEPT RM 255, 256, 258, SOUTH CORRIDOR
24	RECEPT RM 267A GARBAGE DISPOSAL RM 269
25	RECEPT RM 258, 259, SOUTH CORRIDOR
26	RECEPT RM 271B
27	DOCK BZZER & HI LEVEL SUMP ALARM
28	LTS SOUTH TOWER RM 308
29	SOUTH TOWER 220VAC HEATER
30	RECEPT RM 306, AMPLIFIERS FOR 300-301
31	SAME AS 29
32	SOUTH TOWER UNIT HEATER
33	ELECTRIC RANGE RECEPT RM 269
34	SPARE
35	SAME AS 33
36	RECEPT RM 274

LP-15 MECH EQUIP RM287

1	LTS RM 261, 262, 262A
2	SPARE
3	RECEPT RM 262A, LIGHTS RM 262B, 262C
4	LTS RM 202, 203
5	LTS RM 263, 263A
6	RECEPT RM 204, LTS RM 202, 204
7	SPARE
8	LTS IN OUTSIDE CANOPY
9	LTS RM 205B, 206, 208, RECEPT 202, 205A, 205B
10	LTS IN OUTSIDE CANOPY
11	LTS RM 207, 209A, 209B
12	SPARE
13	LTS RM 209A
14	LTS EAST CORRIDOR NORTH
15	FAN RM 209
16	LTS EAST CORRIDOR SOUTH END
17	RECEPT CENTER, EAST RM 269
18	RECEPT RM 261, 262A, 262B, 262C
19	RECEPT CENTER CORRIDOR
20	RECEPT RM 202A, LTS RM 202A
21	RECEPT RM 202, 205A, 205B, 206 RECEPT EAST CORRIDOR NORTH, LTS RM 202
22	RECEPT RM 261, EAST OUTSIDE OF BLDG.
23	RECEPT RM 206, 208, 209A, 209B TWIST-LOCK RECEPT EAST CORRIDOR NORTH END
24	RECEPT RM 201, EAST EXTERIOR
25	RECEPT RM 209A, 209B
26	RECEPT RM 203
27	RECEPT 202A
28	RECEPT RM 203
29	RECEPT RM 202A
30	SPARE
31	RECEPT RM 269 MICROWAVE CABINET
32	RECEPT RM 269
33	RECEPT RM 269
34	RECEPT RM 269
35	RECEPT RM 269
36	RECEPT RM 204 WEST

LP-16 RM 231C

1A	LTS RM 231B, RECEPT RM 231, 231B, 231F
1B	RECEPT RM 231C NORTH EAST
2A	RECEPT NORTH RM 231, SOUTH 231B
2B	RECEPT NORTH RM 231, SOUTH 231B
3A	
3B	
4	EMERG LTS RM 231C, RECEPT RM 231B WEST, 231 EAST
5	EAST LTS RM 231C
6	
7	LTS RM 231, 231A
8	RM 231A OVERHEAD DOOR & DRILL PRESS
9	LTS RM 231, 231 EXIT LTS.
10	SAME AS 8
11	LTS RM 231A, 231F
12	SAME AS 8
13	MOTOR (AIR)
14A	RM 231C CENTER CUBICLES
14B	RECEPT RM 231C WEST
15	SAME AS 13
16A	RECEPT RM 231C SOUTH
16B	RECEPT RM 231C WEST
17	SAME AS 13
18A	WEST LTS RM 231C
18B	
19	RECEPT RM 231, DRINKING FOUNTAIN
20A	NORTHEAST RECEPT STRIP RM 231C
20B	SOUTHEAST RECEPT STRIP RM 231C

LP-17 Rm 231A

1	RECEPT RM 231A
2	RECEPT RM 231A EAST & SOUTH WALLS
3	SPARE
4	SPARE
5	SPARE
6	SPARE
7	SAME AS 5
8	SAME AS 6
9	SPARE
10	SPARE
11	SAME AS 9
12	SAME AS 10

LP-18 MECH EQUIP RM 281

1	NC
2	MAIN
3	NC
4	SAME AS 2
5	NC
6	SAME AS 2
7	RECEPT RM 508 (208VAC)
8	1ST & 3RD REFRIG. IN CORRIDOR 281
9	SAME AS 7
10	2ND REFRIG IN CORRIDOR 281
11	DI PUMP IN RM 300
12	4TH REFRIG IN CORRIDOR 281
13	SPARE
14	UPS EXHAUST FAN IN RM 300
15	SAME AS 13
16	NC
17	NC

LP-19 Rm 260

1	SOUTH DRIVE MAIL TRAILER
2	SPARE
3	SAME AS 1
4	SPARE
5	SAME AS 1
6	SPARE
7	SPARE
8	SPARE
9	SPARE
10	LP-19A
11	SPARE
12	SAME AS 10
13	SPARE
14	SAME AS 10

LP-19 A RM 262

1	1/2 4 PLEX RM 263
2	RECEPT RM 262
3	RECEPT RM 263A
4	RECEPT RM 262B
5	RECEPT RM 263 & 1/2 4 PLEX RECEPT RM 263
6	RECEPT RM 262A
7	RECEPT RM 261
8	SPARE
9	SPARE
10	SPARE
11	SPARE
12	SPARE

LP-19 B MAIL TRAILER

1	HEATING & AIR CONDITIONING
2	LTS
3	SAME AS 1
4	SINK RM LTS, RECEPT, WTR HTR, HEAT TAPE
5	OUTSIDE LTS
6	RECEPT SOUTH
7	SPARE
8	RECEPT NORTH
9	RECEPT UNDER TRAILER

LP-20 RM 241

1	WELDING RECEPT RM 241
2	WELDING RECEPT RM 242
3	SAME AS 1
4	SAME AS 2
5	HC-05
6	RECEPT RM 241
7	SAME AS 5
8	RECEPT RM 241
9	SAME AS 5
10	SAME AS 8
11	NC
12	SAME AS 8
13	NC
14	GFCI RECEPT RM 241
15	NC
16	NC
17	NC
18	NC
19	RM 241 VENT MOTOR
20	LTS RM 241
21	GFCI RECEPT RM 241
22	GFCI RECEPT RM 241
23	RECEPT RM 241
24	GFCI RECEPT RM 241

LP-21 RM 304

1	NORTH HIGH LTS WEST ROW
2	LTS RM 503
3	SOUTH HIGH LTS WEST ROW
4	LTS RM 404, 405A
5	SOUTH HIGH LTS EAST ROW
6	LTS RM 402A, 402B, 403
7	CENTER HIGH LTS CENTER ROW
8	LTS RM 503, DOORS 504 & 505
9	NORTH HIGH LTS EAST ROW
10	RECEPT RM 285
11	LTS RM 302A, 304, SECURITY SYSTEM
12	RECEPT RM 285, 286, STAIR BATTERY LT.
13	LTS RM 302
14	ELEVATOR LTS
15	LTS RM 285
16	RECEPT RM 304
17	LTS RM 275, AIRLOCK
18	RECEPT INST. CABINET BASE RM 302A
19	RECEPT RM 301, 302
20	RECEPT RM 404, 405
21	RECEPT ON BRIDGE
22	RECEPT RM 401, 402B, 503, EMERG. BATTERY LTS 4TH & 5TH LEVEL
23	DISHWASHER RM 304
24	RECEPT RM 275
25	RECEPT RM 302A, CT DOOR ALARM POOL HT. EXCH FLOW D/P CIRCUITRY
26	RECEPT RM 275
27	RECEPT INST. CABINET BASE, DCW, FIRE MAIN ALARMS
28	RECEPT RM 275
29	RECEPT ON BRIDGE
30	HOIST MONORAIL
31	GARBAGE DISPOSAL RM 304
32	SAME AS 30
33	AC UNIT 5TH LEVEL
34	HOT WATER HEATER
35	SAME AS 33

LP-22 EAST TOWER

1	EAST PARKING LOT, SOUTH LIGHT CONTROL
2	EAST ROOF SPOTLIGHT
3	SPARE
4	SOUTH ROOF SPOTLIGHT
5	
6	NORTH ROOF SPOTLIGHT
7	
8	SOUTHEAST PARKING & ROOF LTS CONTROL

LP-23 RM 231

1	RM 231 (MILL & DRILL PRESS)
2	LP-17 RM 231A
3	SAME AS 1
4	SAME AS 2
5	SAME AS 1
6	WEDGE FURNACE RM 231D
7	FORKLIFT CHARGER RM 231D
8	SAME AS 6
9	SAME AS 7
10	SAME AS 6
11	SAME AS 7
12	AUTOMATIC WELDER RM 231
13	RM 231 LATHE
14	SAME AS 12
15	SAME AS 13
16	SAME AS 12
17	SAME AS 13
18	RM 231 BAND SAW
19	RM 231A CAN SEALER
20	SAME AS 18
21	SAME AS 19
22	SAME AS 18
23	SAME AS 19
24	RM 231 208 VAC RECEPT
25	RECEPT RM 231 SOUTH WALL
26	SAME AS 24
27	
28	RECEPT RM 231 EAST WALL
29	
30	
31	
32	
33	
34	
35	
36	

LP-24 RM 299B

1	RECEPT RM 299 EAST
2	SPARE
3	RECEPT RM 299 SOUTH WALL
4	SPARE
5	RECEPT RM 299 CORRIDOR
6	RECEPT RM 299 NORTH
7	SPARE
8	SOUTH OVERHEAD DOOR & RECEPT.
9	RECEPT RM 299B
10	RM 299 OUTSIDE GFCI RECEPT (WEST)
11	RECEPT RM 229A, 299B, LIGHTS RM 299B
12	EAST AIR HANDLER
13	RM 299 CORRIDOR OVERHEAD DOOR & RECEPT.
14	SAME AS 12
15	WATER HEATER RM 299A
16	WEST AIR HANDLER
17	SAME AS 15
18	SAME AS 16

LP-25 RM 299B

1	LP-25A RM 299C
2	LP-25B
3	SAME AS 1
4	NC (Space unavailable due to adjacent breaker)
5	SAME AS 1
6	SAME AS 2
7	NC
8	NC (Space unavailable due to adjacent breaker)
9	NC
10	SAME AS 2
11	NC
12	NC (Space unavailable due to adjacent breaker)
13	NC
14	WASTE TANK TRANSFER PUMP 299A
15	NC
16	AIR DRYER RM 299A
17	LP-25C (RM C299C)
18	NC
19	NC
20	NC
21	LP-25C (RM C299C)
22	NC
23	NC
24	NC
25	LP-25C (RM C299C)
26	
27	
28	

LP-25A RM 299C

1	LTS RM 299C, 299E, 299F
2	TWIST-LOCK CEILING RECEPT RM 299D
3	LTS RM 299D
4	TWIST-LOCK CEILING RECEPT RM 299D
5	CLEAN ROOM CLIMATE CONTROL PANELS
6	SAME AS 2
7	UTILITY 4 PLEX RECEPT ABOVE RM 299E
8	SAME AS 6
9	SPARE
10	RECEPT RM 299E
11	RECEPT RM 299D WEST WALL
12	RECEPT RM 299D EAST & NORTH WALLS
13	TWIST-LOCK RECEPT ABOVE 299E
14	UTILITY 4 PLEX RECEPT ABOVE RM 299F
15	NC
16	RECEPT RM 299F
17	NC
18	NC

LP-25B C299D (SW)

1	SPARE
2	RECEPT RM 299P (SHIELD WALL)
3	SAME AS 1
4	RECEPT RM 299P (SHIELD WALL)
5	SAME AS 1
6	RECEPT RM 299P (SHIELD WALL)
7	WATER FILTER UNIT RM 299M
8	DISINFECTANT UNIT RM 299M
9	SAME AS 7
10	SAME AS 8
11	SAME AS 7
12	SAME AS 8
13	EXHAUST FAN RM 299O
14	RECEPT RM 299P (SHIELD WALL)
15	RECEPT RM 299O
16	RECEPT RM 299P
17	STEAM GENERATOR CONTROLS RM 299O
18	FFU-1, FFU-2 RM 299P & FFU-3 RM 299O
19	RECEPT RM 299P
20	FFU-4, FFU-5 RM 299N & FFU-6 RM 299M
21	RECEPT RM 299M, 299N, 299O
22	LTS RM 299M, 299N, 299O
23	RECEPT RM 299P (SHIELD WALL)
24	LTS RM 299P
25	STERILIZER WASTE PUMP RM 299M
26	PURIFIER WASTE PUMP RM 299M
27	RECEPT RM 299P
28	SPARE
29	RECEPT RM 299P
30	SPARE

LP-25C C299C

1	RECEPT RM 299G, 299H
2	LTS RM 299I, 299J
3	RECEPT RM 299H, 299I, 299J
4	LTS RM 299G, 299H, 299K, 299L
5	FFU RM 299G, 299H
6	STAGE-1 COOL UNIT (16,800 BTU), RMS 299G, H, I, J
7	FFU RMS 299I, 299J, RECEPT RM C299C
8	SAME AS 6
9	RECEPT RM 299J
10	RETURN FAN (RF) RMS 299G, H, I, J
11	ELECTRIC DUCT HEATER (EDH) RMS 299G, H, I, J
12	STAGE-2 COOL UNIT (21,400 BTU), RMS 299G, H, I, J
13	SAME AS 11
14	SAME AS 12
15	SAME AS 11
16	SAME AS 12
17	RECEPT OVERHEAD RM C299C
18	SUPPLY FAN (SF) RMS 299G, H, I, J
19	RECEPT RM 299L
20	SAME AS 18
21	VARIABLE AIR VALVES (VAV) #1-#9 & CONTROL DAMPERS #1-#3 RMS 299G, H, I, J
22	SAME AS 18
23	SPARE
24	AUTOMATIC SLIDING GLASS DOORS
25	POWER TO RM 299S (RECEPTS, LTS, FAN)
26	SPARE
27	RECEPT RM 299R
28	SPARE
29	RECEPT RM 299Q
30	SPARE

LP-26 RM 116

1	WP-11, WP-12, 1P-13
2	WP-7, WP-10, WP-14
3	SPARE
4	WP-21, WP-22, WP-24, WP-25
5	WP-9, JANITOR CLOSET
6	WP-26, WP-27, WP-23
7	WP-6 (A & B) CONTROL
8	WP-3 (A & B) CONTROL
9	WP-6 (A & B) POWER
10	WP-3 (A & B) POWER
11	WP-6 (A & B) POWER
12	WP-3 (A & B) POWER
13	NC
14	NC
15	NC
16	NC
17	NC
18	NC
19	NC
20	TWIST-LOCK RECEPT RM 116
21	NC
22	SAME AS 20
23	NC
24	SAME AS 20

LP-27 RM 101

1	WP-31, WP-32
2	SSP-1
3	SPARE
4	WP-4 (A & B) CONTROL
5	WP-4 (A & B) POWER
6	SPARE
7	WP-4 (A & B) POWER
8	SPARE
9	WP-5 (A & B)
10	NC
11	NC
12	NC
13	NC
14	NC
15	NC
16	NC
17	NC
18	NC
19	NC
20	NC
21	NC
22	NC
23	NC
24	NC

REV. NO	DESCRIPTION	DR. BY	DATE
20	Added drawings for North Office Addition (NOA). Redesignated sheet numbers. Added locations to title block. Misc. load changes.	THS	11/28/07
19	Converted to AutoCad. Added LP-25C, misc. updates.	THS	08/08/06
18	General update	MDR	08/08/05
17	Added 522 (3 of 3)	MDR	11/16/04

seal

2007

DATE: 12/15/00

DRAWN BY: TOM SEEGER

CHECKED BY:

ENGINEER: CPF

CODE:

REVISION NUMBER: 20

REVISION DATE: 11/28/07

FIGURE 2.3.c

UNIVERSITY OF MISSOURI-COLUMBIA FACILITIES OPERATIONS research reactor facility

ELECTRICAL DISTRIBUTION REACTOR/LABORATORY PANELS

MURR NUMBER: 522

SHEET 3 OF 5

LP-31 ENTRANCE RM 114

1	CHEM-PURE RMS
2	J. BOX IN SWITCH IN RM 113
3	WASTE STORAGE & FILTER RMS
4	HT EXHANGER RM 114 LTS
5	WEST PASSAGE LTS
6	HT EXHANGER RM 114 LTS
7	WEST PASSAGE LTS
8	BATTERY CHARGER REC.
9	WORK AREA RM 113 LTS
10	SAME AS 8
11	103 PASSAGE LTS
12	LTS IN TRENCH UNDER REACTOR
13	MECH RM 116, 119, 112 LTS
14	HEAT EXCHANGER RM 114 RECEPT
15	RECEPT IN RM 113 S. WALL
16	RECEPT WASTE STORAGE CHEM-PURE RMS
17	RECEPT IN MECH RM 119, RM 116, & DOOR ALARM
18	RECEPT IN 112 PASSAGE
19	RECEPT RM 113, 103, STAIRS, & Emerg LT
20	FAN, EXH. FAN 15 TUNNEL
21	WASTE TANK AMPLIFIERS, RM 114 DOOR LT, BELL, TUNNEL ALARM
22	LTS IN HOTCELL HC-03
23	RECEPT IN WASTE TANK
24	HOTCELL HC-03
25	LTS & RECEPT WASTE TANK & DI REGEN RM
26	HOTCELL HC-03
27	RECEPT RM 114 ENTRANCE & CT TUNNEL
28	HOTCELL HC-03
29	COMPACTOR
30	HOTCELL HC-03
31	COMPACTOR
32	HOTCELL HC-03
33	COMPACTOR
34	HOTCELL HC-03
35	RECEPT IN WASTE TANK RM
36	HOTCELL HC-03

LP-32 RM 105 (Seal trench)

1	AREA LTS RM 106
2	AREA LTS RM 105, RM 110
3	AREA LTS RM 106
4	AREA LTS RM 110
5	AREA LTS 106
6	AREA LTS 110
7	LTS RM 108, FAN RM 108
8	HOTCELL HC-01 HOIST
9	LTS ABOVE RM 111 LOFT AREA
10	SAME AS 8
11	SUMP PUMP ELEVATOR
12	SAME AS 8
13	COBALT 60 PUMP & LTS
14	RECEPT. RM 106
15	RECEPT. RM 110
16	SPARE
17	
18	ELEVATOR RECEPT.
19	RECEPT. RM 110
20	RECEPT. EXTERNAL HOTCELL HC-01
21	NC
22	LTS HOTCELL HC-01
23	NC
24	RECEPT. TOP HOTCELL HC-01
25	RECEPT. NO. 2 IN HOTCELL HC-01
26	EMERG LT RECEPT. RM 110
27	RECEPT 3 & 4 IN HOTCELL HC-01
28	NC
29	RECEPT. 5 & 6 IN HOTCELL HC-01
30	NC
31	SEAL TRENCH PUMP
32	P-TUBE BLOWER
33	SPARE
34	SAME AS 32
35	
36	SAME AS 32

LP-32A RM 105 (Hotcell)

1	RECEPT. ABOVE HC-01 CONTROL PANEL
2	POWER TO LP-32B SUB PANEL
3	SPARE
4	SAME AS 2
5	HOTCELL HC-01 DOOR
6	HOTCELL WORK TABLE (W. WALL)
7	SAME AS 5
8	HOTCELL HC-01 TRANSFER CART
9	SAME AS 5
10	HOTCELL HC-01 CLUTCH POWER
11	RECEPT. ON STAIRS
12	RECEPT. ON HOTCELL HC-01 W. WALL
13	RECEPT. ON STAIRS
14	HOTCELL HC-01 CAN OPENER
15	HOTCELL HC-01 WELDER
16	HOTCELL HC-01 TRASH DOOR
17	SAME AS 15
18	HOTCELL HC-01 WORK & HYD. CARTS
19	SPARE
20	RECEPT. HOTCELL HC-01 S. WALL

LP-32B RM 105 (Hotcell)

1	NA
2	HOTCELL HC-04
3	4-PLEX RM 106 SOUTH WALL
4	SPARE
5	

LP-33 RM 101 (Beamport, E)

1	LTS RM 101 NORTHEAST
2	SPARE
3	LTS RM 101 SOUTHEAST
4	ELEVATOR RECEPT. G-20.00
5	LTS RM 101 NORTHEAST
6	SPARE
7	LTS RM 101B
8	SPARE
9	LTS RM 101B
10	N. & S. EMERG LTS RM 101, LT RM 101N
11	LTS RM 101B
12	LTS RM 101 SOUTH WALL
13	LTS RM 101B
14	LTS ON BIOLOGICAL SHIELD
15	LTS RM 101A, LTS ABOVE FUEL VAULT, LTS RM 102
16	SPARE
17	RECEPT BIOLOGICAL SHIELD BP-D
18	SPARE
19	RECEPT. RM 101 SOUTH WALL
20	RECEPT. RM 101B WEST WALL
21	RECEPT. RM 101 WEST WALL
22	RECEPT. RM 101A, 101B N. & E. WALLS
23	RECEPT 101C
24	RECEPT. BIOLOGICAL SHIELD BP-D
25	TWISTLOCK RECEPT. RM 101 EAST WALL
26	SPARE
27	NC
28	RECEPT. RM 101 EAST WALL
29	TWISTLOCK RECEPT. RM 101 EAST WALL
30	RECEPT. E. WALL AT PANEL SOUTH
31	NC
32	RECEPT. RM 101 EAST WALL
33	RECEPT. FOR WELDER
34	NC
35	SAME AS 33
36	NC
37	208 BUSS AROUND BIOLOGICAL SHIELD
38	
39	SAME AS 37
40	
41	SAME AS 37

LP-33A RM 101B (Beamport, NE)

1	RECEPT. RM 101A NORTH WALL
2	SPARE
3	LP-33B
4	SAME AS 2
5	SAME AS 3
6	POWER STRIP RM 101A SOUTH WALL RECEPT. RM 101A S. WALL & RECEPT. RM 102
7	SAME AS 3
8	RECEPT. EAST WALL RM 101B
9	220V 3 PHASE RECEPT. COLUMN RM 101B
10	4 PLEX RECEPT. RM 101B S.E. COLUMN E. SIDE
11	SAME AS 9
12	RECEPT RM 101B S.E. COLUMN W. SIDE
13	SAME AS 9
14	SPARE
15	4 PLEX RM 101B S.W. COLUMN N. SIDE
16	SAME AS 14
17	RECEPT. SW COLUMN NORTH SIDE
18	SAME AS 14

LP-33B Rm 101B (Beamport, NW)

1	4 PLEX E. CENTER COLUMN
2	SPARE
3	RECEPT. WEST COLUMN
4	SAME AS 2
5	HOT WATER HEATER
6	SAME AS 2
7	SAME AS 5
8	
9	
10	
11	
12	
13	
14	
15	
16	
17	

LP-34 BASEMENT (Northeast)

1	LTS RM 111 INTERIOR
2	RECEPT.
3	BARRIOR BOX RECEPT.
4	RECEPT.
5	HOOD RECEPT.
6	SPARE
7	EVAPORATOR
8	RECEPT. ABOVE LP-34
9	SAME AS 7
10	RM 111 SUPPLY FAN
11	RM 111 CONDENSOR
12	SAME AS 10
13	SAME AS 11
14	SAME AS 10
15	SAME AS 11
16	FAN CONTROL CIRCUIT
17	SPARE
18	SPARE
19	SAME AS 17
20	SPARE
21	SPARE
22	SAME AS 20
23	SAME AS 21
24	RECEPT. EAST WALL RM 110
25	SPARE
26	HOTCELL HC-02b
27	SAME AS 25
28	SAME AS 26
29	NC
30	SAME AS 26
31	NC
32	HOTCELL HC-02a
33	NC
34	SAME AS 32
35	NC
36	SAME AS 32

LP-35 RM 101(Beamport, S)

1	TWISTLOCK RECEPT. BP-C BIO-SHIELD
2	SPARE
3	SPARE
4	SPARE
5	TRIAX CAB. HANGING TWISTLOCK RECEPT.
6	TRIAX CAB. HANGING TWISTLOCK RECEPT.
7	RECEPT. RM 101 EAST WALL
8	TRIAX AMP RACK RECEPT. BIO-SHIELD
9	TRIAX AMP RACK RECEPT. BIO-SHIELD
10	TRIAX AMP RACK RECEPT. BIO-SHIELD
11	RECEPT. RM 402A
12	RECEPT. RM 101B COLUMN EAST SIDE
13	RECEPT. RM 402B
14	LP-35A (COSTAR TOWER)
15	RECEPT. RM 405A, 405B
16	SAME AS 14
17	NC
18	SAME AS 14
19	RECEPT. RM 404

LP-35A COSTAR TWR (Level 1)

1	LTS RM 101 (Lower basement south)
2	RECEPT. RM 398A (lower level 3 north)
3	RECEPT. RM 101 (lower basement south)
4	LTS RM 398A (lower level 3 north)
5	LTS RM 198A (lower level 1 south)
6	LTS RM 198 (lower level 1 north)
7	RECEPT. RM 198A (lower level 1 south)
8	RECEPT. RM 198 (lower level 1 north)
9	RECEPT. RM 101 (lower basement north)
10	LTS RM 101 (lower basement north)
11	RECEPT. RM 298A (lower level 2 south)
12	RECEPT. RM 298 (lower level 2 north)
13	LTS RM 298A (lower level 2 south)
14	LTS RM 298 (lower level 2 north)
15	NC
16	LTS MOUNTED EXTERNAL 298 & 298A
17	NC
18	NC

LP-41 COOLING TOWER

1	LTS FOR NORTH TOWER BASIN, DAY TANK LT, BASIN SUMP FLOAT LIGHT
2	LTS CT GRADE LEVEL EQUIPMENT RM
3	LTS IN TUNNEL
4	LTS IN ELECTRICAL RM
5	LTS IN TUNNEL
6	BASEMENT LTS
7	LTS IN TOWER BASIN SOUTH
8	LOW SUMP CUTOUT CKT * DO NOT SECURE WHILE REACTOR IS OPERATING (SEE MURR 2617)
9	RECEPT. GRADE LEVEL, EXHAUST FAN
10	HOT WATER CIRC. PUMP
11	ACID SHOWER HEAT TRACE & ALARM
12	YARD LTS.
13	RECEPT GRADE LEVEL
14	YARD LTS.
15	SPARE
16	YARD LTS.
17	STEAM THERMOSTAT, SEC. CHEMICAL ADDITION CONTROL PANEL
18	BASEMENT RECEPT.
19	WEST SUMP PUMP
20	EAST SUMP PUMP
21	SAME AS 19
22	SAME AS 20
23	SUMP CONTROL CIRCUIT
24	SOUTH PARKING LOT PHOTOCELL CONTROL
25	ACID TANK CONTROL PANEL
26	PHICOND INSTRUMENTS, SEC. MAKEUP, ACID ADDITION, BACK-UP VALVE, BLOWDOWN
27	SPARE
28	SPARE
29	SPARE
30	SPARE
31	CT AIR CONDITIONING UNIT
32	SPARE
33	SAME AS 31
34	SPARE
35	SAME AS 31
36	SPARE
37	SPARE
38	SPARE

LP-42 INNER PASSAGE (273)

1	AIR HANDLER & REHEAT COILS RM 258/259
2	AC-1 RM 260 AC
3	SAME AS 1
4	SAME AS 2
5	SAME AS 1
6	SAME AS 2
7	CONDENSING UNIT RM 258/259
8	AC-2 BASEMENT AC
9	SAME AS 7
10	SAME AS 8
11	SAME AS 7
12	SAME AS 8
13	CEILING AIR FILTER RM 258/259
14	HOTCELL BLOWER
15	AIR HANDLER RM 260
16	SAME AS 14
17	NC
18	SAME AS 14
19	NC
20	NC
21	NC
22	NC
23	NC
24	NC

LP-43 INNER PASSAGE (273E)

1	RECEPT. RM 255
2	RECEPT. RM 259
3	RECEPT. RM 255
4	SAME AS 2
5	RECEPT. RM 255
6	RECEPT. RM 259
7	SPARE
8	RECEPT. RM 259
9	RECEPT. RM 255
10	RECEPT. RM 259
11	RECEPT. RM 255
12	SAME AS 10
13	RECEPT. RM 255
14	NC
15	SAME AS 13
16	RECEPTL RM 259
17	RECEPT. RM 255
18	SAME AS 16
19	NC
20	NC
21	RECEPT. RM 257
22	RECEPT. RM 259
23	RECEPT. 257
24	SAME AS 22
25	RECEPT. RM 257
26	RECEPT. RM 260
27	SAME AS 25
28	RECEPT. RM 260
29	RECEPT. RM 257
30	RECEPT. RM 260
31	NC
32	NC
33	NC
34	NC
35	NC
36	NC

LP-44 INNER PASSAGE (278E)

1	RECEPT. RM 247
2	RECEPT. RM 246
3	RECEPT. RM 247
4	RECEPT. RM 246
5	LP-44B
6	SAME AS 4
7	SAME AS 5
8	NC
9	RECEPT. MECH RM 278
10	RECEPT. MECH RM 278
11	NC
12	SAME AS 10
13	NC
14	NC
15	NC
16	NC
17	RECEPT MECH RM 278
18	RECEPT RM 246
19	SAME AS 17
20	SAME AS 18
21	NC
22	NC
23	NC
24	NC
25	NC
26	NC
27	NC
28	NC
29	NC
30	NC
31	NC
32	NC
33	NC
34	NC
35	NC
36	LP-44A
37	NC
38	SAME AS 36
39	NC
40	SAME AS 36
41	
42	RECEPT. MECH RM 276

LP-44A INNER PASSAGE (278)

1	RECEPT. RM 245
2	RECEPT. RM 244
3	RECEPT. RM 245
4	RECEPT. RM 244
5	RECEPT. RM 245
6	SPARE
7	RECEPT RM 245
8	SPARE
9	220V RECEPT RM 245
10	NC
11	SAME AS 9
12	NC
13	220V RECEPT RM 245
14	NC
15	SAME AS 13
16	NC
17	220V RECEPT RM 245
18	SPARE
19	SAME AS 17
20	RECEPT. RM 242A
21	NC
22	RECEPT. RM 242C
23	NC
24	RECEPT. RM 242A, 242C, FILTER UNIT, HOOD

LP-44B INNER PASSAGE (278)

1	NO BREAKER POSITION #1
2	RECEPT. RM 247
3	SAME AS 2
4	RECEPT. RM 247
5	SAME AS 4
6	RECEPT. RM 247
7	SAME AS 6
8	NC

ELP INNER PASSAGE (280)

1	SOUTH LTS, SOUTH PORTAL, HOTCELL VALVE POSITION INDICATION
2	ELP-1
3	OUTER CORRIDOR EXIT LTS. (N,S,E,W)
4	SAME AS 2
5	LTS. E. CORRIDOR, LOBBY, WOMEN'S RM & MEN'S LOCKER RM.
6	FAN FAILURE ALARM
7	CONTAINMENT STAIRWAY LTS.
8	NORTH & WEST OUTER CORRIDOR LTS.
9	CONTAINMENT EXIT LTS & DOOR COMBINATION
10	MECHANICAL RM WEST
11	RECEPT. 289, INTERCOM, EAST LAB PROXY SYSTEM
12	EXIT LTS MECHANICAL RM WEST.
13	PING-1A STACK MONITOR
14	ELP MOUNTED RECEPT.
15	EVACUATION & ISOLATION
16	NUCLEPORE
17	
18	EVACUATION & ISOLATION 16" VALVE COMPRESSOR

ELP-1 INNER PASSAGE (280)

1	RECEPT. RM 216
2	ELP-1A RM 231B
3	RECEPT. RM 213
4	RECEPT. RM 212
5	SPARE
6	SPARE
7	RECEPT. RM 218
8	
9	
10	
11	
12	

ELP-1A RM 231B

1	SPARE
2	SPARE
3	RECEPT. RM 231 NORTH WALL
4	SPARE
5	SPARE

ELP-2 INNER PASSAGE (273)

1	RECEPT. RM 235 (SERVER-30A)
2	ELP-2A
3	RECEPT. RM 234, 235, WEST LAB PROXY SYSTEM
4	SAME AS 2
5	NITROGEN BANKS
6	RECEPT. RM 260 (x4)
7	CAMPUS META SYSTEM
8	FACP, SNAC-4, FIRE PROTECTION CONTROL
9	ENTRY GATE & CARD READER CONTROL POWER
10	FIRE PROTECTION DAMPERS
11	SAME AS 9
12	RECEPT. RM 260 (SERVER-30A)
13	RECEPT. BELOW ELP-2
14	
15	
16	
17	
18	

ELP-2A BASEMENT (E)

1	NA
2	RM 111 ENTRY LTS & ELECT. DOOR STRIKE
3	RM 111 BATTERY BACKUP EMERG. LTS
4	FIRE ALARM
5	LTS ABOVE DOOR FOR RM 101
6	RECEPT. BELOW ELP-2A
7	
8	
9	
10	
11	
12	

DG RM PANEL RM 231E

1	RECEPT. OUTSIDE RM 231D BEHIND 10 X 10 WALL COVER PLATE
2	RECEPT. RM 231E
3	LTS RM 231D, RECEPT. RM 231D
4	RECEPT. RM 231E
5	CONTROL PANEL
6	LTS RM 231E
7	CONTROL PANEL
8	SPARE
9	BATTERY CHARGER
10	SAME AS 8
11	HEAT DETECTOR
12	SAME AS 8
13	BLOCK HEATER
14	EMERG PWR LTS RM 299D, 299E, 299F, 299N, 299P, CARD READER RM 299B
15	SPARE
16	
17	
18	
19	
20	

UPS-1 INNER PASSAGE (280)

1	SPARE
2	UPS-2
3	SPARE
4	SAME AS 2
5	SPARE
6	NMC STACK MONITOR
7	SPARE
8	SPARE
9	SPARE
10	SPARE
11	SPARE
12	SPARE
13	NC
14	SPARE
15	NC
16	SPARE
17	NC
18	SAME AS 16
19	NC
20	NC
21	NC
22	NC
23	NC
24	NC
25	NC
26	NC
27	NC
28	NC
29	Undetermined Load
30	SPARE

UPS-2 RM 302A

1	RECEPT RM 302
2	ANNUNCIATOR (115VAC)
3	SPARE
4	SPARE
5	VENT FANS & CLOCK
6	BACK-UP DOORS
7	115VAC RECEPT.
8	ELEVATOR ALARM SYSTEM
9	PROCESS INSTRUMENTATION
10	UPS-3
11	AIMS CABINET
12	AUX. ANNUNCIATOR PANEL
13	RAD SUMPS AND DRAINS ALARM PANEL
14	4-PLEX RECEPT. W. MEZZANINE

UPS-3 RM 302A

1	AREA RADIATION MONITORS
2	2PS1, 2PS2 & SERVO AMPLIFIER
3	ANNUNCIATOR (24VDC) 2PS3
4	RECORDERS (NI, Stack Monitor, Scaler), SP1 & SP2
5	ROD CONTROL
6	SPARE
7	SPARE
8	GAMMA METRICS NI AMPLIFIERS

REV. NO	DESCRIPTION	DR. BY	DATE
5	Added West Lab bldg. Proxy expansion to ELP-2, Bkr 3; Added East Lab Proxy system to ELP, Bkr 11; Added Rm 102 lights to LP-33, Bkr 15.	THS	08/04/08
4	Added drawings for North Office Addition (NOA). Redesignated sheet numbers. Added locations to title block. Misc. load changes.	THS	11/28/07
3	Converted to AutoCad, misc. load changes.	THS	8/03/06
2	General Update.	MDR	8/8/05
1	Combined LP-41/LP-41A into Single LP.	MDR	1/31/05
0	Added 522 (3 of 3)	MDR	11/16/04

seal 2008

DATE: 12/15/00
DRAWN BY: TOM SEEGER
CHECKED BY: ENGINEER: LPF
CODE:
REVISION NUMBER: 5
REVISION DATE: 08/04/08

FIGURE 2.3.d

UNIVERSITY OF MISSOURI-COLUMBIA
FACILITIES OPERATIONS
research reactor facility

ELECTRICAL DISTRIBUTION
REACTOR/LABORATORY PANELS-2

MURR NUMBER: 522
SHEET 4 OF 5

FIGURE 2.3.e

1LP1	CYCLOTRON EQUIP: ROOM 2009		
1	RECEPT. 2006 (E; W & S), 2007 (E & S)	2	RECEPT. 2008C (W)
3	RECEPT. 2008 (E & W), 2008A (N & W), 2007 (W)	4	RECEPT. 2008 (N, E, & W)
5	RECEPT. 2008A (N)	6	VAV'S
7	RECEPT. 2008A (N)	8	2008A FUME HOOD (E), 2008C (E)
9	RECEPT. 2008A (N)	10	RECEPT. 2008 (S)
11	RECEPT. 2008A (N)	12	2008 AUTOMATIC DOOR
13	RECEPT. 2009A (E, S & W)	14	2008 AUTOMATIC DOOR
15	RECEPT. 2009	16	LIGHTING 2009A
17	RECEPT. 2008A (S & E)	18	SPARE
19	2008A HOTCELL & FOURPLEX ON TOP.	20	SPARE
21	JUNCTION BOX IN CEILING OF 2008A	22	CUSTOMER INTERFACE BOX
23	JUNCTION BOX IN CEILING OF 2008A	24	SPARE
25	JUNCTION BOX IN CEILING OF 2008A	26	SPARE
27	JUNCTION BOX IN CEILING OF 2008A	28	OVEN IN 2008C
29	JUNCTION BOX IN CEILING OF 2008A	30	SAME AS 28
31	JUNCTION BOX IN CEILING OF 2008A	32	JUNCTION BOX IN CEILING OF 2008A
33	SAME AS 31	34	SAME AS 32
35	SAME AS 31	36	SAME AS 32
37	JUNCTION BOX IN CEILING OF 2008A	38	2008A HOT/DISPENSING CELL
39	SAME AS 37	40	SAME AS 38
41	SAME AS 37	42	SAME AS 38

1LP3	ELECTRICAL CLOSET 2016		
1	RECEPT. 2015-02 (E)	2	RECEPT. 2015-52, 2015-53, 2015-64, 2015-65
3	RECEPT. 2015-02 (E)	4	RECEPT. 2015-52, 2015-53, 2015-64, 2015-65
5	RECEPT. 2015-02 (E)	6	RECEPT. 2015-50, 2015-51, 2015-66, 2015-67
7	RECEPT. 2015-02 (W)	8	RECEPT. 2015-50, 2015-51, 2015-66, 2015-67
9	RECEPT. 2015-43 (W)	10	RECEPT. 2015-28, 2015-29, 2015-44, 2015-45
11	RECEPT. 2015-43 (W)	12	RECEPT. 2015-9, 2015-30, 2015-44, 2015-46
13	RECEPT. 2015-43 (W)	14	RECEPT. 2015-28, 2015-30, 2015-45, 2015-46
15	SPARE	16	RECEPT. 2015-25, 2015-27, 2015-48, 2015-49
17	BLANK	18	RECEPT. 2015-25, 2015-26, 2015-47, 2015-48
19	SPARE	20	RECEPT. 2015-26, 2015-27, 2015-47, 2015-49
21	SPARE	22	RECEPT. 2015-9, 2015-10, 2015-21, 2015-22
23	SPARE	24	RECEPT. 2015-9, 2015-10, 2015-21, 2015-22
25	SPARE	26	RECEPT. 2015-7, 2015-8, 2015-23, 2015-24
27	RECEPT. OFFICE 2025 (E & S)	28	RECEPT. 2015-7, 2015-8, 2015-23, 2015-24
29	RECEPT. OFFICE 2024, OFFICE 2025 (N & W)	30	RECEPT. 2015-3, 2015-4, 2015-5, 2015-6
31	RECEPT. OFFICE 2023, OFFICE 2022 (E & S)	32	RECEPT. 2015-3, 2015-4, 2015-5, 2015-6
33	RECEPT. OFFICE 2021, OFFICE 2022 (N & W)	34	SPARE
35	RECEPT. OFFICE 2020, OFFICE 2019 (E & S)	36	SPARE
37	RECEPT. OFFICE 2018, OFFICE 2019 (N & E)	38	CARD ACCESS PANEL
39	RECEPT. OFFICE 2017	40	CUH-1 & 2
41	RECEPT. CONF. RM. 2017 REFRIGERATOR	42	RECEPT. CONF. RM. 2017 ICE MAKER

1LP4	ELECTRICAL CLOSET 2016		
1	RECEPT., CLASSROOM 2014 (N & E)	2	RECEPT. 2015-54, 2015-55, 2015-62, 2015-63
3	RECEPT. CLASSROOMS 2014 (W), 2014A (E & W), 2014B (E)	4	RECEPT. 2015-54, 2015-55, 2015-62, 2015-63
5	RECEPT. CLASSROOM 2014B (E & S)	6	RECEPT. 2015-56, 2015-57, 2015-58, 2015-60, 2015-61
7	CONF. RM 2036 (E & W)	8	RECEPT. 2015-56, 2015-59, 2015-60
9	RECEPT. OFFICE 2033 (W & S), OFFICE 2034 , CONFERENCE ROOM (S)	10	RECEPT. 2015-57, 2015-58, 2015-59, 2015-61
11	RECEPT. OFFICE 2033 (N & E), OFFICE 2032	12	RECEPT. 2015-34, 2015-36, 2015-38, 2015-39
13	RECEPT. OFFICE 2030 (N & W), OFFICE 2031	14	RECEPT. 2015-34, 2015-35, 2015-37, 2015-38
15	RECEPT. OFFICE 2030 (E & S), OFFICE 2029	16	RECEPT. 2015-35, 2015-36, 2015-37, 2015-39
17	RECEPT. OFFICE 2027 (N & W), OFFICE 2028	18	RECEPT. 2015-31, 2015-33, 2015-41, 2015-42
19	RECEPT. OFFICE 2027 (E & W), OFFICE 2026	20	RECEPT. 2015-31, 2015-32, 2015-41, 2015-42
21	RECEPT. ROOM 2015 (E, S, & W)	22	RECEPT. 2015-31, 2015-33, 2015-40, 2015-41
23	SERVICE RECEPT. AND LIGHT PIT 2035A	24	RECEPT. 2015-13, 2015-15, 2015-16, 2015-17, 2015-18
25	SPARE	26	RECEPT. 2015-131, 2015-14, 2015-16, 2015-17
27	SPARE	28	RECEPT. 2015-14, 2015-15, 2015-18
29	SPARE	30	RECEPT. 2015-11, 2015-12, 2015-19, 2015-20
31	SPARE	32	RECEPT. 2015-11, 2015-12, 2015-19, 2015-20
33	SPARE	34	RECEPT., CLASSROOMS 2014/ 2014A (E)
35	SPARE	36	SPARE
37	SPARE	38	SPARE
39	RECEPT. CLASSROOMS 2014/2014B (E)	40	VAV's
41	RECEPT. OVERHEAD PROJECTORS ROOMS 2014, 2014A, 2014B	42	VAV's

2LP2 (Sect. 1)	NE ROOM 3000		
1	RECEPT. C2002 (W) RADIATION MONITOR	2	RECEPT. 2011 (E & W)
3	RECEPT. C2002 (E) RADIATION MONITOR	4	RECEPT. 2011 ICE MACHINE, SWITCH GARGAGE DISPOSAL
5	RECEPT. C2002, C2000, DRINKING FOUNTAIN	6	RECEPT. 2011 WEST MICROWAVE
7	RECEPT. 2005 (N, E, S, W), 2004, 2002	8	RECEPT. 2011 EAST MICROWAVE
9	MONITORING EQUIPMENT	10	RECEPT. 2011 (S) ABOVE COUNTERTOP
11	SPARE	12	RECEPT. 2011 WEST REFRIGERATOR
13	RECEPT. 2041 (N)	14	RECEPT. 2011 EAST REFRIGERATOR RECEPT. GFCI'S ABOVE COUNTERTOP
15	RECEPT. 2041 (E, S, W), C2001 (E)	16	RECEPT. 2040 DATA RACK (N)
17	RECEPT. 2010 (E, W, S), 2041 (W), C2001 (E)	18	RECEPT. 2040 DATA RACK (CN)
19	RECEPT. 3000 (SE, SW), 1000 (E, N)	20	RECEPT. 2040 DATA RACK (CS)
21	RECEPT. 3000 (E, NE, NW)	22	RECEPT. 2040 DATA RACK (S)
23	HWCP-1	24	RECEPT. OUTSIDE BUILDING (S, E, W)
25	RECEPT. 2035 (W)	26	DEIONIZERS
27	RECEPT. 2035 (N, W)	28	SAME AS 26
29	RECEPT. 2035 (S, W)	30	SPARE

2LP2 (Sect. 2)	NE ROOM 3000		
31	RECEPT. 2035 (S)	32	SPARE
33	RECEPT. 2035 (E)	34	RECEPT. 2040 (E, W)
35	FAX MACHINE	36	RECEPT. 2040 (S)
37	RECEPT.	38	SPARE
39	COPIER	40	DUPLEX WATER SOFTENER
41	210 FIRE ALARM PANEL	42	SPARE
43	EF-3	44	PANEL 2LP2A
45	EF-4	46	SAME AS 44
47	LIGHTING CONTACTOR CONTROL	48	SAME AS 44
49	CONTROL PANELS	50	RECEPT. C2001 (E) RADIATION MONITOR
51	CHILL WATER FLOW METER	52	J BOX IN C2001
53	BUILDING MANAGEMENT SYSTEM	54	SPARE
55	RECEPT. C2002 (CE) RADIATION MONITOR	56	RECEPT. 201, V2101A, V210B
57	RECEPT. C2002 (CW) RADIATION MONITOR	58	RECEPT. 201 RECEPTIONISTS DESK
59	SPARE	60	RECEPT. C2000 (S), S2006

2LP2A	RM 1000 (BASEMENT)		
1		2	
3		4	
5		6	
7		8	
9		10	
11		12	
13		14	
15		16	
17		18	
19		20	
21		22	
23		24	
25		26	
27		28	
29		30	

This panel is not currently installed.

1HL2	CYCLOTRON EQUIP. RM 2009		
1	LIGHTS: ROOMS 2009, 2008, 2008A, 2008C, 2008C-1	2	HEPA FILTERS
3	LIGHTS: ROOMS 2008, 2008A	4	HEPA FILTERS
5	SPARE	6	HEPA FILTERS
7	SPARE	8	HEPA FILTERS
9	SPARE	10	HEPA FILTERS
11	SPARE	12	HEPA FILTERS
13	SPARE	14	SPARE
15	SPARE	16	SPARE
17	SPARE	18	SPARE
19	SPARE	20	SPARE
21	SPARE	22	SPARE
23	SPARE	24	SPARE
25	SPARE	26	SPARE
27	SPARE	28	SPARE
29	SPARE	30	SPARE

2HL1	NW ROOM 3000		
1	LIGHTS: 1000, ROOMS 2007, 2040, C2000, C2002, 3000	2	LIGHTS: C2001, C2002 ROOM 2040
3	LIGHTS: ROOMS 2001, 2002, 2003, 2004, 2005, 2006, 2007	4	LIGHTS: C2000, S2006, 1000, ROOMS 2041, 210, VESTIBULES V210A, V210B, 3000
5	SPARE	6	EMERGENCY LIGHTING
7	LIGHTS: C2001, ROOMS 2041, 2010,	8	LIGHTS:
9	LIGHTS: ROOMS 2041,	10	LIGHTS: ROOM 210,
11	LIGHTS: ROOMS 2011, 2012, 2014, 2014A, 2014B, 2017, 2018, 2019, 2020, 2021, 2022, 2023, 2024, 2025	12	LIGHTS: OUTSIDE (E, N, S, W)
13	LIGHTS: ROOMS 2026, 2027, 2028, 2029, 2030, 2031, 2032, 2033, 2034, 2035, 2036	14	LIGHTS: COURTYARD, CANOPY, OUTSIDE (W)
15	LIGHTS: ROOM 2015 (S)	16	SPARE
17	LIGHTS: ROOM 2015 (N)	18	SPARE
19	SPARE	20	SPARE
21	SPARE	22	SPARE
23	SPARE	24	SPARE
25	SPARE	26	SPARE
27	SPARE	28	SPARE
29	SPARE	30	SPARE

DESIGNATIONS:

1000	BASEMENT
2XXX	GRADE LEVEL
3000	UPPER LEVEL
S	STAIRWELL
C	CORRIDOR
V	VESTIBULE

REV. NO	DESCRIPTION	DR. BY	DATE
0	Added drawings for North Office Addition (NOA). Redesignated sheet numbers. Added locations to title block.	THS	11/28/07

seal

2007

DATE: 11/28/07

DRAWN BY: TOM SEEGER

CHECKED BY: ENGINEER: CPF

CODE:

REVISION NUMBER: 0

REVISION DATE: 11/28/07

FIGURE 2.3.e

UNIVERSITY OF MISSOURI-COLUMBIA

FACILITIES OPERATIONS

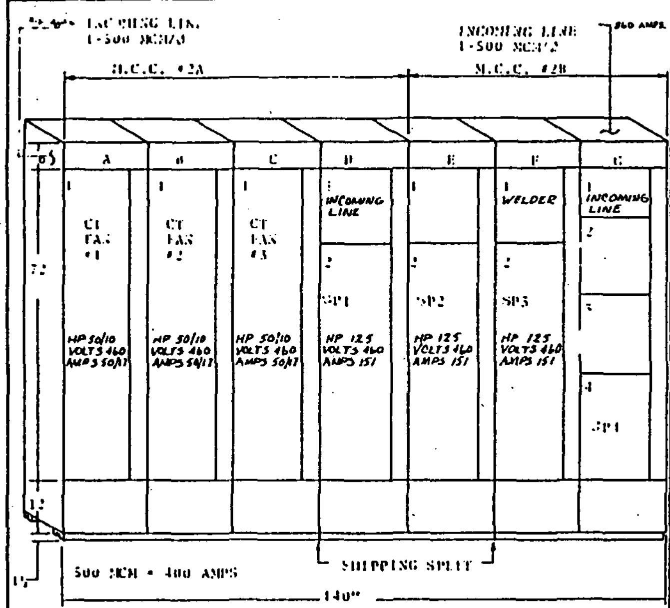
research reactor facility

ELECTRICAL DISTRIBUTION

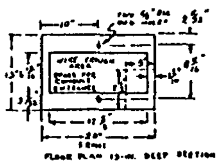
NORTH OFFICE ADDITION PANELS

MURR NUMBER: 522

SHEET 5 OF 5



Unit No.	Description	Size	Motor H.P.	Circuit Breaker Fr.	Circuit Breaker Trip	Heater CR123C CR123F*	PB Start	PB Stop	Sel. Sw.	PL R	PL G	Aux. Int. A	Aux. Int. B	Total Int.	Connection Diagram	Nameplate Marking 1st Line	2nd Line	3rd Line	Mfg Code
A1	2S-1W(REV)C/B COMB	3	30	EF	70	*48.7B	2		2	1	1			4	P7	FAN	COOLING TOWER #1		YA
B1		3	30	EF	70	*48.7B	2		2	1	1			4	P7	FAN	COOLING TOWER #2		YA
C1	6X SPACE	3	40	EF	70	*48.7B	2		2	1	1			4	P7	FAN	COOLING TOWER #3		YA
D1	1½ X SPACE																		
D2	2S-PW C/B COMB.	4PW	125	EJ	225	*91.4B			1	1		2	2	4	P8	COOLING TOWER	WATER PUMP P-1		YB
E1	1½ X SPACE																		
E2	2S-PW C/B COMB.	4PW	125	EJ	225	*91.4B			1	1		2	2	4	P8	COOLING TOWER	WATER PUMP P-2		YB
F1	1½ X SPACE															WELDER			
F2	4½ X SPACE	4PW	125	EJ	225	*91.4B			1	1		2	2	4	P8	COOLING TOWER	WATER PUMP P-3		YB
G1	INC. LINE LUGS															INCOMING LINE			
G2	FVNR C/B COMB.	1	1	EF	15	2.20A			1	1		2	2	4	P4	SPARE			
G3		1	1	EF	15	2.20A			1	1		2	2	4	P4	SPARE			
G4		3	30	EF	70	*48.7B			1	1		2	2	4	P4	COOLING TOWER	WATER PUMP P-4		



NOTES:

1. VERTICAL BUS ASSEMBLY SHALL BE RATED 600A.
2. UNITS A1 AND B2 SHALL BE EQUIPPED WITH 2-4 POLE RELAY #261-A14AT2.
3. UNITS D2, E2, G2, G3 AND G4 SHALL BE EQUIPPED WITH A SINGLE POLE RELAY #CR120E01002.
4. SELECTOR SWITCHES SHALL BE MARKED ON-OFF-AUTO.

Figure 2.4

SPECIFICATIONS

FINISH - LIGHT GRAY (ASA-61) WITH DARK GRAY (ASA-33) EDGING

NEMA CLASSIFICATION:
CLASS I TYPE B WIRING, NEMA ______ ENCLOSURE

POWER SUPPLY:
3 PHASE 3 WIRE 480 VOLTS 60 CYCLE

CONTROL:
SOURCE INDIV. C.P.T. 120 VOLTS 60 CYCLE

UNIT IDENTIFICATION: ENGRAVED NAMEPLATES

HORIZONTAL MAIN BUS RATED 600 AMPS
MAXIMUM 600 VOLTS AND BRACED FOR 25,000 AMPS

VERTICAL BUS RATED 600 AMPS, MAXIMUM 600 VOLTS AND BRACED FOR 25,000 AMPS RMS ASYMMETRICAL.

REVISIONS

3	MS/EFH	1,4,5-'64	5	[illegible]	[illegible]-7-72
REV PER CUST. MKD. DWG					
4	Vicki Snodgrass	3-15-72	6	D.C.N.	4-28-95
ADDITION OF 10 NG W EQUIP			REMOVED: CHEM FEED PUMPS		

TITLE

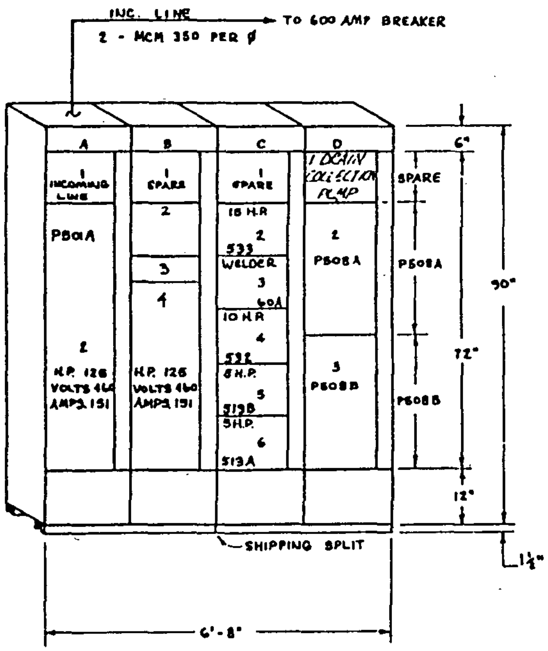
MOTOR CONTROL CENTER #2

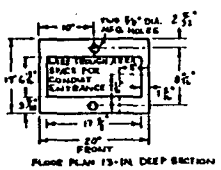
RESEARCH REACTOR FACILITY
UNIVERSITY OF MISSOURI

DRAWN BY Vicki Snodgrass APPROVED [illegible]

SHEET NO. 2 OF 6 DWG. NO. 2

1558





UNIT NO.	DESCRIPTION	SIZE	MOTOR H.P.	CIRCUIT BREAKER		HEATER CR123C CR123P*	PB		SEL. SW.	PL		AUX. INT.		TOTAL INT.	CONNECTION DIAGRAM	NAMEPLATE MARKINGS			MFG. CODE
				FR.	TRIP		START	STOP		R	G	A	B			1st LINE	2nd LINE	3rd LINE	
A1	MAIN LUG COMPT.																		
A2	P501A	4	125	JJ	225	*84.88			1	1		1		1	1022	P 501			
B1	SPARE															SPARE			
B2	SPARE																		
C1	SPARE																		
C2																			
C3	WELDER				60											WELDER			
C4	P532	1	10	EF	40	16.38			1	1		1		1	41	P 532			
C5	P513B	1	5	EF	15	8.67A			1	1		1		1	41	P 513B			
C6	P513A	1	5	EF	15	8.67A			1	1		1		1	41	P 513A			
D1	DRAIN COLLECTION PUMP																		
D2	P508A		60	EF	100										1076	P508A			
D3	P508B		60	EF	100										1076	P508B			
B3	VALVE OF AIRWAY		7.5	EF	20											VALVE OF AIR COMP			
B4	P501D	4	125	JJ	225	*84.88			1	1		1		1	1022				

NOTES:

1. PART WINDING STARTERS TO HAVE 500VA. CONTROL TRANSFORMERS.

Figure 2.5

SPECIFICATIONS

FINISH - LIGHT GRAY (ASA-61) WITH DARK GRAY (ASA-33) EDGING

NEMA CLASSIFICATION:
CLASS I TYPE B WIRING, NEMA I ENCLOSURE

POWER SUPPLY:
3 PHASE 3 WIRE 480 VOLTS 60 CYCLE

CONTROL:
SOURCE INDIV. C.P.T. 120 VOLTS 60 CYCLE

UNIT IDENTIFICATION: CARDHOLDERS

HORIZONTAL MAIN BUS RATED 600 AMPS
MAXIMUM 600 VOLTS AND BRACED FOR 25,000 AMPS

VERTICAL BUS RATED 300 AMPS, MAXIMUM 600 VOLTS AND BRACED FOR 25,000 AMPS RMS ASYMMETRICAL

REVISIONS

1	Vickie Snodgrass	3-16-72	3	D WILLIAMS	3-18-76
	ADDITION OF 10MW EQUIP.			CHANGED D1	
2	Vickie Snodgrass	1-24-73	4	D.G.H.	1-28-75
	AS INSTALLED			VALVE OF AIRWAY	

TITLE

MOTOR CONTROL CENTER #5

RESEARCH REACTOR FACILITY
UNIVERSITY OF MISSOURI

DRAWN BY Vickie Snodgrass APP

SHEET NO. 1 OF 2 DWG. NO. #1

2.5 Instrumentation

2.5.1 Introduction

The instrumentation, control and safety systems are interconnected to provide reactor surveillance and control.

Detectors provide an indication of neutron flux activity. This flux or radiation signal is routed to the appropriate monitor which provides a readout of the flux level, period or radiation activity.

In addition to supplying the operator with information, the monitors also are interconnected with an alarm, and/or reactor safeguard equipment to automatically alert the operator to excessive nuclear activity, stop rod withdrawal or, if required, run in the rods or scram the reactor.

A process instrumentation system is also provided to give the operator an indication of the level of reactor processes other than nuclear functions. Process transducers provide a signal to the temperature and flow instrumentation which supplies direct readout and recorded data of water temperatures, pressure, and flow. These data are also used to automatically control the water systems through valve interlock instrumentation. The valve instrumentation also provides a position indication for the process valves.

The reactor control system is a relay and switch logic system used to prohibit accidental or incorrect operation which might result in an unsafe condition.

2.5.2 Power Range Monitors

2.5.2.1 Wide Range Monitor, Channel 4 (see Figure 2.6, Dwg. No. 965-R3). When the range switch is in the 10 MW position, relay K2 will be energized, placing R2 in the feedback circuit. This will double the allowable input current from 5 MW (1×10^{-4} Ampere) to 10 MW (2×10^{-4} Ampere). Resistor R2 (1×10^{4} ohms) is to be changed in value to 1.58×10^{4} ohms, 0.1%. It will be a Dale Resistor, type RN65C.

The rod run-in and scram trip percent values to the safety system will not have to be changed when operating in either the 5 or 10 MW position. The power level interlock circuit (Figure 2.7) prevents operating the reactor with systems on line for 5 MW (Mode II) and the range switch in the 10 MW position.

2.5.2.2 Power Range Monitors, Channels 5 and 6 (see Figures 2.8 and 2.9). Power Range Monitors, Channels 5 and 6, are identical. Channel 5 changes will be described.

Refer to Test and Feedback Unit, Z14 on Figure 2.9.

The value of DC amperes approximate full scale current range is determined by feedback connections of Table 1.

Two test and feedback units for each channel 5 and 6 are to be used; one for 5 MW and one for 10 MW. Connections for 5 MW will be:

E1 to E2
E4 to E5
6X to 6Y
16X to 16Y
22X to 22Y,

placing unit in the 2.48×10^{-5} to 3.08×10^{-5} Ampere range using resistors R1F, R4F and capacitor C3. Input current is to be 2.5×10^{-5} Ampere for 5 MW.

Connections for 10 MW will be:

E1 to E2
E4 to E5
4X to 4Y
14X to 14Y
21X to 21Y,

placing unit in the 4.06×10^{-5} to 5.9×10^{-5} Ampere range using resistors R1D, R4D and capacitor C2. Input current is to be 5×10^{-5} Ampere for 10 MW.

Electrical interlock circuit will be changed to prohibit use of wrong test and feedback module when in the 5 or 10 MW mode.

The electrical interlock must be completed or an instrument anomaly occurs which in turn provides a scram input into the safety system.

Electrical interlock for .1 and 5 MW is as follows (Figure 2.8): J14-6, F14-5, 1S8-11, J14-11, J14-12 on the 5 MW test and feedback unit (Z-14).

Electrical interlock for 10 MW is as follows: J14-6, J14-5, 1S8-12, J14-10, J14-12 on the 10 MW test and feedback unit (Z-14).

For channel 6, switch 1S8-13 and 1S8-14 are used in identical manner. Switch 1S8 is the Power Level Selection Control Switch. The test and feedback units of channels 5 and 6 will be color coded and clearly labeled for 5 and 10 MW.

"2.5.2.3 Gamma-Metrics Power Range Monitors

NEW SECTION ADDED BY 2001 REV.

Gamma-Metrics power range channels do not have interchangeable Test and Feedback modules designed to create full scale power range meter deflection for two separate modes of reactor operation, Mode I (10 MW) and Mode II (5 MW). Because there is no possibility of having the wrong test and feedback module installed for either mode of operation, this power range channel does not require an electrical interlock."

2.10).

~~outlet~~

ature

~~temp~~ monitoring. These assemblies are identified as EP No. 980A, B, C and D; ~~., B, C and D,~~

~~EP/9~~ EP No. 990A, B, C and D."

2001

"Specifications for temperature assemblies 980A and B are as follows:

1. Output: 4-20 MADC
2. Range: 75-175 °F
3. Accuracy: ± 0.2% of span
4. Power requirements: 12-45 VDC

~~perature assemblies~~

2001

"Specifications for temperature assemblies 980C and D, and 990A, B, C, and D are as follows:"

2001

1. Accuracy: ± 0.1% of full scale.
2. Zero Shift: ±.05 microvolt/°F maximum.
3. Span Shift: ± .001%/°F.
4. Cold Junction Error: ± 1°F maximum deviation from National Bureau of Standards.

5. Stray Rejection:
   Transverse, AC: 34 db at 60 Hz
   Common Mode, AC: 10 VAC or 110 VAC through 1 megohm, 60 Hz, for 0.1% span output shift.
6. Supply Voltage: 90 VDC.
7. Output: 10-50 MADC into 600 ohms ± 10%.
8. Frequency Response: 3 db down at 1 Hz for 100% peak-to-peak input.
9. Factory calibrated.
10. Temperature, ambient: -40° to 200°F at the transmitter.
11. Vibration: no effect for .2 g, 10 to 60 Hz.
12. Burnout: none.
13. Range: 50-150°F.
14. Temperature element is directly coupled to transmitter and transmitter to supply output signal of 10-50 MADC for temperatures of 50-150°F.
15. Waterproofed conduit connectors.
16. DC input power and output signals to be carried on same two copper wires with one power supply for all temperature readouts.
17. No immediate transducer between units and receiver.

Well Specifications:

1. Two wells, 6 inch immersion, 304 stainless steel, no lag.
2. Two wells, 4 inch immersion, 304 stainless steel, no lag.
3. Two wells, 6 inch immersion, 6061 aluminum.
4. Two wells, 3 inch immersion, 6061 aluminum.
5. Mounting thread: 3/4 inch NPT hex (inches) 1-1/4.

2.5.3.2 Differential Pressure Transmitters 928A and B (see Figures 2.1, 2.11, 2.12).

Differential pressure transmitters 928A and B will be installed for monitoring HX 503A and B differential pressures. A low differential pressure will alarm annunciator 6-2 in the 5 or 10 MW mode. In the 5 MW mode (see Figure 2.11), 1S8-8 will complete a circuit allowing DPS 928A or B to clear annunciator 6-2. In the 10 MW mode, 1S8-8 is opened and both 928A and B must be satisfied to clear annunciator 6-2.

In December 2006, DP 928A and DPS 928B were removed as part of the primary coolant heat exchanger replacement project which consisted of replacing the shell and tube-type heat exchangers with plate-type heat exchangers. New dual tap flanges were installed for flow orifices 913A and 913B, which allowed an additional flow transmitter to be connected to each heat exchanger leg, thus eliminating the need for DPS 928A and DPS 928B.

2002
2006

2.5.3.3 Differential Pressure Transmitter 929 (see Figures 2.1, 2.11, 2.12 and 2.13). Differential pressure transmitter 929 will be installed to verify flow through the core. On low flow through the core the following will occur:

1. Annunciation of reactor loop low flow scram (3-1) (Figure 2.11).
2. De-energization of relays 2K13 and 2K28 (Figure 2.13) which in turn provide scram input signals to the safety system (Figure 2.14).
3. Valve 546 A and B operation (Open).

Specifications for EP 928A and B and EP 929:

1. Two wire transmitters operate on a single standard DC power supply.
2. Remote low set points:
   a. Individual unit for each differential pressure transmitter.
   b. Adjustable through entire span.
   c. Accuracy: ± 0.2% setpoint and repeatability.
   d. Front panel adjustment.
   e. Alarm contacts open output circuit in event of low flow and/or loss of power supplying units.
   f. Two alarm contacts per unit.
3. Differential pressure transmitter:
   a. Solid state circuitry.
   b. Accuracy: ± 0.5% of calibrated span includes linearity, hysteresis, repeatability, and dead band.
   c. Span and zero externally adjustable.
   d. Sensitivity: 0.1% of span.
   e. Temperature coefficient: 0.02% per °F.
   f. Ambient temperature 0-180°F.
   g. Output: 10-50 MADC.
   h. Constructed of 316 stainless steel: housing, measuring element and all other components in contact with H_2O.
   i. Pressure connections: 1/2" NPT.
   j. Static pressure rating: 1500 psig.
   k. Vents: for easy clean out.
   l. Electrical: two 1/2" NPT tapped holes.

In December 2006, DPS 928A and DPS 928B were removed as part of the primary coolant heat exchanger replacement project which consisted of replacing the shell and tube-type heat exchangers with plate-type heat exchangers. 2006

2.5.3.4 Pool Water Flow Instrumentation (see Figures 2.1, 2.7, 2.11, and 2.12). Pool system flow is monitored by a single flow element supporting two flow transmitters, 912F and 912D. Flow transmitter 912F, square root "transmitter" 919F, pool alarm unit 920D, and recorder 915D form one flow 2002

2002
2006

instrument leg. Flow transmitter 912D, square root "transmitter" 919B, dual alarm 920C, and recorder 915C form a second flow instrument leg. Flow of heat exchanger 521A is monitored by flow transmitter 912D, square root "transmitter" 919B, dual alarm 920C and recorder 915C. 2002

Output logic of the dual electronic alarm unit EP 920C and D determine the scram functions, alarms and control circuitry for 5 or 10 MW operation. EP 920C output controls K31, K68 and K72. EP 920D output controls K67, K69 and K73.

2.5.3.5 Reactor Water Flow Instrumentation (see Figures 2.1, 2.7, 2.11 and 2.12). The additional heat exchanger 503B is monitored for flow by flow element 913B, flow transmitter 912E, square root "transmitter" 919E, electronic alarm 920B, and recorder 915B. 2002

Flow of heat exchanger 503A is monitored by flow element 913A, flow transmitter 912A, square root "transmitter" 919A, electronic alarm 920A, and recorder 915A. In December 2006, as part of the primary coolant heat exchanger replacement project, which consisted of replacing the shell and tube-type heat exchangers with plate-type heat exchangers, new dual tap flanges were installed for flow orifices 913A and 913B. This allowed an additional flow transmitter to be connected to each heat exchanger leg. The following paragraphs describe how flow through the heat exchangers is now monitored. 2002

Flow through heat exchanger 503A is monitored by flow element 913A, flow transmitters 912A and 912E, square root transmitters 919A and 919E, electronic alarm units 920A and 920C, and recorder 915A/B. 2006

Flow through heat exchanger 503B is monitored by flow element 913B, flow transmitters 912G and 912H, square root transmitters 919G and 919H, electronic alarm units 920E and 920G, and recorder 915G/H.

Output logic of the dual electronic alarm units EP 920A, EP 920C, EP 920E, and EP 920G determine the scram functions, alarms and control circuitry for 5 and 10 MW operation.

2.5.3.6 Specifications for Flow Measurement Equipment

EP No. 912F:

Differential pressure transmitter, body of type 316 stainless steel. Unit supplied with stainless steel 1/2" NPT process connections and 3-way valve manifold with fittings, type M1, Anderson, Greenwood and Co. Accuracy ± 0.5% of span, normal operating conditions 150°F and 100 psig water. Unit has scale of 0-600 inches H_2O. Transmission signal 10-50 MADC. Type 553, GE/MAC. Cat. No. 50-553122CAAN2.

2002

EP No. 912E:

Differential pressure transmitter, body of 316 stainless steel. Unit supplied with stainless steel 1/2" NPT process connections and 3-way valve manifold with fittings, type M1, Anderson, Greenwood and Co. Accuracy ± 0.5% of span, normal operating conditions of 150°F and 100 psig water. Unit has scale of 0-250" H_2O. Transmission signal 10-50 MADC. Type 553, GE/MAC. Cat. No. 50-553122CAAF2.

Flow Integrators EP No's. 919E and 919F:

Two flow integrators (square root transmitters), 10-50 MADC input, 10-50 MADC output, rack mounting, accuracy ± 0.5% of full scale at 20% output."

~~input, 10-50 at 20% output, type 565 GE/MAC, Cat. No. 50-565100AAAC1, with GE Type 1952K40 cable and fuse.~~ 2002

Dual Electronic Alarm Unit, EP No. 920C and D:

Dual unit electronic alarm, 10-50 MADC input. Accuracy ± 0.5%, 1-100% adjustable span, rack mounting. Each channel has single low alarm capability. GE/MAC type 560. Cat. No. 50-560330AAC1. GE Type 1952 K40 cable assembly with fuse.

2.5.4 Reactor Control System (see Figure 2.7)

2.5.4.1 Power level selection indicators in conjunction with power level switch (1S8) allows a visual indication of power level selected.

2.5.4.2 Power level selection switch (1S8) will have the following contact schedule:

1S8 (SBM)
POWER LEVEL

Contact	Position 50 kW	Position 5 MW	Position 10 MW
1	X		
2	X	X	
3			X
4		X	X
5	X		
6	X	X	
7			X
8		X	
9	X		
10	X	X	
11	X	X	
12			X
13	X	X	
14			X
15	X		
16		X	
17			X
18	X		
19		X	
20			X

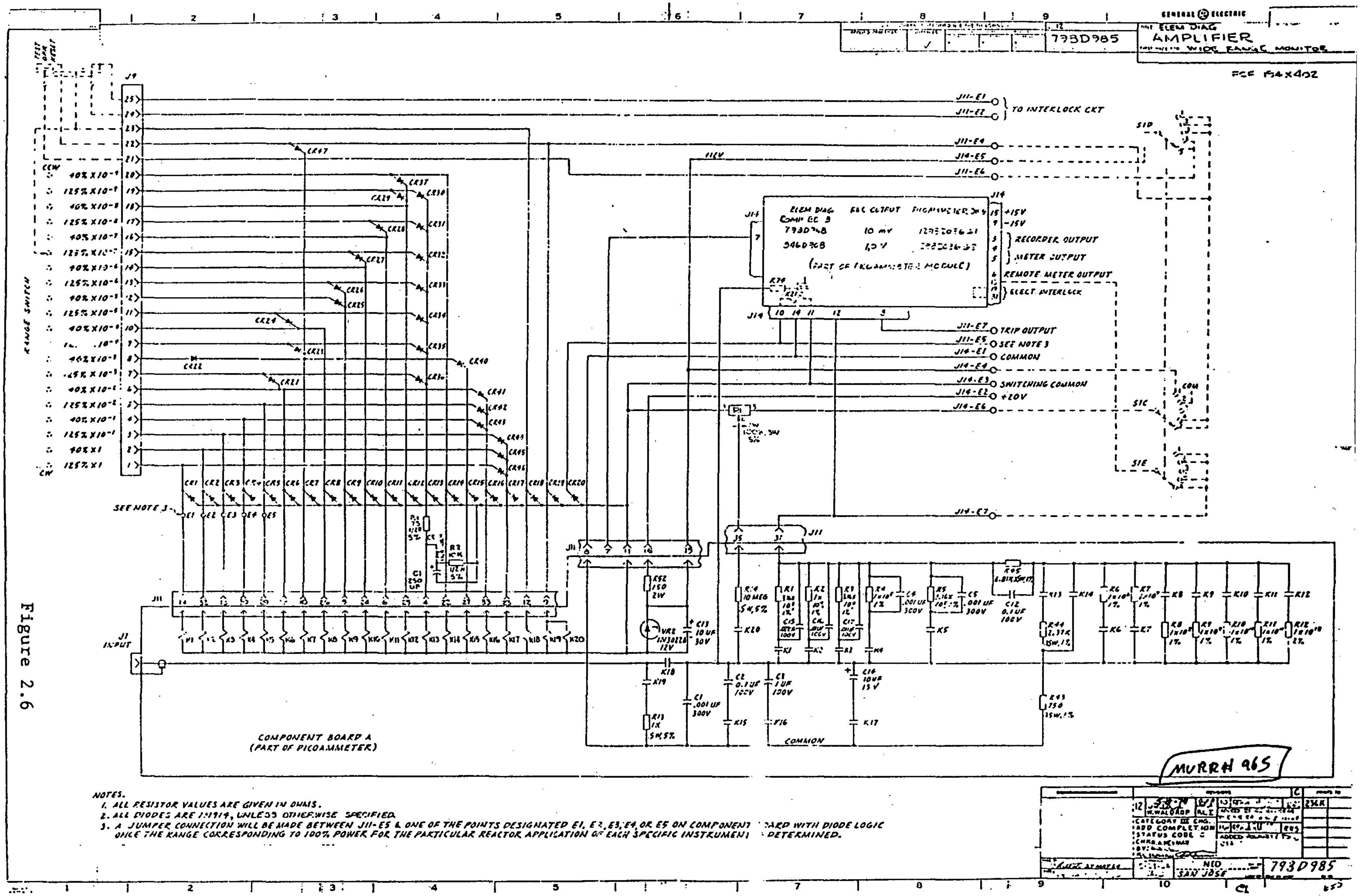

GENERAL ELECTRIC
ELEM DIAG
AMPLIFIER
WIDE RANGE MONITOR
793D985
TO INTERLOCK CKT
RECORDER OUTPUT
METER OUTPUT
REMOTE METER OUTPUT
ELECT. INTERLOCK
TRIP OUTPUT
SEE NOTE 3
COMMON
SWITCHING COMMON
+20V
RANGE SWITCH
J1 INPUT
COMPONENT BOARD A
(PART OF PICOAMMETER)
NOTES.
1. ALL RESISTOR VALUES ARE GIVEN IN OHMS.
2. ALL DIODES ARE 1N914, UNLESS OTHERWISE SPECIFIED.
3. A JUMPER CONNECTION WILL BE MADE BETWEEN J11-E5 & ONE OF THE POINTS DESIGNATED E1, E2, E3, E4, OR E5 ON COMPONENT BOARD WITH DIODE LOGIC ONCE THE RANGE CORRESPONDING TO 100% POWER FOR THE PARTICULAR REACTOR APPLICATION OF EACH SPECIFIC INSTRUMENT IS DETERMINED.
MURRH 965
NED
SAN JOSE
793D985

Figure 2.6

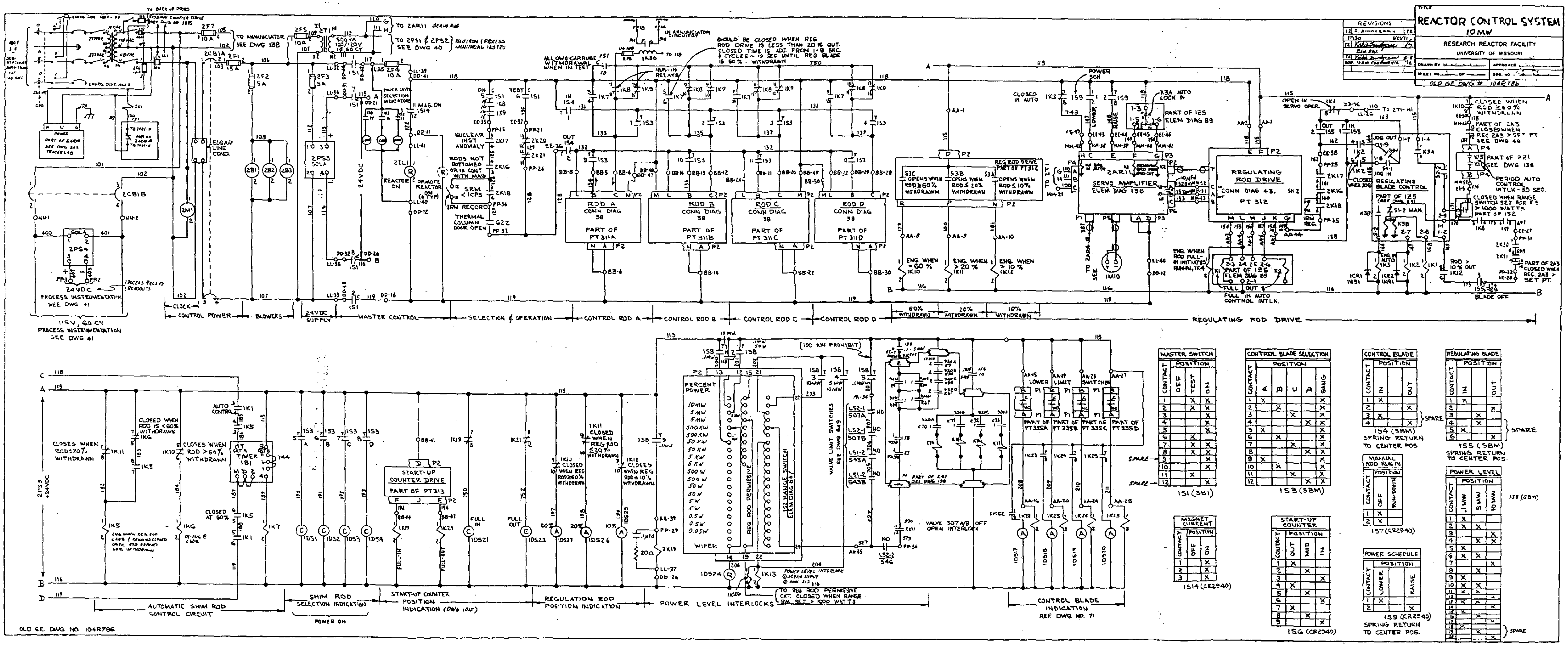

REACTOR CONTROL SYSTEM
10 MW
RESEARCH REACTOR FACILITY
UNIVERSITY OF MISSOURI
OLD GE DWG NO. 104R786
CONTROL POWER
BLOWERS
24VDC SUPPLY
MASTER CONTROL
SELECTION & OPERATION
CONTROL ROD A
CONTROL ROD B
CONTROL ROD C
CONTROL ROD D
REGULATING ROD DRIVE
AUTOMATIC SHIM ROD CONTROL CIRCUIT
SHIM ROD SELECTION INDICATION
START-UP COUNTER POSITION INDICATION (DWG 1015)
REGULATION ROD POSITION INDICATION
POWER LEVEL INTERLOCKS
CONTROL BLADE INDICATION REF DWG NO. 71
MASTER SWITCH
CONTROL BLADE SELECTION
CONTROL BLADE
REGULATING BLADE
MAGNET CURRENT
START-UP COUNTER
POWER SCHEDULE
POWER LEVEL

Figure 2.7

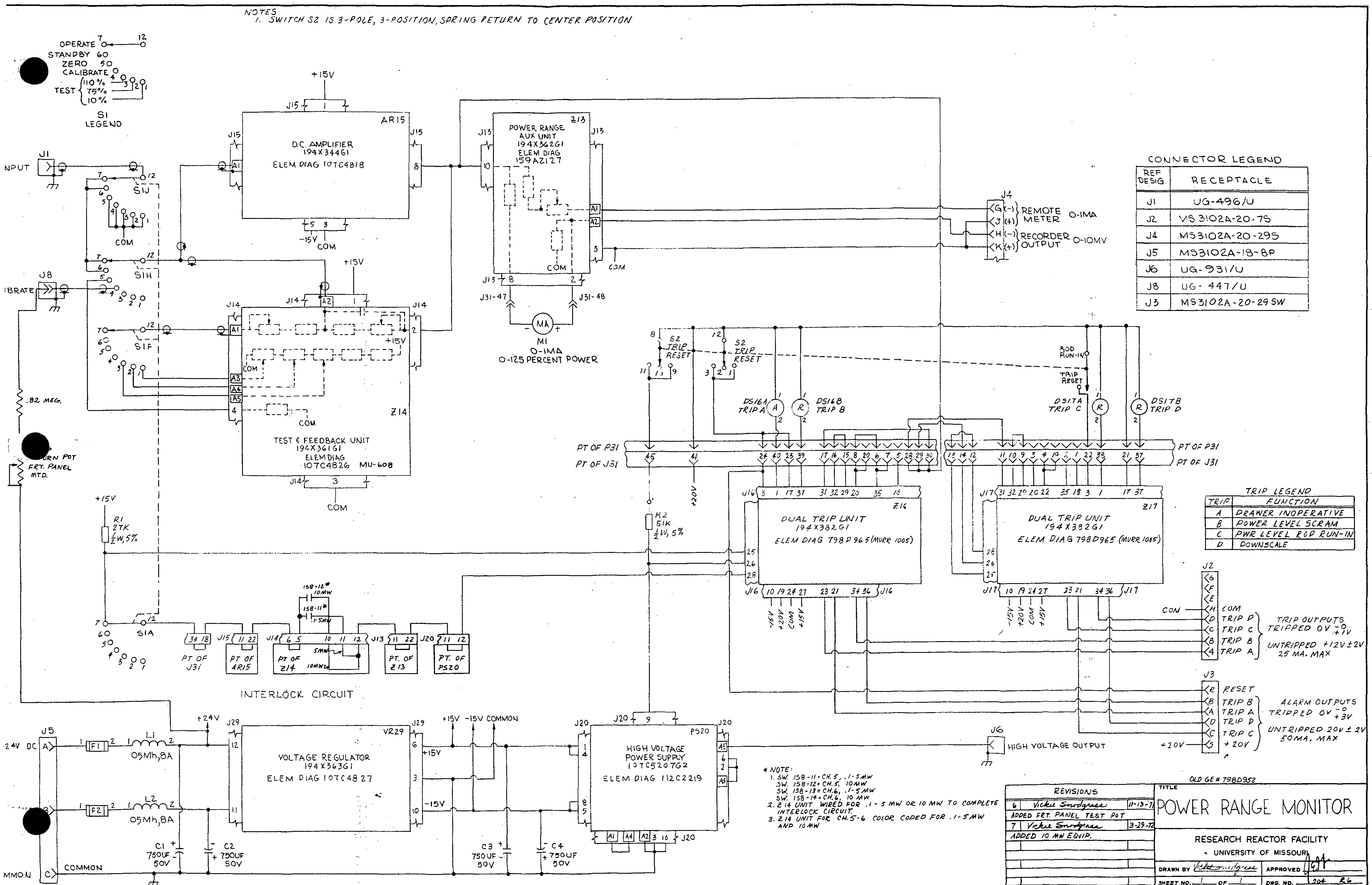

NOTES
1. SWITCH S2 IS 3-POLE, 3-POSITION, SPRING RETURN TO CENTER POSITION
OPERATE
STANDBY
ZERO
CALIBRATE
TEST 110% 75% 10%
S1 LEGEND
+15V
D.C. AMPLIFIER 194X344G1 ELEM DIAG 107C4818
AR15
POWER RANGE AUX UNIT 194X362G1 ELEM DIAG 159A2127
Z13
TEST & FEEDBACK UNIT 194X361G1 ELEM DIAG 107C4826 MU-608
Z14
M1 0-1MA 0-125 PERCENT POWER
REMOTE METER 0-1MA
RECORDER OUTPUT 0-10MV
S2 TRIP RESET
ROD RUN-IN
TRIP RESET
DS16A TRIP A
DS16B TRIP B
DS17A TRIP C
DS17B TRIP D
PT OF P31
PT OF J31
DUAL TRIP UNIT 194X382G1 ELEM DIAG 798D965 (MURR 1005)
Z16
Z17
R1 2.7K 1/2W, 5%
K2 51K 1/2W, 5%
.82 MEG.
TURN POT FRT. PANEL MTD.
INTERLOCK CIRCUIT
PT OF J31
PT OF AR15
PT OF Z14
PT. OF Z13
PT. OF PS20
CONNECTOR LEGEND
REF DESIG | RECEPTACLE
J1 | UG-496/U
J2 | VS 3102A-20-7S
J4 | MS3102A-20-29S
J5 | MS3102A-18-8P
J6 | UG-931/U
J8 | UG-447/U
J3 | MS3102A-20-29SW
TRIP LEGEND
TRIP | FUNCTION
A | DRAWER INOPERATIVE
B | POWER LEVEL SCRAM
C | PWR LEVEL ROD RUN-IN
D | DOWNSCALE
TRIP OUTPUTS TRIPPED 0V +0 -1V UNTRIPPED +12V ± 2V 25 MA. MAX
RESET
ALARM OUTPUTS TRIPPED 0V -0 +3V UNTRIPPED 20V ± 2V 50MA. MAX
HIGH VOLTAGE OUTPUT
24V DC
COMMON
L1 .05Mh, 8A
L2 .05Mh, 8A
C1 750UF 50V
C2 750UF 50V
C3 750UF 50V
C4 750UF 50V
+24V
+15V -15V COMMON
VOLTAGE REGULATOR 194X363G1 ELEM DIAG 107C4827
VR29
HIGH VOLTAGE POWER SUPPLY 107C5207G2 ELEM DIAG 112C2219
PS20
* NOTE:
1. SW. 158-11 = CH.5, .1-5MW
SW. 158-12 = CH.5, 10MW
SW. 158-13 = CH.6, .1-5MW
SW. 158-14 = CH.6, 10 MW
2. Z14 UNIT WIRED FOR .1-5 MW OR 10 MW TO COMPLETE INTERLOCK CIRCUIT.
3. Z14 UNIT FOR CH.5-6 COLOR CODED FOR .1-5MW AND 10MW
REVISIONS
6 Vickie Snodgrass 11-13-7
ADDED FRT. PANEL TEST POT
7 Vickie Snodgrass 3-29-7
ADDED 10 MW EQUIP.
OLD GE # 798D952
TITLE
POWER RANGE MONITOR
RESEARCH REACTOR FACILITY
UNIVERSITY OF MISSOURI
DRAWN BY
APPROVED
SHEET NO. 1 OF 1
DWG. NO. 204 R6

Figure 2.8

GENERAL ELECTRIC 107C4826

TEST & FEEDBACK UNIT

FIRST MADE FOR POWER RANGE MONITOR

FCF 194X361

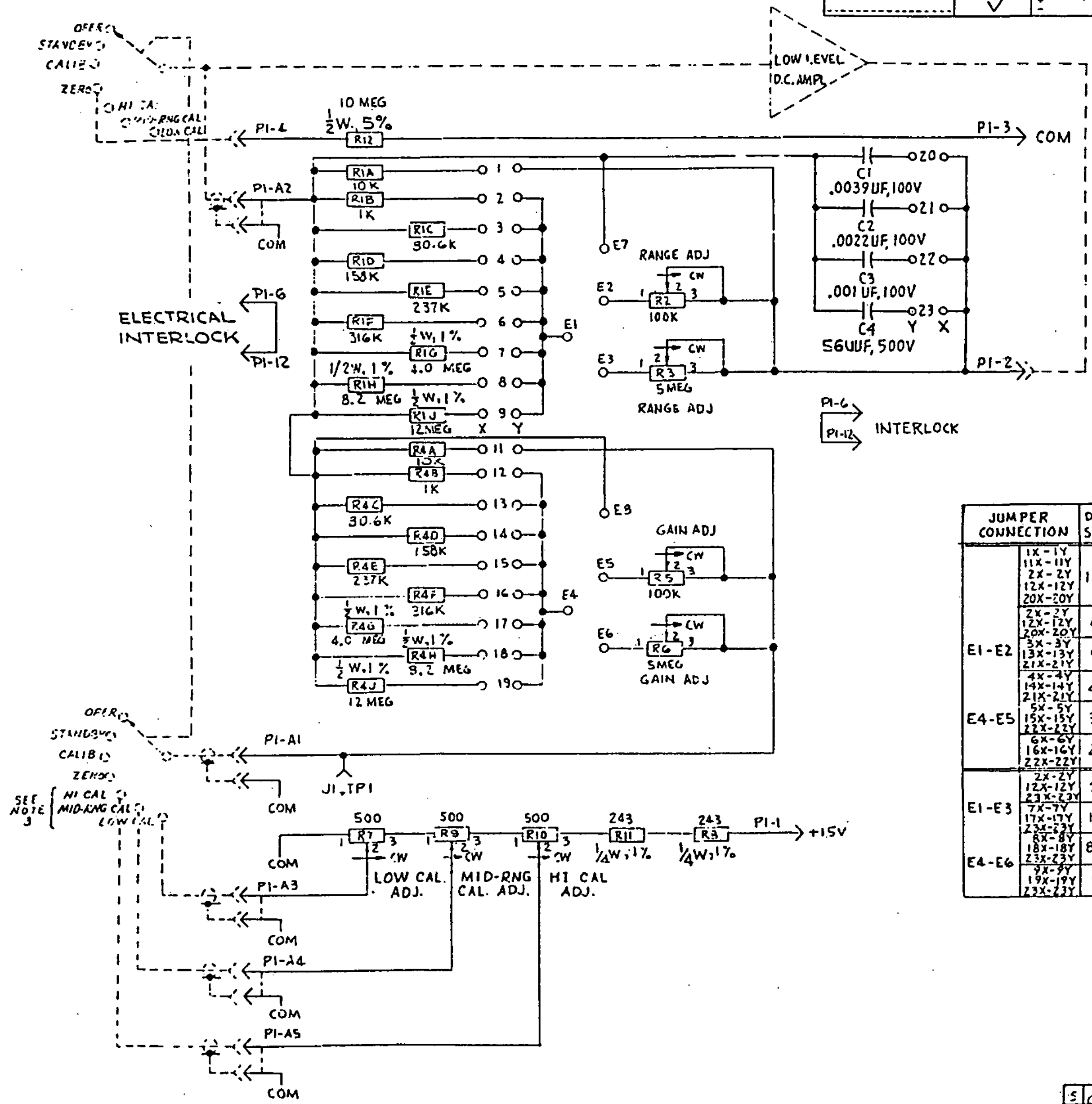


Figure 2.9

NOTE

1. ALL RESISTORS ARE $\frac{1}{8}$ W, TOLERANCE ±1% & VALUE GIVEN IN OHMS UNLESS OTHERWISE SPECIFIED
2. RESISTORS R1A-R1J INCL., R2 & R3 ARE FEEDBACK RESISTORS, CONNECT PER TABLE 1 FOR INDICATED CURRENT RANGE.
   RESISTORS R4A-R4J INCL., R5 & R6 SET OVERALL GAIN OF D.C. AMPLIFIER TO UNITY WHEN CALIB. SWITCH IS IN TEST POSITIONS. CONNECT PER TABLE 1.
3. *NOMENCLATURE SHOWN ON THE 3 CALIBRATE POSITIONS OF SELECTOR SWITCH S1 IS GENERATED AND IS PRESENTED FOR REFERENCE. SEE THE APPLICABLE POWER RANGE MONITOR ELEM DIAG FOR CORRECT SELECTOR SWITCH POSITION NOMENCLATURE FOR A SPECIFIC INSTRUMENT.*

TABLE 1

JUMPER CONNECTION		DC AMPS APPROX FULL SCALE CURRENT RANGE		RESISTOR DESIG
E1-E2	1X-1Y 11X-11Y 2X-2Y 12X-12Y 20X-20Y	1.1×10^{-3}	2.0×10^{-3}	R1A R4A
	2X-2Y 12X-12Y 20X-20Y	1.1×10^{-4}	1.1×10^{-3}	R1B, R4B
	3X-3Y 13X-13Y 21X-21Y	5.9×10^{-5}	1.1×10^{-4}	R1C, R4C
E4-E5	4X-4Y 14X-14Y 21X-21Y	4.06×10^{-5}	5.9×10^{-5}	R1D, R4D
	5X-5Y 15X-15Y 22X-22Y	3.08×10^{-5}	4.06×10^{-5}	R1E, R4E
	6X-6Y 16X-16Y 22X-22Y	2.48×10^{-5}	3.08×10^{-5}	R1F, R4F
E1-E3	2X-2Y 12X-12Y 23X-23Y	2.22×10^{-6}	2.48×10^{-5}	R1B, R4B
	7X-7Y 17X-17Y 23X-23Y	1.18×10^{-6}	2.22×10^{-6}	R1G, R4G
E4-E6	8X-8Y 18X-18Y 23X-23Y	8.06×10^{-7}	1.18×10^{-6}	R1H, R4H
	9X-9Y 19X-19Y 23X-23Y	6.3×10^{-7}	3.06×10^{-7}	R1J, R4J

MURR #608

REVISIONS

1 CHANGED TABLE 1 CHG'D. DESCR. OF ADJ. R7, R9, R10

2 DELETED E7 & E8 FROM TABLE. ON SW POS. 110% WAS 130%, 75% WAS 100%

3 ADDED CONSTRUC & JUMPER CONN.

4 2.22×10⁻⁶ RANGE JUMPER CONN. WAS 20X-20Y; C4 WAS 100PF

5 ADDED NOTE 3

6 CHG TOL AT R1H, R1G, R1J, R4G, R4H, R4J & VALUE AT R1H, R1G, R4G & R4H

A10

22 JAN 64

APED SAN JOSE

107C4826

C1

JAN 64

4

2 3 4 5 6

(255)

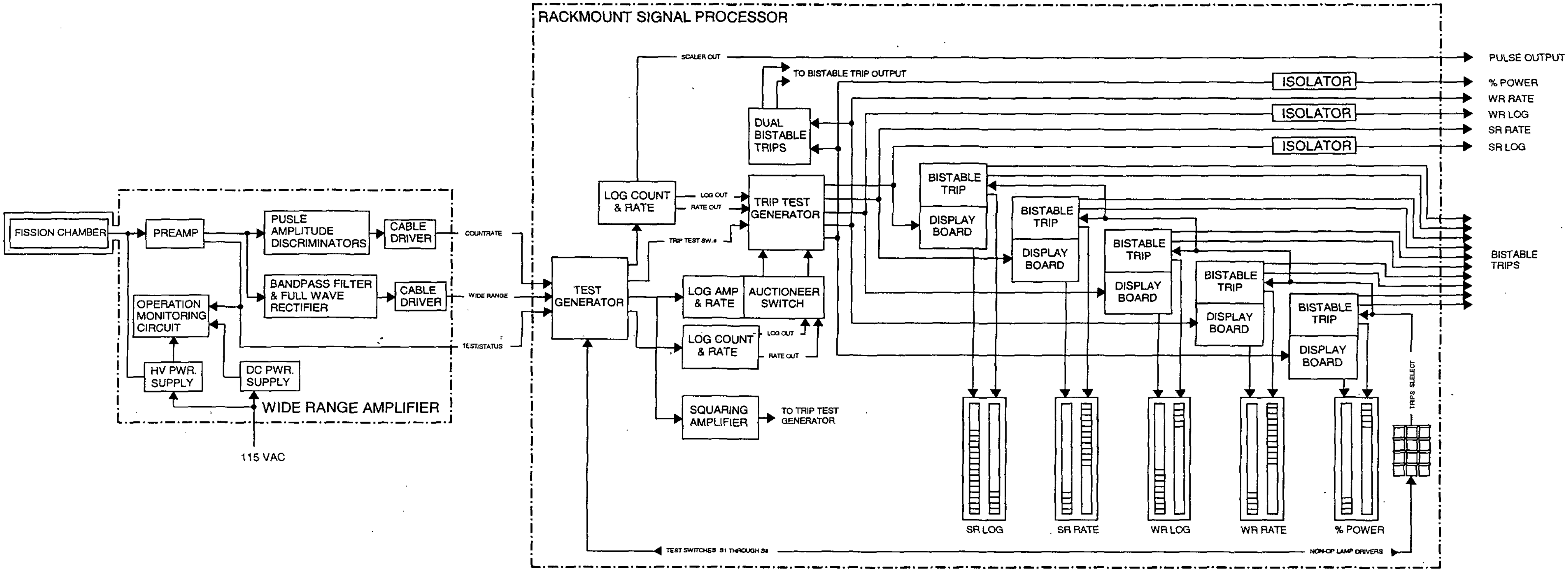
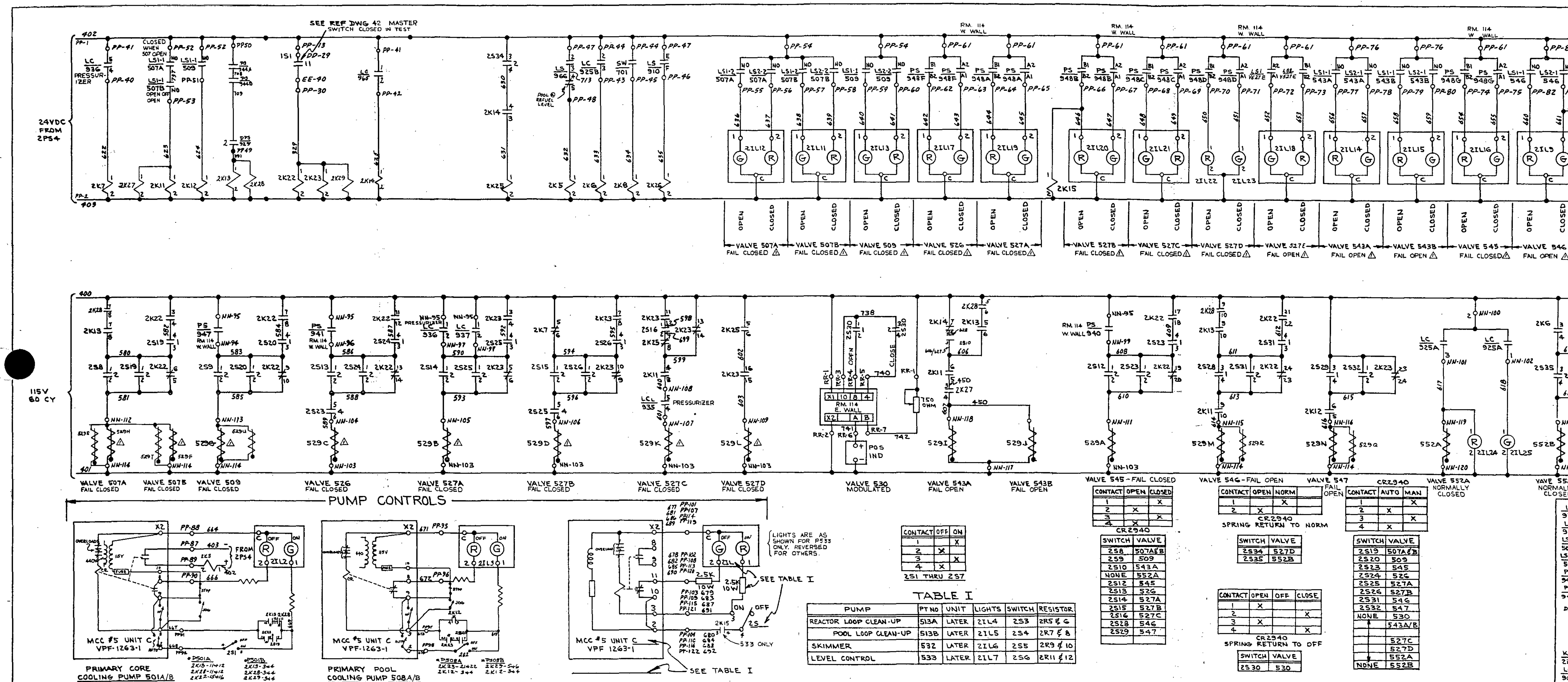


FIGURE 2.8.a

GAMMA-METRICS NEUTRON FLUX MONITOR BLOCK DIAGRAM

2001



CONTACT	OFF	ON
1		X
2	X	
3		X
4	X	

2S1 THRU 2S7

TABLE I

PUMP	PT NO	UNIT	LIGHTS	SWITCH	RESISTOR
REACTOR LOOP CLEAN-UP	513A	LATER	2IL4	2S3	2R5 & 6
POOL LOOP CLEAN-UP	513B	LATER	2IL5	2S4	2R7 & 8
SKIMMER	532	LATER	2IL6	2S5	2R9 & 10
LEVEL CONTROL	533	LATER	2IL7	2S6	2R11 & 12

VALVE 545 - FAIL CLOSED

CONTACT	OPEN	CLOSED
1		X
2	X	
3		X
4	X	

CR2940

SWITCH	VALVE
2S8	507A&B
2S9	509
2S10	543A
NONE	552A
2S12	545
2S13	526
2S14	527A
2S15	527B
2S16	527C
2S28	546
2S29	547

VALVE 546 - FAIL OPEN

CONTACT	OPEN	NORM	
1		X	
2	X		

CR2940
SPRING RETURN TO NORM

SWITCH	VALVE
2S34	527D
2S35	552B

CONTACT	OPEN	OFF	CLOSE
1	X		
2			X
3	X		
4			X

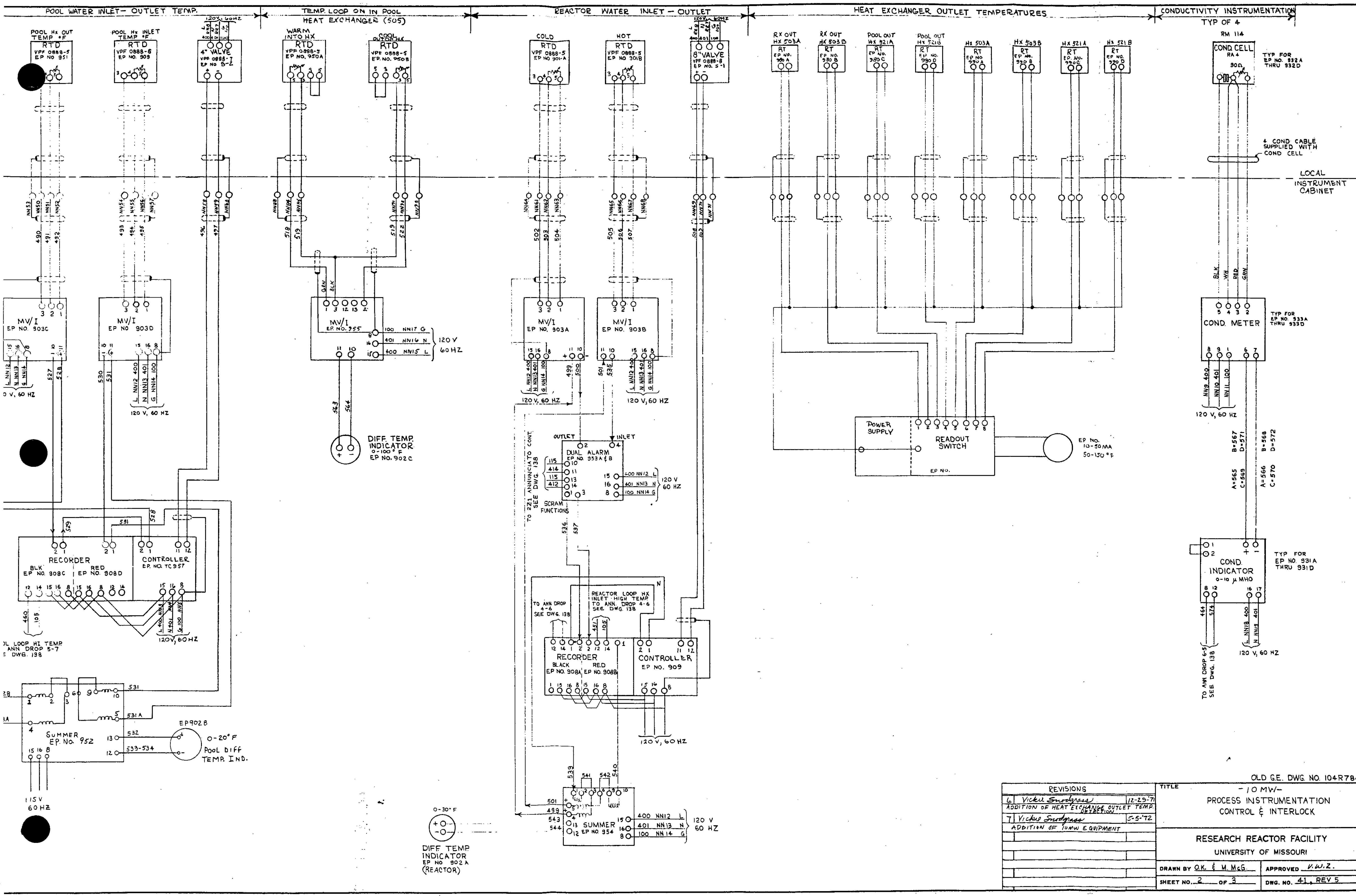
CR2940
SPRING RETURN TO OFF

SWITCH	VALVE
2S30	530

CR2940

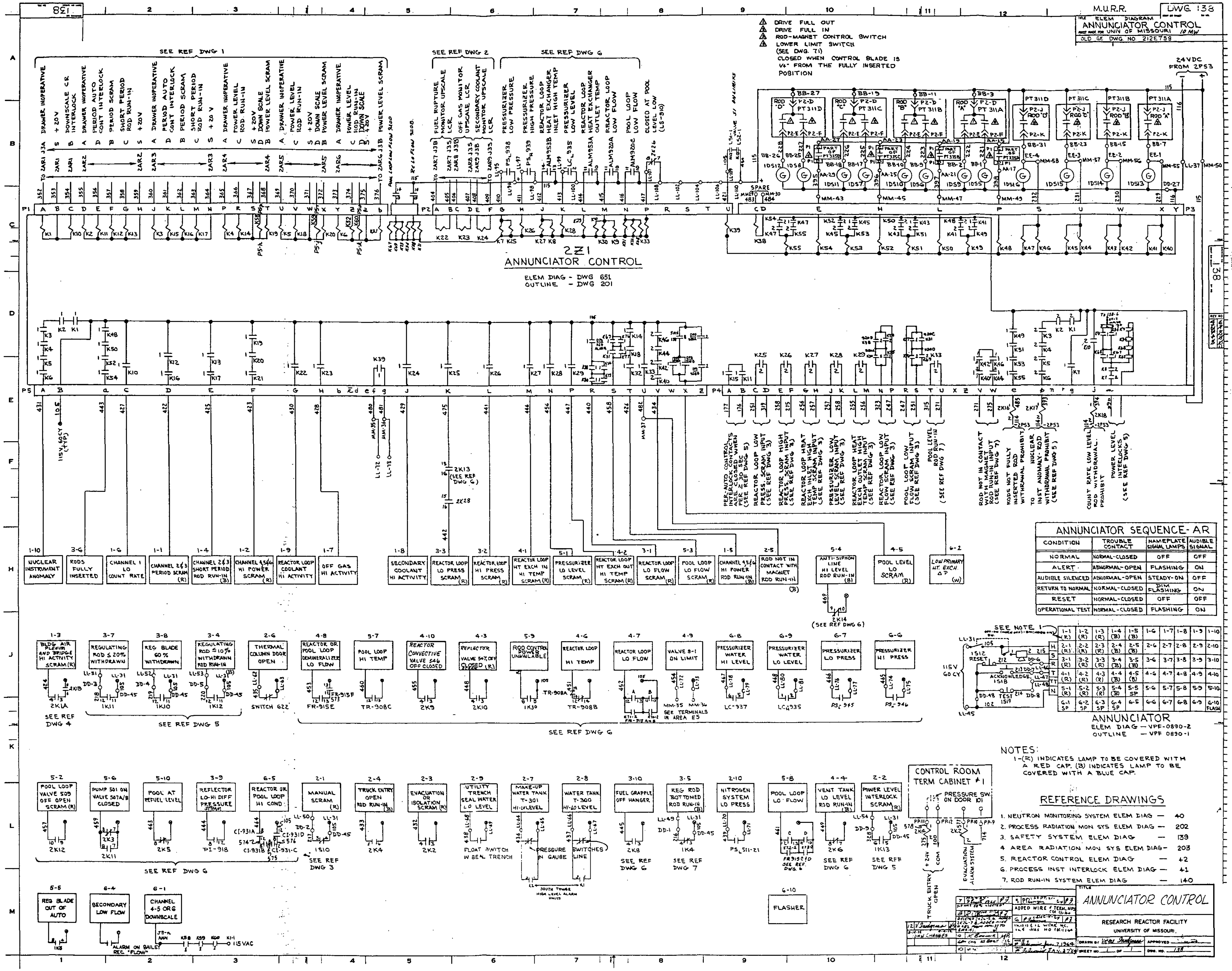
CONTACT	AUTO	MAN
1		X
2	X	
3		X
4	X	

SWITCH	VALVE
2S19	507A&B
2S20	509
2S23	545
2S24	526
2S25	527A
2S26	527B
2S31	546
2S32	547
NONE	530
	543A/B
	527C
	527D
	552A
NONE	552B



POOL WATER INLET- OUTLET TEMP.
TEMP. LOOP ON IN POOL HEAT EXCHANGER (505)
REACTOR WATER INLET - OUTLET
HEAT EXCHANGER OUTLET TEMPERATURES
CONDUCTIVITY INSTRUMENTATION
TYP OF 4
LOCAL INSTRUMENT CABINET
DIFF. TEMP. INDICATOR 0-100° F EP NO. 902C
READOUT SWITCH
POWER SUPPLY
COND. METER
COND. INDICATOR 0-10 μ MHO
DUAL ALARM EP NO. 953A & B
SUMMER EP NO. 952
SUMMER EP NO 954
DIFF TEMP INDICATOR EP NO 902 A (REACTOR)
OLD G.E. DWG. NO. 104R784
REVISIONS
6 Vickie Snodgrass 12-29-71 ADDITION OF HEAT EXCHANGE OUTLET TEMP. DETECTION
7 Vickie Snodgrass 5-5-72 ADDITION OF 10MW EQUIPMENT
TITLE -10 MW- PROCESS INSTRUMENTATION CONTROL & INTERLOCK
RESEARCH REACTOR FACILITY
UNIVERSITY OF MISSOURI
DRAWN BY O.K. & M. McG.
APPROVED V.W.Z.
SHEET NO. 2 OF 3
DWG. NO. 41, REV 5

Figure 2.10



M.U.R.R.
DWG. 138
ELEM DIAGRAM
ANNUNCIATOR CONTROL
UNIV OF MISSOURI 10 MW
OLD GE DWG NO 212E759
DRIVE FULL OUT
DRIVE FULL IN
ROD-MAGNET CONTROL SWITCH
LOWER LIMIT SWITCH (SEE DWG. 71) CLOSED WHEN CONTROL BLADE IS 1/4" FROM THE FULLY INSERTED POSITION
24VDC FROM 2PS3
SEE REF DWG 1
SEE REF DWG 2
SEE REF DWG 6
2Z1
ANNUNCIATOR CONTROL
ELEM DIAG - DWG 651
OUTLINE - DWG 201
ANNUNCIATOR SEQUENCE-AR
CONDITION | TROUBLE CONTACT | NAMEPLATE SIGNAL LAMPS | AUDIBLE SIGNAL
NORMAL | NORMAL-CLOSED | OFF | OFF
ALERT | ABNORMAL-OPEN | FLASHING | ON
AUDIBLE SILENCED | ABNORMAL-OPEN | STEADY-ON | OFF
RETURN TO NORMAL | NORMAL-CLOSED | DIM FLASHING | ON
RESET | NORMAL-CLOSED | OFF | OFF
OPERATIONAL TEST | NORMAL-CLOSED | FLASHING | ON
NUCLEAR INSTRUMENT ANOMALY
RODS FULLY INSERTED
CHANNEL 1 LO COUNT RATE
CHANNEL 2&3 PERIOD SCRAM (R)
CHANNEL 2&3 SHORT PERIOD ROD RUN-IN (B)
CHANNEL 4,5&6 HI POWER SCRAM (R)
REACTOR LOOP COOLANT HI ACTIVITY
OFF GAS HI ACTIVITY
SECONDARY COOLANT HI ACTIVITY
REACTOR LOOP LO PRESS SCRAM (R)
REACTOR LOOP HI PRESS SCRAM (R)
POOL LOOP LO FLOW SCRAM (R)
POOL LEVEL LO SCRAM (R)
ANNUNCIATOR
ELEM DIAG — VPF-0890-2
OUTLINE — VPF-0890-1
NOTES:
1-(R) INDICATES LAMP TO BE COVERED WITH A RED CAP. (B) INDICATES LAMP TO BE COVERED WITH A BLUE CAP.
CONTROL ROOM TERM CABINET #1
PRESSURE SW. ON DOOR 101
REFERENCE DRAWINGS
1. NEUTRON MONITORING SYSTEM ELEM DIAG — 40
2. PROCESS RADIATION MON SYS ELEM DIAG — 202
3. SAFETY SYSTEM ELEM DIAG — 139
4. AREA RADIATION MON SYS ELEM DIAG — 203
5. REACTOR CONTROL ELEM DIAG — 42
6. PROCESS INST INTERLOCK ELEM DIAG — 41
7. ROD RUN-IN SYSTEM ELEM DIAG — 140
SEE REF DWG 4
SEE REF DWG 5
SEE REF DWG 6
SEE REF DWG 3
FLASHER
ANNUNCIATOR CONTROL
RESEARCH REACTOR FACILITY
UNIVERSITY OF MISSOURI

Figure 2.11

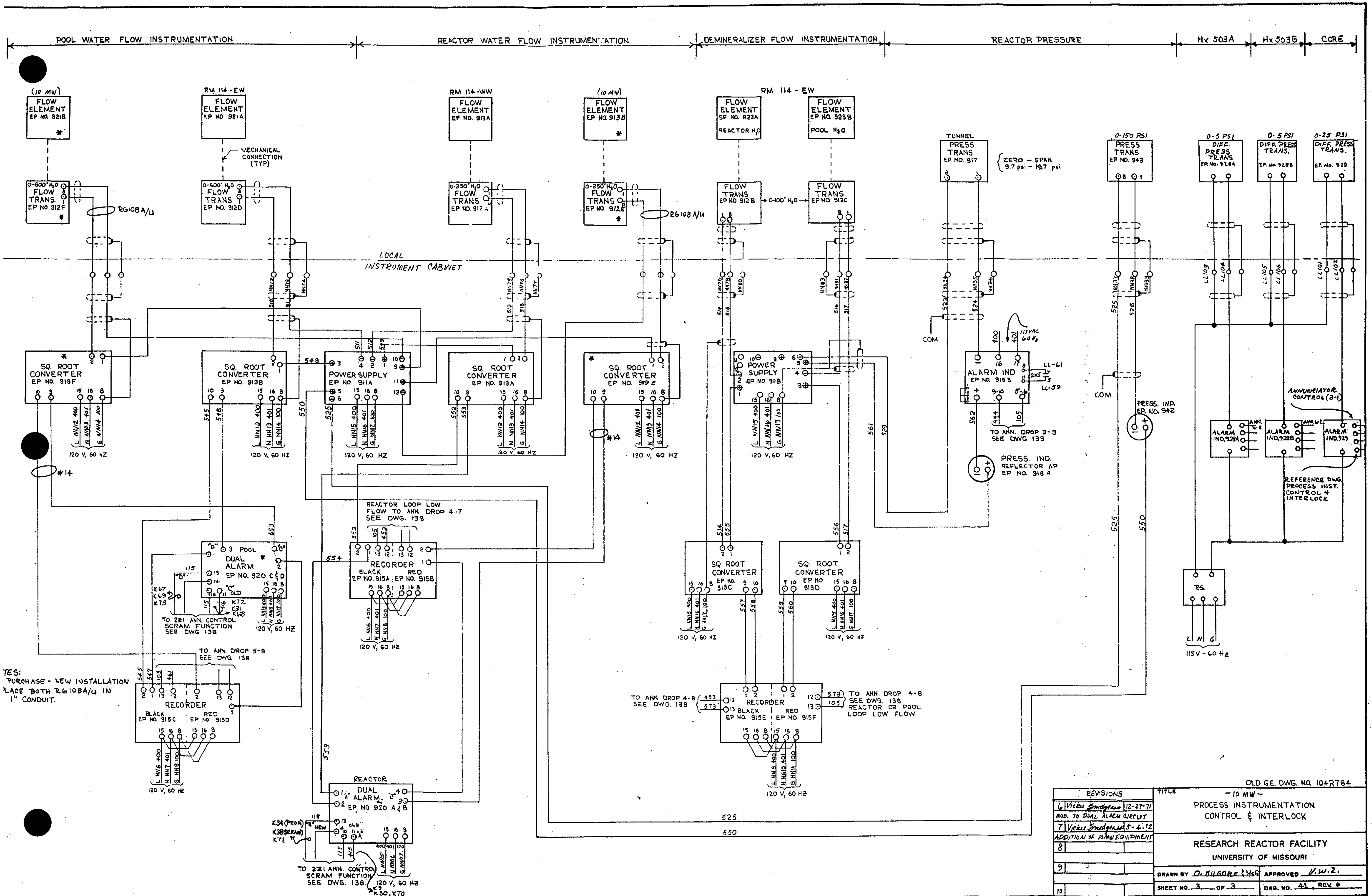
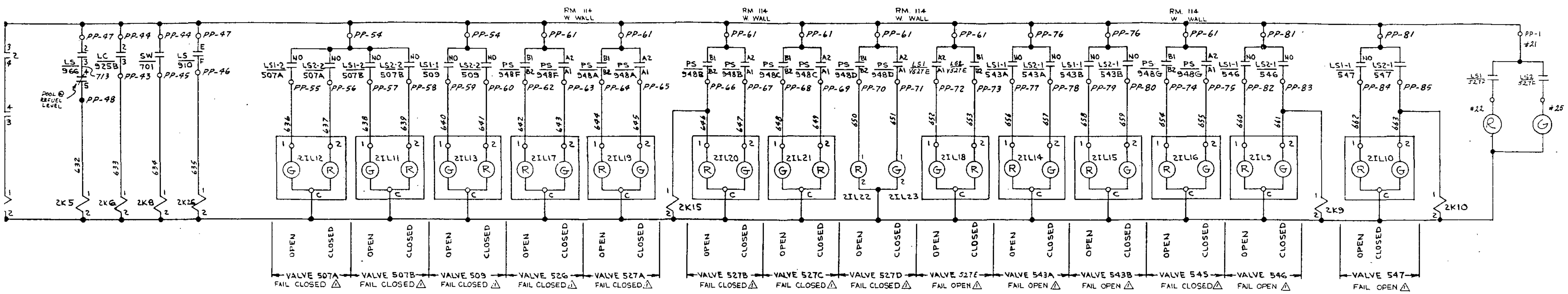
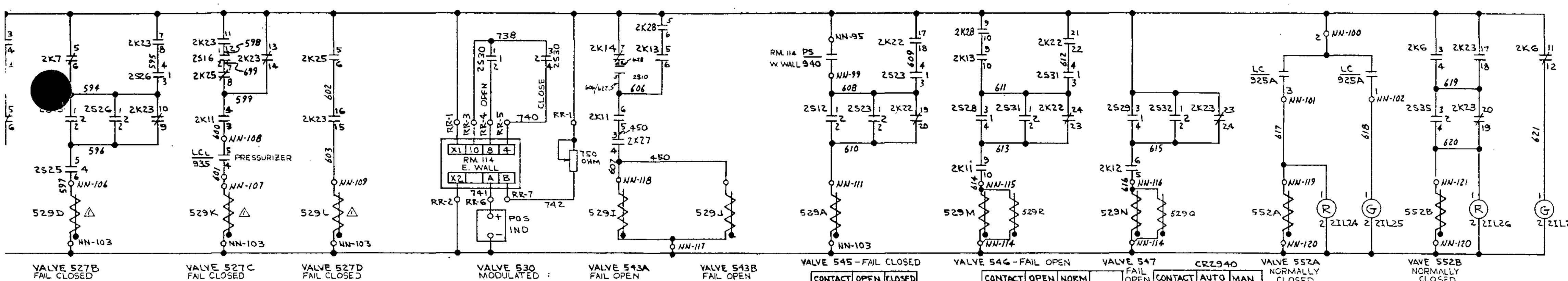
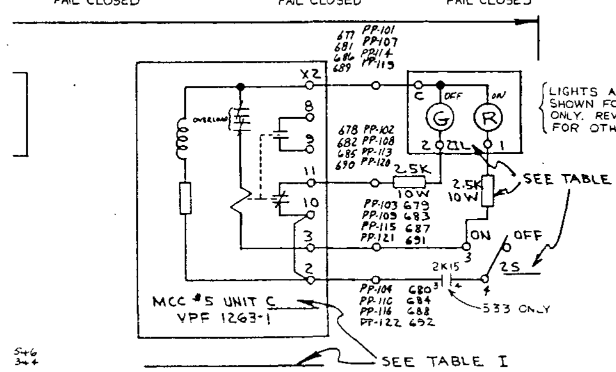

POOL WATER FLOW INSTRUMENTATION
REACTOR WATER FLOW INSTRUMENTATION
DEMINERALIZER FLOW INSTRUMENTATION
REACTOR PRESSURE
Hx 503A
Hx 503B
CORE
FLOW ELEMENT EP NO. 921B
FLOW ELEMENT EP NO. 921A
FLOW ELEMENT EP NO. 913A
FLOW ELEMENT EP NO. 913B
FLOW ELEMENT EP NO. 923A
FLOW ELEMENT EP NO. 923B
RM 114 - EW
RM 114 - WW
MECHANICAL CONNECTION (TYP)
FLOW TRANS. EP NO. 912F
FLOW TRANS EP NO. 912D
FLOW TRANS EP NO. 912B
FLOW TRANS EP NO. 912C
RG108A/U
LOCAL
INSTRUMENT CABINET
TUNNEL PRESS TRANS EP NO. 917
ZERO - SPAN 9.7 psi - 19.7 psi
0-150 PSI PRESS TRANS EP NO. 943
0-5 PSI DIFF. PRESS TRANS
0-25 PSI DIFF. PRESS TRANS.
SQ. ROOT CONVERTER EP NO. 919F
SQ. ROOT CONVERTER EP NO. 919B
POWER SUPPLY EP NO. 911A
SQ. ROOT CONVERTER EP NO. 919A
SQ. ROOT CONVERTER EP NO. 919E
POWER SUPPLY EP NO 911B
ALARM IND EP NO. 918B
COM
PRESS. IND. EP. NO. 942
ANNUNCIATOR CONTROL (3-1)
ALARM IND, 928A
ALARM IND, 928B
ALARM IND, 929
TO ANN. DROP 3-9 SEE DWG 138
PRESS. IND. REFLECTOR ΔP EP NO. 918A
REFERENCE DWG. PROCESS INST. CONTROL + INTERLOCK
120 V, 60 HZ
POOL DUAL ALARM EP NO. 920 C & D
TO 221 ANN. CONTROL SCRAM FUNCTION SEE DWG 138
REACTOR LOOP LOW FLOW TO ANN. DROP 4-7 SEE DWG. 138
RECORDER BLACK EP NO. 915A RED EP NO. 915B
SQ. ROOT CONVERTER EP NO. 913C
SQ. ROOT CONVERTER EP NO. 913D
115V - 60 Hz
TO ANN. DROP 5-8 SEE DWG. 138
NOTES:
PURCHASE - NEW INSTALLATION
PLACE BOTH RG108A/U IN 1" CONDUIT.
RECORDER BLACK EP NO. 915C RED EP NO. 915D
TO ANN DROP 4-8 SEE DWG. 138
RECORDER BLACK EP NO. 915E RED EP NO. 915F
TO ANN. DROP 4-8 SEE DWG. 138 REACTOR OR POOL LOOP LOW FLOW
REACTOR DUAL ALARM EP NO 920 A & B
TO 221 ANN. CONTROL SCRAM FUNCTION SEE DWG. 138
OLD G.E. DWG. NO. 104R784
REVISIONS
6 Vicki Snodgrass 12-27-71 MOD. TO DUAL ALARM CIRCUIT
7 Vicki Snodgrass 5-4-72 ADDITION OF 10MW EQUIPMENT
TITLE - 10 MW - PROCESS INSTRUMENTATION CONTROL & INTERLOCK
RESEARCH REACTOR FACILITY
UNIVERSITY OF MISSOURI
DRAWN BY O. KILGORE E.McG
APPROVED V.W.Z.
SHEET NO. 3 OF 3
DWG. NO. 41, REV 6

Figure 2.12







CONTACT	OFF	ON
1		X
2	X	
3		X
4	X	

2S1 THRU 2S7

TABLE I

PUMP	PT NO	UNIT	LIGHTS	SWITCH	RESISTOR
REACTOR LOOP CLEAN-UP	513A	LATER	2IL4	2S3	2R5 & 6
POOL LOOP CLEAN-UP	513B	LATER	2IL5	2S4	2R7 & 8
SKIMMER	532	LATER	2IL6	2S5	2R9 & 10
LEVEL CONTROL	533	LATER	2IL7	2S6	2R11 & 12

CONTACT	OPEN	CLOSED
1		X
2	X	
3		X
4	X	

CR2940

SWITCH	VALVE
2S8	507A&B
2S9	509
2S10	543A
NONE	552A
2S12	545
2S13	526
2S14	527A
2S15	527B
2S16	527C
2S28	546
2S29	547

CONTACT	OPEN	NORM	
1		X	
2	X		

CR2940
SPRING RETURN TO NORM

SWITCH	VALVE
2S34	527D
2S35	552B

CONTACT	OPEN	OFF	CLOSE
1	X		
2			X
3	X		
4			X

CR2940
SPRING RETURN TO OFF

SWITCH	VALVE
2S30	530

CR2940

CONTACT	AUTO	MAN
1		X
2	X	
3		X
4	X	

SWITCH	VALVE
2S19	507A&B
2S20	509
2S23	545
2S24	526
2S25	527A
2S26	527B
2S31	546
2S32	547
NONE	530
	543A/B
	527C
	527D
	552A
NONE	552B

DEVICE FUNCTION SCHEDULE

LC/937 (1-2) CLOSES ON HIGH LEVEL
LC/936 (5-4) OPENS ON LOW LEVEL
LS1-1/507A/B (NO) OPENS WHEN VALVE OFF OPEN
LS1-1/509 (NO) OPENS WHEN VALVE OFF OPEN
PS/944A OPENS ON LOW PRESS
PS/944B OPENS ON LOW PRESS
DPS 929 OPENS ON LOW ΔP (LOW FLOW) ACROSS THE CORE.
K9/221 OPENS ON LOW FLOW (AL 920A)
LC/965 (1-2) OPENS ON LOW LEVEL

PS/947 OPENS ON DECREASING PRESSURE
LC/966 (2-3, 4-5) CLOSES AT REFUELING POOL LEVEL
SW/701 OPENS WHEN FUEL GRAPPLE REMOVED FROM HANGER
PS/941 CLOSES ON LOW PRESSURE
PS/940 CLOSES ON HI PRESSURE
LC/925A (2-3) CLOSES ON LOW LEVEL
LC/925B (2-3) CLOSES ON LOW LEVEL
LC/936 (2-1) CLOSES ON HIGH LEVEL
LC/937 (2-1) CLOSES ON HIGH LEVEL
LCL/935 (5-4) OPENS ON LOW LEVEL
LS/910 CLOSES ON LOW LEVEL

PS/948 (TYP) (A2-A1) CLOSES ON DECREASING PRESSURE
PS/948 (TYP) (B1-B2) CLOSES ON INCREASING PRESSURE
LS1 (NO) (TYP) LIMIT SWITCH OPENS WHEN VALVE OFF OF OPEN POSITION.
LS2 (NO) (TYP) LIMIT SWITCH OPENS WHEN VALVE OFF OF CLOSED POSITION

REFERENCE DRAWINGS

1. REACTOR CONTROL SYSTEM — DWG. NO. 42
2. PNEUMATIC CONTROL SYSTEM DIAGRAM — DWG. NO. 172
3. ELECTRIC CONTROL SYSTEM DIAGRAM — DWG. NO. 171
4. RELAY FUNCTIONS — DWG. NO. 229

— OLD G.E. DWG. NO. 104R784 —

NOTES

△ VALVE RETURNS TO POSITION SHOWN IN EVENT OF AN ELECTRICAL OR PNEUMATIC FAILURE.

TITLE ELEM DIAG
PROCESS INSTRUMENTATION
CONTROL & INTERLOCK

RESEARCH REACTOR FACILITY
UNIVERSITY OF MISSOURI

DRAWN BY M. MCGINNESS | APPROVED
SHEET NO. 1 OF 3 | DWG. NO. 41 REV 5

REVISIONS

1	Vickie Snodgrass	5-5-72	5		
ADDITION OF 10MW EQUIP.					
2			6		
3			7		
4			8		

Figure 2.13

2006

2.5.5 Safety System (see Figure 2.14 - Dwg. No. 139).

This section (2.5.5) changed by response in Addendum 4, Appendix A.

The safety system circuit has nine input logic points designated TBI-1 through TBI-9. Only TBI-6 and TBI-7 logic circuits will be changed for operation at 10 MW.

2.5.5.1 TBI-6 Leg

Relay 2K28 contacts 13-14 will be added as a backup for relay 2K13 as discussed in Section 3.8.

Reactor loop low flow alarms 920A (K30-2) and 920B (K38-2) are bypassed at 5 MW by either K76-2 or K75-2 when flow through either heat exchanger is correct. When at 10 MW, bypasses are eliminated and both 920A and B must be satisfied to clear the scram input to TBI-6. Pool loop low flow alarms 920C (K31-2) and 920D (K69-2) are bypassed at 5 MW by either K76-2 or K77-2 when flow through either heat exchanger is correct. When at 10 MW, bypasses are eliminated and both 920 C and D must be satisfied to clear the annunciation and scram input to TBI-6. Refer to Figure 2.7, Reactor Control System for illustration of the interlock logic.

Presently three paddle flow switches (designated 928A, 928B and 929) are installed in the primary loop as a backup for a complete loss of flow. These flow switches have proven to be unreliable and are difficult to calibrate and repair. It is proposed that the paddle switches be replaced by three differential pressure switches; one sensor measuring the differential pressure across the core, and the others measuring the differential pressure across each of the primary loop heat exchangers. The core low differential pressure would actuate relays 2K13 and 2K28; either one would scram the reactor. The heat exchanger low differential pressure would actuate the control room annunciator to alert the operator of an abnormal flow condition.

In December 2006, DPS 928A and DPS 928B were removed as part of the primary coolant heat exchanger replacement project, which consisted of replacing the shell and tube-type heat exchangers with plate-type heat exchangers. 2006

2.5.5.2 TBI-7 Leg

The only changes are the addition of relays 2K1B and 1K26, both of which were discussed in Section 3.8.

References

1. "Specifications for Upgrading Mechanical Equipment at Research Reactor Facility," University of Missouri, June 26, 1972.
2. "Specifications for upgrading Electrical Equipment at Research Reactor Facility," University of Missouri, March, 1972.

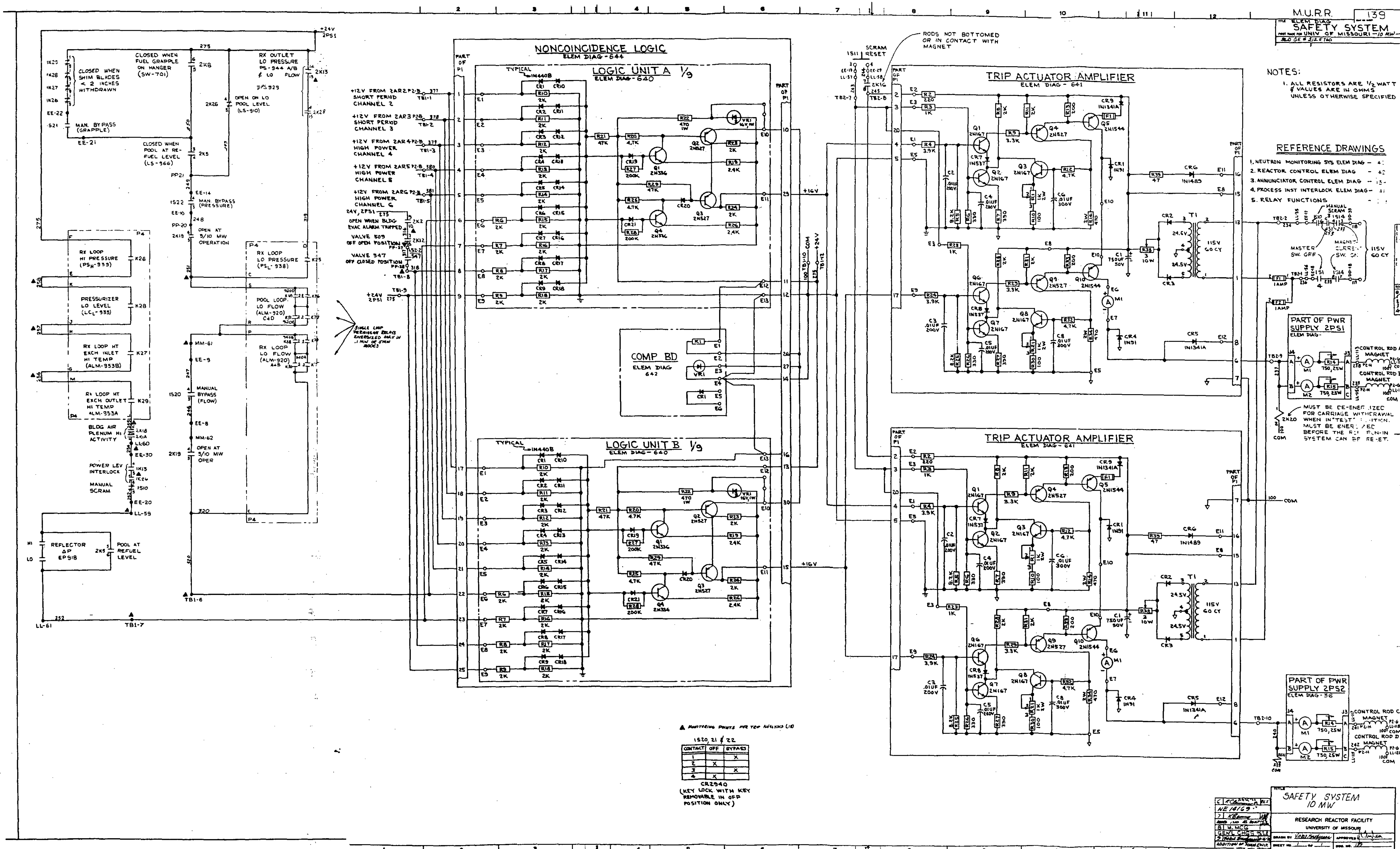

M.U.R.R. 139
SAFETY SYSTEM
NONCOINCIDENCE LOGIC
ELEM DIAG - 644
LOGIC UNIT A 1/9
ELEM DIAG - 640
LOGIC UNIT B 1/9
ELEM DIAG - 640
COMP BD
ELEM DIAG 642
TRIP ACTUATOR AMPLIFIER
ELEM DIAG - 641
SCRAM RESET
RODS NOT BOTTOMED OR IN CONTACT WITH MAGNET
PART OF PWR SUPPLY 2PS1
PART OF PWR SUPPLY 2PS2
NOTES:
1. ALL RESISTORS ARE 1/2 WATT & VALUES ARE IN OHMS UNLESS OTHERWISE SPECIFIED
REFERENCE DRAWINGS
1. NEUTRON MONITORING SYS ELEM DIAG
2. REACTOR CONTROL ELEM DIAG
3. ANNUNCIATOR CONTROL ELEM DIAG
4. PROCESS INST INTERLOCK ELEM DIAG
5. RELAY FUNCTIONS
REFLECTOR ΔP EPS18
POOL AT REFUEL LEVEL
SAFETY SYSTEM 10 MW
RESEARCH REACTOR FACILITY
UNIVERSITY OF MISSOURI

Figure 2.1-

3.0 SAFETY ANALYSIS FOR TEN MEGAWATT OPERATION

3.1 Introduction

The following analyses treat questions presented in Addenda One and Two to the Hazards Summary Report which require further study for ten megawatt operation as well as several safety questions which have arisen since the original licensing study for the MURR.

3.2 Justification of the Acceptability of the 125% Power Level Scram for 10 Megawatt Operation

Ref: Hazards Summary Report Addendum 1, Section 3.7

The primary reason for a high power scram is to protect the reactor core and associated equipment by providing a safe shutdown with no burnout in the event of overpower operation.

Using applicable burnout correlations, steady state burnout heat fluxes were computed for 10 MW conditions of operation. In order to be conservative, all conditions were set at their safety setpoint values. Also, for the final analysis, the most conservative (i.e., smallest) burnout heat flux was used. The average 10 MW heat flux was multiplied by 1.10 (for +10% uncertainty). Finally, the peak 11 MW heat flux was divided into the burnout heat flux to give the maximum, worst case, operating power multiplier for no burnout. The burnout power level thus obtained greatly exceeds the 25% overpower scram setting. Thus the 125% scram trip is justified.

To obtain the departure from nucleate boiling heat flux, ϕ_{DNB}, various correlations were used for 10 MW operating conditions. The applicable equations are summarized at the end of this section. A summary of the results is presented in Table 1.

Table 3.2.1 Burnout Correlations for 10 MW Conditions

Correlation	ϕ_{DNB} (BTU/ft^2-hr)
Bernath (2)	2.35×10^6
McAdams (3)	2.09×10^6
Gunther (4)	2.45×10^6
Lowdermilk (5)	2.75×10^6

The calculation of the hot channel-hot spot heat flux follows.

$$\phi_{(10\ MW)} = P_u P_r P_a P_\theta P_n \bar{\phi}_{(10\ MW)}$$

where: $\phi_{(10\ MW)}$ = worst case, hot spot heat flux (BTU/hr-ft^2)

$\bar{\phi}_{(10\ MW)}$ = average core heat flux at 10 MW (1.723×10^5 BTU/hr-ft^2)

P_u = reference multiplier for 10 MW, 10% uncertainty in power (1.10)

P_r = radial peaking factor (2.643 (1))

P_a = axial peaking factor (1.432 (1))

P_θ = circumferential peaking factor (1.04 (1))

P_n = non-uniform loading peaking factor (1.112 (1))

Thus for a worst case, non-uniformly loaded core, the 10 MW hot spot heat flux would be

$$\phi_{(10\ MW)} = (1.10)\,(2.643)\,(1.432)\,(1.04)\,(1.112)\,(1.723 \times 10^5)$$
$$= 8.30 \times 10^5$$

The maximum steady-state operating power for no burnout for the various correlations is thus

$$\text{Power} = 10\,\frac{(\phi_{DNB})}{(\phi_{(10MW)})}$$

Correlation	Power
Bernath	28.3 MW
McAdams	25.2 MW
Gunther	29.5 MW
Lowdermilk	33.1 MW

Thus, burnout may occur when a steady-state power of 25.2 MW is exceeded. The 125% scram trip is therefore reasonable.

Correlation Equations for Burnout Heat Flux Calculations

A. Bernath Correlation

$$\phi_{DNB} = \left[5710\left(\frac{De}{Dh}\right)^{0.6} + 48\,\frac{\overline{U_L}'}{(De)^{0.6}}\right]$$

$$\left[102.6\ \ln P' - 97.1\,\frac{P'}{P' + 15} - \frac{\overline{U_L}'}{2.22}\left(\frac{Dh}{De}\right)^{0.6} + 32 - \bar{t}_b\right]$$

where

De = equivalent diameter = 0.16/12 ft

Dh = wetted perimeter (of hot channel) = 0.121 ft

$\overline{U_L}'$ = fluid velocity = 19.2 fps (assumed)

P' = average pressure - 55 psia (assumed)

$\overline{t_b}$ = bulk fluid temperature = 175°F (assumed)

B. McAdams Correlation

$\phi_{DNB} = (400{,}000 + 4800\,\Delta t_{sub})(U_L')^{1/3}$

where Δt_{sub} = degrees of subcooling below saturation temperature of fluid

C. Gunther Correlation

$\phi_{DNB} = 7000\,(U_L')^{1/2}(\Delta t_{sub})$

D. Lowdermilk Correlation

$$\phi_{DNB} = \frac{270\ G^{0.85}}{De^{0.12}(L/De)^{0.85}}$$

where G = mass fluid flow rate in lb/hr-ft^2 surface

= 4.23×10^6 (19.2 fps at 175°F)

L = channel length = 2 ft

References

1. Julian, C., Evaluation of Power Peaking Factors in the MURR 6.2 Kg. Core, Internal Report, May 1972 (See Section 3.2 of this report).
2. Etherington, Nuclear Engineering Handbook, pp. 9-78, equation 54.
3. Ibid, equation 50.
4. Ibid, equation 51.
5. W. H. Lowdermilk, C. D. Lanzo, B. L. Siegel, "Investigation of Boiling Burnout and Flow Stability for Water Flowing in Tubes," NACA TN-4382 (September 1958).

3.3 Evaluation of Power Peaking Factors in the MURR 6.2 Kg Core

Reference: Hazards Summary Report Addendum 1, Section 3.9.

The original MURR design study, as presented in the Hazards Summary Report, presented power peaking values based on a 5.0 kilogram ^{235}U uniformly loaded core. Correction factors derived from one-dimensional calculations were used to extrapolate the power peaking values to various core configurations. During the course of the physics evaluation of the MURR's 6.2 kilogram uranium-aluminide core, a second study was performed which focused directly on the "as constructed" MURR and included the input of observed operating characteristics to date. The results reported (1) are outlined here.

The primary tools employed for reactor physics studies at the MURR are the Exterminator-II two-dimensional multigroup neutron diffusion code and the MURR four-group macroscopic cross section set. Exterminator-II is listed as abstract number 156 with the Argonne National Laboratory Code Center. The code is capable of modeling extremely complex reactor systems in X-Y, R-Z and R-θ geometries. This code calculates the spatial and energy dependent neutron flux, the effective multiplication factor (K_{eff}) and several other reactor parameters of the system modeled. This code has been in use at the MURR for two years and has produced consistently reliable predictions of various reactor physics parameters of the MURR.

The MURR 4-group cross section set was compiled by Internuclear Company and is listed in reference (2). The three epithermal groups were generated by the MUFT-4 (3) program and the thermal group was derived from Maxwellian averaged microscopic cross sections as presented in Appendix A of reference (2).

To obtain a conservative estimate for power peaking in a 6.2 Kg MURR core, we may consider an Exterminator R-Z calculation of the flux distribution in a "smeared" core, i.e., one in which the cross sections have been homogenized over the core region. For a clean critical situation with the control blades 12" withdrawn, the hot channel is at the inner fuel plate and the hot spot is 5" below core center line.

Define the axial peaking factor as

$$P_a = \frac{\text{Maximum Power in the Hot Channel}}{\text{Average Power in the Hot Channel}}$$

For a clean uniformly loaded core, this is just the ratio of the peak to average flux in the hot channel. This value is $P_a = 1.432$.

Define the radial peaking factor as

$$P_r = \frac{\text{Average Power in the Hot Channel}}{\text{Average Power in the Core}}$$

Again for the clean critical situation, this is just the ratio of the corresponding thermal fluxes. Exterminator calculates this value to be 2.643. The smeared core actually extends inside the physical location of the inner plate, however. To account for this, the radial flux profile at the axial peak position, as calculated by Exterminator, was examined and a correction factor derived which is just the ratio of the thermal flux at the actual innermost fuel plate location to the thermal flux at the inner edge of the smeared core. This factor was computed to be 0.84 and when multiplied by 2.643 gives a clean core radial peaking factor P_r of 2.220.

A third factor which must be included is the circumferential peaking factor. The thermal flux tends to peak in the aluminum side plates and the water gap between elements. For the inner fuel plate, this value has been computed to be $P_\theta = 1.04$ from modeling the core in R-θ geometry with no rods. Therefore, the overall power peaking factor for a clean, uniformly loaded, critical 6.2 Kg MURR core is $P_r P_a P_\theta = 2.220 \times 1.432 \times 1.04 = 3.306$.

A fourth factor which must be introduced is a correction for non-uniform loading. In order to use fuel economically and to maintain sufficient excess reactivity to reactivity to recover from unforeseen shutdowns under high xenon conditions, it has proven desirable to mix load old and new fuel elements. The MURR's UAl_x standard elements contain 775 grams ^{235}U each. The worst non-uniform power peaking situation foreseeable is a mixed loading of seven fully depleted elements, i.e., approaching the MURR burnup limit of 99 megawatt-days per element, and one clean 775-gram element.

This configuration was studied in detail at the MURR and values were obtained for nonuniform power peaking factors. Define P_n, the nonuniform power peaking factor, as

$$P_n = \frac{\text{R-}\theta\text{ peak/average power ratio in nonuniform core}}{\text{R-}\theta\text{ peak/average power ratio in corresponding uniformly loaded core}}$$

For the case of seven depleted elements and one clean element it was found that $P_n = 1.112$.

Then for the worst case of a clean critical MURR core in the worst possible nonuniform loading, the power peaking factor is $P_nP_rP_aP_\theta = 1.112 \times 2.220 \times 1.432 \times 1.04 = 3.676$.

This is considered to be a realistic power peaking factor derived from calculations on the exact expected MURR conditions and should be used for further heat transfer and fluid flow analysis on the MURR reactor under 10 megawatt power with a 6.2 kilogram ^{235}U core.

References

1. Letter of August 5, 1970, from the MURR to AEC Division of Reactor Licensing regarding a request for change in Technical Specifications.
2. "Missouri University Research Reactor Design Data," Volume II, Internuclear Company, Clayton, Missouri, 1962.
3. "MUFT-4, Fast Neutron Spectrum Code for the IBM 704," WAPD-TM-72, Bohl, et. al.

3.4 <u>Evaluate the Consequences of the Valve to the Inpool Convective Loop Failing to Open Following a Loss of Primary Coolant Flow and a Reactor Scram</u>

Ref: Hazards Summary Report Addendum 1, Section 3.13

It is assumed that after 10 MW operation for a period of time such that fission product saturation is reached, the reactor experiences a scram as a result of loss of primary flow with the following events occurring:

1. V-507 A and B close.
2. V-543 A and B open.
3. V-546 fails to open, prohibiting the dissipation of decay heat through the inpool heat exchanger.

For these conditions, it will be shown that although there may be local boiling, there will be no net formation of steam in the reactor vessel and associated piping. Temperatures will remain well below 1184°F at which temperature fission product release from the fuel is appreciable (1).

The fission product energy release rate from a reactor which has been operated to near equilibrium fission product concentration is equal to approximately 6% of the operating power (2). Thus, the core will be producing decay heat at a rate of 0.6 MW which is equal to 568.7 BTU/second. Of this heat 62.7% or 357 BTU/sec is dissipated in the core region (3). It will be assumed that the average temperature of the core coolant at the time of the scram is 175°F.

The amount of water in the pressure vessel and associated isolated piping under the proposed accident condition is approximately 650 pounds. The saturation temperature for the water is 235°F (4). Therefore, the amount of heat input required to raise the total volume of water to saturation is

$$650 \text{ lb} \times \frac{1 \text{ BTU}}{\text{lb - °F}} \times (235°\text{F} - 175°\text{F}) = 39{,}000 \text{ BTU}$$

Note that raising the water to its saturation temperature does not imply boiling since the saturation temperatures and pressures apply to equilibrium at a flat interface between vapor and liquid. Adjacent to a heated surface, the liquid superheat may be 30°F to 50°F (5).

In order to prevent the net formation of steam in the pressure vessel and associated piping, the heat transfer rate to the pool must be adequate (when the contained water reaches saturation temperature) to remove any further heat generated. It will be assumed that the heat release rate is constant until the saturation temperature is reached. This is conservative in that actually the heat production rate decreases with time (2). The minimum time to saturation may be calculated as follows.

$$\frac{39{,}000 \text{ BTU required}}{357 \text{ BTU/sec}} = 109.2 \text{ sec}$$

The heat release rate 109.2 seconds after shutdown is equal to

$$P/P_O\,(= 0.3) * P_O\,(=0.6) * 946.5\,\frac{\text{BTU/sec}}{\text{MW}} = 170.4 \text{ BTU/sec (5).}$$

The film coefficient for heat transfer from the approximately 200°F pressure vessel to the 120°F pool is 233 BTU/hr-ft^2 °F (3,6). The total heat transfer area is 54 ft^2 (3). Hence the required average piping outside wall temperature for heat removal of 170.4 BTU/sec is

$$\left(T_{\text{outerwall}} - 120°\text{F}\right) = \frac{170.4 \text{ BTU/sec x } 3600 \text{ sec/hr}}{54 \text{ ft}^2 \text{ x } 233 \text{ BTU/hr - ft}^{2\circ}\text{F}}$$

$$T_{\text{outer wall}} = 168.75°\text{F}$$

The inner wall temperature can be calculated for k = 1390 (BTU-in)/(ft^2-Hr-°F) thickness of aluminum (7).

$$T_{\text{inner}} - T_{\text{outer}} = \frac{q\,x}{kA} = \frac{170.4 \text{ x } 3600 \text{ x } 0.39 \text{ in}}{1390 \text{ x } 54}$$

$$T_{\text{inner wall}} = 171.93°\text{F}$$

Thus, the inner wall temperature necessary to transfer the required amount of heat is approximately 63°F below the saturation temperature of the contained water. Therefore, there should be no net formation of steam.

Next, consider the possibility of fuel meltdown caused by departure from nucleate boiling in the core and the corresponding high fuel temperatures resulting from the reduction in heat transfer across the vapor film. It has been shown (1) that for temperatures less than 1184°F, there is no appreciable release of fission products from UAL_x fuel.

For the accident configuration proposed, the core, pressure vessel, and associated piping may be considered to be a natural circulation vaporizer (thermosiphon reboiler).

The circulation in this type of vaporizer is due to the difference in density between heated liquid (perhaps containing vapor) and a cooler liquid. Assume an equivalent diameter for flow of the hot liquid of 0.84 inches (4 channels) and a temperature difference of approximately 30°F from hot to cold channels. Under similar conditions, Perry (8) indicates that the maximum heat flux before departure from nucleate boiling is approximately 55,000 BTU/hr-ft^2. The total heat transfer area in the core is 184.28 ft^2 (9). Thus, the total heat flux for departure from nucleate boiling under these conditions is 2856.3 BTU/sec which is well above the maximum initial heat generation rate of 568.7 BTU/sec. Thus, film boiling should not occur and fuel element temperature should remain well below the fuel melting point.

<u>References</u>

1. In-1437, Metallurgy and Materials Science Branch Annual Report, Fiscal Year 1970, pp 27-34.
2. Glasstone S and A Sesonske, <u>Nuclear Reactor Engineering</u>, Van Nostrand Co, 1967 ed, p 100.
3. Hazards Summary Report, Addendum I, February 1966, p 40.
4. Keenan and Keys, <u>Thermodynamic Properties of Steam</u>, p 31.

5. Bennett CO and JE Meyers, Momentum Heat and Mass Transfer, McGraw-Hill, 1962, p 355.
6. Perry's Chemical Engineer's Handbook, 4th Ed, Table 10-8.
7. Ibid, pp 23-40.
8. Ibid, pp 10-23.
9. Pearce WR and ER Schmidt, "Reactor Heat Transfer and Fluid Flow--Final Design Conditions," Internuclear TM-WRP- 62-16, September 1962.

3.5 Analysis of Rapid Step Reactivity Insertions from Full Power in the MURR

Previous studies (1,2) have evaluated extensively the expected results of a sudden positive step insertion of reactivity in the MURR. Addendum II to the Hazards Summary Report (2) concluded that the MURR could withstand a positive step insertion of 0.008 ΔK without fuel damage. This study was based on an initial power level of 10 MW, nominal flow, pressure, and reactor temperature conditions and the calculated core temperature and void coefficients of -7×10^{-5} $\Delta K/K/°F$ and -2×10^{-3} $\Delta K/\%$ void respectively. During the initial startup and calibration of the MURR, these two parameters were observed to differ from the calculated values, and the MURR technical specifications were changed to require these numbers to be more negative than -3×10^{-5} $\Delta K/K/°F$ and -1.2×10^{-3} $\Delta K/\%$ void respectively. Core voiding and temperature increase are the two major negative reactivity feedback mechanisms which halt the rapid power escalation following a positive step reactivity insertion, therefore it was concluded that the maximum tolerable step insertion and hence the maximum experiment worth, should be reduced to +0.004 ΔK.

During the low power testing program for the MURR's 6.2 kilogram uranium-aluminide core, the temperature and void coefficients were carefully remeasured and found to be very close to the original calculated values. The quantities observed were -7.0×10^{-5} $\Delta K/K/°F$ and 2.51×10^{-3} $\Delta K/\%$ void respectively (3).

As part of the safety evaluation for power upgrade to 10 mw, a third study was undertaken to determine the maximum step reactivity insertion the MURR can withstand with no core damage. The MURR was modeled with a transient code ideally suited for this type of study; the Chic-Kin (4) computer code originating at Bettis Atomic Power Laboratory. This code combines hydraulic and heat transfer analysis with reactor kinetics to predict the power, temperature and pressure changes during reactor transients for either pin or plate type fuel. Rather than considering transients from nominal conditions, for this study the reactor was modeled with all critical parameters set to their scram values, i.e., the worst possible conditions for full power operation of the MURR (5). From previous work (6), the most conservative steady state power level at which burnout could occur was determined to be 25.23 mw. Therefore for a power transient starting from 11 mw, fuel plate failure would be conservatively predicted at steady state operation with power increased to a factor of 2.3 of its initial value. Figure 3.5.1 presents the normalized power increase factor versus time after the step insertion from the Chic-Kin code. Consistent with previous studies (1,2), it may be assumed that the MURR fuel can withstand the prompt power burst, since it is of such short time duration, and that fuel failure will occur at the hot spot only when the reactor continues in sustained operation with a normalized power increase factor greater than or equal to 2.3.

Figure 3.5.1 thus indicates that the MURR can withstand a positive step insertion of 0.006 ΔK. Experimental evidence indicates that one of the two short period trip circuits or one of three high power trips in the MURR safety system will initiate a scram within at least 115 milliseconds. Sufficient redundancy certainly exists to ensure that a post burst scram will occur. Experimentally observed (7) rod worth data and rod drop times enabled the modeling of a scram at 150 milliseconds after the step insertion by the Chic-Kin code. Figure 3.5.2 presents the expected reactivity insertion rate versus time after initiation of the scram. Figure 3.5.1 demonstrates that such a scram will safely shut the reactor down with no fuel damage.

Assumed parameters for this study are a core temperature and void coefficient of -6.0×10^{-5} ΔK/°F and -2.0×10^{-3} ΔK/% void. Experimental results have shown (3) that the MURR 6.2 kilogram core has temperature and void coefficients more negative than those cited. In summary, under the worst possible conditions, the MURR reactor can withstand a positive step insertion of +0.006 ΔK reactivity without core damage.

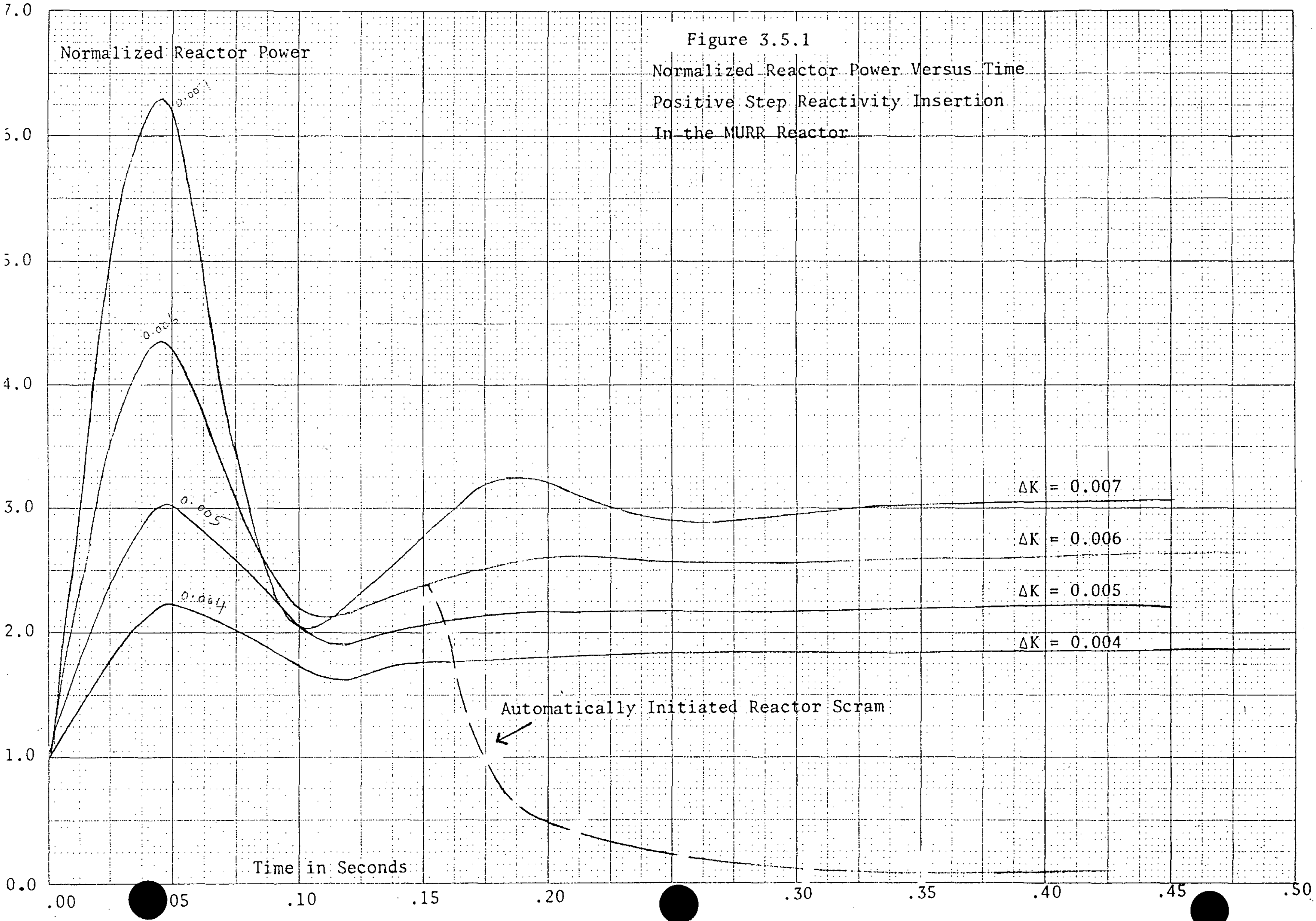
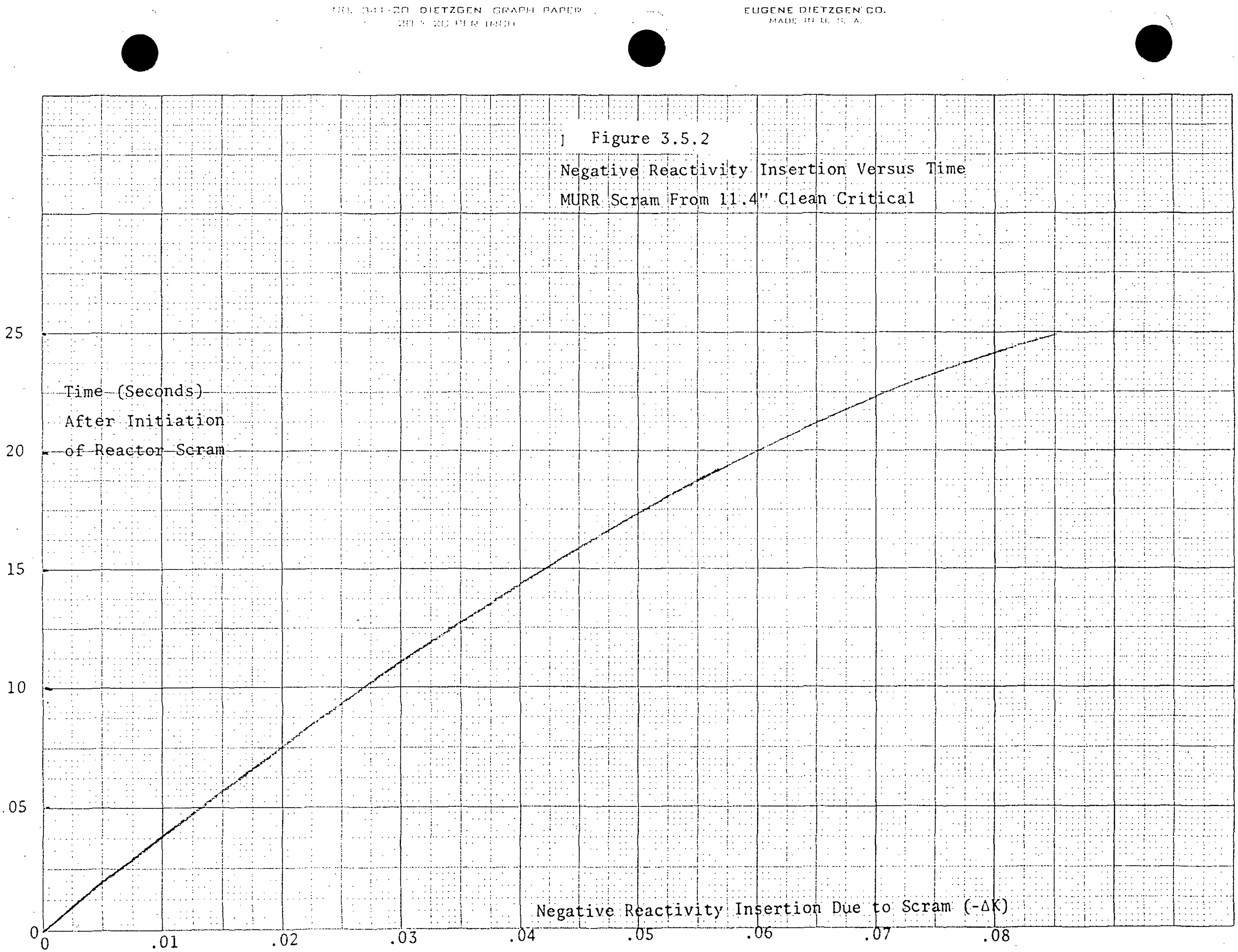


Figure 3.5.1
Normalized Reactor Power Versus Time
Positive Step Reactivity Insertion
In the MURR Reactor
Normalized Reactor Power
0.007
0.006
0.005
0.004
ΔK = 0.007
ΔK = 0.006
ΔK = 0.005
ΔK = 0.004
Automatically Initiated Reactor Scram
Time in Seconds
7.0
6.0
5.0
4.0
3.0
2.0
1.0
0.0
.00
.05
.10
.15
.20
.30
.35
.40
.45
.50



NO. 341-20 DIETZGEN GRAPH PAPER
20 X 20 PER INCH
EUGENE DIETZGEN CO.
MADE IN U. S. A.
Figure 3.5.2
Negative Reactivity Insertion Versus Time
MURR Scram From 11.4" Clean Critical
Time (Seconds)
After Initiation
of Reactor Scram
25
20
15
10
.05
0
0
.01
.02
.03
.04
.05
.06
.07
.08
Negative Reactivity Insertion Due to Scram (-ΔK)

References

1. Hazards Summary Report, Addendum One, pp 53-79.
2. Hazards Summary Report, Addendum Two, pp 23-58.
3. Low Power Testing Program for the Missouri University Research Reactor 6.2 Kilogram Core, p 74.
4. JA Redfield, CHIC-KIN - A Fortran Program for Intermediate and Fast Transients in a Water Moderated Reactor, United States Atomic Energy Commission Report WAPD-TM-479 (1965).
5. Section 3.9, this report.
6. Section 3.2, this report.
7. Low Power Testing, Op. Cit., p 49.

3.6 Evaluation of the Need for An Emergency Shutdown System

The need for an emergency shutdown system was initially evaluated as the answer to question 3.23 (page 105) in Addendum I of the Hazards Summary Report and again as the answer to question 14 (page 37) in Addendum II of the Hazards Summary Report.

As indicated in these Addenda the need for an emergency shutdown system would be dictated by the reliability (or lack thereof) of the control rods under normal operation conditions and an analysis of those conditions which would cause binding of the rods.

This question has been thoroughly analyzed and it is concluded that an emergency shutdown system is not required. A summary of the analysis leading to this conclusion is presented below.

The reliability of the control rods has been demonstrated by four years of operation with no control rod failure. The original design of the control rod offset mechanism did prove to be faulty, but these mechanisms were removed and replaced with a new type mechanism in early 1968. The reactor has experienced over 600 startups and 19,000 operating hours since the mechanism replacement with no control rod problems. To ensure continued reliability of the control rods the following preventive maintenance program is in effect:

1. Annually the control rods and their offset mechanisms are thoroughly inspected. The inspection of the control blades includes checks for warpage, cladding integrity, wear marks and blade curvature. The offset mechanism inspection begins with a complete disassembly of all moving parts. The guide tube bearings are inspected and any questionable bearings are replaced. All bearing surfaces are inspected for excessive wear or other conditions which could cause binding. Before the mechanism and blade are reinstalled the blade is adjusted on a test stand to ensure that it will ride in the center of the rod gap and has a minimal (<.050") run out. This inspection is done each quarter for one of the four rods so that each rod (and mechanism) are inspected annually.

2. Semi-annually the drive mechanism for each rod is thoroughly cleaned, inspected and regreased.

Conditions which could cause binding of the control rods are:

1. Distortion of the pressure vessel and/or beryllium reflector,
2. Debris collecting in the control blade gap or off-set mechanism. As discussed in Addendum II (page 37), the distortion of the pressure vessel due to pressure surges is highly unlikely because of the pressure relief valves installed in the primary coolant system. Distortion of the beryllium reflector, however, is possible due to the neutron (E>1 Mev) induced growth of the beryllium.

Studies by Idaho Nuclear (1) have shown that at temperatures less than 150°C beryllium exhibits a constant growth in relation to neutron fluence (E>1 Mev) of 0.02% per 10^{21} n/cm^2. The peak neutron (E>1 Mev) flux in the beryllium reflector is 5.08 x 10^{13} n/cm^2-sec at 5 MW. Using this flux, the above growth rate and the operating history should have experienced a radial growth of only .002". At 10 MW the reflector should experience an annual radial growth of only .001". The possibility of bowing due to non-uniform growth has not been analyzed because of the relatively large thickness (2.71") of the reflector. The clearance of the control blade gap is checked quarterly during the control rod inspections by a feeler gauge, and no gap reduction has been observed to date. Since the blade is .250" thick and the gap is .5625" thick, this quarterly check should provide an adequate warning that the blade gap is being reduced well before the gap is reduced sufficiently to cause binding.

The possibility of debris collecting in the offset mechanism is essentially eliminated by an aluminum shield tube which fits over the offset mechanism and encloses the drive rod connecting the lead screw and the offset mechanism. There is no protective cover over the blade gap, but the probability that more than one gap be plugged simultaneously is very remote. A strictly enforced precaution that people who are working near the pool must empty their pockets and use safety lines on all tools further reduces the probability of debris collecting in the blade gaps or offset mechanism.

Physics tests have demonstrated that the 6.2 Kg fuel loading which is now being used adds sufficient reactivity such that it is now necessary for only three rods to simultaneously (completely) stick before shutdown capability is lost. This is different from the four stuck rods necessary with a 5.2 Kg loading. This analysis has shown, however, that the probability of one rod sticking is quite remote and thus the probability of three rods sticking is extremely remote.

Since the need for an emergency shutdown system is not apparent, the plans for constructing such a system (Addendum II, page 38) have been canceled.

References

1. "Reactor Engineering Branch Annual Report Fiscal Year 1968," Corporation, IN-1228, February 1969.

3.7 Atmospheric Dilution of Gaseous Effluents and Effective Stack Height

Page 29 of Addendum Two, Hazards Summary Report, May 1966, presents calculations relating to atmospheric dilution factor for the Research Reactor Facility. Various parameters of the previous study which were based upon calculated values can now be replaced with measured values resulting from operating experience.

The measured discharged activity level of Argon-41 at five mw operation is approximately 5×10^{-6} µCi/ml as it leaves the facility stack. When the pneumatic tube system is first turned on this activity increases to about 1.3×10^{-5} µCi/cc for about five seconds then the activity returns to normal. At ten mw power level it is expected that these values will double.

The reactor has operated an average of about 96 hours per week during the past year, therefore the yearly average Argon-41 concentration at the stack calculates to be approximately 3×10^{-6} µCi/ml. The quoted values are rounded off to one significant figure because of error due to normal variations in wind speed and counting statistics. On the basis of standard deviations the error is about 25%.

Because of the momentum of the air as it leaves the building exhaust stack, the effective stack height is actually higher than fifty-five feet and is a variable depending upon wind conditions. By using smoke and streamers, the effective stack height has been experimentally determined as a function of wind velocity. The result of this measurement is shown plotted in Figure 3.7.1. Buoyant forces due to temperature difference between atmospheric and building air were considered to be negligible.

The concentration of Argon-41 to persons inside the reactor site exclusion area will be the greatest when the wind velocity is such as to keep the plume closest to the ground.

From atmospheric data obtained from the U.S. Weather Bureau, the average wind velocity in Columbia, Missouri, over a ten year period is recorded as 10 mph. The Research Reactor is located in a small valley south of Columbia and is affected by the terrain to the extent that wind patterns recorded by instruments on the reactor building show an average wind velocity of 5.8 mph over a ten month period.

Results are calculated using the standard Gaussian diffusion equation for average conditions as follows:(1)

$$\chi = \frac{Q}{2\pi\,\sigma_z\,\sigma_y\,\bar{u}}\exp\left[-\frac{1}{2}\left(\frac{y^2}{\sigma_y^2}+\frac{h^2}{\sigma_z^2}\right)\right]$$

χ = downwind concentration in $\mu Ci/m^3$

Q = source strength in $\mu Ci/sec$

$\bar{u}$ = average wind velocity

y = horizontal distance from plume centerline

h = effective stack height

Values of σ_y and σ_z are calculated from Sutton's equations as follows:

$$\sigma_y^2 = \frac{1}{2} C_y^2 x^{(2-n)}$$

$$\sigma_z^2 = \frac{1}{2} C_z^2 x^{(2-n)}$$

x = downwind distance from source in meters

C_z and C_y are atmospheric diffusion coefficients

n = atmospheric stability parameter

Under average conditions the atmospheric dilution as a function of distance from the source and at ground level is shown in Table 1 where:(2)

$\bar{u}$ = 2.5 meters/sec

n = 0.15

h = 23 meters

C_y = 0.45 meters

C_z = 0.8 meters

Table 3.7.1

Atmospheric Dilution Factor

Distance from Source	Dilution Factor
50 meters	7×10^3
100 meters	1.5×10^4
150 meters	3×10^4
200 meters	5×10^4
500 meters	3×10^5

The result of this calculation shows that under average conditions, without considering factors such as wind variability, terrain and building effect to increase the dilution factor, the average radioactive concentration of Argon-41 is well below MPC limits for restricted and nonrestricted areas.

It is also to be noted that the University of Missouri owns land such that public dwelling areas are not closer than 1/3 mile from the reactor building.

Gaseous activity above background has never been detected around or in the near vicinity of the reactor building.

References

1. Conley LA, Croke EJ, et al, Isopleth-Area Tables, ANL/ES-8, Meteorology, October 1971.
2. International Symposium on Fission Product Release and Transport under Accident Conditions, CONF-650407, Vol. 1, USAEC, Oak Ridge National Laboratory, April 1965.

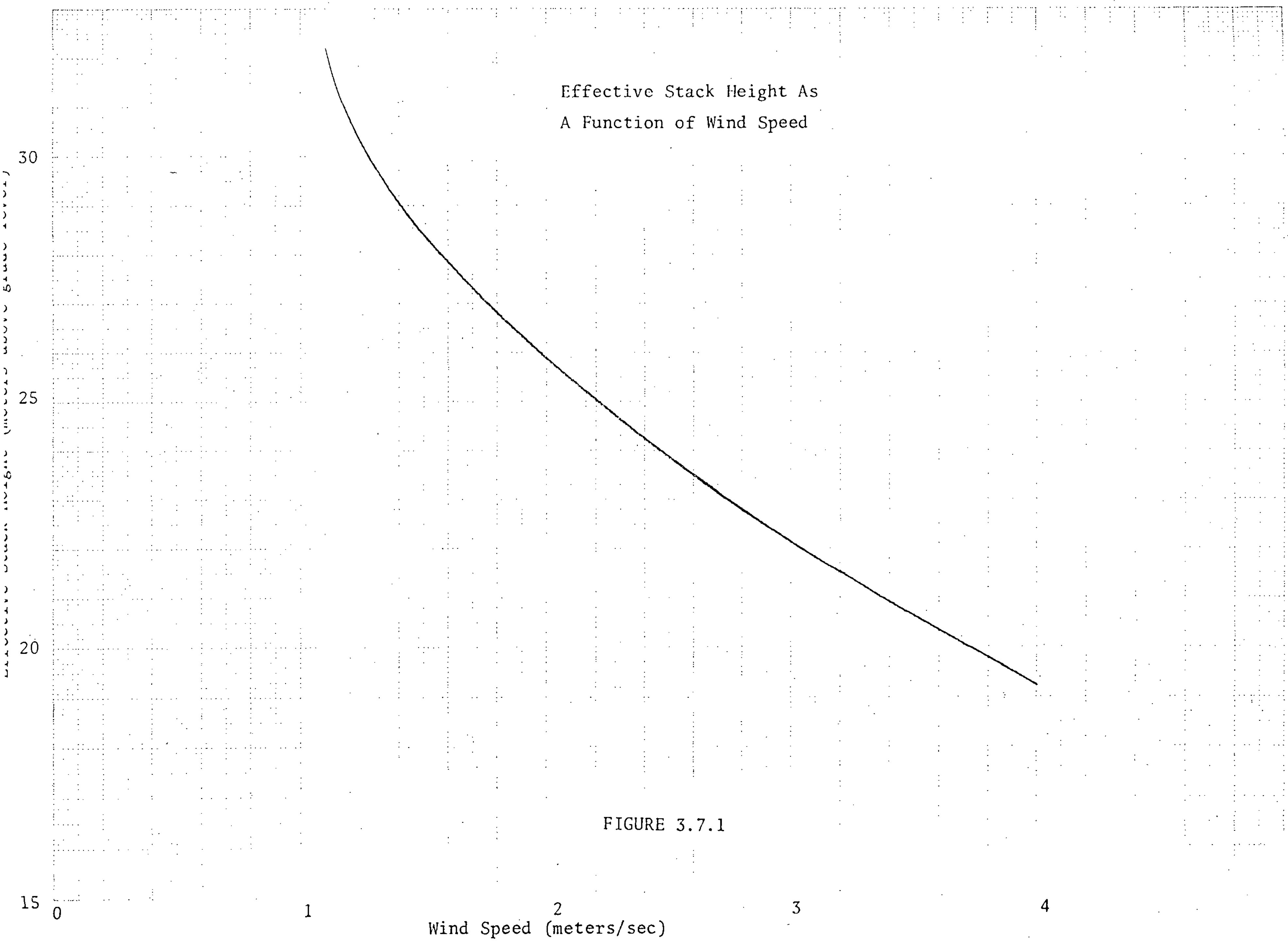

Effective Stack Height As
A Function of Wind Speed
30
25
20
15
0
1
2
3
4
Wind Speed (meters/sec)

FIGURE 3.7.1

3.8 Single Failure Criterion Analysis

3.8.1 Introduction

As part of the overall safety review conducted for upgrading the reactor to 10 MW, the plant protection systems were analyzed for compliance with IEEE Standard 279-1971: Criterion for Protection Systems for Nuclear Power Generating Stations. It was recognized that the difference in construction and operating requirements between the MURR and a power operating station would result in the Standard not being totally applicable, however, many sections could be used as a guide. Particular attention was paid to meeting the single failure criterion in the following systems:

1. Nuclear Instrumentation
2. Safety System
3. Rod Control
4. Reactor Control System
5. Building Isolation and Evacuation
6. Emergency Power
7. Process Instrumentation and Control

The remainder of this section discusses the plant changes made or proposed to cause all protective systems to meet the single failure criterion.

3.8.2 Nuclear Instrumentation

The nuclear instrumentation system as now constructed and as proposed for 10 MW operation (see Section 2 for 10 MW alterations) complies with the single failure criterion.

3.8.3 Safety System

The power level interlock circuit (see Section 2, Reactor Control System) is single failure prone by relay 1K13 when in the 0.1 MW mode of operation. Failure of 1K13 could allow operation at power levels in excess of 100 kW should administrative procedures be violated. Relay 1K26 is proposed to be added as a backup to 1K13. Relay 2K13 which causes an automatic isolation of the primary cooling system upon sensing a low pressure of flow condition has no backup. Therefore, relay 2K28 will be added to the process control and safety systems to operate simultaneously with relay 2K13.

Automatic initiation of a reactor isolation and scram as a result of high radiation in the building air plenum or above the pool occurs through operation of relay 2K1A. Also, 2K1A was a normally energized relay and therefore not fail-safe on loss of electrical power. An additional relay, designated 2K1B, has been added as a backup to relay 2K1A and the circuit logic changed to make both relays fail-safe on loss of power.

Completion of the above changes satisfies single failure criterion for the safety system.

3.8.4 Rod Control

The rod control system satisfies the single failure criterion.

3.8.5 Reactor Control System

The reactor control system will satisfy the single failure criterion with the addition of relay 1K26 discussed in paragraph 3.8.3

3.8.6 Building Isolation and Evacuation System

The Building Isolation and Evacuation System contained several components which were not fail-safe on loss of electrical power and which could cause failure of the system by a single failure. Refer to the Annual Report for 1972 for a discussion of the system modification to correct these deficiencies.

3.8.7 Emergency Power

The electrical power distribution system complies with the single failure criterion.

3.8.8 Process Instrumentation and Control

With the addition of relay 2K28 described in paragraph 3.8.3 the Process Instrumentation and Control Systems will meet the single failure criterion.

3.9 Steady State Heat Transfer and Hydraulic Analysis of the MURR Primary Cooling System

3.9.1 Introduction

The MURR design data (1) presented a lengthy analysis of the heat transfer and hydraulics of the reactor for various conditions but since the design data model differs slightly from the "as constructed" piping, it was felt necessary to confirm the results of the first study during the analysis for power upgrade to 10 mw. The following report is intended to outline the methods used and present the results and conclusions of this study.

3.9.2 Hydraulic Analysis

The plate heat transfer study determines the primary coolant pressure necessary to suppress boiling in the reactor core under the most adverse conditions. The pressurizer tank is, however, far removed from the core so it is necessary to analyze the relationship between pressure applied to the system by the pressurizer and actual pressure in the core. Examination of the primary piping system reveals the following hydraulic model between these two points.

1. Pressurizer outlet
2. 5 feet of 8 inch pipe
3. Expansion from 8 inch to 12 inch pipe
4. 80 feet of 12 inch pipe
5. Four 12 inch 90 degree elbows
6. Three 12 inch 45 degree elbows
7. One 12 inch butterfly valve (valve 507B)
8. One 12 inch swing check valve (valve 502)
9. Entrance to pressure vessel (treat as 200 pipe diameters)
10. 4.375 feet of annular pressure vessel
11. Entrance to plate type fuel elements
12. 25.5 inches of fuel element plates

The friction loss in this piping was found from standard turbulent flow equations with friction factors determined from the Moody Chart (2) and absolute roughness values assumed to be those used in the design data (1). Figure 3.9.1 relates the calculated pressure drop from pressurizer to core exit as a function of primary coolant flow. Under normal flow conditions of 3600 gpm, the pressure drop from pressurizer to core outlet was found to be 19.4 psi.

Steady State Heat Transfer

A computer program was written to analyze a single MURR coolant channel and compute the steady state temperature distribution. The required inputs are coolant flow rates, reactor power level, power peaking factors, coolant inlet temperature, and several other parameters. The outputs are axial temperature profiles in the coolant, plate surface, clad-fuel interface, and fuel centerline. The program approximates the continuous heat transfer process by assuming uniform heat generation in each of 24 discrete axial sections one inch in length. An axial normalized power profile calculated from the Exterminator II (4) two-dimensional neutron diffusion code for the case of a clean critical MURR core is input to the heat transfer program. An average value for this power generation function for the axial step in question is multiplied by radial, azimuthal, and non-uniform fuel loading power peaking factors and the average plate surface heat flux in the MURR for 11 megawatt operation to obtain the heat flux input to the coolant from each fuel plate for the axial step in question. The product of this heat flux and the incremental plate surface area is then the heat input to the coolant in a single axial section. Values of the coolant specific heat and mass flow rate are computed and a heat balance performed to determine the coolant temperature at the end of the axial step. The resulting coolant temperature and heat flux are used to calculate the plate surface temperature using a convective heat transfer coefficient calculated from the Dittus-Boelter equation (5). Plate internal interface temperatures are then computed by solution of the conduction heat transfer Fourier equation. The same process is then repeated for successive axial steps until the temperature profiles for the entire coolant channel length have been computed. Among the numerous configurations considered, the worst case situation is the most important. This case assumes that the primary coolant enters the core at the reactor inlet high temperature scram point with the flow reduced to the low flow scram point. Pressurizer pressure was reduced to its scram value, the reactor power was taken to be 110% of nominal, i.e., 11 mw, and the power peaking factors used were for the worst projected non-uniform loading (3). The hot channel between the two innermost plates was considered to be reduced to 0.072".

Figure 3.9.2 presents the results of this study as a parametric graph of reactor inlet water temperature versus pressurizer pressure as a function of coolant flow rate. These curves represent combinations of the three parameters required to cause onset of nucleate boiling in the MURR under worst conditions. Boiling is conservatively assumed to occur when the plate temperature is 20°F above local coolant saturation temperature.

A safety limit for the MURR is defined to be an extreme variation in one of these three parameters which causes boiling while the other two parameters remain at their nominal operating value. Examination of figure 3.9.2 reveals the following safety limits.

Variation	Reactor Inlet Temperature (°F)	Pressurizer Pressure (psig)	Primary Coolant Flow (gpm)
Nominal	102	60	3600
Temperature	181	60	3600
Pressure	120	30.5	3600
Flow	120	60	2150
Scram Value	155	50	1500(1)

(1)Received from either loop

In order to avoid these safety limits it is recommended that for 10 megawatt operation, the reactor inlet high temperature scram and pressurizer low pressure scram be retained at their present values of 155°F and 50 psig respectively, and that the reactor low flow scram be set at 1500 gpm in either loop. These values would imply a worst case plate surface temperature at the core exit of 295°F or 18°F above saturation temperature. This is well within the design criteria of maintaining the hot spot temperature at no greater than 20°F above local saturation temperature which was assumed in the original design study (1) of the MURR.

Again, it must be emphasized that these conditions represent the worst case for all parameters involved.

For the more realistic case of nominal 3600 gpm flow, 60 psig pressurizer pressure and an 0.080" coolant channel, the hot spot plate surface temperature will be 279.5°F or approximately 8°F below local saturation temperature. This still assumes 11 megawatt reactor power, 155°F reactor inlet water temperature, and the most adverse power peaking conditions.

A similar analysis was done for the present 5 megawatt operating mode of the MURR. Figure 3.9.3 presents the parametric relationship between pressurizer pressure, reactor inlet temperature, and coolant flow rate. From this curve, the following safety limits and safety set points, i.e., scram points, were derived.

Variation	Reactor Inlet Temperature (°F)	Pressurizer Pressure (psig)	Primary Coolant Flow (gpm)
Nominal	120	60	1800
Temperature	210	60	1800
Pressure	120	9.5	1800
Flow	120	60	900
Scram Value	155	50	1500

For all conditions nominal, the heat transfer program predicts a hot spot temperature of 244°F. This is 58°F below local saturation temperature, thus the MURR reactor is far subcooled during 5 megawatt operation.

In summary, this study indicates that even under the most unfavorable conditions of all parameters, the MURR reactor will not experience boiling of the primary coolant. There are, of course, many conservative assumptions made in this study, however, the largest safety margin is derived from the fact that the most adverse power peaking conditions can only occur for a brief period at the start of core life. Thus, the MURR is well protected from boiling as a result of overpower operation under any conceivable set of circumstances.

References

1. Emmons AH, EL Cox and DG Fitzgerald, University of Missouri Research Reactor Facility Hazards Summary Report, University of Missouri, 1965, pp 5-14 to 5-31.
2. El-Wakil MM, Nuclear Power Engineering, McGraw-Hill, 1962, p 536.
3. Julian CA, Power Peaking in the MURR 6.2 Kg Core, MURR Internal Report, Section 3.2.
4. "Exterminator-2: A Fortran IV Code for Solving Multigroup Neutron Diffusion Equations in Two Dimensions," ORNL-4078, April 1966.
5. El-Wakil, op. cit., p 250.

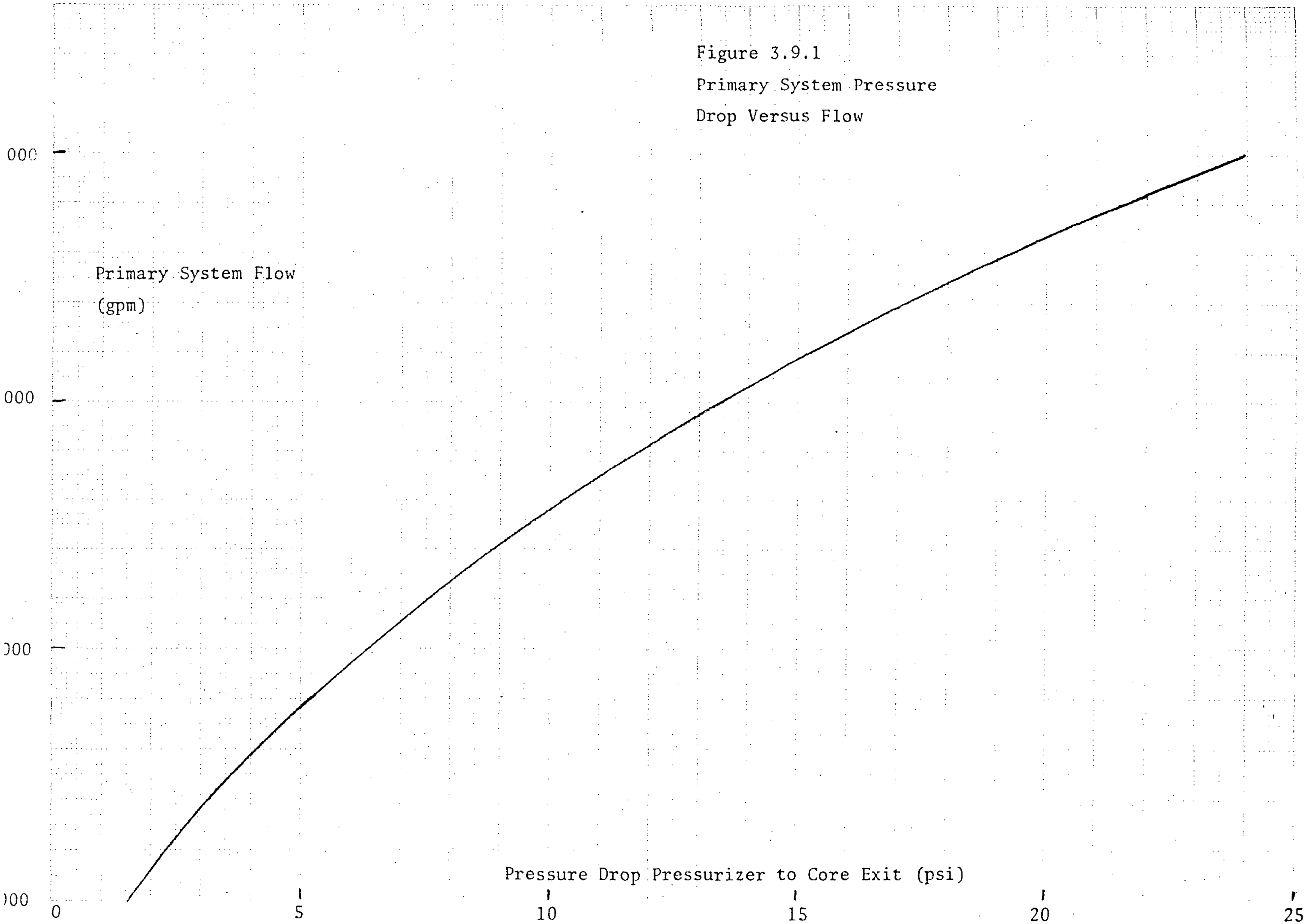


Figure 3.9.1
Primary System Pressure
Drop Versus Flow
Primary System Flow
(gpm)
000
000
)00
)00
Pressure Drop Pressurizer to Core Exit (psi)
0
5
10
15
20
25

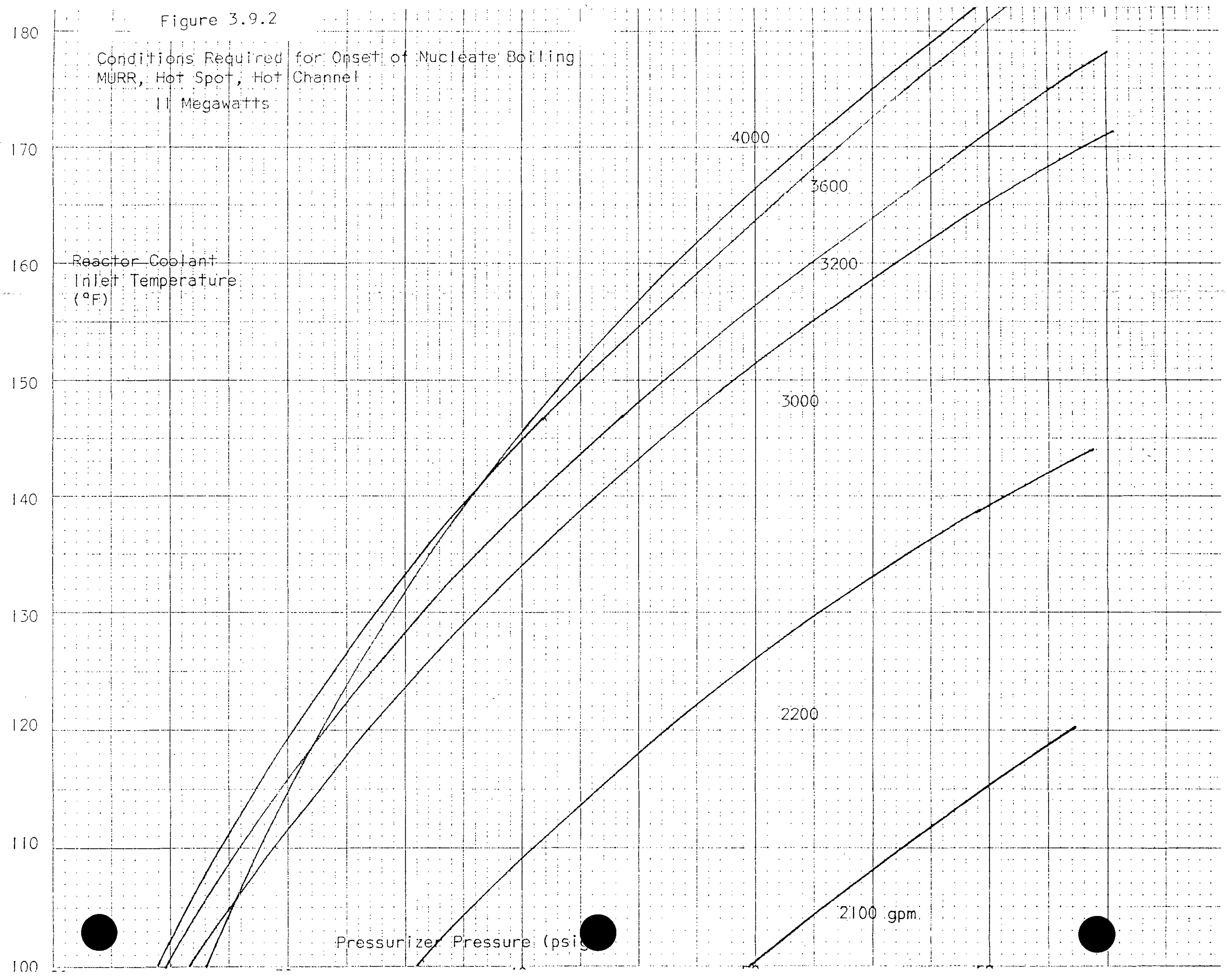


Figure 3.9.2

Conditions Required for Onset of Nucleate Boiling
MURR, Hot Spot, Hot Channel
11 Megawatts

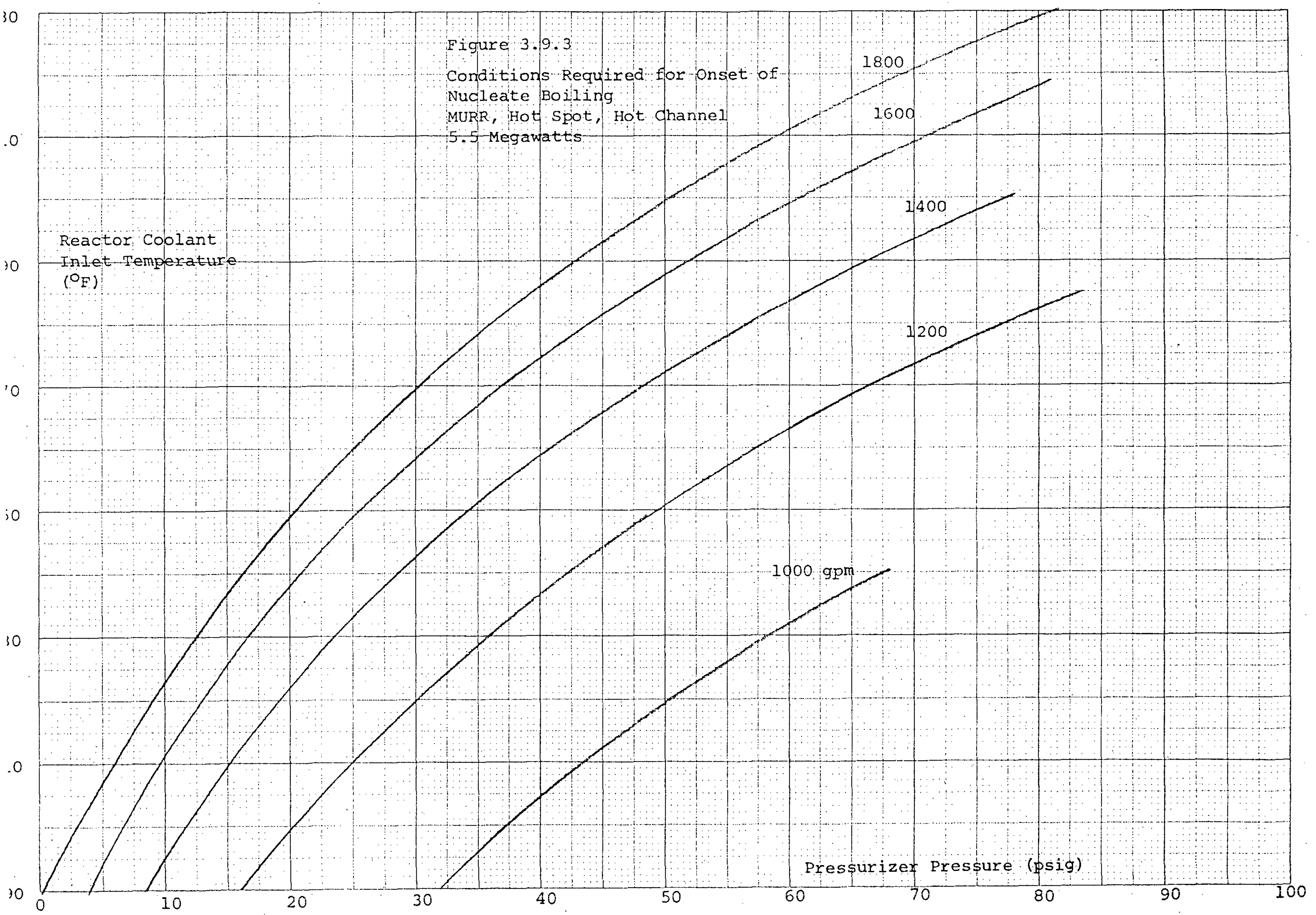


Figure 3.9.3

Conditions Required for Onset of
Nucleate Boiling
MURR, Hot Spot, Hot Channel
5.5 Megawatts

2000

3.10 Safety Analysis of the Center Test Hole

The MURR annular core surrounds an experimental irradiation facility designated as the center test hole. At ten mw operating power the peak unperturbed flux in this region is expected to be approximately 7×10^{14} nv. The center test hole is cooled by bulk pool water down flowing to the reflector plenum. The relatively stable bulk pool prevents the center test hole temperature from responding quickly to core temperature changes and the core negative temperature coefficient of reactivity more than offsets any positive effect from the center test hole.

"Because of its spatial importance the reactor is sensitive to reactivity changes in the center test hole and consequently this space receives particularly rigorous construction and administrative safety controls. Only movable experiments in the center test hole shall be removed or installed with the reactor operating. All other experiments in the center test hole shall be removed or installed with the reactor shut down. Secured experiments shall be rigidly held in place during reactor operation. The center test hole experiment holder has been described to the commission in the 1972-73 Reactor Operations Annual Report and a Licensee Event Report dated November 17, 1982. Additional modifications are described in the 2000 Reactor Operations Annual Report."

2000

The original hazards summary report evaluated the transient analysis based upon a negative temperature coefficient of -7×10^{-5} ΔK/K°F and placed a limit of .007 ΔK on the worth of experiments in the center test hole. During the initial start up and calibration the temperature coefficient was experimentally determined to be -3.2×10^{-5} and the associated transient analysis limited the maximum experiment worth in the center test hole to 0.004 ΔK. During the low power testing program for the 6.2 Kg core the temperature and void coefficients were carefully remeasured and found to be very close to the original calculated values. Two independent measurements were made to confirm that the temperature and void coefficients were indeed close to the original calculations. The final experimental values were -7.0×10^{-5} ΔK/K°F for the temperature coefficient and -2.51×10^{-3} ΔK/% void for the void coefficient.(3)

A transient analysis for 10 MW operation has been performed to determine a safe step reactivity insertion in the center test hole and is described in section 3.5 of this submittal.

Each experiment is carefully reviewed to ensure safety and its reactivity worth is mathematically determined. Prior to loading in the reactor, each proposed center test hole loading is reviewed and the reactivity worth of all samples is also determined.

While highly unlikely it is possible to construct the situation in which all of the experiments in the center test hole could be rapidly removed and therefore a restriction is put on the limit of the net reactivity worth for all the experiments in the center test hole in accordance with the analysis of section 3.5.

The most likely accident is the possible failure of any single experiment in the center test hole. The worst case might be the sudden bursting of the sample can and discharge of its contents and possible damage to adjacent sample cans. Experiments shall be limited such that the failure of any single experiment cannot introduce a reactivity change greater than 0.006 ΔK. The limit for each individual experiment places importance on a critical review by the reactor operations staff, the Reactor Advisory Committee and the Director. It is, however, a review they are qualified to make.

3.11 Experimental Programs

All experiments, including previously approved programs that are still active, will be reviewed by the reactor staff and where appropriate by the Reactor Safety Subcommittee to ensure they can be safely conducted at 10 MW operation. If necessary, experiments will be modified or removed to accommodate the higher power level.

3.12 Analysis of Primary System Flow Coast Down in the MURR

In the event of a loss of flow in the primary system of the MURR a reactor scram will occur, valve 546 will automatically open to allow circulation through the in-pool heat exchanger (HX-505) for removal of the residual decay heat from the reactor core, and valves 507A and 507B will automatically close to isolate the in-pool primary system from the rest of the primary loop.

Under forced convection operation the MURR core is cooled by down flow, however, natural convection flow through the in-pool heat exchanger for removal of decay heat after a scram is upward through the core. Therefore, during the critical seconds after a loss of flow scram, the coolant flow through the MURR core must reverse directions.

This critical time period has been examined (1) using the reactor transient analysis computer code PARET (2) and the hydraulic loop transient flow analysis code TINKER (1) derived from HAFMAT (3). PARET is a well known transient code combining reactor kinetics with hydraulic and heat transfer analysis, and is ideally suited for heat transfer analysis in the MURR core. The TINKER code was used to model the entire in-pool primary loop and predict the flow coast down and reversal.

It was found that flow reversal occurs at about 6.3 seconds after a loss of flow from 10 megawatt operation. During this period the maximum hot spot heat flux was found to be a factor of 2.5 below the departure from nucleate boiling heat flux. Thus, it is concluded that no core damage will occur as a result of a loss of flow scram from 10 MW operation.

References

1. Smith, D. W. and D. R. Edwards, "An Analysis of the Flow Coast Down Problem for the MURR," University of Missouri, Rolla, Missouri, 1972.
2. C. F. Obenchain, "PARET--A Program for the Analysis of Reactor Transients," USAEC Report IDO-17282 (1969).
3. Wunderlich, L. H., "HAFMAT: Steady-State Flow Distribution Program," Knolls Atomic Power Laboratory Report KAPL-M-LXW-1 (1962).

3.13 Evaluation of Proposed Changes to the MURR License R-103

The proposed amendment to License R-103 and change to the Technical Specifications for the MURR contain several alterations to the present license. This report presents justification for the significant changes.

3.13.1 Sealed Antimony 124-Beryllium Neutron Source Strength

The MURR startup neutron source consists of a mixture of antimony and beryllium powders doubly encapsulated in 304 stainless steel with the outer cylinder being 5.5 inches long and 1.25 inches in diameter. It is presently licensed for a maximum ^{124}Sb activity of two curies. The source was used in the initial startup program of the MURR, however, (γ,n) and (n,2N) reactions in the beryllium reflector as a result of previous reactor operation have far eclipsed the startup neutron population which could be introduced by this source. Therefore, it is no longer necessary for reactor startup. The source is now used for subcritical multiplication measurements in spent fuel storage racks and shipping casks. The neutron emission rate has been found to be too low for accurate measurements in far subcritical systems. It is therefore requested that the maximum source strength be increased to 100 Curies of antimony-124.

The saturation activity of the source when placed in the MURR beryllium reflector source irradiation position is calculated to be over 1,000 Curies.

3.13.2 Core Excess Reactivity

Previous technical specifications limited the core excess reactivity above cold clean critical to no greater than 0.098 ΔK/K. Core excess reactivity is of no safety consequence as long as sufficient negative reactivity is present to shut the reactor down. The technical specifications require that sufficient shutdown margin exists such that the reactor will be subcritical by at least 0.02 ΔK with the most reactive shim blade fully withdrawn. This is sufficient to ensure that the reactor can be shutdown under any circumstances.

3.13.3 Reactivity Addition Rate from Regulating Blade

Previous technical specifications required that the regulating blade reactivity insertion rate be no greater than 2.5×10^{-4} ΔK/second. The total worth of the regulating blade is restricted to no more than 0.006 ΔK. Section 3.5 of this report concludes that the MURR could withstand an instantaneous step reactivity insertion of +0.006 ΔK without fuel damage, thus the rate of reactivity insertion by the regulating blade is of no safety consequence.

3.13.4 Core Void Coefficient of Reactivity

The technical specification requirement for the MURR core void coefficient is changed from more negative than -1.2×10^{-3} ΔK/% void to more negative than -2.0×10^{-3} ΔK/% void. Latest experimental measurements (1) yield a value of -2.51×10^{-3} ΔK/% void, thus the MURR is well within the new limit. The new limiting value of -2.0×10^{-3} was used in the step reactivity insertion analysis of section 3.5.

3.13.5 Core Temperature Coefficient of Reactivity

The core temperature coefficient of reactivity is to be limited to more negative than -6.0×10^{-5} ΔK/°F rather than -3.0×10^{-5}. Repeated measurements (1) of the MURR core temperature coefficient of reactivity consistently yield a value of -7.0×10^{-5} ΔK/°F, thus the MURR is well within this limit.

3.13.6 Experiment Reactivity Limits

Previous technical specification requirements limited the sum of the absolute values of the reactivity worths of all experiments to less than 0.025 ΔK. Experiments in the MURR are required to be designed such that the failure of any experiment shall not result in the introduction of reactivity exceeding 0.006 ΔK. Each experiment is additionally limited to a reactivity worth of not greater than 0.006 ΔK. The previous value of 0.025 ΔK has no basis as a safety limit in the MURR and is therefore an unnecessary requirement.

References

1. Low Power Testing Program for the Missouri University Research Reactor 6.2 Kilogram Core, p 74.

4.0 TESTING PROGRAM

4.1 Introduction

The testing program will consist of two phases; the first will include preoperational inspection, testing and calibrating of systems affected by the upgrade modifications and the second phase will encompass the initial operation of the reactor and the approach to full power operation at 10 MW.

The general procedures included below are intended as a guide to illustrate the scope of the testing to be conducted. Detailed procedures and check lists will be used and filed with the facility records for all preoperational and operational testing. These procedures are not outlined in the sequence in which they will be performed.

4.2 Pre-operational Testing

4.2.1 Primary Coolant System Hydrostatic Pressure Test

4.2.1.1 Test Purpose

To demonstrate the primary system integrity, after additions to the system for 10 MW.

4.2.1.2 Test Method

Fill, vent and apply pressure to the primary system, utilizing an accumulator tank filled with water and a nitrogen supply with appropriate regulator.

4.2.1.3 References

1. MURR Dwg. 156
2. Valve line-up procedure

4.2.1.4 Prerequisites

All valves, flanges and system components checked for leakage and operability

4.2.1.5 Conditions Prior to Test

1. Heise gauge installed into system ______
2. System filled and vented ______
3. Hydrostactic test rig, accumulator filled with water and nitrogen supply with regulator installed into system ______
4. Approved valve line-up procedure completed ______

4.2.1.6 Test Procedure

1. Using hydro test rig, slowly increase pressure to just below relief valve settings (Or perform test with reliefs removed and tested separately). ______
2. Check pressure at ________ psi on Heise. ______
3. Check for leaks and correct same. ______
4. Hold pressure for ________ without appreciable drop. ______
5. Remove hydro test rig and Heise gauge and return system to normal. ______

Prepared by ______________________ Date ____________
Approved by ______________________ Date ____________
Performed by ______________________ Date ____________
Witnessed by ______________________ Date ____________

4.2.2 Primary Coolant System Checkout

4.2.2.1 Test Purpose

The purpose of this test is to demonstrate the satisfactory operation of the primary coolant pump 501B and heat exchanger 503B and associated loops.

4.2.2.2 Test Method

Circulate water through the system varying the output to attain a flow versus discharge pressure plot and checking the various system combinations.

4.2.2.3 References

1. Reactor operating procedures
2. MURR Dwg. 156
3. Worthington pump characteristic curve, MURR Print 731

4.2.2.4 Prerequisites

1. Hydrostatic test completed ______
2. System flushed free of debris ______
3. All gauges calibrated ______
4. Flow elements 913A and 913B calibrated ______
5. Motor rotation checked proper ______
6. Pump rotated by hand for mechanical freedom ______
7. Pump 501B properly lubricated in accordance with manufacturer's instructions ______
8. Pump and motor properly aligned ______

4.2.2.5 Conditions Prior to Test

1. Both pumps power on ______
2. Control room power on ______
3. Power on for primary instrumentation ______
4. Primary system filled and vented ______
5. Valve checksheet completed ______

4.2.2.6 Test Procedure

1. Place system in service in accordance with latest revision of Standard Operating Procedure. As soon as pump(s) are running, verify flow; if not, secure immediately. ______
   a. Pump speed ________ rpm
      Motor volts ________
      Motor starting current ________ amps
      Motor running current ________ amps

b. Bearing vibration

	Inboard	Outboard	
Horizontal left	_______ mils	_______ mils	_____
Horizontal right	_______ mils	_______ mils	_____
Vertical top	_______ mils	_______ mils	_____
Vertical bottom	_______ mils	_______ mils	_____

c. Bearing temperatures

Check bearing temperatures frequently during pump testing _____

2. The following procedure involves gathering data while varying the flow using various pumps and heat exchanger combinations.

RUN #1

New pump (501B) and both heat exchangers on _____

Record flow _______ gpm

ΔP across 503A _______ psi, 503B _______ psi _____

Pump suction pressure ______, discharge pressure ______ _____

Close inlet valve 510F to heat exchanger 503B _____

Vary the output of pump by throttling pump bypass valve 538B and pump outlet valve 510E to obtain the most efficient operation of the pump characteristic curve _____

Record flow _______ gpm

ΔP across 503A _______ psi _____

Pump suction pressure _______ _____

Pump discharge pressure _______ _____

Open 503B inlet valve 510F _____

Close 503A inlet valve 510B _____

Record flow _______ gpm _____

ΔP across 503B _______ _____

Pump suction pressure ______, discharge pressure ______ _____

Secure pump 501B _____

Open 503A inlet valve 510B _____

Start pump 501A _____

RUN #2

Pump 501A and both heat exchangers on ______

Record flow ________ gpm

ΔP across 503A ________ psi, 503B ________ psi ______

Pump suction pressure ______, discharge pressure ______ ______

Close inlet valve 510F to heat exchanger 503B ______

Vary the output of pump by throttling pump bypass valve 538A and pump outlet valve 510C to obtain the most efficient operation on the pump characteristic curve ______

Record flow ________ gpm

ΔP across 503A ________ psi ______

Pump suction pressure ______, discharge pressure ______ ______

Open 503B inlet valve 510F ______

Close 503A inlet valve 510B ______

Record flow ________ gpm ______

ΔP across 503B ________ ______

Pump suction pressure ______, discharge pressure ______ ______

Secure pump 501A ______

Open 503A inlet valve 510B ______

3. The next test is to determine the effect of each pump check valve shutting when the opposite pump is secured. ______

With both pumps and heat exchangers on in accordance with SOP, and approximately 1800 gpm flow from each pump, secure pump 501A. ______

Note pipe noise and movement. ______

Start pump 501A. ______

Secure pump 501B. ______

Note pipe noise and movement. ______

If tests are satisfactory, return system to normal shutdown condition.

4.2.3 Pool Coolant System Hydrostatic Pressure Test

4.2.3.1 Test Purpose

To demonstrate the ability of the added equipment to withstand and hold a pressure of 125 psig.

4.2.3.2 Test Method

Isolate the new equipment, fill, vent and apply pressure using an accumulator tank filled with water and an N_2 supply with a regulator.

4.2.3.3 References

MURR drawing 156

Approved valve line-up procedure

4.2.3.4 Prerequisites

All new valves checked for leakage and operability. ______

All new equipment as far as valves, take off, and piping that pertains to pressure under operating conditions installed. ______

4.2.3.5 Conditions Prior to Test ______

Heise gauge installed in test loop. ______

Loop filled and vented. ______

The pressurizing rig which consists of a tank of about 10 gallon capacity filled with water and N_2 supply with regulator installed. ______

Approved valve line-up procedure completed. ______

4.2.3.6 Test Procedure

Using pressurizing rig, slowly increase pressure to 125 psig as read on calibrated Heise gauge. ______

Check for leaks and repair same. ______

Hold pressure 2 hours without appreciable drop. ______

Remove pressurizing rig and Heise gauge. ______

Return valves and loop to normal. ______

Prepared by ____________________ Date ____________
Approved by ____________________ Date ____________
Performed by ____________________ Date ____________
Witnessed by ____________________ Date ____________

4.2.4 Pool Coolant System Checkout

4.2.4.1 Test Purpose

The purpose of this test is to demonstrate the satisfactory operation of the pool pump and heat exchanger.

4.2.4.2 Functional Description

The added pump, heat exchanger and their associated equipment provides the pool with the ability to increase reactor thermal power from 5 MW to 10 MW, also providing back-up equipment for 5 MW or below. The equipment will be installed in parallel with present equipment. The flow path is identical to that described in Reactor Operating Procedures Pages IV-4 and IV-5 Revised 10-71 except a parallel path is provided through pump 508B and heat exchanger 521B and the flow control from 400 to 600 gpm is controlled manually instead of by a motor operated valve.

4.2.4.3 Test Method

Circulating pool water through the system with varying pump and heat exchanger combinations while varying the flow output of the pumps.

4.2.4.4 References

Reactor Operating Procedures
MURR Drawing 156
Worthington Vendor Print 1201-1C which is the pump characteristic curve and is filed under MURR Print 732.

4.2.4.5 Prerequisites

Hydrostatic test completed. ______
System flush free of debris. ______
All gauges calibrated. ______

Flow element 921B calibrated. ______

P-508B and isolation valve 509 interlock tested. ______

Pump rotated by hand for mechanical freedom. ______

Pump 508B properly lubricated in accordance with vendor's instructions. ______

ΔP meters installed across heat exchangers 521A and 521B. ______

Pump and motor aligned in accordance with manufacturer's instructions. ______

4.2.4.6 Condition Prior to Test

Power to pumps 508A & B. ______

Control room power on. ______

Power on pool instrumentation. ______

Pool filled and vented to normal level. ______

Air to air operated valve on. ______

Approved manual valve checksheet completed. ______

4.2.4.7 Test Procedure

1. Place master switch 1S1 in test position. ______
2. Turn on pool flow recorder. ______
3. Turn on pool temperature recorder. ______
4. Place valve 509 auto-manual switch to manual. ______
5. Place valve 509 open-close switch to open. Valve should open. ______
6. Place valve 547 auto-manual switch to manual. ______
7. Place valve 547 open-close switch to close position. Valve should close. ______
8. Jog pump 508B with local switch. Check proper rotation. ______
9. Start pump. ______
10. As soon as pump is running verify flow. ______
11. Record

    Pump speed ________ rpm ______

    Motor volts ________ volts ______

    Motor starting current ________ amps ______

Motor running current ________ amps ______

Pool flow ________ gpm

Bearing vibration

	Inboard	Outboard	
Horizontal left	________ mils	________ mils	______
Horizontal right	________ mils	________ mils	______
Vertical top	________ mils	________ mils	______
Vertical bottom	________ mils	________ mils	______

12. Check bearing temperature frequently during pump testing. ______

4.2.4.8 The following procedure involves gathering data while varying the flow using various pump and heat exchanger combinations.

1. Run new pump with both heat exchangers on. ______
2. Record flow ________ gpm

   ΔP across new heat exchanger ________ old heat exchanger ________

   Pump suction pressure ______ discharge pressure ______ ______
3. Close inlet valve 539D to new heat exchanger 521B. ______
4. Vary the output by throttling pump bypass valve 531B and pump outlet valve 522E to obtain the most efficient operation on the pump characteristic curve. ______
5. Record flow ________ gpm ______

   ΔP across heat exchanger ________ ______

   Pump suction pressure ________ ______

   Pump discharge pressure ________ ______
6. Open heat exchanger inlet valve 539D. ______
7. Close heat exchanger inlet valve 539B. ______
8. Record flow ________ gpm ______

   ΔP across heat exchanger ________ ______

   Pump suction pressure ______ discharge pressure ______ ______
9. Secure pump 508B. ______
10. Open heat exchanger inlet valve 539B. ______
11. Start pump 508A. ______

12. While varying pump output using motor operated valve 530 record minimum flow _______ maximum flow _______
ΔP across both heat exchanger 521A _____ and 521B _____ _____
Pump suction pressure _____ discharge pressure _____ _____
13. Close heat exchanger inlet valve 539D. _____
14. Vary output using V-530. _____
15. Record maximum flow _______ minimum flow _______ _____
ΔP across 521A maximum _______ minimum_______ _____
Pump suction pressure _____ discharge pressure _____ _____
16. Open heat exchanger inlet valve 539D. _____
17. Close heat exchanger inlet valve 539B. _____
18. Vary flow using V-530. _____
19. Record maximum flow _______ minimum flow _______ _____
ΔP across 521B maximum _______ minimum_______ _____
20. Open heat exchanger inlet valve 539B. _____

4.2.4.9 The next test is to determine the effect of each pump check valve shutting when the opposite pump is secured.

1. With both pump and heat exchangers on and the flow adjusted giving approximately 1000 gpm, secure pump 508A. Note pipe noise or movement. _____
2. Start pump 508A. _____
3. Secure pump 508B. Note pipe noise or movement. _____
4. If tests are satisfactory, return system to normal shutdown condition. _____

4.2.5 Pre-operational Test of Secondary System

4.2.5.1 Test Purpose

The purpose of this test is to insure that the secondary system is leak tight, that the system will operate in its intended manner, and to determine the operating characteristic curve of SP-3.

4.2.5.2 Test Method

The test method for leak checking the system will be to operate the secondary system (SP-3) with a shutoff head by closing the CT distribution box valves and visually inspecting all piping, valves and fittings for leaks.

The method for determining the pumps (SP-3) characteristic curve will be to plot flow vs. pump discharge pressure by varying the position of the discharge valve of SP-3.

4.2.5.3 Prerequisites

Prior to installation all valves will be checked for mechanical operation by visually observing the disc while operating the valve handle to insure the valve operates correctly and the system will be checked during installation to insure that no foreign materials are inadvertently introduced into the piping.

Prior to running the pump characteristic curve, all gauges will be calibrated and the flow sensor will be calibrated.

4.2.5.4 Conditions prior to test

1. The system is filled and vented. ______
2. All valves are checked according to an approved procedure. ______

4.2.5.5 Test Procedure

1. Close valves S-18, S-19, S-20, S-21, S-22 and S-17. ______
2. Energize secondary flow recorder. ______
3. Establish communications with the control room. ______
4. Start SP-3. ______
5. With system lines up normally, measure starting and running currents on SP-3. ______
6. Check all piping, valves and fittings for leaks. ______
7. Secure SP-3. ______
8. Open S-18, S-19, S-20, S-21, S-22 and S-17. ______

9. Shut SP-3 discharge valve S-7. ______
10. Start SP-3. ______
11. Open S-7 slowly plotting flow vs. pump discharge pressure until valve is full open. Test for normal operation of system. ______
12. Attach a thermometer to motor case at bearing housing and run system until temperature of bearings stabilize and record same. ______

4.2.6 Preoperational Test of Motors and Motor Controllers

4.2.6.1 Test Purpose

The purpose of this test is to verify the proper installation and operation of new motors, controllers and associated circuitry.

4.2.6.2 Test Method

The test method consists of conducting tests in accordance with AEC requirements. Tests shall include:

1. Grounds
2. Short-circuits
3. Meggering
4. Motor rotation
5. Start-stop and automatic operation of controls
6. Voltage checks at terminals of equipment
7. Current checks at terminals of equipment

4.2.6.3 Prerequisites

Upon completion of installation of all units, i.e., motors, motor controllers, start-stop switches, etc., all wiring will be confirmed to drawings.

4.2.6.4 Conditions Prior to Test

1. All main power switches on motor controllers will be "off" and "red tagged."
2. A visual inspection will be made for loose, grounded leads or other abnormalities.

4.2.6.5 Test Procedure for P-501B, P-508A&B, SP-3 and CT Fan 3

1. Megger power leads ______
2. Remove "red tag" and turn main breaker power on ______
3. Measure input voltage (460 VAC) ______
4. "Jog" at motor. Observe for correct rotation ______
5. If incorrect rotation, correct wiring for proper operation ______
6. Turn motor on at controller and verify that "Stop" at the controller stops motor ______
7. Turn motor on at controller and verify that "Stop" at the motor stops motor ______
8. Turn motor on and off from control room; verify proper control of motor from control room controls ______
9. Turn motor on and measure currents of each motor phase ______
   Ø 1 ______ Ø 2 ______ Ø 3 ______
10. Check for vibration of motor ______
11. Check temperature of motor bearings, insuring normal temperatures ______

4.2.6.6 P-501B

Steps 1-11 of 4.2.6.5

1. ______
2. ______
3. ______ VAC
4. ______
5. ______
6. ______
7. ______
8. ______
9. ______ Ø 1 ______ Ø 2 ______ Ø 3 ______
10. ______
11. ______

4.2.6.7 P-508A

Steps 1-11 of 4.2.6.5

1. ______
2. ______
3. ______ VAC
4. ______
5. ______
6. ______
7. ______
8. ______
9. ______ Ø 1 ______ Ø 2 ______ Ø 3 ______
10. ______
11. ______

4.2.6.8 P-508B

Steps 1-11 of 4.2.6.5

1. ______
2. ______
3. ______ VAC
4. ______
5. ______
6. ______
7. ______
8. ______
9. ______ Ø 1 ______ Ø 2 ______ Ø 3 ______
10. ______
11. ______

4.2.6.9 SP-3

Steps 1-11 of 4.2.6.5

1. ______
2. ______
3. ______ VAC

4. ______
5. ______
6. ______
7. ______
8. ______
9. ______ Ø 1 ______ Ø 2 ______ Ø 3 ______
10. ______
11. ______

4.1.6.10 Cooling Tower Fan 3

Steps 1-11 of 4.2.6.5

1. ______
2. ______
3. ______ VAC
4. ______
5. ______
6. ______
7. ______
8. ______
9. ______ Ø 1 ______ Ø 2 ______ Ø 3 ______
10. ______
11. ______

In addition to steps 1-11 of 4.2.6.5, check for fast and slow speeds and forward and reverse operations at the controller and control room. ______

Test completed ____________

Date ____________

Performed by ________________

4.2.7 Preoperational Test of Safety System

4.2.7.1 Test Purpose

The purpose of this test is to verify the proper installation and operation of the modified safety system.

4.2.7.2 Test Method

Systems will be placed in service to place the safety system in operation. Each leg and function of the safety system will be verified by loss of magnet current. Redundancy test of 2K13, 2K28 will be accomplished prior to this test (see preoperational test of process control and interlock).

4.2.7.3 Test Procedure

See MURR Dwg. No. 41 Rev. 5, 138 and 139

4.2.7.3.1 Verification of DPS 944A and B and DPS 929

1. Open connectors on control rods A, B, C and D input cables ______
2. Place control rod test plugs on control cables for control rods A, B, C and D ______
3. Place power level switch 1S8 to 5 MW ______
4. Install a short from 2K28-13 to TB1-6 ______
5. Place systems into service, clear annunciator and initiate magnet current ______
6. Short DPS 929 alarm contacts ______
7. Short DPS 944B alarm contacts ______
8. Decrease Rx flow to alarm point of 944A ______
9. Verify annunciator 3-3 and loss of magnet current ______
10. Remove short on DPS 944B ______
11. Place short on DPS 944A ______
12. Clear annunciator ______
13. Place magnet current on ______
14. Decrease Rx flow to alarm point of 944B ______
15. Verify annunciator 3-3 and loss of magnet current ______

16. Remove short from DPS 929 ______
17. Place short on 944B ______
18. Clear annunciator ______
19. Place magnet current on ______
20. Decrease Rx flow to the alarm point of DPS 929 ______
21. Verify annunciator 3-3 and loss of magnet current ______
22. Remove shorts from DPS 944A & B ______
23. Remove short from 2K28-13 to TB1-6 ______

4.2.7.3.2 Verification of Rx Loop Low Flow Alarms 920A&B

1. Conditions for this test same as A, steps 1-3 ______
2. Short 2K13-14 to 2K19-5 ______
3. Install shorted relay in K69 and K31 ______
4. Place HX 503A into service with flow ______
5. Clear annunciator and place magnet current on ______
6. Decrease flow of HX 503A to recorder 915A alarm point ______
7. Verify annunciator 4-7 ______
8. Decrease flow of HX 503A further to 920A alarm point ______
9. Verify annunciator 3-1 and loss of magnet current ______
10. Remove HX 503A from service ______
11. Place HX 503B into service with flow ______
12. Clear annunciator and place magnet current on ______
13. Decrease flow of HX 503B to recorder 915B alarm point ______
14. Verify annunciator 4-7 ______
15. Decrease flow of HX 503B further to 920B alarm point ______
16. Verify annunciator 3-1 and loss of magnet current ______
17. Place 1S8 to 10 MW position ______
18. Place HX 503A&B into service with flow ______
19. Clear annunciator and place magnet current on ______
20. Decrease flow of HX 503A to recorder 915A alarm point ______
21. Verify annunciator 4-7 ______
22. Decrease flow of HX 503A further to 920A alarm point ______
23. Verify annunciator 3-1 and loss of magnet current ______

24. Place HX 503A&B into service with flow ______
25. Clear annunciator and place magnet current on ______
26. Decrease flow of HX 503B to recorder 915B alarm point ______
27. Verify annunciator 4-7 ______
28. Decrease flow of HX 503B further to 920B alarm point ______
29. Verify annunciator 3-1 and loss of magnet current ______
30. Remove shorted relays K69 and K31 ______
31. Remove short from 2K13-14 to 2K19-5 ______

4.2.7.3.3 <u>Verification of Pool Loop Low Flow Alarms 920C&D</u>

1. Conditions for this test same as A, steps 1-3 ______
2. Short 2K13-14 to 2K19-5 ______
3. Install shorted relay in K38 and K30 ______
4. Place HX 521A into service with flow ______
5. Clear annunciator and place magnet current on ______
6. Decrease flow of HX 521A to recorder 915C alarm point ______
7. Verify annunciator 5-8 ______
8. Decrease flow of HX 521A further to 920D alarm point ______
9. Verify annunciator 5-3 and loss of magnet current ______
10. Remove HX 521A from service ______
11. Place HX 521B into service with flow ______
12. Clear annunciator and place magnet current on ______
13. Decrease flow of HX 521B to recorder 915D alarm point ______
14. Verify annunciator 5-8 ______
15. Decrease flow of HX 521B further to 920C alarm point ______
16. Verify annunciator 5-3 and loss of magnet current ______
17. Place 1S8 to 10 MW position ______
18. Place HX 521A&B into service with flow ______
19. Clear annunciator and place magnet current on ______
20. Decrease flow of HX 521A to recorder 915C alarm point ______
21. Verify annunciator 5-8 ______
22. Decrease flow of HX 521A further to 920D alarm point ______
23. Verify annunciator 5-3 and loss of magnet current ______

24. Place HX 521A&B into service with flow ______
25. Clear annunciator and place magnet current on ______
26. Decrease flow of HX 521B to recorder 915D alarm point ______
27. Verify annunciator 5-8 ______
28. Decrease flow of HX 521B further to 920C alarm point ______
29. Verify annunciator 5-3 and loss of magnet current ______
30. Remove shorted relays K38 and K30 ______
31. Remove short from 2K13-14 to 2K10-5 ______
32. Remove control rod test plugs from control cables of control rods A, B, C and D ______
33. Connect control rods A, B, C and D to respective cables ______
34. Preoperational test of safety system completed ______

Date ____________

Performed by __________________

4.2.8 Preoperational Test of Reactor Control System

4.2.8.1 Test Purpose

The purpose of this test is to verify the proper installation and operation of the modified reactor control system.

4.2.8.2 Test Method

Systems will be placed into operation as required. Each leg and function of the modified control system will be verified by power level interlock scram, annunciator 2-2 and other associated annunciators.

4.2.8.3 Test Procedure

(See MURR Dwgs 138, 139 and 42 Rev. 13)

1. Short TB1-9 of the safety system to
   a. TB1-6 safety system ______
   b. TB1-8 safety system ______
   c. LL-60 ______
2. Short LL59 to LL61 ______
3. Open connectors on control rods A, B, C and D input cables ______
4. Place control rod test plugs on control cables for control rods A, B, C and D ______
5. Install shorted relay in K7 and K8 ______
6. Short contacts of 2K11-11 to AA-35 ______
7. Place power level switch to .1 MW position ______
8. Verify that .1 MW selection indicating light illuminates ______
9. Verify 2K19 is energized ______
10. Verify that "warning light" comes on at 500 kW - 40% position ______
11. Verify that magnet current cannot be reset with no reactor or pool flow ______
12. Place HX 503A and HX 521B in service with proper flow ______
13. Verify annunciator 2-2 reset and turn magnet current on ______
14. Decrease flow of HX 503A to 920A alarm point ______

15. Verify power level interlock annunciator 2-2, 3-1 and loss of magnet current (scram) ______
16. Increase HX 503A flow to proper level ______
17. Reset annunciator and magnet currents ______
18. Decrease flow of HX 521B to alarm point of 920C ______
19. Verify annunciator 2-2, 5-3 and loss of magnet current ______
20. Remove HX 503A and HX 521B from service ______
21. Place HX 503B and HX 521A into service with proper flow ______
22. Reset annunciator and magnet current ______
23. Decrease flow of HX 503B to alarm point of 920B ______
24. Verify annunciator 2-2, 3-1 and loss of magnet current ______
25. Increase HX 503B to proper flow ______
26. Reset annunciator and magnet current ______
27. Decrease flow of HX 521A to alarm point of 920D ______
28. Verify annunciator 2-2, 5-3 and loss of magnet current ______
29. Place 1S8 in the 5 MW position ______
30. Verify that 2K19 is de-energized ______
31. Verify that the 5 MW indicator is illuminated ______
32. Verify that the warning light illuminates at 5 MW, 125% position of range selector switch ______
33. Place HX 503A and HX 521A into service with proper flow ______
34. Reset annunciator and turn magnet current on ______
35. Decrease flow of HX 503A to alarm point of 920A ______
36. Verify annunciator 2-2, 3-1 and loss of magnet current ______
37. Increase HX 503A to proper flow ______
38. Reset annunciator and magnet current ______
39. Decrease flow of HX 521A to alarm point of 920D ______
40. Verify annunciator 2-2, 5-3 and loss of magnet current ______
41. Remove HX 503A and HX 521A from service ______
42. Place HX 503B and HX 521B into service with proper flow ______
43. Reset annunciator and turn magnet current on ______
44. Decrease flow of HX 503B to alarm point of 920B ______
45. Verify annunciator 2-2, 3-1 and loss of magnet current ______

46. Increase HX 503B to proper flow ______
47. Reset annunciator and turn magnet current on ______
48. Decrease HX 521B flow to alarm point of 920C ______
49. Verify annunciator 2-2, 5-3 and loss of magnet current ______
50. Place switch 1S8 to 10 MW position ______
51. Verify that the 10 MW indicator is illuminated ______
52. Verify that the warning light illuminates at 10 MW percent power of the range switch ______
53. Verify annunciator 2-2 and magnet current cannot be reset ______
54. Place HX 503A&B, HX 521A&B into service with proper flow ______
55. Place range switch to 10 MW ______
56. Reset annunciator and turn magnet current on ______
57. Decrease flow of HX 503A to alarm point of 920A ______
58. Verify annunciator 2-2, 3-1 and loss of magnet current ______
59. Increase HX 503A to proper flow ______
60. Reset annunciator and magnet current on ______
61. Decrease flow of HX 503B to alarm point of 920B ______
62. Verify annunciator 2-2, 3-1 and loss of magnet current ______
63. Increase HX 503B to proper flow ______
64. Reset annunciator and magnet current on ______
65. Decrease flow of HX 521A to alarm point of 920D ______
66. Verify annunciator 2-2, 5-3 and loss of magnet current ______
67. Increase HX 521A to proper flow ______
68. Reset annunciator and magnet current on ______
69. Decrease flow of HX 521B to alarm point of 920C ______
70. Verify annunciator 2-2, 5-3 and loss of magnet current ______
71. Remove shorted relay K7 and K8 and install operational relay K7 and K8 ______
72. Increase HX 521B to proper flow ______
73. Reset annunciator and magnet current on ______
74. Move alarm set point of 953B to initiate an alarm ______
75. Verify annunciator 2-2, 4-1 and loss of magnet current ______
76. Reset alarm set point of 953B to proper value ______

77. Reset annunciator and magnet current on ______
78. Decrease pressurizer pressure to PSL 938 alarm point ______
79. Verify annunciator 2-2, 3-3 and loss of magnet current ______
80. Increase pressurizer pressure to normal operating range ______
81. Reset annunciator and magnet current on ______
82. Remove systems from service ______
83. Remove short of TB1-9 of safety system to
    a. TB1-6 safety system ______
    b. TB1-8 safety system ______
    c. LL 60 ______
84. Remove short from LL59 to LL61 ______
85. Remove control rod test plugs from control cables for rods A, B, C and D ______
86. Connect cables of control rods for proper operation ______
87. Remove short from 2K11-11 to AA35 ______

Date completed ____________

Tested by ________________________

4.2.9 Preoperational Test of Neutron Monitoring System WRM (Channel 4) and PRM (Channels 5 and 6)

4.2.9.1 Test Purpose

The purpose of this test is to insure that the changes made in the neutron monitoring system are correct and operational.

4.2.9.2 Test Method

The test method consists of verifying that the WRM and PRM's are calibrated properly and to insure that modules (Z14 units) of the PRM's cannot be interchanged for incorrect power levels desired.

4.2.9.3 Test Procedure

4.2.9.3.1 Wide Range Monitor, Channel 4

(See MURR Dwg 965 - Rev. 3)

1. Place front panel selector switch to zero 1 ______
2. Remove VR unit ______
3. Disconnect signal input cable ______
4. Connect current source to signal input connector ______
5. Install VR unit ______
6. Place front panel switch to "operate" ______
7. Perform WRM accuracy check using Table 1 ______
8. Place front panel selector switch to zero 1 ______
9. Remove VR unit ______
10. Remove calibration input current ______
11. Connect signal input from chamber for operation ______
12. Install VR unit ______
13. Place front panel switch to "operate" ______
14. WRM is now ready for operation ______

4.2.9.3.2 Power Range Monitors 5 and 6

(See MURR Dwgs No. 204-R6, 608 and 42-R13)	Ch 5	Ch 6
1. Place front panel selector switches to zero	____	____
2. Remove VR units	____	____
3. Remove Z14 unit covers and verify that the 5 and and 10 MW Z14 units are connected correctly		
5 MW Z14 Units		
E1 to E2	____	____
E4 to E5	____	____
4X to 4Y	____	____
14X to 14 Y	____	____
21X to 21Y	____	____
10 MW Z14 Units		
E1 to E2	____	____
E4 to E5	____	____
6X to 6Y	____	____
16X to 16 Y	____	____
22X to 22Y	____	____
4. Place covers on Z14 units	____	____
5. Install 5 MW Z14 units	____	____
6. Place power selector switch 1S8 to the 5 MW position	____	____
7. Observe that the nuclear instrument anomaly can be cleared and magnet current is available	____	____
8. Remove 5 MW Z14 units and insert 10 MW Z14 units	____	____
9. Observe that with 1S8 still in the 5 MW position the nuclear instrument anomalies cannot be cleared	____	____
10. Insure that magnet current cannot be had due to scram of channels 5 and 6	____	____
11. Place 1S8 in the 10 MW position	____	____
12. Insure that nuclear instrument anomaly can now be cleared and magnet current obtained	____	____
13. Place 1S8 in the 5 MW position	____	____
14. Install the 5 MW Z14 modules	____	____

		Ch 5	Ch 6
15.	Perform the 5 MW accuracy checks using Tables 2 and 3	______	______
16.	Note the remote R2 potentiometer setting for 5 MW	______	______
17.	Place 1S8 in the 10 MW position	______	______
18.	Install the 10 MW Z14 modules	______	______
19.	Perform MURR semi-annual calibration of PRM for 10 MW	______	______
20.	Perform the 10 MW accuracy checks using Tables 4 and 5	______	______
21.	Note the remote R2 potentiometer for 10 MW	______	______
22.	Place front panel selector switches to zero	______	______
23.	Remove VR units	______	______
24.	Disconnect calibration current from signal input	______	______
25.	Connect signal input from chamber for normal operation	______	______
26.	Insert VR units	______	______
27.	Place front panel selector switch to operate	______	______
28.	Power range monitors are now operational	______	______

Table 4.2.1

WRM - Channel 4 - Accuracy Check

Input Voltage	Input Resistance	Input Current	Indicated Reading M1-Recorder-Remote MTr	Specified Reading
20	10^5	2×10^{-4}		10 MW Blk ± 2%
10	10^5	1×10^{-4}		5 MW Blk ± 2%
3.16	10^5	3.16×10^{-5}		5 MW Red ± 2%
10	10^6	1×10^{-5}		500 kW Blk ± 2%
3.16	10^6	3.16×10^{-5}		500 kW Red ± 2%
10	10^7	1×10^{-6}		50 kW Blk ± 2%
3.16	10^7	3.16×10^{-7}		50 kW Red ± 2%
10	10^8	1×10^{-7}		5 kW Blk ± 2%
3.16	10^8	3.16×10^{-8}		5 kW Red ± 2%
10	10^9	1×10^{-8}		500 W Blk ± 2%
3.16	10^9	3.16×10^{-9}		500 W Red ± 3%
10	10^{10}	1×10^{-9}		50 W Blk ± 3%
3.16	10^{10}	3.16×10^{-10}		50 W Red ± 3%
10	10^{11}	1×10^{-10}		5 W Blk ± 3%
3.16	10^{11}	3.16×10^{-11}		5 W Red ± 3%
10	10^{12}	1×10^{-11}		.5 W Blk ± 3%
3.16	10^{12}	3.16×10^{-12}		.5 W Red ± 5%
1	10^{12}	1×10^{-12}		.05 W Blk ± 5%

Prior Calibration R50 value ____________

New Calibration R50 value ____________

Operational R50 value ____________

Tested by ______________________________

Date __________________

Remarks:

Table 4.2.2

Channel 5

5 MW Accuracy Check

Accuracy - Check the PRM using the current inputs specified below. Provide input currents using a John Fluke Power Supply and Precision Resistance.

Input Voltage	Series P	Input Current	Voltage AR15-TP1	Voltage Tolerance
25	10^6	2.5×10^{-5}		-10.00 ± 0.08 V
20	10^6	2.0×10^{-5}		8.00 ± 0.08 V
15	10^6	1.5×10^{-5}		6.00 ± 0.08 V
10	10^6	1.0×10^{-5}		4.00 ± 0.08 V
5	10^6	0.5×10^{-5}		2.00 ± 0.08 V
2.5	10^6	0.25×10^{-5}		1.00 ± 0.08 V

Notes:

1. Previous remote R2 setting ____________
2. New remote R2 setting ____________

Tested by ________________________

Date __________________

Table 4.2.3

Channel 6

5 MW Accuracy Check

Accuracy - Check the PRM using the current inputs specified below. Provide input currents using a John Fluke Power Supply and Precision Resistance.

Input Voltage	Series P	Input Current	Voltage AR15-TP1	Voltage Tolerance
25	10^6	2.5×10^{-5}		-10.00 ± 0.08 V
20	10^6	2.0×10^{-5}		8.00 ± 0.08 V
15	10^6	1.5×10^{-5}		6.00 ± 0.08 V
10	10^6	1.0×10^{-5}		4.00 ± 0.08 V
5	10^6	0.5×10^{-5}		2.00 ± 0.08 V
2.5	10^6	0.25×10^{-5}		1.00 ± 0.08 V

Notes:

1. Previous remote R2 setting ____________
2. New remote R2 setting ____________

Tested by ____________________

Date ________________

Table 4.2.4

Channel 5

10 MW Accuracy Check

Input Voltage	Input R	Input I	Voltage AR15-TP1	Voltage Tolerance
50	10^6	5×10^{-5}		-10.00 ± 0.08 V
40	10^6	4×10^{-5}		8.00 ± 0.08 V
30	10^6	3×10^{-5}		6.00 ± 0.08 V
20	10^6	2×10^{-5}		4.00 ± 0.08 V
10	10^6	1×10^{-5}		2.00 ± 0.08 V
5	10^6	0.5×10^{-5}		1.00 ± 0.08 V

10 MW Remote R2 potentiometer setting ____________

Tested by ________________________

Date ______________________

Table 4.2.5

Channel 6

10 MW Accuracy Check

Input Voltage	Input R	Input I	Voltage AR15-TP1	Voltage Tolerance
50	10^6	5×10^{-5}		-10.00 ± 0.08 V
40	10^6	4×10^{-5}		8.00 ± 0.08 V
30	10^6	3×10^{-5}		6.00 ± 0.08 V
20	10^6	2×10^{-5}		4.00 ± 0.08 V
10	10^6	1×10^{-5}		2.00 ± 0.08 V
5	10^6	0.5×10^{-5}		1.00 ± 0.08 V

10 MW remote R2 potentiometer setting ____________

Tested by ______________________

Date ________________

4.2.10 Preoperational Test of Process Control and Interlock

4.2.10.1 Test Purpose

The purpose of this test is to verify the correct operation of relay 2K28 and 2K13.

4.2.10.2 Test Method

The test method will be to short the contacts concerned on relay 2K13, verify operation of 2K28, then the contacts concerned of 2K28 will be shorted and verification of 2K13 relay will be observed. Systems will be placed in normal operation.

4.2.10.3 Test Procedure

(See MURR Dwg 41, Rev. 5, Sheet 1)

1. Short 2K13 - 5 and 6 ______
2. Short 2K13 - 7 and 8 ______
3. Short 2K13 - 9 and 10 ______
4. Short 2K13 - 11 and 12 ______
5. Short 2K13 - 13 and 14 ______
6. Short 2K13 - 15 and 16 ______
7. Move the set point of EP 929 alarm to cause an alarm ______
8. Verify the following occurs
   a. Valves 543 A&B fail open ______
   b. Valves 507 A&B fail closed ______
   c. Valve 546 fails open ______
   d. Pump 501 stops ______
   e. Magnet rod control current is lost (scram) ______
   f. Annunciator 3-3 operates ______
9. Clear alarm set point of EP 929 ______
10. Remove all shorts from relay 2K13 contacts ______
11. Short 2K28 - 5 and 6 ______
12. Short 2K28 - 7 and 8 ______
13. Short 2K28 - 9 and 10 ______

14. Short 2K28 - 11 and 12 ______
15. Short 2K28 - 13 and 14 ______
16. Short 2K28 - 15 and 16 ______
17. Move the set point of EP 929 alarm to cause an alarm ______
18. Verify the following occurs
    a. Valves 543 A&B fail open ______
    b. Valves 507 A&B fail closed ______
    c. Valve 546 fails open ______
    d. Pump 501 stops ______
    e. Magnet rod control current is lost (scram) ______
    f. Annunciator 3-3 operates ______
19. Clear alarm set point of EP 929
20. Remove all shorts from relay 2K28 contacts ______

Date Test Completed ____________

Performed by ________________________

4.2.11 Preoperational Test of Process Instrumentation

4.2.11.1 Test Purpose

The purpose of this test is to verify correct calibration and operation of the process instrumentation.

4.2.11.2 Test Method

The test method will be described on individual units referring to manufacturer and MURR test procedures.

4.2.11.3 Test Procedure

See MURR Dwgs No 41, Rev. 5, sheet 2; 41, Rev. 6, sheet 3

4.2.11.4 Heat Exchanger Outlet Temperatures

EP No. 980A, B, C and D and 990A, B, C and D

1. Prior to installation a temperature curve will be made of all eight units, using a temperature bath ______
2. After installation of the RT units, a low and high temperature check will be made in a temperature bath to confirm correct installation, and that temperature indications fall on the original curves. ______

4.2.11.5 Differential Pressure Transmitter EP No. 928A and 928B

	928A	928B
1. Apply a ΔP of 0 psi	______	______
2. Make zero adjustment	______	______
3. Apply a ΔP of 5 psi	______	______
4. Make the span adjustment	______	______
5. Repeat steps 1-4 until no further adjustment is required	______	______
6. Place the power level switch 1S8 to 10 MW	______	______
7. Apply a ΔP of ______ psi to 928 A&B	______	______
8. Lower the ΔP of 928A to alarm set point	______	______

9. Verify that annunciator 6-2 is activated ______ ______
10. Increase the ΔP of 928A to 5 psi ______
11. Lower the ΔP of 928B to alarm set point ______
12. Verify that annunciator 6-2 is activated ______
13. Increase the ΔP of 928B to 5 psi ______
14. Place the 1S8 switch to 5 MW ______ ______
15. Lower ΔP of 928A to zero ______
16. Lower ΔP of 928B to alarm set point ______
17. Verify annunciator 6-2 is activated ______
18. Lower ΔP of 928B to zero and raise ΔP of 928A to 5 psi ______
19. Lower ΔP of 928A to alarm set point ______
20. Verify that annunciator 6-2 is activated ______

4.2.11.6 <u>Differential Pressure Transmitter EP No. 929</u>

1. Apply a ΔP of 0 psi ______
2. Make the zero adjustment ______
3. Apply a ΔP of 25 psi ______
4. Make the span adjustment ______
5. Repeat steps 1-4 until no further adjustments are required ______
6. Apply a ΔP of 25 psi ______
7. Lower the ΔP to the alarm set point ______
8. Confirm that the alarm set point is activated ______

4.2.11.7 <u>Pressure Transmitter EP No. 917</u>

1. Perform calibration checks on the following instruments as specified in applicable GEMAC instruction manuals.
   a. Pressure Transmitter 917 ______
   b. Alarm Indicator 918B ______
   c. Pressure Indicator 918A ______
2. Verify that "Reflector Hi Diff Pressure" annunciation and a loss of magnet current to the control rods occurs when the alarm indication exceeds the low and high set points ______

4.2.11.8 Pool Water Flow Instrumentation

1. Perform calibration checks on the following instruments as specified in applicable GEMAC instruction manuals.
   a. Flow element 921B ______
   b. Flow transmitter 921F ______
   c. Square Root converter 919F ______
   d. Power supply 911A ______
   e. Recorder 915B ______
   f. Dual alarm 920A and B ______
2. Verify that "Reactor Loop Low Flow" annunciation and a loss of magnet current to the control rods occurs when the alarm indication exceeds the set point ______

Date Test Completed ________________

Performed by ___________________________

4.3 Operational Testing and Approach to Ten Megawatt Power Levels

This procedure outlines the steps to be taken in addition to the MURR Standard Operating Procedures in preparation for 10 MW operation. There will be no deviation from the limits or requirements of the MURR technical Specifications. Deviations from approved Operating Procedures or power increase procedures will not be made without the approval of the Facility Director during the power increase program.

The power testing program will not be started until final completion of all preoperational test procedures as described in section 4.2 of this report.

The reactor is so constructed to be operated at 5 MW on each cooling leg of the system. The reactor will be operated for one week (100 hours) at 5 MW on each system; the reactor will be operated for eight hours after initial startup at 2 MW and for four hours at 4 MW prior to increasing power to 5 MW for the balance of the 100 hour test.

Following reactor operation at 5 MW on each system of one week (100 hours) the following program will be accomplished.

1. All systems will be place in the 10 MW operational mode.
2. All safety system circuits will be checked for proper operation in the 10 MW mode.
3. With all systems on line for 10 MW operation the power will be escalated in the following manner
   a. Normal startup to 1 MW, period not be less than 50 seconds.
   b. Operation at 1 MW for one hour.
   c. Increase power to 5 MW in 1 MW increments remaining at each power level for one hour.
   d. Operation at 5 MW for 12 hours.
   e. Increase power to 9 MW in 1 MW increments remaining at each ;power level for 12 hours.
   f. Increase power level to 9.5 MW and hold for 12 hours.
   g. Increase power to 10 MW and hold for 24 hours.

4. The following requirements will be adhered to during the power increase program of 4.3.2.
   a. Power will not be increased on less than a 100 second period above 1 MW.
   b. Power levels will be determined by heat balance immediately prior to each incremental power increase and if necessary nuclear instrumentation adjusted to agree with the calculated reactor power.
   c. A primary coolant sample is to be taken and analyzed for iodine at each power level increment above 5 MW and every 8 hours above 9 MW.
   d. The reactor will be operated in manual control for the first two hours at each power level above 5 MW and the power reduced .5 MW to switch control to automatic.
   e. Special care will be taken to observe for anomalies in flows, temperatures, high radiation areas, and reactor stability at all incremental power levels.
   f. The Reactor Supervisor or Facility Director shall be present and their approval received for each incremental power increase.
   g. The reactor shall be shut down upon the discovery of any suspected unsafe condition or significant deviation from calculated operating parameters in an unsafe direction.
   h. Two licensed reactor operators shall be in the control room at all times during operations above 5 MW, one of which shall be a senior reactor operator.

5.0 ENVIRONMENTAL REPORT

5.1 Introduction

The University of Missouri Research Reactor is located at Columbia, the county seat and largest city in Boone County, Missouri. The Reactor was licensed in 1963 with initial criticality on October 13, 1966. The present licensed power level of 5 MW was attained on June 30, 1967. Since September 29, 1969, the Reactor has been operating routinely at 5 MW for approximately 100 hours per week.

No additional site preparation or external construction is planned for the increase in licensed power to 10 MW. Therefore, the impact on the environment will be minimal. This section describes the operating history with respect to the release of radionuclides to the environment and provides data on the site in addition to that covered in the original application for 5 MW operation.[1]

5.2 Site Description

5.2.1 Plant Location and Land Usage

The Reactor is located approximately 1/2 mile south of the southern border of the residential area of Columbia, Missouri. Stadium Boulevard marks the general southern border of the city. A 5000 ft. radius has been chosen to represent the Reactor survey area. Flat Branch Creek essentially bisects the area north of Stadium Boulevard running jointly with the Missouri-Kansas-Texas Railroad line. These run together within a relatively wooded area between housing developments up to 200 ft. of Providence Road, outside of the survey area. The City of Columbia has long-range plans for the development of this Flat Branch area into a park and recreational area.

Fraternity and sorority housing, dormitories, Veterans Administration Hospital, and the University Medical Center are located in the survey area east of Providence Road and north of Stadium Boulevard. A portion of the survey site west of Providence Road and north of Stadium Boulevard is primarily residential. The primary business areas are in the central and northern portions of the city and are located outside of the survey area. The football stadium, Warren E. Hearnes Multi-Purpose Auditorium, and General Services complex are situated south on Stadium Boulevard and east of Providence Road (Route K). The home for the President of the University is positioned on a bluff overlooking Hinkson Creek and a large cornfield adjacent to the creek. The home is 1/4 of a mile straight south of the sports stadium. The rest of this area north of Hinkson Creek is woodland with the exception of an old field and a pasture, in which two horses are grazed. These fields are east of the Reactor at the extreme edge of the survey site. Another abandoned field is located just east of Providence Road and north of Hinkson Creek. Neither of these two fields have been in agricultural use for 2 to 5 years.

Approximately 30 acres of corn is planted annually in the rich flood plain just south of Hinkson Creek and east of Providence Road. The field continues south to a fence row forming a demarcation between it and an old field of 3 to 6 years. Several homes are located along Victoria Street, a gravel road leaving Route K southeast of the Reactor and reentering Route K near a trailer court. The land through this area south of the flood plain is extremely rocky, and is covered by woods or abandoned clearings. A small residential development can be found 1 mile south of the Reactor turning east off Route K.

The largest area of land within the survey site is west of Route K and south of Stadium Boulevard (Highway 740). Rollins Athletic Field, the baseball field and athletic house are southwest at the intersection of Stadium Boulevard. University Hall, containing offices for the University-Wide Administration, is just west of these fields. The A. L. Gustin, Jr. Golf Course is situated south of Highway 740, east of the railroad lines, west of the Botany Woods, and north of University agricultural research fields. The golf course is bordered on all sides by woodlands. The wooded strip between the course and Research Park (where the Reactor is located) is known as Botany Woods, and is controlled by the Biology Department of the University. The municipal sewage treatment facilities are in two parts; the first is located west of the railroad tracks on Stadium Boulevard. Sewage is pumped to the final plant west of the railroad lines at the extreme southern border of the reactor survey site. This sewage treatment plant in turn dumps into Hinkson Creek. The Forum Shopping Center lies partially within the site area just west of a residential housing development. The remainder of the land west of the railroad lines is densely wooded with several homes in the area. The University agricultural research fields are south of the golf course. This land is the highly fertile flood plain just north of the Hinkson. The field is in three parts running east to west. The first and second fields are separated by a wooded slope, the third by Flat Branch Creek. All this area is in corn with a small plot of the third field in beans. The topography south of Hinkson Creek is rolling and heavily wooded. There are several old fields that have been abandoned for 6 to 20 years at the southern end of the Reactor area. Some of this area is beginning to be developed into housing areas as a continuation of the residential developments already present south of the survey site.

Jewell Cemetery and a neighboring trailer park are west on Route K and south of two abandoned fields previously used as strip quarries. A portion of the field just north of the trailer park is being used for construction dumping.

The Reactor is only one of several facilities located within Research Park. The Science Instrument Shop and three greenhouses are south of the Reactor. The U.S.D.A. Biological Control of Insects Research Laboratory is just north of the Reactor. The Dalton Research Center is northeast of the Reactor and south of the Psychology Research Building and Nutrition feedlot. The feedlot contains 150 sheep involved in ruminant nutrition research. The Laboratory Animal Center containing 100 to 150 dogs used by the Medical School is situated west of the Psychology Research Building. The Research Park Development Building is located north of Psychology Research and south of the baseball stadium.

Approximately 50 head of beef cattle are grazed 2 miles southeast of the Reactor on Nifong Road. One mile south of the Reactor 10 dairy cattle (Jersey) are maintained in a small dairy operation. The raising of livestock in the areas adjacent to the Reactor survey site is limited and residential developments are increasing in number.

5.2.2 Demography

The Reactor site is located 1/2 mile south of Stadium Boulevard (Highway 740) on Route K. A 5000 ft. radius around the site includes an area of residential Columbia 1/2 mile north of Stadium Boulevard to Lathrop Road. This area north of Highway 740 is 1 1/2 miles wide from Crestland Avenue to the Medical Center. The population of this portion is 5,070 people living in housing, apartments and University dormitories. The portion of the Reactor survey area south of Highway 740, north of Hinkson Creek, and east of the Missouri-Kansas-Texas Railroad tracks supports no population. Populations within the site survey south of Hinkson Creek can only be estimated from the 1970 Census of Housing because of the division of the area into larger census tracts from the census blocks than is the case for the city. This portion of the population is roughly estimated at 600 people. This population resides in residential housing west of the railroad tracks, residential and mobile homes near Jewell Cemetery and scattered farm homes east of Route K and south of Hinkson Creek. Approximately 5,700 people of the 58,804 people of Columbia, Missouri, live within the 5,000 ft. radius of the Reactor.[2]

5.2.3 Physiography

Boone County lies in the northern Ozark border of the State of Missouri; one of several geographic regions into which the state is divided. It is superior in agricultural productivity to the Ozark uplands to the south, yet inferior to the plains of northern and western Missouri. Surface configuration is the primary consideration when studying the agricultural pattern. Crop production is confined primarily to valley bottoms and lower slopes. Corn and winter wheat, in general, are the most prevalent crops of the region. Much importance is also given to livestock because much of the land is poorly adapted to growing crops.[3]

Boone County can be divided into four physiographic divisions; the Centralia uplands, a flat upland plain 850 ft. above sea level, Columbia dissected uplands, Ashland hills lying between the Columbia dissected uplands, and the Missouri river flood plain, the latter being in the extreme southern tip of the county[4] (Figure 5.1).

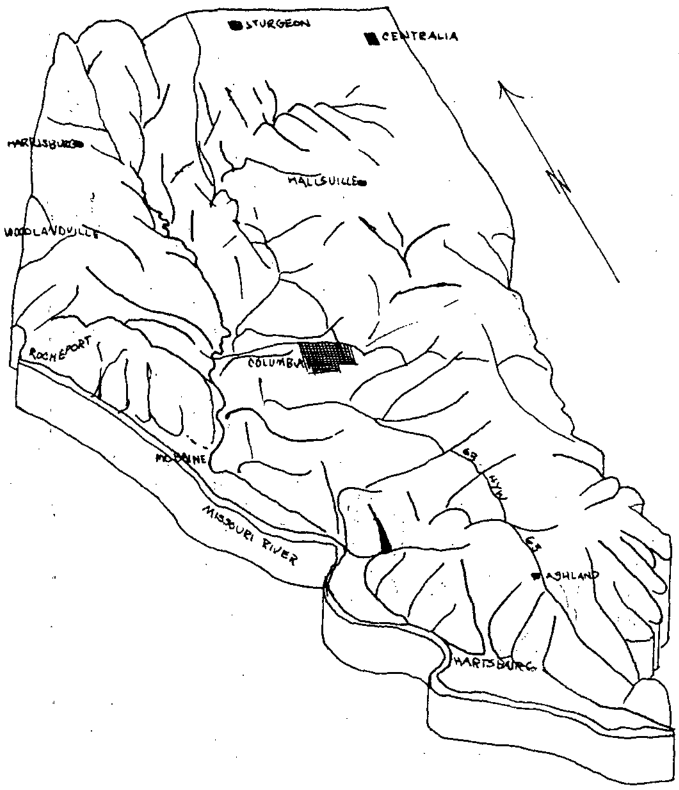


Figure 5.1 Relief and Drainage in Boone County[4]

5.2.4 Geology

The Reactor site and accompanying area is representative of central Boone County in that most of its topography is sloped along streams running through the area. The elevation varies from 650 ft. at the flood plains of Hinkson Creek where the agriculture fields, old fields, and Research Park are located, to 700 ft. at the peak of the slopes. The site is at a point just south (approximately 5 miles) of a change in the geology of central Boone County. The site is located on the edge of the Osagean series which is characterized by its crinoidal, very cherty, crystalline, and fossiliferous limestones having a thickness of 200 ft. To the north, the Desmoinesian series of the Pennsylvania system is represented by the cabaniss subgroup. This consists of sandstone, siltstone, shale, underclay, limestone, and coal beds.[5] Coal is a mineral of Columbia and is mined a few miles north of the city from this Pennsylvania system. The coal beds may range from 34 to 48 inches in this area of Missouri.[6] The Kansasian glacial stage of the Pleistocene layed a glacial till (clay, sand, and pebbles) in Boone County which was later covered by loss deposition following glaciation.

The soils are alluvial and tend to be stratified along the slopes. The golf course, Research Park and fields that have been cleared once represented a continuation of the woodland which is now restricted to the slopes with steep, stony terrains. The agricultural fields and Research Park are covered with Shirley silt loam of high fertility and favorable depth, with soils of the latter area being somewhat more intermediate. These areas represent Hinkson Creek flood plains and bottom areas. Flat Branch is much narrower, cutting between the slopes and joining the Hinkson Creek. The golf course and southern area of residential Columbia included in the survey area is a thin silt loam suitable for pastureland. The sloped woodlands are covered by loose, thin silt loam of low capabilities.[7] (Figure 5.2).

*Legend

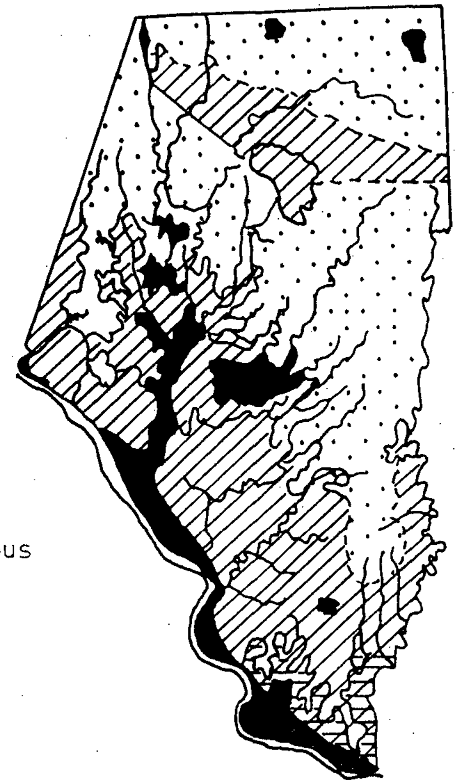
Alluvium

Pennsylvanian largely shales with some sandstone and limestone

Mississippian, mainly cherty limestone

Ordovician, mainly siliceous dolomitic limestone

---- Estimated boundary



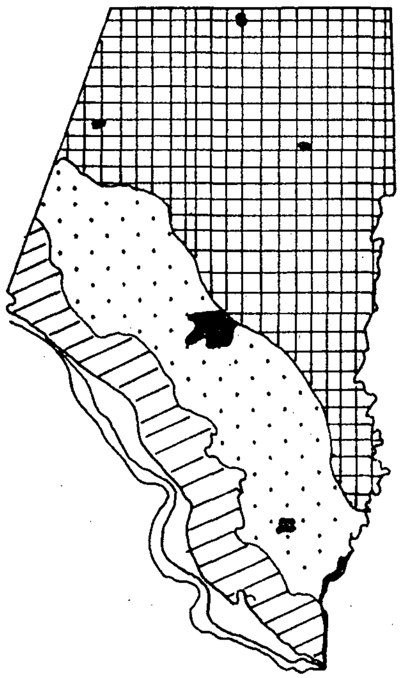
*Legend

50 inches

50 to 85 inches

85 to 240 inches

Depths represent thickness of loess mantle on ridgetops.



*From Krusekopf and Scrivner, 1962, Soil Survey of Boone Co., Missouri

Figure 5.2 Soils of Boone County, Missouri[7]

5.2.5 Seismology

Missouri can be divided into two parts when considering its seismology. Northern and western Missouri, excluding the Ozark uplift, is typical of the stable midcontinent United States. The Ozark area of southeastern Missouri has been the site of frequent uplifts. Considerable faulting has occurred there since pre-Cambrian times, giving the Ozarks its ruptured dome pattern. Earthquake epicenters are concentrated along the Mississippi Valley within the New Madrid and Charleston areas and are associated with the Ste. Genevieve fault system. The Cap-au-gra and Crystal City faults of the St. Louis area are sites that have had seismic activity.[8,9] The intensity of the midcontinent earthquakes are quite mild, although they are usually felt over wide areas. However, in 1811 one of the highest intensity seismic events of the United States was recorded when an earthquake occurred in the New Madrid area.[6,10]

Although seismic activity occurs in the Ozark region, it is relatively mild when earthquakes outside the midcontinent are considered. Boone County shows considerable stability and has had little or no evidence of seismic activity.

5.2.6 Hydrology

5.2.6.1 Surface Water

The trunk stream of the northern Ozark region is the Missouri river. The present Boone County surface is largely erosional with Perche Creek being the principal stream of the Boone County drainage basin. Hinkson Creek is one of the major tributaries of Perche Creek. The Hinkson originates in the Centralia uplands near Hallsville, Boone County, Missouri. It is approximately 15 miles from its origin to the Reactor area. By the time the Hinkson has reached the Reactor site, it has drained approximately 50 square miles. At its point of origin the creek drains the level prairies of Putnam silt loam, it then flows into the rocky soils of the Columbia dissected uplands. All of the tributaries of the stream are short with a steep gradient

and very few are named.[11] Grindstone Creek and Flat Branch Creek are the major tributaries entering at Columbia. The Grindstone joins Hinkson Creek 1 mile east of the Reactor and 1/2 mile south of Stadium Boulevard. Flat Branch Creek enters the Hinkson 3/4 of a mile southwest of the Reactor, southeast of the sewage treatment plant.
The stream bed is crooked and rocky with frequent riffles, many pools, and rocky or dirt banks. The average fall is 25 ft. per mile along the course of Hinkson Creek flowing in a southwestern direction joining Perche Creek 1 mile north of Brushwood, this later enters the Missouri River.[11]

The formation of Hinkson Creek occurred after the regression of the Kansanian glacier. The stream ran into the preglacial valley that was filled with glacial drift, taking advantage of these irregularities. The downward cutting was controlled by the level of the Missouri River, which was considerably higher than today due to the overland debris from the glacier. The area within the Reactor site offers an excellent example of "incised meanders" caused by the downward lateral cutting of the Hinkson as the Missouri River was lowered. During this incising the Hinkson entrenched itself from 80 to 120 ft. below the surrounding uplands.[12]

Hinkson Creek is a polluted stream carrying a relatively large amount of suspended materials. The majority of the tributaries near Columbia and the creek itself receive runoff and sewage from the city and surrounding areas. The Hinkson drains 70.2 square miles of Boone County.[13] The maximum recorded flow for the stream was 9100 cu ft/sec with no flow typically recorded several times within a year. The mean flow throughout 1970 was 89.81 cu ft/sec[13] (Table 5.1).

5.2.6.2 <u>Ground Water</u>

Boone County obtains much of its water supply from wells of Mississippian limestones and underlying Ordovician dolomites.[14] As a general rule the glacial deposits of Boone County are not good aquifers despite the presence of glacial sands and gravels in portions of upland Boone County.[14] Emince and

Potosi dolomites are dependable producers as are Gunder sandstone, Van Buren-Gasconade dolomites, Roubidoux sandstone, and Jefferson City dolomite. St. Petersburg sandstone, Chouteau limestone, Burlington limestone and glacial drift are merely local producers.[4] The Devonian and Pennsylvanian strata have little or no productive capabilities. Private wells range in depth from 200 to 450 ft. giving yields of 5 to 15 gallons per minute, with all deep wells of record producing fresh water.[14]

TABLE 5.1
SUMMARY OF HINKSON CREEK FLOW AT
COLUMBIA, MISSOURI (1969-1970)

	Average Flow (Cu. Ft. per Sec.)
October 1969	275.0
November 1969	20.2
December 1969	14.8
January 1970	14.0
February 1970	8.22
March 1970	15.0
April 1970	223.0
May 1970	254.0
June 1970	119.0
July 1970	1.60
August 1970	12.9
September 1970	120.0
Grand Average for Period	89.81

Based on Water Resources Data for Missouri, 1970; United States Department of the Interior Geological Survey (taken on left bank 400 ft. downstream from bridge on County Highway K, 0.5 mile south of Columbia, Missouri).

The City of Columbia receives its water supply from six wells located approximately seven miles southwest of Columbia near McBaine, Missouri. These wells are alluvial, located on the Missouri flood plain. Four wells are maintained for standby use in the western portion of the city. Two are situated near Broadway and Stadium Boulevard, a third 1 1/2 miles west of Stadium Boulevard and Broadway, and the fourth is 1/2 mile southwest of Columbia near Turner Road. The city has four other wells, three in the eastern portion of the city adjacent to the water and light plant, the fourth 1/2 mile north of the city. These are not used on a standby basis. The average depth of the standby wells located near the city is 1,325 ft. The average depth of the wells near McBaine is 100 ft.

The University of Missouri has four wells located on the campus at Columbia. Three of these wells have depths of 1,200 ft. while the fourth has a depth of 1,400 ft. Two of the wells are located within the 5,000 ft. radius of the survey area.

5.2.7 Climate

Columbia has a characteristic long, warm summer with a lower relative humidity than the areas of southern Missouri. The winters are cold and are marked by their low precipitation. The average high for July is 78.7°F with an average low of 30.3°F for January. Precipitation averages 36.96 inches yearly with an average yearly temperature of 55°F.[15]

The Columbia area is considered to be located at the prairie-forest border with a climate more closely resembling the prairie region. Thus, moisture becomes a very important consideration of the ecosystem when viewing secondary succession. Climate fluctuations have noticeable affect on plant succession.

Brees'[15] study and his construction of a mean monthly water balance diagram can give a clearer understanding of the stress encountered by plant communities in the area (Table 5.2 and Figure 5.3). Potential evapotranspiration is the amount of water that can be evaporated from a soil. If the actual evapotranspiration is less than the potential, there is a moisture deficiency.

Low winter precipitation can place a considerable amount of stress on plant communities of the area. Summer deficits rely on winter excess to recharge the loss. Because of the normal low-winter precipitation, winter rains become important for both agriculture and vegetation lacking a forest canopy.

5.2.8 Aquatic Biota

Hinkson Creek is a relatively shallow stream with variations in its appearance associated with annual precipitation. Its greatest flow periods occur between April and June and again through September and October. Summer and winter flow rates are decreased nearly ten-fold. The stream bottom is variable; loose rock and gravel make up a good percentage, while silt and sand coverings of pool bottoms are frequent.

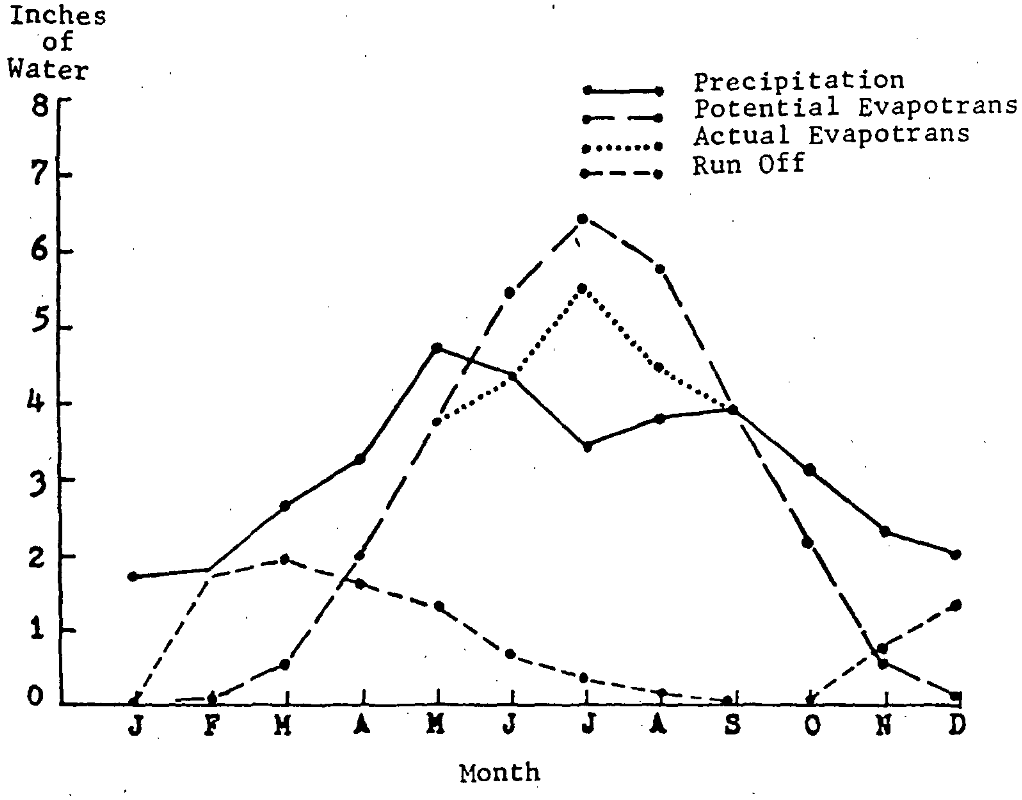
No true plankton communities exist in streams although plankton is present originating in quiet back waters and pools. Periphytes represent the primary level within the stream community because of their ability to attach to the surface of the stream bottoms. Many algae are characteristically sessile allowing diatoms and other organisms to grow in crustose masses attaching to their surface.[16] Land runoff is often the major contributor to the nutrients of streams. Terrestrial debris along with algae are the major food stuff of a stream community.[16] Diverse diatom populations are present; dominants belonging to the genus Navicula and Tabellaria. The algae forms are important in production of O_2, which contributes to the total oxygen dissolved within the stream. Thirty-two different genera of algae forms were identified in Hinkson Creek; the principal dominants were diatoms. These form the base of a stream community.

TABLE 5.2
TEMPERATURE AND WATER BALANCE AVERAGE VARIATION FOR COLUMBIA, MISSOURI OVER A 30-YEAR PERIOD[15]

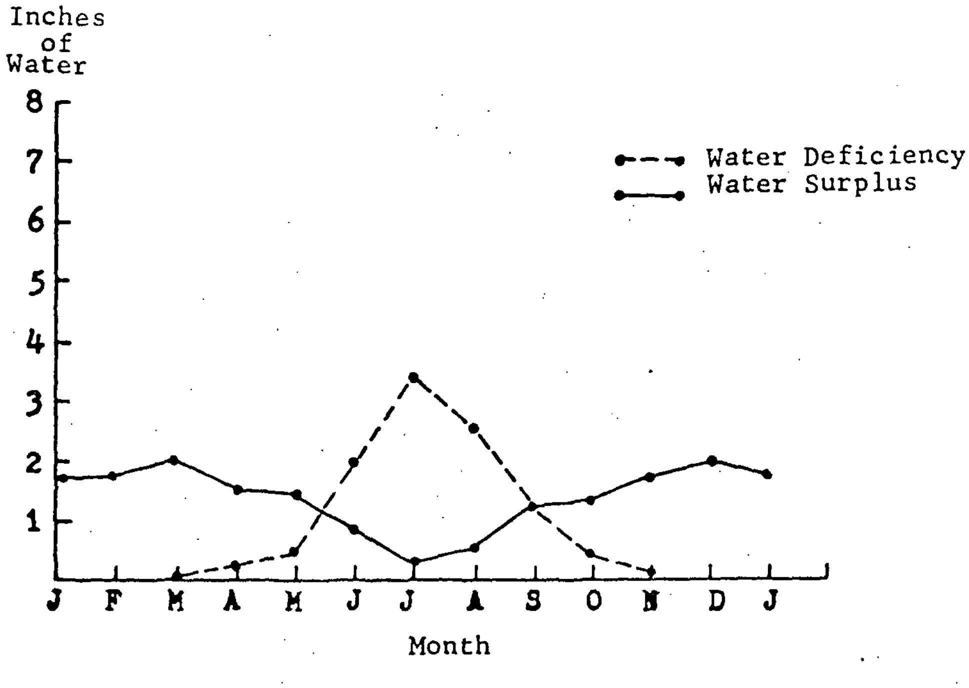
	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Total
Temp. °F	30.3	33.8	41.9	54.6	64.4	74.0	78.7	77.2	69.3	58.7	43.3	33.8	
Precipitation	1.71	1.81	2.65	3.30	4.70	4.34	3.41	3.81	3.88	3.12	2.27	1.96	36.96
Potential Evapo-transpiration	.00	.04	.55	1.98	3.71	5.47	6.48	5.79	3.86	2.19	.55	.04	30.66
Actual Evapo-transpiration	.00	.04	.55	1.98	3.71	4.34	5.51	4.43	3.86	2.19	.55	.04	27.20
Runoff	.00	1.74	1.92	1.62	1.30	.65	.33	.16	.00	.00	.78	1.35	9.85

Figure 5.3

THIRTY YEAR
MEAN WATER BALANCE FOR COLUMBIA, MISSOURI [15]



THIRTY YEAR
AVERAGE SURPLUS AND DEFICIENCY FOR COLUMBIA, MISSOURI



Bacterial communities are responsible for the decomposition of organic matter within a stream which can be considerable in a polluted environment. Bacteria also contribute as food for zooplankton organisms.[17] Estimates of the amount of organic matter available for decomposition is calculated by Biochemical Oxygen Demand (BOD). This can be compared with the Chemical Oxygen Demand (COD) that estimates the amount of organic compounds within the water.[18] These give estimates of the integrity of the stream. The presence of coliform organisms is used as an index of excretional pollution.[17] The relative abundance of the organisms indicate the safeness for human use. Hinkson Creek is relatively polluted with the decomposition contributing to oxygen loss and other detriments affecting the biota.

Protozoan, rotifers, and other microorganisms of this type are fewer in number in stream communities. They are confined primarily to decaying vegetation located along stream banks or pools within the course of the stream.

The macroinvertebrates constitute organisms which are essentially confined to bottom deposits on the surface of gravel and rocks. Mayflies, dragonflies, damsel flies, water bugs, caddisflies, beetles, midges, crustaceas and gastropoda represent the macroinvertebrates found in the survey. Mayflies and caddisflies are few in number due to their intolerance of a polluted environment.[19] Many of the larvae forms require a cool environment, spending this portion of their life cycle in the stream in the winter months.[18] Fifty-eight species of aquatic insects and crustaceans are known to reside in Hinkson Creek during some portion of the year. The gastropods are the dominant mollusca living within the stream. <u>Amnicola</u> and <u>Physa</u> are dominants located along bottoms, increasing in density under mats of vegetation near the water's edge. The invertebrates in general feed by scraping algae off of the surface of stone and rocks.

Although fish are the most motile of the aquatic organisms they are limited by the other organisms in lower trophic levels. They feed on organic detritus, larger invertebrates and other fish. They are the most affected in general by pollution.[19] Fish populations are not high in Hinkson Creek although twenty-five different species are present. Although sport fish are found in the creek, sport fishing is minimal.

TABLE 5.3
AQUATIC ORGANISMS RESIDING IN HINKSON CREEK*

Protozoa	
Frontonia	Ameoba
Euglena	Astasia
Dileptis	Paramecium
Stylonychia	Loxodes
Chlamydomas	
Chryanophyta (Blue Green Algae)	
Rivuluria	Lynbya
Chroocacius	Phormidium
Anabaenna	
Chlorophyta (Green Algae)	
Protococcus	Oedogonium
Ulothrix	Ulvales
Closterium	Hydrodictyon
Coelastrum	Ophrocytium
Zygnema	Spirogyia
Chrysophyta (Yellow Green Algae)	
Tubellaria	Melosira
Navicula	Stephanodiscus
Diatoma	Gamphonema
Meridion	Cymbella
Epithema	Nitzschia
Frustulia	Tribonema
Asterionella	Cladophora
Amphiprora	Tabellaria
Rotifera	
Euchlanis	Trichocera
Oorystacae	Epiphanes
Dimorphococcus	
Mollusca	
Gastropoda (Snails)	
Amnicola	
Physa	

*Collected by Reactor Staff using Aufwuch sampler, June 1972.

TABLE 5.4

FRESH WATER INVERTEBRATES

FROM STREAM SURVEY OF HINKSON CREEK*

Ephemeroptera (Mayflies)
- Baetidae
  - Caenis sp.
  - Centroptilum sp.
- Heptageniidae
  - Stenonema tripunctatum
  - Stenonema femoratum
  - Heptagenia maeulipennz
- Ephemeridae
  - Hexagenia munda munda
  - Hexagenia limbata

Odanata (Dragonflies and Damsel Flies)
- Coenargrionidae
  - Agia apicalis tebialis
  - Agia sedula
  - Agia moesta putrida
  - Enallagma sp.
  - Iohnura verticalis
- Libellalidae
  - Libellala vibrans
- Agrionidae
  - Agrioni aequabile

Hemiptera (Water Bugs)
- Hydrometridae
  - Hydrometra sp.
- Gerridae
  - Gerris remigis
  - Trepobates sp.
- Gelastocoridae
  - Gelastocoras oculatus
  - Enallagma sp.
  - Lethocerus sp.
- Veliidae
  - Rhaguelia sp.
  - Microvelia sp.

Trichoptera (Caddisflies)
- Hydropsychidae
  - Cheumatopsyche sp.
  - Hydropsyche sp.
  - Deplectrona sp.
- Limnephilidae
  - Caborius sp.

Coleoptera (Beetles)
- Haleplidae
  - Peltodytes simplex
  - Peltodytes tortulosus
- Elmdae
  - Ancyronyx sp.
  - Duliraphia sp.
  - Stenelinus sp.
- Cyriridae
  - Gyrinus sp.
- Dryopidae
  - Helichus sp.
- Dytiscidae
  - Hydroparus
  - Laccophilis
- Hydrophilidae
  - Troposternus ellipticus
  - Troposternus lateralis nimbatus
  - Erochrus
  - Paracymus
  - Helophosus

Diptera (Flies, Mosquitoes and Midges)
- Tipulidae
  - Tipula abdominalis
  - Hexatoma
- Simuliidae
  - Simulium vittatum
- Culicidae
  - Anopheles punctipennus
  - Culex territans
  - Culex pipens quinquefasciatus
- Tentanoceridae
  - Sepedon fuscipennis
- Tentepedidae
  - Tendipes (kiefferulus) dux
  - Tendipes (Tendipes) riparius
  - Anatopyria sp.

Crustacea
- Amphipoda
  - Talitridae
  - Hyallela azteca
  - Gammaradae
  - Grangonyz forbesi
  - Gammrus limaeus
- Isopoda
  - Asellidae
  - Asellus brevicaudus
  - Lirceus lineatus
- Decapoda
  - Astacidae
  - Orconectus sp.
- Hydracaina
  - Arrenuridae
  - Arrenurus sp.
- Collembola
  - Isotomidae
  - Isotomurus plaustris

*Data based on collections over several years by classes in the Department of Entomology under the direction of Dr. Wilbur Enns.

TABLE 5.5

SPECIES LIST OF FISH COMMON TO HINKSON CREEK*

Lepisosteidae
- Lepisosteus osseus - Long Nos Gar

Culpeidae
- Durosoma cepedianum - Gizzard Shad

Catostomidae
- Catostomus commersoni - Common White Sucker

Ictaluridae
- Ictalurus punctatus - Channel Catfish
- Ictalurus milas - Black Bullhead
- Ictalurus natalis - Yellow Bullhead

Catostomidae
- Moxostoma erythrurum - Golden Redhorse

Cyprinidae
- Cyrenus carpio - Common Carp

Cottidae
- Semotilus atromaculatus - Creek Chub
- Campotoma anomalum - Stone Roller
- Notropis winutus - Common Shiner
- Notropis lutrensis - Plains Red Shiner
- Notemigonus erysolencus - Golden Shiner
- Pimephales notutus - Blunt Head Minnow
- Pimephales promelas - Fathead Minnow

Percidae
- Perca flavescens - Perch
- Etheostoma negrum - Johnny Darter
- Percina caprodes semitasuata - Log Perch
- Etheostoma spectable spectable - Orange Throat Darter

Atherinidae
- Labidesthes sicculus - Brook Silversides

Serranidae
- Micropterus salmoides - Largemouth Bass
- Lepomis cyanellus - Fewwn Aundiah
- Lepomis humilis - Vluwfill
- Pomoxis annularis - White Crappie

*Expected species as per Dr. Arthur Witt, Department of Biological Sciences.

TABLE 5.6

ANALYSIS OF HINKSON CREEK WATER AT COLUMBIA, MISSOURI*

Analysis	Average
Nitrate, ppm NO_3	.573
Sulfate, ppm SO_4	68.25
Phosphate, ppm PO_4	.073
Ammonia, ppm NH_3	.495
Chloride, ppm Cl	24.69
Alkalinity, ppm	139.75
Total Hardness, ppm $CaCO_3$	184
Total Solids, ppm	433.2
Total Volatile Solids, ppm	95.6
Total Fixed Solids, ppm	337.5
Total Dissolved Solids, ppm	402.8
Volatile Dissolved Solids, ppm	127.2
Fixed Dissolved Solids, ppm	275.6
Chemical Oxygen Demand, mg/L	22.4
Biochemical Oxygen Demand, mg/L	4.65
Coliform Count, MPN cells/100 ml	16.869×10^4
Turbidity, Jackson Candle Scale	52.64

*Based on sampling of 5 stations located on stream as it run through Columbia, Missouri, by Civil Engineering 445 under the direction of Darrell L. King, Ph.D., during 1968.

TABLE 5.7

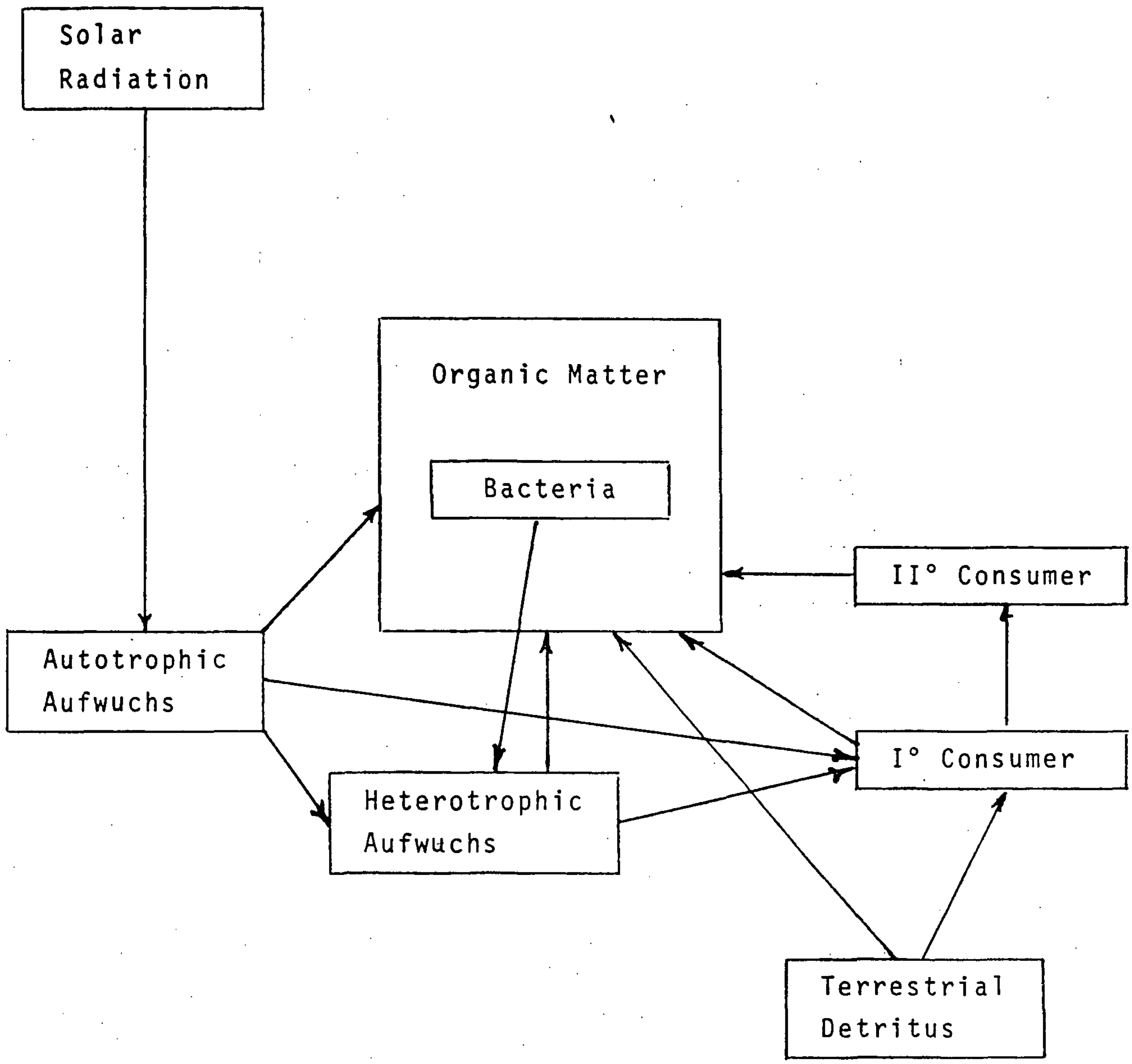
AUFWUCH, BACTERIA AND MACROINVERTEBRATE ANALYSIS*

Aufwuch Measurements (surface 1.4 cm^2)		Average
Relative Chlorophyll Conc.		37.31 mg/l
Total Dry Wt.		196.93 mg
Total Ash-Free Dry Wt.		50.037 mg
Chlorophyll/Organic Wt.		2.306
Organic Matter		
B.O.D.		4.375 mg/l
C.O.D.		22.4 mg/l
Coliform Test, MPN cells/100 ml		16.869×10^4
Wet Wt. of Bottom Organisms	Riffle	Pool
(grams/sq. ft.)	.1075	.91138

*Samples collected during April 1968, under the direction of Dr. Darrell King.

FIGURE 5.4

SIMPLIFIED FOOD WEB OF HINKSON CREEK



5.2.9 Terrestrial Flora

The natural residing vegetation of the Reactor site can be seen on the sloped woodland areas. These were formerly a portion of the extensive forests covering the Hinkson flood plain, which has been cleared for urban, agricultural, and University use. Many of the poorer agricultural fields have been abandoned and are now in a state of succession seeking the climax seen in the woodland.

Extensive studies have been done on the woodland slope west and adjacent to the Reactor. This area, known as Botany Woods, is owned by the University of Missouri and is used by the Biology Department for teaching purposes.

From Spellman's[20] study of the Botany Woods area Acer saccharum, Celtis occidentalis, Quercus muhlenbergii, and Quercus alba are dominants, with the latter being located on the top of the bluffs. These dominants represent the typical upland flora. Acer saccharum and Quercus muhlenbergii indicate a near climax status. The upland forest is more similar to the oak-hickory forests to the northeast than to Ozark oak-hickory stands to the south.[20] This represents a physiographic climax due primarily to topography, soils, and climate.[20,21,22] Productivity and biomass estimates have been taken of the upper and lower slopes indicating community dynamics and stratification typical of the woodland slopes within the Reactor survey site.[23]

The abandoned limestone strip pits and old fields with then rocky soils are in an intermediate state of secondary succession. Succession has been studied in old fields of mid-Missouri by Drew[21] and Huber[24] indi cating species dominance within the successional sequence. Annuals: common ragweed, yellow foxtail grass (Setaria glacica) and bull nettle (Solanum carolinense) are dominant species found with the last crop and persist for 2 to 3 years.[24] The perennials that follow, goldenrod (Solidago nemoralis) and white aster (Aster pilosus) increase in density. Panic grass (Panicum lanuginosum) and Broomsedge are also appearing at this time. For 5 to 20 years goldenrod and white aster remain dominant while annuals disappear as dominants after 3 years. In this

20-year period, cinquefoil (Potentilla simplex) and legumes increase in density.[21,24]

Following pioneering intermediate shrubs appear smooth sumac (Rhus glabra), winged sumac (Rhus copallina), blackberry (Rubus allegheniensis) and buckbrush (Symphoricarpos orbiculatus).[21,24] They begin shading to be later replaced by seedlings of tree species found among the shrubs. This is as far as the progression of succession has moved in the old fields of the area. Most of the fields have been abandoned from 5 to 20 years and are far from a climax status. The highest diversity of plant species occurs in the successional communities as expected with less diversity in communities near equilibrium. The flora contributes to a relatively diverse habitat for the fauna of the area.

TABLE 5.8
TREE AND SHRUB SPECIES FOUND WITHIN
THE REACTOR SURVEY SITE[24]

Acer saccharum	Sugar Maple
Ailanthus altissima	Tree of Heaven
Carya cordiformis	Bitternut Hickory
Carya ovata	Shagbark Hickory
Carya texana	Black Hickory
Carya tomentosa	Mockernut Hickory
Celastrus scandens	Bittersweet
Celtis occidentalis	Hackberry
Cercis canadensis	Redbud
Cornus drummondii	Dogwood
Cornus florida	Flowering Dogwood
Crataegus sp.	Hawthorn
Diospyros virginiana	Persimmon
Euonymus atropurpureus	Wahoo
Fraxinus americana	White Ash
Fraxinus quadrangulata	Blue Ash
Juglans nigra	Black Walnut
Juniperus virginiana	Red Cedar
Morus rubra	Red Mulberry
Parthenocissus quinquefolia	Virginia Creeper
Prunus americana	Wild Plum
Prunus serotina	Black Cherry
Quercus alba	White Oak
Quercus imbricaria	Shingle Oak
Quercus muhlenbergii	Chinkapin Oak
Quercus rubra	Red Oak
Quercus stellata	Post Oak
Quercus velutina	Black Oak
Rhus aromatica	Aromatic Sumac
Rhus copallina	Winged Sumac
Rhus glabra	Smooth Sumac
Rhus radicans	Poison Ivy
Rosa multiflora	Multiflora Rose
Rubus allegheniensis	Blackberry
Rubus occidentalis	Black Raspberry
Sambucus canadensis	Elderberry
Sassafras albidum	Sassafras
Smilax tamnoides	Green Briar
Tilia americana	Basswood
Ulmus americana	American Elm
Ulmus rubra	Red Elm
Viburnum prunifolium	Black Haw
Viburnum rufidulum	Rusty Black Haw
Vitis aestivalis	Summer Grape

TABLE 5.9
SEEDLING SPECIES FOUND WITHIN
THE REACTOR SURVEY SITE[24]

Campsis radicans	Trumpet Creeper
Carya cordiformis	Bitternut Hickory
Carya texana	Black Hickory
Ceanothus americanus	New Jersey Tea
Celastrus scandens	Bittersweet
Celtis occidentalis	Hackberry
Cornus drummondii	Dogwood
Cornus florida	Flowering Dogwood
Crataegus sp.	Hawthorn
Diospyros virginiana	Persimmon
Fraxinus pennsylvanica	Green Ash
Fraxinus quadrangulata	Blue Ash
Gleditsia triacanthos	Honeylocust
Juglans nigra	Black Walnut
Juniperus virginiana	Red Cedar
Parthenocissus quinquefolia	Virginia Creeper
Pinus echinata	Shortleaf Pine
Prunus serotina	Black Cherry
Quercus muhlenbergii	Chinkapin Oak
Quercus rubra	Red Oak
Quercus velutina	Black Oak
Rhamnus lanceolata	Buckthorn
Rhus aromatica	Aromatic Sumac
Rhus glabra	Smooth Sumac
Rhus radicans	Poison Ivy
Rosa carolina	Pasture Rose
Rosa multiflora	Multiflora Rose
Rubus allegheniensis	Blackberry
Rubus flagellaris	Dewberry
Rubus occidentals	Black Raspberry
Sassafras albidum	Sassafras
Smilax tamnoides	Green Briar
Symphoricarpos orbiculatus	Buckbrush
Ulmus americana	American Elm
Vitis aestivalis	Summer Grape

TABLE 5.10
HERB SPECIES FOUND WITHIN THE
REACTOR SURVEY SITE[24]

Acalypha virginica	Three-Seeded Mercury
Ambrosia artemisiifolia	Common Ragweed
Andropogon virginicus	Broomsedge
Antennaria plantaginifolia	Pussy Toes
Aristida longespica	Triple-Awned Grass
Aster pilosus	White Aster
Atrichum sp.	Atrichum
Bidens polylepis	Bur Marigold
Bromus tectorum	Brome Grass
Carex complanata	Sedge
Carex flaccosperma	Sedge
Cassia fasciculata	Partridge Pea
Cirsium altissimum	Tall Thistle
Danthonia spicata	Poverty Grass
Daucus carota	Queen Anne's Lace
Desmodium paniculatum var. dillenii	Tick Trefoil
Erigeron canadensis	Horse Weed
Erigeron strigosus	Daisy Fleabane
Eupatorium serotinum	Thoroughwort
Galium circaezans	Bedstraw
Geum canadense	White Avens
Helianthus hirsutus	Sunflower
Heuchera richardsonii	Alum Root
Hypericum punctatum	St. John's Wort
Juncus tenuis	Bog Rush
Lactuca canadensis	Wild Lettuce
Leptoloma cognatum	Fall Witch Grass
Lespedeza cuneata	Bush Clover
Lesedeza intermedia	Bush Clover
Lespedeza stipulacea	Korean Clover
Lespedeza violacea	Bush Clover
Lespedeza virginica	Bush Clover
Muhlenbergia tenuiflora	Muhlenbergia
Oenothera biennis	Evening Primrose

TABLE 5.10 (Cont'd)

Oxalis stricta	Yellow Wood Sorel
Panicum lanuginosum var. fasciculatum	Panicum
Panicum linearifolium	Panicum
Paspalum ciliatifolium	Paspalum
Potentilla simplex	Cinquefoil
Prunella vulgaris	Selfheal
Pycnanthemum tenuifolium	Mountain Mint
Rudbeckia hirta	Coneflower
Sanicula canadensis	Black Snakeroot
Solidago altissima	Tall Goldenrod
Solidago nemoralis	Old Field Goldenrod
Strophostyles helveola	Wild Bean
Strophostyles leiosperma	Wild Bean
Teucrium canadense	Germander
Trifolium repens	White Clover

TABLE 5.11

SUMMARY OF DATA FROM THE BOTANY WOODS BASED ON A COMPLETE INVENTORY[20]

Species	Total No. of Stems	Percent of Total	Stems per Acre	Basal Area	Precent of Total	Basal Area per Acre
Acer saccharum	227	17.40	51.6	43.2	8.11	9.8
Celtis occidentalis	182	13.90	41.4	53.5	10.04	12.2
Quercus muehlenbergii	162	12.40	36.8	86.7	16.28	19.7
Ulmus americana	114	8.70	25.9	32.1	6.02	7.3
Quercus alba	101	7.70	22.9	83.8	15.73	19.0
Aesculus glabra	67	5.10	15.2	7.1	1.33	1.6
Fraxinus quadrangulata	56	4.30	12.7	20.5	3.84	4.7
Ulmus rubra	56	4.30	12.7	34.7	6.51	7.9
Cercis canadensis	52	4.00	11.8	5.9	1.10	1.3
Ulmus thomasi	50	3.80	11.4	14.3	2.68	3.3
Carya ovata	35	2.70	8.0	11.8	2.21	2.7
Quercus rubra	31	2.40	7.0	32.7	6.14	7.4
Juglans nigra	22	1.70	5.0	17.2	3.23	3.9
Carya cordiformis	21	1.60	4.8	6.4	1.20	1.5
Tilia americana	19	1.50	4.3	10.0	1.87	2.3
Acer saccharinum	17	1.30	3.9	8.3	1.55	1.9
Ostrya virginiana	13	1.00	2.9	0.9	0.16	0.2
Quercus macrocarpa	12	0.92	2.7	13.9	2.61	3.2

Species	Total No. of Stems	Percent of Total	Stems per Acre	Basal Area	Precent of Total	Basal Area per Acre
Gleditsia triacanthos	12	0.92	2.7	5.7	1.07	1.3
Asimina triloba	11	0.84	2.5	0.4	0.07	0.1
Quercus velutina	9	0.69	2.0	8.0	1.50	1.8
Platanus occidentalis	9	0.69	2.0	28.9	5.42	6.6
Morus rubra	8	0.61	1.8	0.7	0.13	0.2
Carya tomentosa	6	0.46	1.4	1.5	0.28	0.3
Quercus imbricaria	3	0.23	0.7	2.8	0.53	0.6
Fraxinum pennsylvanica	2	0.15	0.5	0.8	0.15	0.8
Acer negundo	2	0.15	0.5	0.1	0.01	0.1
Amelanchier arborea	2	0.15	0.5	0.1	0.01	0.1
Juglans cinerea	1	0.08	0.2	0.1	--	--
Maclura pomifera	1	0.08	0.2	0.3	0.05	0.1
Gymnocladus dioicus	1	0.08	0.2	0.6	0.11	0.1
Crataegus sp.	1	0.08	0.2	0.1	--	--
TOTALS	1305	100.00	294.6	532.5	100.00	120.2

TABLE 5.12
PRODUCTIVITY AND BIOMASS OF TWO FOREST COMMUNITIES
(BOTANY WOODS)*

The following summary is based on data from upper and lower slope communities on Hinkson Creek near the University Reactor.

SITE I (Lower slope)

Size Class -in	Density -A	Annual Inc.-A	Total Biomass-A	BAR
0.5	2300	20.5	50	2.4
0.6-3.5	830	3283.5	11080	3.3
3.6-10.5	170	1986.5	29503	14.8
10.6-16.5	34	1500.0	33370	22.1
16.5-20.5	7	425.0	18474	43.4
20.5	3	98.0	7120	72.7
Total	3344	7313.5	99597	

Leaf fall: 3215.5 lbs/A
Herbs & grasses: 237.5 lbs/A
Total growth/yr (woody + herbaceous) = 10766.5
Litter biomass: 15308 lbs/A
K = A/A+L = 0.17 t.t. = 5.8 yrs

SITE II (Upper slope)

Size Class -in	Density -A	Annual Inc.-A	Total Biomass-A	BAR
0.5	5533	276.0	968	3.5
0.6-3.5	506	1802.0	5816	3.2
3.6-10.5	143	3529.6	25446	7.2
10.6-16.5	82	1881.0	65169	34.5
16.6-20.5	17	1421.0	26001	18.3
Total	6281	8909.6	123400	

Leaf fall: 2562.6 lbs/A
Herbs & grasses: 654.3 lbs/A
Total growth/yr (woody + herbaceous) = 11126.2
Litter biomass: 12266.7
K = A/A+L = 0.17 t.t. = 5.8 yrs

*Determined by students in Botany 406 under the direction of Dr. Clair Kucera.

5.2.10 Terrestrial Fauna

The resident animal population of the survey area is fairly typical of northeastern Missouri. Much of the area is wooded offering a good wildlife habitat. The Hinkson Creek bottomland and forested slopes allow a diversity of the animal population not present in other surrounding areas of the city. The major limitation of species diversity is human intrusion.

Resident bird populations are quite diverse because of the forest, field, and creek habitat. Eighty-five species of birds are known to reside within the survey site. Nest building generally occurs in early April with egg laying in early May.[25] Eastern Missouri is located within the Mississippi flyway. This particular flyway is excellent for waterfowl, especially mallards, Canadian geese and pintails because of the river basins lying along the way.[26] Waterfowl probably would not be seen this close to the city, except when rarely forced down by severe weather conditions. Forty-seven species of amphibians and reptiles are associated with the ponds, creeks, fields and wooded environment. Thirty species of mammals are known residents of this area. Secluded portions of the creek offer a stream habitat for raccoons, mink and beaver. One species of bat was the only animal of the area found to be an endangered species. This species, Myotis sodalis,[27] the Indiana bat, is typical of the five mouse-eared bats of the region. It hibernates with colonies in limestone caves in the southern part of the State during the winter months. They disperse in the summer and spring, the females living in nursery colonies located in cliff crevices, hollow trees, or any other inaccessible retreat. The males stay in places similar to the females, yet remain segregated throughout the summer. The bats are insectivorous, eating caddisflies, winged ants, moths, flies and other insects of this size. Mating occurs in the fall before hibernation and again in spring following the hibernation. A single young is born annually following a 50 to 60 day gestation period. The predators are raccoons, cats, rats, hawks, owls, snakes-usually foraging them from their summer roosts.[27]

TABLE 5.13
RESIDENT TERRESTRIAL ANIMALS
OF THE REACTOR SURVEY SITE

Amphibians & Reptiles[28,29,30]

Amphibians

Psuedacris triseriata triseriata - Western Chorus Frog
Hyla crucifer - Spring Peeper
Acris crepitans blanchardi - Blanchards Cricket Frog
Rana cutesheiana - Bullfrog
Rana clamitans melanota - Green Frog
Bufo woodhousei Fowleri - Fowler's Toad
Bufo americanas - American Toad
Ambystoma trigrinum trigrinum - Eastern Tiger Salamander
Ambystoma maculatum - Spotted Salamander
Ambystoma texanum - Small-Mouthed Salamander
Necturus maculosus - Mudpuppy

Reptiles

Chelydra serpentina serpentina - Common Snapping Turtle
Sternothaerus odoratus - Stinkpot
Terrapene ornata ornata - Ornate Box Turtle
Graptemys geographica - Map Turtle
Graptemys kohni - Mississippi Map Turtle
Chrysemys picta belli - Western Painted Turtle
Pseudemys scripta elegans - Red-Earred Turtle
Trionyk spinifer hartwegi - Western Spinney Soft Shelled Turtle
Trionyk mutica mutica - Smooth Soft Shelled Turtle
Sceloporus undulatus hyncinrinus - Northern Fence Lizard
Ophisaurus attenuatus attenuatus - Western Slender Glass Lizard
Cnemidophorus sexlineatus - Six-Lined Racerunner
Eumeces fascatus - Five-Lines Lizard
Gumeces laticeps - Broadheaded Skink
Natrix grahami - Graham's Water Snake
Natrix rhombifera rhomifera - Diamond Back Water Snake
Natrix sipendon sipedon - Northern Water Snake
Storeria dekayi wrightorum - Midland Brown Snake

TABLE 5.13 (Cont'd)

Reptiles - cont'd

Storeria dekayi texana - Texas Brown Snake
Storeria occipitomaculata occipitomaculata - Northern Red-Bellied Snake
Thamnophis sauritus proximus - Western Ribbon Snake
Thamnophis sirtalis sirtalis - Eastern Garter Snake
Thamnophis sirtalis parietalis - Red-Sided Garter Snake
Thopidoclonion lineatum lineatum - Northern Lined Snake
Diadephis punctatus arnyi - Prairie Ringneck Snake
Carphophis amoenus vermis - Western Worm Snake
Coluber constricter flaviventis - Eastern Yellow-Bellied Racer
Opheodrys aestivus - Rough Green Snake
Elaphe obsoleta obsoleta - Black Rat Snake
Pituophis melanoleucus sayi - Bull Snake
Lampropeltis calligaster calligaster - Prairie King Snake
Lampropeltis getulus holbrooki - Speckled King Snake
Lampropeltis doliata suspila - Redmilk Snake
Agkistrodon contortrix mokeson - Northern Copperhead
Crotalus horridus horridus - Timber Rattlesnake

TABLE 5.13 (Cont'd)
Resident Mammals[31]

Didelphis marsupialis - Opossum
Scalopus aquaticus - Eastern Mole
Cryptotis parud - Little Short-Tailed Shrew
Blarind brevicauda - Large Short-Tailed Shrew
Myotis lucifugus - Little Brown Bat
Myotis sodalis - Indiana Bat
Eptesicus fuscus - Big Brown Bat
Lasiurus borealis - Red Bat
Procyon lotor - Raccoon
Mustela vison - Mink
Mephitis mephitis - Striped Skunk
Vulpes fulva - Red Fox
Urocyon cinereoargenteous - Gray Fox
Marmota monax - Groundhog
Sciorus carolinensis - Gray Squirrel
Sciorus niger - Fox Squirrel
Glaucomys volans - Flying Squirrel
Castor canadensis - Beaver
Recthrodontomys megalotis - Prairie Harvest Mouse
Peromyscus maniculatus - Prairie White Footed Mouse
Peromyscus leucopus - Woodland White Footed Mouse
Synaptomys cooperi - Lemming Mouse
Microtus ochrogaster - Meadow Mole
Pitymus nemoralis - Pine Mouse
Ondatra zibethicus - Muskrat
Rattus norvegicus - Norway Rat
Mus musculus - House Mouse
Sylvilagus floridanus - Cottontail
Odocuileus virginianus - White Tailed Deer

TABLE 5.13 (Cont'd)
Permanent Bird Residents[31]

Ardea herodias - Great Blue Heron
Butorides virescens - Green Heron
Nycticorax nycticorax - Black Crowned Night
Nyctanossa violacea - Yellow Crowned Night
Botaurus lentigenosus - American Bittern
Cathartes aura septentnionalis - Turkey Vulture
Accipiter velox - Sharp Skinned Hawk
Accipiter cooperi - Cooper's Hawk
Buteo borealis borealis - Red Tailed Hawk
Buteo lineatus lineatus - Red Shouldered Hawk
Buteo platypterns - Broadwinged Hawk
Circus hudsonius - Marsh Hawk
Porzana carolina - Sora
Oxyechus vociferus - Killdeer
Philohela minor - Woodcock
Gullinago delicata - Wilson's (Common) Snipe
Zenaidura macroura carolinensis - Mourning Dove
Strix varia varia - Barred Owl
Otus asio asio - Screech Owl
Bubo virginianus virginianus - Great Horned Owl
Myiarchus crinitus - Crested Flycatcher
Sayornis phoebe - Phoebe
Caprimulugus voiciferus - Whippoorwill
Chordeiles minor - Night Hawk
Choetura pelagica - Chimney Swift
Archilochus colubris - Ruby Throated Hummingbird
Megaceryle alcyon alcyon - Belted King Fisher
Colaptes auratus - Yellow Shafted Flicker
Hylatomux pileatus - Pileated Woodpecker
Centurus carolinus - Red Bellied Woodpecker
Melanerpes erythrocephalus - Red Headed Woodpecker
Dendrocopus villorus - Hairy Woodpecker
Dendrocopus pubescens - Downy Woodpecker
Tyrannus tyrannus - Eastern Kingbird
Myiarchus crinitus - Great Crested Flycatcher

TABLE 5.13 (Cont'd)
Permanent Bird Residents[31]

Sayornis phoebe - Eastern Phoebe
Contopus virens - Wood Pewee
Eremophila alpestyris - Horned Lark
Stelgidopteryx ruficollis serripennis - Rough Winged Swallow
Progne subis subis - Purple Martin
Cyanocitta cristata - Blue Jay
Carvus brachyrhynchos - Common Crow
Parus atricapillus - Black Capped Chickadee
Parus bicolor - Tufted Titmouse
Sitta carolinensis - White Breasted Nuthatch
Troglodytes aedon - House Wren
Thryothorus ludovicianus - Carolina Wren
Mimus polyglottus polyglottus - Mockingbird
Dumetella carolnensis - Catbird
Toxostoma rufum rufum - Brown Thrasher
Turdus migatorius - Robin
Hylocichla mustelina - Wood Thrush
Sialia sialis - Eastern Bluebird
Polioptila coerulea coerulea - Blue Gray Gnatcather
Columbia livia - Rock Dove
Colinus virginianaus - Bobwhite
Bombycilla cedorum - Cedar Waxwing
Lanius ludovicianus - Loggerhead Shrike
Vireo griseus - White-Eyed Vireo
Vireo olivaceus - Red-Eyed Vireo
Vireo gilvus vilvus - Warbling Vireo
Helmitheros vermivorus - Worm Eating Warbler
Verwora pinus - Blue Winged Warbler
Parula americana - Parula Warbler
Dendroica petechia - Yellow Warbler
Seiurus aurocapillers - Ovenbird
Seiurus motocilla - Louisiana Water Thrush
Oporonis formosus - Kentucky Warbler
Geothypis trichas - Yellow Throat Warbler
Icteria virens virens - Yellow Breasted Chat

TABLE 5.13 (Cont'd)
Permanent Bird Residents - cont'd

Passer comesticus domesticus - House Sparrow
Sturnella magna - Eastern Meadowlark
Agelaius phoeniceus - Red Winged Blackbird
Icterus spurius - Orchard Oriole
Icterus galbula - Baltimore Oriole
Quiscalus quiscula - Common Grackle
Molothrus ater ater - Cowbird
Piranga rubra rubra - Summer Tanager
Richmondema cardinales - Cardinal
Pheurticus ludovicianus - Rose Breasted Crossbeak
Passerina cyanea - Indigo Bunting
Spinus tristis tristis - Goldfinch
Pipilo erythophthalmus - Rufous-Sided Towhee
Spizella passerina passerina - Chipping Sparrow
Spizella pusilla pusilla - Field Sparrow

5.3 Effluents

5.3.1 Total Radioactivity Released to the Environment:

5.3.1.1

Period	Total Activity (curies) Liquid	Gaseous
July 1, 1966 - June 30, 1967	0.002	17.
July 1, 1967 - June 30, 1968	0.084	289.
July 1, 1968 - June 30, 1969	0.689	503.
July 1, 1969 - June 30, 1970	0.945	821.
July 1, 1970 - June 30, 1971	0.570	897.
July 1, 1971 - June 30, 1972	0.048	903.

5.3.1.2 Solid Wastes

All low level solid wastes from the Research Reactor Facility is released to a licensed commercial firm for burial. Spent fuel is shipped to licensed reprocessors.

5.3.2 Identification of Principal Radionuclides and Estimated Quantities in Liquid and Gaseous Effluents

5.3.2.1 Liquid

Period	Nuclide	Activity (curies)
July 1, 1966 - June 30, 1967	Na-24	0.001
	Cr-51	<0.001
July 1, 1967 - June 30, 1968	Sb-124	0.045
	Na-24	0.037
	Tc-96	0.001
	La-140	0.001
	Cd-109	<0.001
July 1, 1968 - June 30, 1969	Sb-124	0.355
	H-3	0.124
	Na-24	0.079
	Zn-65	0.067
	Nb-95	0.063
	Cr-51	0.001

July 1, 1969 - June 30, 1970	Na-24	0.366
	H-3	0.329
	Sb-124	0.225
	Other	0.025
July 1, 1970 - June 30, 1971	H-3	0.507
	Sc-46	0.019
	Sb-124	0.014
	Cr-51	0.012
	Na-24	0.011
	Co-60	0.003
	Mn-54	0.002
	Other	0.002
July 1, 1971 - June 30, 1972	H-3	0.040
	Co-60	0.002
	Na-24	0.002
	Sb-124	0.001
	Sc-46	0.001
	Mn-54	0.001
	Other	0.001

5.3.2.2 Gaseous

The principal nuclide in gaseous effluents has been argon-41. Relatively small quantities (estimated to be less than 1 curie per year) of tritium have been released.

5.3.3 Evaluation of Environmental Impact of Increased Stack Release Flow Rate

Stack release limits set for MURR in Technical Specification(1) Number 3.7: "Facility Gaseous and Particulate Radioactive Release" are based on activity concentrations. An increase in stack flow affects the total allowable release of activity, and thus this evaluation is made to assess the environmental impact the increase will have on the nearest resident and on the population surrounding the MURR. The change in stack height and exhaust exit path is also considered. The safety significance of the impact is discussed in relation to background radiation and in relation to a previous environmental impact appraisal made by NRC.(2)

Data and Assumptions

The data and calculations in Table 1 describe the physical information of the stack release point. Argon-41 is the principal isotope released in gaseous effluents from MURR. The Technical Specification limit for Ar-41 release is 350 times the MPC listed in Appendix B, Table II, Column I of 10CFR20, or:

$$
\begin{aligned}
Q &= 350 \text{ x MPC x flowrate} \\
&= (350)\,(4 \times 10^{-8}\ \mu\text{Ci/ml})\,(36500\ \text{ft}^3/\text{min})\,(2.831 \times 10^{4}\ \text{ml/ft}^3) \\
&= (1.4 \times 10^{4}\ \mu\text{Ci/min})\,(1 \times 10^{-6}\ \text{Ci}/\mu\text{Ci})\,(1\ \text{min}/60\ \text{sec}) \\
&= 2.4 \times 10^{-4}\ \text{Ci/sec}
\end{aligned}
$$

In the previous environmental assessment,(2) the NRC used meteoro-logical data collected at the Callaway Plant, located near Fulton. These data were collected between May 5, 1973, and May 4, 1975, and were judged by the NRC to be "reasonably representative of long-term conditions expected at the MURR site." This current assessment utilizes meteorological data gathered in Columbia, MO from 1960 to 1969.(3) The Columbia data was judged to be more appropriate for use in assessing airborne releases from MURR because of the longer data period and the proximity of the data site to MURR. Table 2 lists wind data (stability, class, speed and frequency) for each of the sixteen compass points.

TABLE 1

PHYSICAL INFORMATION FOR STACK RELEASE POINT

Elevation above sea level = 687 feet

Diameter = 40 inches

New Max flowrate = 36500 ft^3/min

$$\text{Area Cross Section} = \pi r^2$$

$$= \pi \left(\frac{40 \text{ inches}}{2 \cdot 12 \text{ inches/ft}} \right)^2$$

$$= 8.73 \text{ ft}^2$$

$$\text{Air Velocity (v)} = \frac{36500 \text{ ft}^3/\text{min}}{8.73 \text{ ft}^2} \cdot 0.304 \text{ m/ft} \cdot \frac{1 \text{ min}}{60 \text{ sec}}$$

$$= 21.2 \text{ m/sec}$$

TABLE 2
METEOROLOGICAL DATA--COLUMBIA, MO (1960-1969)[3]

Stability Class Information NNE

Class[a]	% Class[b]	Wind Speed[c] (m/sec)	%NNE[d] Wind	%'s[e] Comb.
A	0.4	2.3	3.4	1.4E-04
B	4.7	2.8	2.7	1.3E-03
C	11.5	4.0	3.5	4.0E-03
D	53.6	5.7	4.2	2.3E-02
E	17.6	3.8	3.2	5.6E-03
F	12.2	2.4	4.9	6.0E-03

Stability Class Information NE

Class[a]	% Class[b]	Wind Speed[c] (m/sec)	%NE[d] Wind	%'s[e] Comb.
A	0.4	2.1	1.7	6.8E-05
B	4.7	2.7	2.6	1.2E-03
C	11.5	3.7	2.7	3.1E-03
D	53.6	5.2	3.9	2.1E-02
E	17.6	3.6	2.8	4.9E-03
F	12.2	2.5	4.9	6.0E-03

Stability Class Information ENE

Class[a]	% Class[b]	Wind Speed[c] (m/sec)	%ENE[d] Wind	%'s[e] Comb.
A	0.4	2.0	7.8	3.1E-04
B	4.7	2.8	5.1	2.4E-03
C	11.5	3.9	4.3	4.9E-03
D	53.6	4.9	4.7	2.5E-02
E	17.6	3.4	4.2	7.4E-03
F	12.2	2.5	6.8	8.3E-03

Stability Class Information **E**

Class[a]	% Class[b]	Wind Speed[c] (m/sec)	%E[d] Wind	%'s[e] Comb.
A	0.4	2.0	4.3	1.7E-04
B	4.7	2.9	5.3	2.5E-03
C	11.5	3.8	4.4	5.1E-03
D	53.6	4.9	4.4	2.4E-02
E	17.6	3.5	5.0	8.8E-03
F	12.2	2.5	7.9	9.6E-03

Stability Class Information **ESE**

Class[a]	% Class[b]	Wind Speed[c] (m/sec)	%ESE[d] Wind	%'s[e] Comb.
A	0.4	2.0	3.4	1.4E-04
B	4.7	2.9	4.7	2.2E-03
C	11.5	3.9	4.8	5.5E-03
D	53.6	5.3	6.1	3.3E-02
E	17.6	4.0	6.1	1.1E-02
F	12.2	2.6	4.5	5.5E-03

Stability Class Information **SE**

Class[a]	% Class[b]	Wind Speed[c] (m/sec)	%SE[d] Wind	%'s[e] Comb.
A	0.4	2.2	2.6	1.0E-04
B	4.7	2.9	4.6	2.2E-03
C	11.5	4.1	6.4	7.4E-03
D	53.6	5.7	7.8	4.2E-02
E	17.6	4.1	8.2	1.4E-02
F	12.2	2.5	4.3	5.2E-03

TABLE 2 - CONT'D

Stability Class Information			SSE	
Class(a)	% Class(b)	Wind Speed(c) (m/sec)	%SSE(d) Wind	%'s(e) Comb.
A	0.4	2.3	4.3	1.7E-04
B	4.7	3.0	6.5	3.1E-03
C	11.5	4.1	8.7	1.0E-02
D	53.6	5.6	9.3	5.0E-02
E	17.6	4.1	12.0	2.1E-02
F	12.2	2.7	7.2	8.8E-03

Stability Class Information			S	
Class(a)	% Class(b)	Wind Speed(c) (m/sec)	%S(d) Wind	%'s(e) Comb.
A	0.4	2.1	6.0	2.4E-04
B	4.7	3.0	10.8	5.1E-03
C	11.5	4.2	14.4	1.7E-02
D	53.6	5.6	11.8	6.3E-02
E	17.6	4.0	17.6	3.1E-02
F	12.2	2.6	12.0	1.5E-02

Stability Class Information			SSW	
Class(a)	% Class(b)	Wind Speed(c) (m/sec)	%SSW(d) Wind	%'s(e) Comb.
A	0.4	2.4	6.0	2.4E-04
B	4.7	3.1	8.6	4.0E-03
C	11.5	4.1	9.7	1.1E-02
D	53.6	5.6	5.5	2.9E-02
E	17.6	3.9	7.4	1.3E-02
F	12.2	2.6	6.3	7.7E-03

TABLE 2 - CONT'D

Stability Class Information			SW	
Class[a]	% Class[b]	Wind Speed[c] (m/sec)	%SW[d] Wind	%'s[e] Comb.
A	0.4	1.8	5.2	2.1E-04
B	4.7	3.0	9.2	4.3E-03
C	11.5	4.1	7.5	8.6E-03
D	53.6	5.4	3.5	1.9E-02
E	17.6	3.9	4.3	7.6E-03
F	12.2	2.5	6.0	7.3E-03

Stability Class Information			WSW	
Class[a]	% Class[b]	Wind Speed[c] (m/sec)	%WSW[d] Wind	%'s[e] Comb.
A	0.4	2.2	6.0	2.4E-04
B	4.7	3.0	10.8	5.1E-03
C	11.5	4.3	9.0	1.0E-02
D	53.6	5.9	4.9	2.6E-02
E	17.6	3.9	5.7	1.0E-02
F	12.2	2.5	5.9	7.2E-03

Stability Class Information			W	
Class[a]	% Class[b]	Wind Speed[c] (m/sec)	%W[d] Wind	%'s[e] Comb.
A	0.4	1.8	3.4	1.4E-04
B	4.7	2.8	6.7	3.1E-03
C	11.5	3.9	6.2	7.1E-03
D	53.6	6.0	4.7	2.5E-02
E	17.6	3.7	5.3	9.3E-03
F	12.2	2.5	6.1	7.4E-03

TABLE 2 - CONT'D

Stability Class Information		WNW		
Class(a)	% Class(b)	Wind Speed(c) (m/sec)	%WNW(d) Wind	%'s(e) Comb.
A	0.4	2.1	4.3	1.7E-04
B	4.7	2.8	5.4	2.5E-03
C	11.5	4.3	5.1	5.9E-03
D	53.6	6.7	7.9	4.2E-02
E	17.6	4.0	5.5	9.7E-03
F	12.2	2.5	5.0	6.1E-03

Stability Class Information		NW		
Class(a)	% Class(b)	Wind Speed(c) (m/sec)	%NW(d) Wind	%'s(e) Comb.
A	0.4	2.2	4.3	1.7E-04
B	4.7	2.9	4.4	2.1E-03
C	11.5	4.3	4.7	5.4E-03
D	53.6	7.1	8.8	4.7E-02
E	17.6	4.2	5.1	9.0E-03
F	12.2	2.5	3.6	4.4E-03

Stability Class Information		NNW		
Class(a)	% Class(b)	Wind Speed(c) (m/sec)	%NNW(d) Wind	%'s(e) Comb.
A	0.4	2.3	1.7	6.8E-05
B	4.7	2.7	2.9	1.4E-03
C	11.5	4.1	3.0	3.5E-03
D	53.6	6.6	5.8	3.1E-02
E	17.6	4.0	3.6	6.3E-03
F	12.2	2.4	3.0	3.7E-03

TABLE 2 - CONT'D

Stability Class Information			N	
Class[a]	% Class[b]	Wind Speed[c] (m/sec)	%N[d] Wind	%'s[e] Comb.
A	0.4	2.4	7.8	3.1E-04
B	4.7	2.7	4.8	2.3E-03
C	11.5	4.0	4.8	5.5E-03
D	53.6	6.0	6.2	3.3E-02
E	17.6	3.8	4.0	7.0E-02
F	12.2	2.5	5.8	7.1E-03

[a] Stability class as defined by Pasquill's Categories.[4]

[b] Annual frequency distribution of stability class for all directions, or the total probability of occurrence for that class.

[c] Average wind speed for stability class and wind direction.

[d] Annual frequency distribution of wind direction for the specific stability class, or the probability of the wind direction given that the stability class exists.

[e] %'s Comb. = (% class/100) x (% NNE/100), or the joint probability of the specific stability class and the specific direction occurring at the same time.
Example: A conditional probability is one in which the probability of the events depends upon whether the other event has occurred.[5]
P (A) = probability of Class A conditions = 0.4%
P (N/A) = probability of wind direction from N given Class A conditions = 7.8%.
P (AN) = probability of having Class A conditions and wind direction from N.
P (AN) = P (A) P (N/A) = 3.1×10^{-4}.

Listed in Table 3 are the equations used to calculate the Ar-41 concentration and dose, along with the associated assumptions used for each case, at a distance, x, downwind from the stack release point. Calculations are based on the Pasquill-Gifford Method of determining stack release concentrations (effective stack height). Data for σ_y and σ_z were obtained from Ref. 4 and the DCF from Ref. 6.

(1) Effective Stack Height[4] (H):

$$H = h + d\left(\frac{\upsilon}{\mu}\right)^{1.4}\left(1 + \frac{\Delta T}{T}\right) \qquad \text{(Eq. 1)}$$

where, h = actual height (m)
= difference in elevation from release point to downwind site of dose calculation
d = diameter of release point (m)
μ = average wind speed for specific stability class (m/sec)
υ = exit velocity (m/sec)
ΔT = temperature difference between stack air and surrounding air
= assumed to be 0
T = absolute temperature of stack air

Therefore,

$$H = h + d\left(\frac{\upsilon}{\mu}\right)^{1.4} \qquad \text{(Eq. 1a)}$$

(2) Concentration Calculation:

$$\frac{\chi}{Q} = \frac{1}{\pi\sigma_y\sigma_z\mu}\exp\left[-\frac{1}{2}\left(\frac{y^2}{\sigma_y^2} + \frac{H^2}{\sigma_z^2}\right)\right] \qquad \text{(Eq. 2)}$$

where, χ = concentration at downwind site of dose calculation (µCi/ml or Ci/m^3)
Q = release rate (Ci/sec)
σ_y = lateral dispersion coefficient at downwind site of dose calculation (m)
σ_z = vertical dispersion coefficient at downwind site of dose calculation for specific stability class (m)
μ = average wind speed for specific stability class (m/sec)

y = distance from plume centerline (m) for maximum concentration, assume to be 0
H = effective stack height (m)

For maximum concentration:

$$\frac{\chi}{Q} = \frac{1}{\pi\sigma_y\sigma_z\mu} \exp\left[-\frac{1}{2}\left(\frac{H}{\sigma_z}\right)^2\right] \qquad \text{(Eq. 2a)}$$

Further, for case of ground release (H=0),

$$\frac{\chi}{Q} = \frac{1}{\pi\sigma_y\sigma_z\mu} \qquad \text{(Eq. 2b)}$$

Considering decay, the equation becomes

$$\frac{\chi}{Q} = \frac{e^{-\lambda t}}{\pi\sigma_y\sigma_z\mu} \qquad \text{(Eq. 2c)}$$

where, λ = decay constant for Ar-41 (sec^{-1})
t = time (sec)
= x/μ

(3) Annual Dose Calculation (D):

$$D = DCF \sum_i \chi_i (\%\ comb)_i \qquad \text{(Eq. 3)}$$

where, DCF = dose conversion factor
= $8.84 \times 10^{-3} \dfrac{\text{mrem m}^3}{\rho\text{Ci}-\text{y}}$ for Ar-41(6)
i = summation over all stability classes
$(\%\ comb)_i$ = relative frequency for stability class, i, and specific wind direction

Maximum Individual Dose

To determine the maximum individual dose, the south wind direction was chosen as being the most probable and annual doses determined at maximum release rate for two different distances: 150 m north to the exclusion boundary,[7] and 760 m north to the nearest residence. Elevations for these two sites were estimated from a University of Missouri topographical map (shown in Fig. 1). Data and the maximum calculated dose estimates for these sites are given in Table 4, with an example calculation given in Table 5. The maximum average annual dose at 150 m was calculated at ~ 2 mrem/y and ~ 18 mrem/y at 760 m. the difference in relative plume height at these sites is what leads to this difference in dose rates.

TABLE 4. MAXIMUM AVERAGE ANNUAL INDIVIDUAL DOSE

Location at 150 m directly North
Elevation at man height: 636 ft.

Class	Eff height (m)	σ_y (m)	σ_z (m)	χ/Q (s/m^3)	χ (μCi/ml) (Ci/m^3)	Dose w/%'s (mrem/y)
A	42	35	23	3.6E-05	8.6E-09	0.0
B	31	25	15	3.3E-05	7.9E-09	0.4
C	25	19	11	2.5E-05	6.0E-09	0.9
D	22	12	7	4.5E-06	1.1E-09	0.6
E	26	9	5	1.9E-09	4.6E-13	0.0
F	35	6.6	3.2	2.7E-28	6.5E-32	0.0
					TOTAL	1.9 mrem/y

Location at 760 m directly North
Elevation at man height: 700 ft.

Class	Eff height (m)	σ_y (m)	σ_z (m)	χ/Q (s/m^3)	χ (μCi/ml) (Ci/m^3)	Dose w/%'s (mrem/y)
A	23	160	300	3.2E-06	7.8E-10	0.0
B	12	110	90	1.1E-05	2.5E-09	0.1
C	6	81	50	1.9E-05	4.5E-09	0.7
D	3	54	25	4.2E-05	1.0E-08	5.7
E	7	41	18	1.0E-04	2.5E-08	6.7
F	15	30	11	1.4E-04	3.5E-08	4.5
					TOTAL	17.7 mrem/y

Distance: 760 m North
Elevation: 700 ft.

Class E: $\mu = 4.0$ m/sec
$\sigma_y = 41$ m
$\sigma_z = 18$ m

Effective Stack Height:

$$H = 687 + \left(\frac{40}{12}\right)\left(\frac{21.2}{4.0}\right)^{1.4} - 700$$

$$= 21 \text{ ft} \cdot \frac{1\text{m}}{3.2808 \text{ ft}}$$

$$= 6.5 \text{ m}$$

Ar-41 Concentration:

$$\frac{\chi}{Q} = \frac{1}{\pi\,(41)\,(18)\,(4)} \exp\left[-\frac{1}{2}\left(\frac{6.5}{18}\right)^2\right]$$

$$= (1.08 \times 10^{-4})\,(0.94)$$

$$= 1.0 \times 10^{-4} \,\frac{\text{sec}}{\text{m}^3}$$

$$\chi = \left(1.0 \times 10^{-4} \,\frac{\text{sec}}{\text{m}^3}\right)\left(2.4 \times 10^{-4} \text{ Ci/sec}\right)$$

$$= 2.4 \times 10^{-8} \,\frac{\text{Ci}}{\text{m}^3}$$

$$= 2.4 \times 10^{-8} \,\frac{\mu\text{Ci}}{\text{ml}}$$

Class E occurs 17.6% of the time and of that time the wind blows from the South 17.6% of time.

$$\%\text{'s Comb} = \left(\frac{17.6}{100}\right)\left(\frac{17.6}{100}\right) = 0.031$$

$$\text{Dose}_E = \left(2.4 \times 10^{-8} \,\frac{\text{Ci}}{\text{m}^3}\right)(0.031)\left(8.84 \times 10^{-3} \,\frac{\text{mrem - m}^3}{\text{pCi y}}\right)\left(10^{12} \,\frac{\text{pCi}}{\text{Ci}}\right)$$

$$= 6.7 \text{ mrem/y}$$

Maximum Population Dose Estimate

Population dose estimates were made assuming ground release conditions. Population density data was generated[8] using 1980 census data, 1985 update data, and growth projections provided by City of Columbia officials. Estimates for population doses were based on the projected 1990 population densities (see Table 6).

The maximum average annual dose was determined at the center of each population zone, except for the 16 zones at 0-1 miles. Because residences are no closer than 760 m, the midpoint was chosen at 0.75 miles (1200 m) from MURR. In addition, radioactive decay was considered in these calculations due to the significant amount of time required for the plume to move to these distances. Otherwise, calculations were made as were the individual dose estimate calculations. Data for σ_y and σ_z is given in Table 7, the summary of annual doses in Table 8, and the population dose estimate in Table 9. For the population out to 10 miles, the maximum annual population dose is estimated to be 145 person-rem.

TABLE 6. PROJECTED 1990 POPULATION DENSITIES
(NUMBER OF PEOPLE)

	Midpoint Distances (m)					
Wind Direction	1200	2400	4000	5600	7200	12000
NNE	238	437	368	315	262	206
NE	101	845	469	105	210	204
ENE	132	534	449	440	76	305
E	94	1189	1270	3905	220	315
ESE	186	1138	2025	849	51	3850
SE	406	2096	1664	1021	474	402
SSE	354	2747	1676	542	428	920
S	364	2649	2293	644	157	5750
SSW	1131	3163	1843	1404	1513	6135
SW	2699	5137	2491	2387	1571	2877
WSW	1997	4803	1067	1146	1055	5610
W	49	1446	525	385	153	234
WNW	52	592	1182	325	364	316
NW	36	126	644	222	103	315
NNW	288	665	229	30	154	210
N	339	851	974	432	210	255

TABLE 7

σ_y's (TOP) AND σ_z's (BOTTOM) FOR
POPULATION DISTANCES AND STABILITY

Stability Class	Midpoint Distances (m)					
	1200	2400	4000	5600	7200	12000
A	220	400	620	900	1050	1800
	800	5000	9700	14000	19000	33000
B	170	310	480	690	820	1300
	150	470	1100	2200	3300	6600
C	130	220	340	480	600	900
	75	130	200	270	320	500
D	80	140	220	300	400	610
	34	53	72	91	100	140
E	60	110	170	220	300	460
	23	40	50	60	70	84
F	42	80	120	160	200	300
	14	22	30	33	40	48

TABLE 8

A SUMMARY OF DOSE RATE ESTIMATES (MREM/Y) BASED ON WIND DIRECTION & DISTANCE

Wind Direction	Midpoint Distances (m) 1200	2400	4000	5600	7200	12000
NNE	4.5	1.4	0.7	0.4	0.3	0.1
NE	4.3	1.4	0.6	0.4	0.2	0.1
ENE	6.1	2.0	0.9	0.6	0.3	0.2
E	6.7	2.1	1.0	0.6	0.4	0.2
ESE	5.2	1.7	0.8	0.5	0.3	0.1
SE	5.9	1.9	0.9	0.5	0.3	0.2
SSE	8.4	2.7	1.3	0.8	0.5	0.2
S	13.0	4.2	1.9	1.2	0.7	0.3
SSW	6.3	2.0	0.9	0.6	0.4	0.2
SW	5.1	1.7	0.8	0.5	0.3	0.1
WSW	5.7	1.8	0.8	0.5	0.3	0.1
W	5.6	1.8	0.8	0.5	0.3	0.1
WNW	5.5	1.8	0.8	0.5	0.3	0.1
NW	4.7	1.5	0.7	0.4	0.3	0.1
NNW	3.7	1.2	0.6	0.3	0.2	0.1
N	5.5	1.8	0.8	0.5	0.3	0.1

TABLE 9

PERSON-REM ESTIMATES (PERSON-REM/Y)

Wind Direction	Midpoint Distances (m) 1200	2400	4000	5600	7200	12000
NNE	1.1	0.6	0.2	0.1	0.1	0.0
NE	0.4	1.2	0.3	0.0	0.1	0.0
ENE	0.8	1.0	0.4	0.2	0.0	0.0
E	0.6	2.5	1.3	2.4	0.1	0.1
ESE	1.0	1.9	1.6	0.4	0.0	0.5
SE	2.4	4.0	1.5	0.6	0.2	0.1
SSE	3.0	7.5	2.1	0.4	0.2	0.2
S	4.7	11.1	4.5	0.8	0.1	1.9
SSW	7.2	6.5	1.7	0.8	0.5	1.0
SW	13.9	8.5	1.9	1.1	0.5	0.4
WSW	11.3	8.7	0.9	0.6	0.3	0.8
W	0.3	2.6	0.4	0.2	0.0	0.0
WNW	0.3	1.1	1.0	0.2	0.1	0.0
NW	0.2	0.2	0.5	0.1	0.0	0.0
NNW	1.1	0.8	0.1	0.0	0.0	0.0
N	1.8	1.5	0.8	0.2	0.1	0.0
Subtotals	50.0	59.8	19.2	8.2	2.4	5.3
					TOTAL	144.8

Consideration of Normal Operational Releases

For the past five years, MURR has released ~ 1000 Ci/y of Ar-41 with a stack flowrate of ~ 16,500 ft³/min. Production of Ar-41 is expected to remain the same, and so the average Ar-41 concentration is anticipated to be:

$$3.7\text{E-}6\ (\mu\text{Ci/ml}) \cdot 16500/36500 = 2\text{E-}6\ \mu\text{Ci/ml}$$

which is ~ 13% of the Technical Specifications Limit. Because the dose estimates calculated thus far are proportional to the total amount of Ar-41 released, the dose estimates for actual operating conditions are easily calculated using the ratios of the stack release flow rates (given Ar-41 production remains constant). The actual operational dose estimates are:

Individual @ 150 m = 0.2 mrem/y
Individual @ 760 m = 2 mrem/yr
Population to 10 miles = 15 person-rem

Comparison of Risk

In the Safety Evaluation made by the NRC in support of Amendment No. 12,[2] an individual located at the nearest resident was estimated to receive an annual average total body dose of 13 mrem per year based on the 1977/78 release of 1925 Ci/y and 29 mrem/y for the maximum estimate. In the same NRC evaluation, the population dose for implementing Amendment No. 12 was estimated to be 20 person-rem. Although assumptions, data, and conditions for calculation are not fully described in the NRC Amendment No. 12, estimated doses are greater than those predicted by the current assessment, which utilizes a more realistic model (effective stack height and stability class weighting) and better site-specific data (meteoro-logical data and updated population densities). The NRC concluded "that there would be no significant environmental impact attributable" to an increase in stack release limit to 350 MPC. With lower doses estimated for the current change in stack height and flowrate, it is also concluded that no significant environmental impact exists. The same conclusion applies to instantaneous release limits.

Another method of assessing risk from the estimated doses is to compare them to natural background dose rates. The average whole body dose to an individual in the U.S. is 360 mrem/yr.[9] The estimated doses in terms of percent of natural background are:

	Maximum Case	Normal Operation
Individual @ 150 m	0.5%	0.1%
Individual @ 750 m	5.0%	0.6%
Population	0.4%	< 0.1 %

Variations of this magnitude can be found in annual dose for populations living in different areas of the U.S. with no observable effects.

Conclusion

The estimated dose rates calculated using improved methods and data were no greater than those calculated from previous appraisals where impact was judged by the NRC to be not significant in environmental impact. Therefore, there is no significant reduction in safety as the result of the changes in the MURR stack release conditions.

References for Section 5.3.3

(1) Appendix A: Technical Specifications for University of Missouri Research Reactor Facility--Facility Operating License No. R-103.

(2) NRC Amendment No. 12 for R-103, July 5, 1979.

(3) Callaway Environmental Report, Operational License Stage, Vol. 1, Tables 2.3-19 and 2.3-20.

(4) Cember, Herman, Introduction to Health Physics, Second Edition, Pergamon Press, 1983, pp. 340-352.

(5) DeGroot, Morris H., Probability and Statistics, Addison-Wesley Publishing Company, Inc., 1975, pp. 49-50.

(6) Regulatory Guide 1.109: "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluation Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977.

(7) NRC Amendment No. 8 for R-103, February 24, 1978.
(8) Environmental Report for Upgrade of MURR, 1987 (Draft).
(9) NCRP Report No. 94: "Exposure of the Population in the United States and Canada from Natural Background Radiation," December 1987.

5.4 Environmental Monitoring

5.4.1 Preoperational Phase[32]

A series of environmental samples were collected during the period of time from September 1965 to July 1966 in order to measure and establish baseline values for the level of ambient radioactivity in the vicinity of the Research Reactor Facility. The locations of the sampling stations are illustrated in Figure 5.5. The radioactive content of the environment was determined by analysis of alpha, beta and gamma-ray emissions from samples of grass, soil, water and air. In addition, the exposure dose rate due to the gamma radiation background was measured.

The pertinent data summarizing the results of the preoperational environmental monitoring are given in Tables 5.14 through 5.16 and Figures 5.6 through 5.8.[32]

5.4.2 Operational Phase (October 1966 to July 1971)[33]

The environmental monitoring program has continued for the period of time from initial operation in October 1966 to the present in order to determine if operation of the Research Reactor Facility is contributing to any increase in environmental radioactivity.

The reactor was operated at 5 MW from June 1967 to May 1974. This power level was maintained for approximately 40 hours per week during the period of time from July 1967 to September 1969, and approximately 100 hours per week for the period from September1969 to May 1974.

Slight changes have been made in the locations of several sampling stations in order to accomodate changes in the terrain resulting from weathering and construction (Figure 5.9). In addition to the routine sampling, occassional nonroutine samples are collected and analyzed. For example, nonroutine samples for a special study were collected from the following locations:
1) Grindstone Creek, about one-half mile south of the intersection of highways WW and Business 63 South, east of Columbia; and

TABLE 5.14
SAMPLES COLLECTED DURING PREOPERATIONAL SURVEY

Grass	38
Soil	38
Water	5
Air	5

TABLE 5.15
ACTIVITIES IN WATER

Sample	Alpha (pCi/l)	Beta (pCi/l)
1	0.47	5.7
2	0.24	5.0
3	0.34	3.6
4	0.11	1.0
5	0.21	2.8

TABLE 5.16
ACTIVITY IN AIR

Sample	Beta (pCi/m^3)
1	7×10^{-1}
2	11×10^{-1}
3	14×10^{-1}
4	11×10^{-1}
5	10×10^{-1}

2) Flat Branch Creek, the outfall of the creek, north of the intersection of Providence and Stewart Roads, in Columbia.

The results from environmental sample analyses do not differ significantly from the preoperational data. Averages and range of values per sample type are comparable. Gamma-ray spectra of water and air samples do not contain any radioactivity that might have been released in the effluents discharged from the Research Reactor Facility as the result of operation. Numbers of each sample type collected and analyzed are given in Table 5.17. The results are summarized in Table 5.18 and Figure 5.10. From the analyses of the data it appears that the operation of the Facility has not appreciably changed the radioactivity content in the environment. Activities determined in samples collected during the operational phase are similar to those of the preoperational survey. The gamma-ray spectra of samples collected during the operational phase surveys do not indicate the presence of such contaminants as Sb-124, Zn-65 and Mn-54, which have been discharged into the sanitary sewer system from the facility.

Direct radiation levels in the environment of the Reactor Facility vary between 0.01 and 0.02 mR/hr. These radiation levels are attributable to natural background.

TABLE 5.17
SAMPLES COLLECTED DURING PERIOD
OCTOBER 1966 - JULY 1971

Air		50
Station #3	18	
Other Stations	12	
Exclusion Area	20	
Water		32
Station #4	12	
Station #8	12	
Station #9	4	
Grindstone Creek	2	
Flat Branch Creek	2	
Grass		12
Soil		12
Sludge		8
Station #8	4	
Station #9	4	
Gravel		4
Grindstone Creek	2	
Flat Branch Creek	2	

TABLE 5.18

SUMMARY OF DATA

Sample, Station, Radiation	Operational Survey October 1966 - July 1971 Average	High	Low	Preoperation Survey September 1965 - July 1966 Average	High	Low	Units
Grass, all stations, α	2.6	4.00	1.10	2.00	3.20	1.30	pCi/gram
Grass, all stations, β	170.0	250.00	90.00	180.00	230.00	110.0	pCi/gram
Soil, all stations, α	5.1	22.00	3.00	6.20	20.00	3.60	pCi/gram
Soil, all stations, β	14.0	68.00	2.20	18.00	73.00	2.00	pCi/gram
Water, station #4, α	0.3	0.52	0.09	0.27	0.47	0.11	pCi/liter
Water, station #4, β	3.4	4.40	1.30	3.60	5.70	1.00	pCi/liter
Air, station #3, β	1.3	1.40	0.90	1.00	1.40	0.70	pCi/m^3
Air, other stations, β	1.2						pCi/m^3
Air, CAM, γ	2.0						pCi/m^3
Water, station #8, inlet, β	4.1						pCi/liter
Water, station #8, outlet, β	2.3						pCi/liter
Water, station #9, inlet, β	7.4						pCi/liter
Water, station #9, outlet, β	7.8						pCi/liter
Water, Grindstone, γ	2.5						pCi/liter
Water, Flat Branch, γ	8.0						pCi/liter
Sludge, station #8, γ	4.4						pCi/gram
Sludge, station #9, γ	4.2						pCi/gram
Gravel, Grindstone, γ	5.5×10^{-4}						pCi/liter of gravel
Gravel, Flat Branch, γ	4.3×10^{-4}						pCi/liter of gravel

5.5 Environmental Effects

5.5.1 Gaseous Releases

There have been no discernible effects on the environment due to gaseous effluents released during the period of operation of the reactor to date. Raising the power level from 5 MW to 10 MW may double the Ar-41 production rate but this is not expected to cause significant changes. The environmental monitoring program will continue in order to verify this conclusion.

5.5.2 Liquid Releases

The quantities of liquid effluents, all of which have been released to the sanitary sewer system, have produced no discernible effects on the environment.

The proposed increase in reactor power could result in an increase in activated products in the reactor coolant but this increase is not expected to be more than a factor of two. The impact on the environment is not expected to be significant but environmental monitoring will continue in order to identify changes if they occur.

5.5.3 Solid Wastes

The proposed increase in reactor power is not expected to result in a significant increase in solid wastes from the Reactor Facility. As noted above, all radioactive solid wastes from the facility, including the reactor and laboratories, are transferred to a licensed commercial firm for burial.

5.5.4 Fuel Shipments

Currently, the reactor is using fuel at a rate of approximately one core (8 elements) per nine months. New fuel is purchased in lots of three cores (24 elements) and received at the Reactor Facility at a rate of about one core per month over a three month period. Therefore, new fuel is purchased approximately every twenty-seven months.

The expended fuel is shipped for reprocessing in lots of two cores (16 elements) at intervals of approximately eighteen months.

After the proposed increase in reactor power, the rate of burnup will double; therefore, the average rate of receipt of new fuel and the average rate of shipment of expended fuel will double.

All fuel shipments are received from and sent to licensed firms via licensed commercial carriers. The increased usage and transportation are not expected to have a significant impact on the environment.

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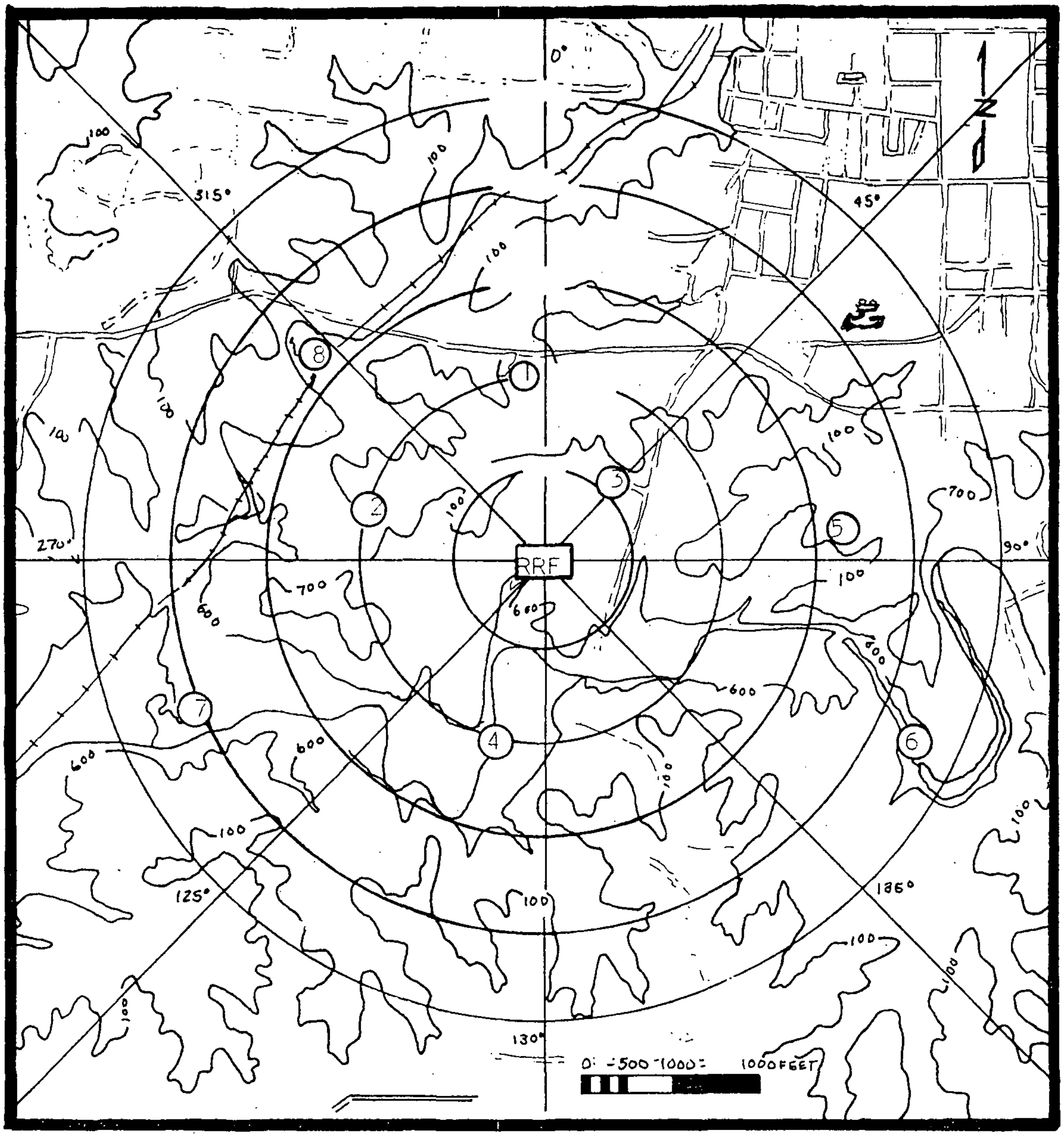


LOCATION OF PREOPERATIONAL SAMPLE STATIONS
RESEARCH REACTOR FACILITY
UNIVERSITY OF MISSOURI

FIGURE 5.5

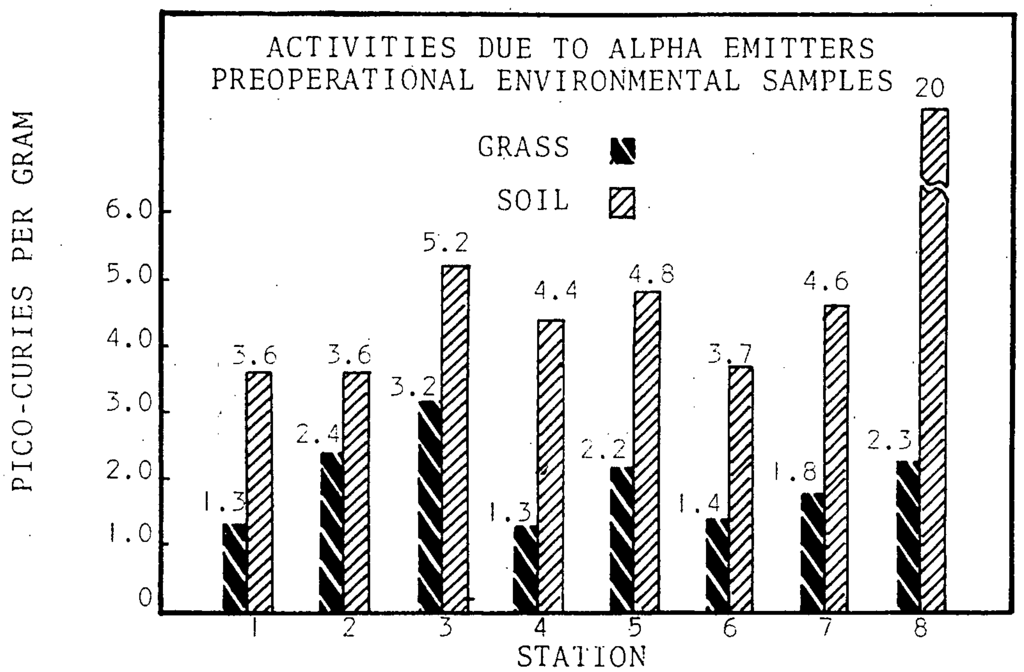


FIGURE 5.6

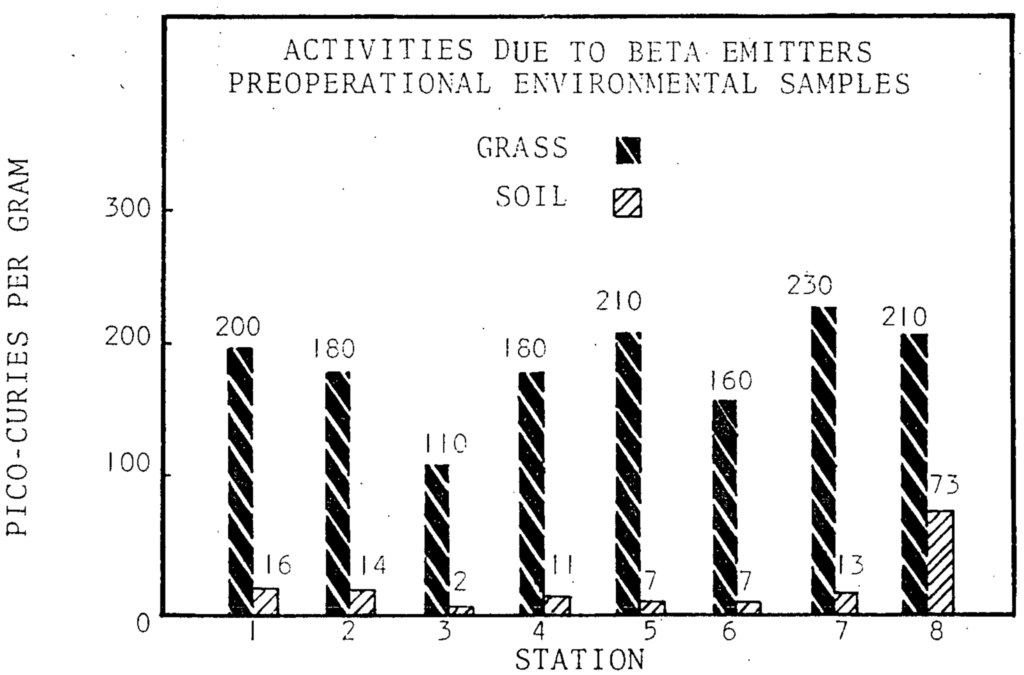


FIGURE 5.7

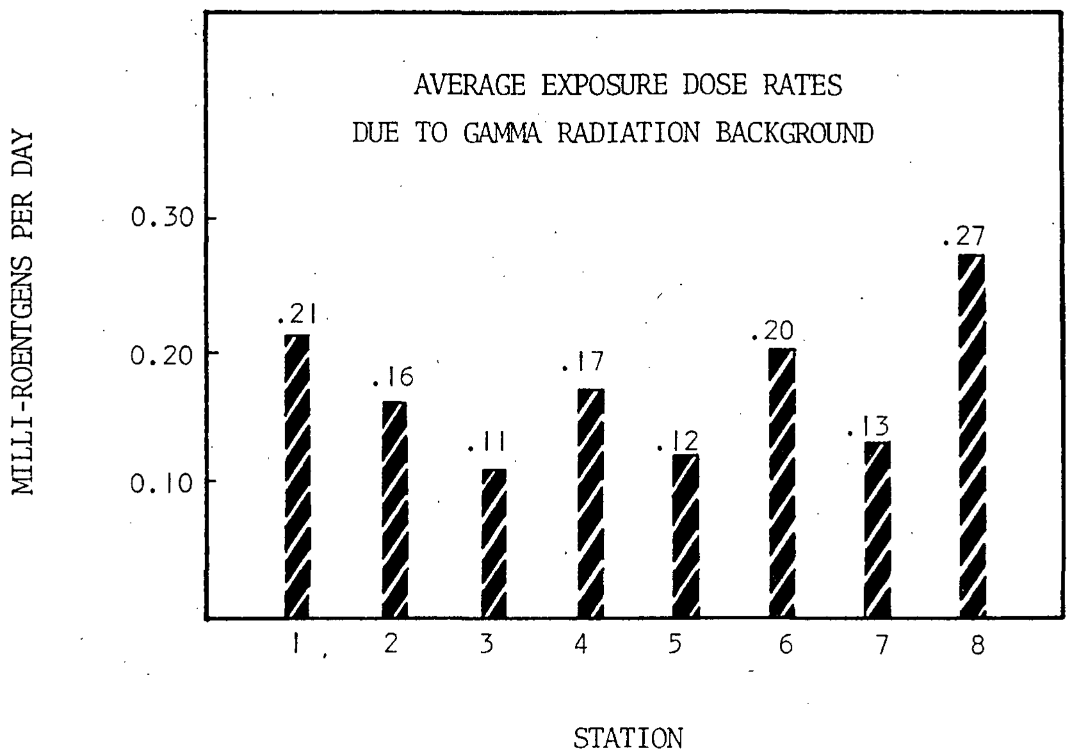
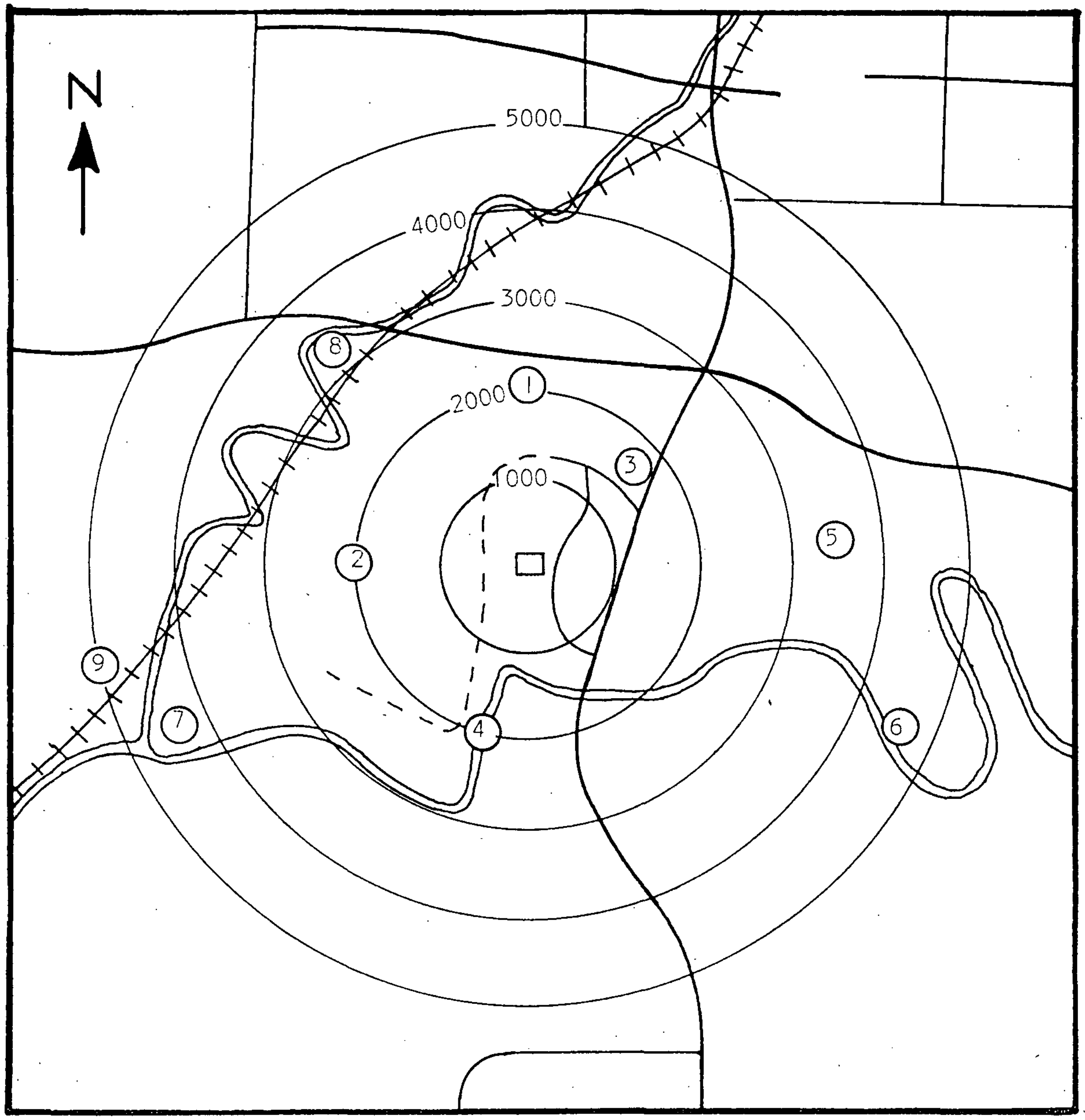
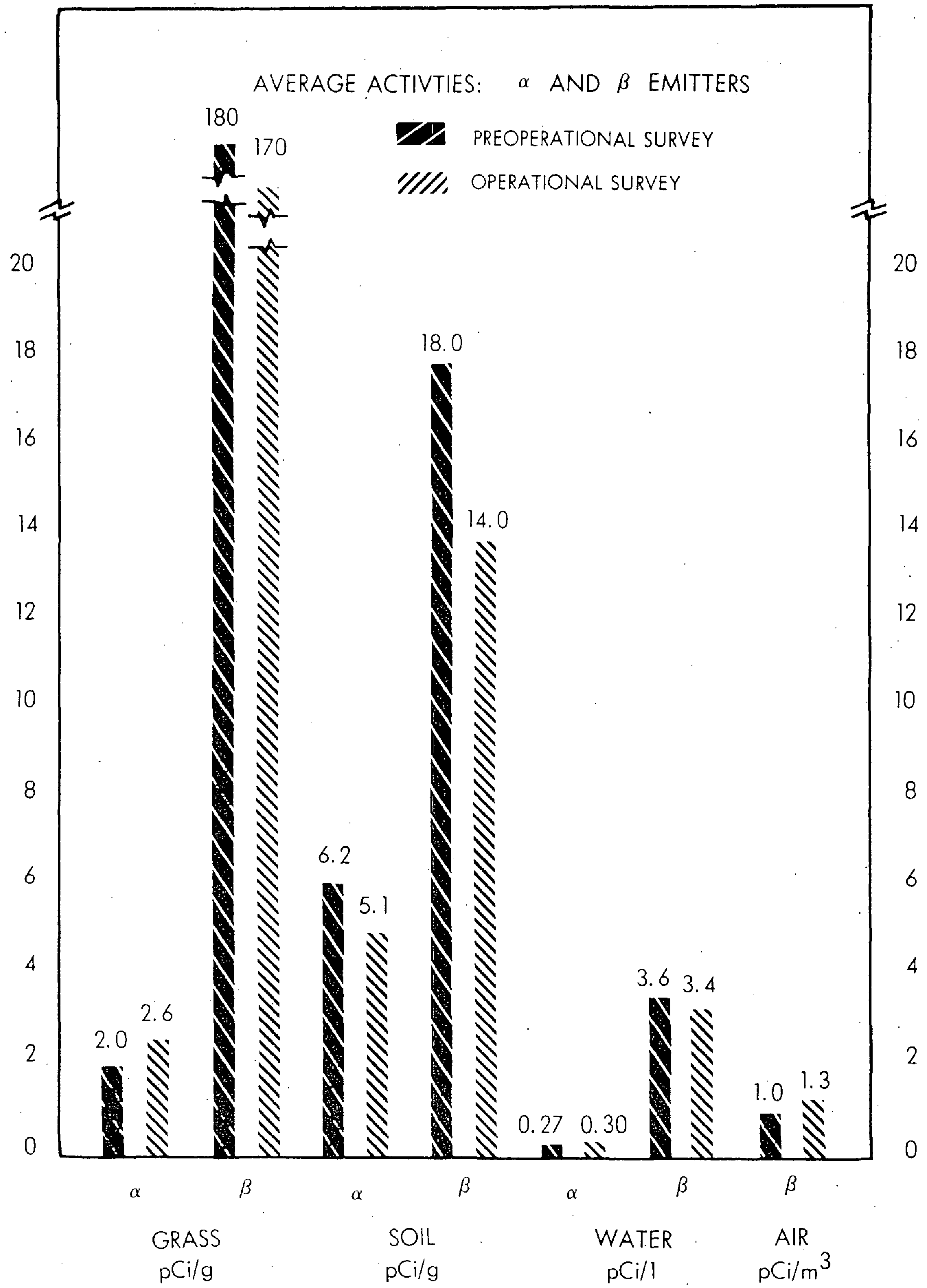

AVERAGE EXPOSURE DOSE RATES
DUE TO GAMMA RADIATION BACKGROUND
MILLI-ROENTGENS PER DAY
0.30
0.20
0.10
.21
.16
.11
.17
.12
.20
.13
.27
1
2
3
4
5
6
7
8
STATION

FIGURE 5.8



LOCATION OF SAMPLE STATIONS
RESEARCH REACTOR FACILITY
UNIVERSITY OF MISSOURI

FIGURE 5.9



AVERAGE ACTIVTIES: α AND β EMITTERS
PREOPERATIONAL SURVEY
OPERATIONAL SURVEY
180
170
18.0
14.0
6.2
5.1
2.0
2.6
0.27
0.30
3.6
3.4
1.0
1.3
0
2
4
6
8
10
12
14
16
18
20
α
β
α
β
α
β
β
GRASS pCi/g
SOIL pCi/g
WATER pCi/1
AIR pCi/m3

FIGURE 5.10

ADDENDUM NO. 4
University of Missouri
Research Reactor Facility

Compiled and Edited by
The Staff
Research Reactor Facility

Submitted by
The University of Missouri
Columbia, Missouri

UNIVERSITY OF MISSOURI
RESEARCH REACTOR FACILITY

Submittal in Reply to Questions
by the AEC Letter February 23, 1973

October 1973

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I. INTRODUCTION

A. Background

By letter dated August 25, 1972, the University of Missouri requested changes to License R-103 and acceptance of new Technical Specifications. The Commission by letter dated February 23, 1973, identified in six paragraphs the major areas of concern where additional information and analysis were needed for the further evaluation of the University of Missouri application.

This additional information is supplied in this report and in the accompanying document containing new Technical Specifications. For convenience, each area of concern is stated at the beginning of a separate section and each section provides a general response to each concern with supporting analysis, as appropriate, contained in appendices.

B. Commission letter dated February 23, 1973.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

FEB 23 1973

Docket No. 50-186

University of Missouri
ATTN: Dr. E. L. Cox
Director
Research Reactor Facility
Columbia, Missouri 65201

Gentlemen:

By letter dated August 25, 1972, you requested an amendment to License No. R-103 to authorize (1) an increase in reactor steady state power level to 10,000 kilowatts (thermal), (2) use of reactor startup source of 100 curies of antimony-beryllium, and (3) deletion of record keeping and reporting requirements from the license with these requirements incorporated into revised Technical Specifications.

You also proposed new Technical Specifications that would replace the present Technical Specifications in their entirety.

By letter dated November 21, 1972, you expressed a need for clarification of the status of the Commission's review of your application and discussed your plans for additional analysis.

As discussed with you, your staff, and your consultant, we have identified the following major areas of concern:

1. Your accident analysis (loss of primary flow without scram) shows a potential 304 Rem thyroid dose at 500 feet from the reactor building if this accident were to occur when operating at 10 MW. This potential exposure is unacceptable. In addition, our calculations that are based on your postulated accident, show that the potential exposures to personnel inside the reactor building (operators, experimenters, and visitors) would also be unacceptable. Safety Evaluations of recent research reactor licensing actions are enclosed for guidance in this respect.

You may propose design modifications that would effect a reduction of the probability of occurrence and consequences of releases of radioactivity to acceptable levels. As an example, it is necessary in this connection that the loss of flow protective instrumentation for the MURR meets all applicable criteria of IEEE 297, and the General Design Criteria 20 through 25 in 10 CFR Part 50. You should provide the appropriate analyses that demonstrate that you are meeting these requirements for the design of the loss of flow protective instrumentation and any other function needed for mitigating the consequences of accidents that could result in fission product releases.

2. Your analysis of systems performance should be extended to include an evaluation that demonstrates that an acceptable capability is available for cooling the core in the event of an assumed double ended rupture of the largest cooling pipe. In this connection, it is noted that your present analysis does not eliminate the single failure potential of V 546 in the core convective coolant system and does not show conclusively that its failure would not result in core damage. Your analysis of V 546 failure does not consider fuel element plate warping or cladding separation, and does not show how the circulation or mixing is accomplished in the full 650 lb of water above the core with V 546 closed. Your analysis should be extended to include the above considerations.

3. With respect to the conduct of experiments, specifically fueled experiments with radioactive iodine inventories up to 520 curies [technical specification 3.6(a)], we find them unacceptable unless justified by appropriate basis. The information that you have included in your change request dated March 10, 1972, does not demonstrate that the probability of failure of such experiments that could result in fission product release is acceptably low. You may propose design modifications for this experiment that would significantly reduce the likelihood of a potential fission product release from the fueled experiment.

4. Releases of radioactive materials resulting from reactor operations should be as low as practical. In this respect, you should provide necessary information to demonstrate that all practical measures have been or will be taken to minimize the discharges of Argon 41.

5. The Technical Specifications that you propose as Safety Limits do not meet regulatory requirements since they do not provide the necessary high degree of confidence that clad integrity will be maintained within the entire envelope of the true values of all interrelated process variables. In this respect, we have suggested that you follow the guidance provided in proposed ANS 15.1 and other approved technical specifications to assist you in developing acceptable Safety Limits.

6. You should provide a safety analysis for the irradiation of the antimony-beryllium source to 100 curies of activity.

Additional information and analysis necessary to resolve the above deficiencies are needed to permit us to proceed with our evaluation of your application.

Sincerely,

Robert J Schemel, for
Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosures:
DL Safety Evaluations for
MIT and Georgia Tech

II. DESIGN BASIS ACCIDENT

"Your accident analysis (loss of primary flow without scram) shows a potential 304 Rem thyroid dose at 500 feet from the reactor building if this accident were to occur when operating at 10 MW. This potential exposure is unacceptable. In addition, our calculations that are based on your postulated accident, show that the potential exposures to personnel inside the reactor building (operators, experimenters, and visitors) would also be unacceptable. Safety Evaluations of recent research reactor licensing actions are enclosed for guidance in this respect.

You may propose design modifications that would effect a reduction of the probability of occurrence and consequences of releases of radioactivity to acceptable levels. As an example, it is necessary in this connection that the loss of flow protective instrumentation for the MURR meets all applicable criteria of IEEE 279, and the General Design Criteria 20 through 25 in 10CFR Part 50. You should provide the appropriate analyses that demonstrate that you are meeting these requirements for the design of the loss of flow protective instrumentation and any other function needed for mitigating the consequences of accidents that could result in fission product releases."(1)

The University of Missouri proposed two design modifications which affect the accident analysis for the research reactor. The instrumentation and safety systems have received thorough analysis by the reactor staff and by an outside consultant. As a result of this analysis the instrumentation and safety systems are to be modified as shown in Appendix A to conform with applicable criteria contained in the General Design Criteria 20 through 25 of 10CFR Part 50 and in IEEE Standard 279.

Since the requirements of the General Design Criteria 20 through 25 of 10CFR50 are included in the criteria of IEEE 279, only the conformance with each section of IEEE 279 is discussed in Appendix A. These changes will essentially eliminate the probability of an accident leading to the release of fission products.

In addition, the reactor primary coolant anti-siphon system will be modified so as to maintain the integrity of the primary system under all conditions. It was the anti-siphon system that provided a path for fission products released in the primary system to pass to reactor containment. By this modification the reactor coolant system now constitutes a primary barrier to the release of fission products to the containment. A description of the modified anti-siphon system and its operation is contained in Appendix B.

A thorough analysis of the loss of flow and loss of coolant accident conditions has been conducted, and the results are summarized in Appendix D and E. This analysis demonstrates that under the worst conditions the core will not be covered and that under all conditions in which water is present the fuel cladding integrity will be maintained.

In light of the foregoing, it is incumbent upon us to show that if there were a release of fission products, the health and safety of the public will not be jeopardized. The accident that is considered to constitute the most severe consequences to the health and safety of the public is the release of fission products from the accidental melting of four fuel plates postulated to occur as a result of blockage of flow. We have shown (Appendix C) that the probability of such a situation occurring is extremely remote. In such an event, the fission products must escape the first barrier of the reactor primary system and in addition must escape from the reactor containment building.

It is appropriate to indicate that the containment building isolation system has been modified to ensure further proper containment by eliminating the four-inch penetration from the pneumatic tube system, by addition of a redundant 16 inch isolating valve on the building exhaust system, and by modifications to make the instrumentation system conform with IEEE 279. A description of these modifications is contained in Appendix A.

The potential whole body and thyroid dose in the reactor containment building has been calculated based upon the design basis accident and subsequent fission product release. The result is a thyroid dose of 43.7 mrem and a whole body dose of < 1.0 mrem. The details and assumptions pertaining to this calculation are contained in Appendix C.

It is relevant to indicate at this point that the siting of the research reactor alleviates even this exposure. The location of the reactor in the University Research Park and the control of occupancy in, and access to, Research Park precludes extended exposure to any radiation hazard in this area.

Personnel in reactor containment are now protected against the release of fission products by the modification of the anti-siphon line making the primary system first containment for any fission products released. The exposure experienced would be that resulting from contained fission products in the anti-siphon air storage tank and the piping associated with this system.

For purposes of shielding the air storage tank is to be located beneath the pool surface. Any radioactivity in this tank would be seen at a very low level by the reactor pool radiation monitor and cause a reactor scram and building isolation. The radiation hazard to personnel is considered minimal.

There are three independent sensors which detect and provide warning of the presence of fission product activity in the primary system. The primary coolant fission product monitor is the primary instrument channel. Indications of abnormal radiation levels are also detected on the radiation monitors in equipment room 114 and at the pool surface.

In the unlikely event that radioactivity does escape to the containment building, two radiation detectors in the containment ventilation lines (fifth level) will automatically initiate a reactor scram and building isolation.

III. LOSS OF COOLANT ANALYSIS

"Your analysis of systems performance should be extended to include an evaluation that demonstrates that an acceptable capability is available for cooling the core in the event of an assumed double ended rupture of the largest cooling pipe. In this connection, it is noted that your present analysis does not eliminate the single failure potential of V546 in the core convective coolant system and does not show conclusively that its failure would not result in core damage. Your analysis of V546 failures does not consider fuel element plate warping or cladding separation, and does not show how the circulation or mixing is accomplished in the full 650 lb of water above the core with V546 closed. Your analysis should be extended to include the above considerations." (1)

The loss of coolant analysis postulated as resulting from a double-ended rupture of the largest cooling pipe at the worst location is contained in Appendix E. The analysis demonstrates that the core will not be damaged in the event this accident should occur.

As an alternative to providing the detailed analysis of the two phase fluid flow condition existing within and above the core following a failure of valve V546 the University will install a redundant valve in parallel with V546. The additional instrumentation for control and indication is shown in Appendix A.

IV. FUELED EXPERIMENTS

"With respect to the conduct of experiments, specifically fueled experiments with radioactive iodine inventories up to 520 curies [technical specification 3.6(a)], we find them unacceptable unless justified by appropriate basis. The information that you have included in your change request dated March 10, 1972, does not demonstrate that the probability of failure of such experiments that could result in fission product release is acceptably low. You may propose design modifications for this experiment that would significantly reduce the likelihood of a potential fission product release from the fueled experiment."(1)

The fueled experiments proposed by the University change request of March 10, 1972, and as included in the proposed Technical Specification 3.6(a) are being delayed until such time as the reactor staff is free from the more important task of upgrading the reactor to 10 MW. The University does not therefore request any change to the presently authorized fueled experiments. Technical Specification 3.6(a) has been revised.

V. ARGON-41 RELEASES

"Releases of radioactive materials resulting from reactor operations should be as low as practical. In this respect, you should provide necessary information to demonstrate that all practical measures have been or will be taken to minimize the discharges of Argon 41."(1)

The University of Missouri has adopted as a policy the philosophy that the releases of radioactive materials from MURR operation shall be as low as practical. Emphasis was first placed on reducing the liquid releases to the sanitary sewer. By extensive modifications to the demineralizer systems and the liquid waste storage and cleanup systems, by the addition of a drain collection system for reusing radioactive reactor grade water, and through changes in operating procedures and administrative controls, the radioactive liquid releases to sewage have decreased from about 1.0 Curie per year to an average, for the past two years, of 0.056 Curies per year. About 90% of the radioactivity released to the sewer has been tritium.

Efforts over the past six months to reduce the ^{41}Ar release in the facility exhaust have resulted in more than a factor of two reduction. This has been accomplished by reducing the air volume being irradiated on the sides of the graphite stack of the thermal column facility, adding a neutron absorbing material on the top of the graphite stack to prevent neutrons from reaching the air space above the thermal column and by sealing the collimator of the thermal neutron radiographic facility to prevent release of gaseous activity. Therefore, the rate at which ^{41}Ar will be discharged during 10 MW operation will be less than what has been previously released for 5 MW operation (see paragraph 5.3.1.1 of Addendum 3 to HSR).

Greater than 98% of the remaining ^{41}Ar production occurs within the three pneumatic tube terminals located in the graphite reflector region and the one terminal located in the bulk pool.

The high gamma ray and neutron heating rates in the pneumatic tube terminals require that cooling air be continuously circulated past sample containers to prevent damage to many sample materials. Since the system air flow rate is about 170 cu ft/min, it would be impractical to use bottled gas not containing argon or to hold up the exhausting air sufficiently long to allow for appreciable radioactive decay.

The closed loop nitrogen or carbon dioxide circulating type systems have the disadvantage that a small amount of air contamination presents a risk to personnel within the laboratory of exposure to high concentrations of ^{41}Ar. The MURR exhausting type system maintains a vacuum relative to the laboratory and therefore any leakage in the system does not present a risk to the experimenters.

Several alternatives exist to reduce the ^{41}Ar discharges. These alternatives include a reduction in the number of pneumatic tubes, the replacement of the existing tubes with smaller ones, and relocation of the tubes to areas of lower neutron flux. All of these alternatives, however, reduce the experimental capability of the reactor and will be avoided if possible.

It is felt that all known practical measures to reduce ^{41}Ar releases have been taken. The University will continue to explore new techniques and will implement all practical measures to reduce the ^{41}Ar discharges further.

VI. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

"The Technical Specifications that you propose as Safety Limits do not meet regulatory requirements since they do not provide the necessary high degree of confidence that clad integrity will be maintained with the entire envelope of the true values of all interrelated process variables. In this respect, we have suggested that you follow the guidance provided in proposed ANS 15.1 and other approved technical specifications to assist you in developing acceptable Safety Limits."(1)

A summary of the analysis used and the results obtained in developing new MURR Safety Limits are presented in Appendix F. The bases for selecting Limiting Safety System Settings is presented in Appendix H. After reviewing the proposed Standard 15.1 and several recently approved technical specifications and after consultation with members of the Directorate of Licensing staff, the MURR technical specifications were revised to the format presented in the document, "MURR Technical Specifications," which accompanies this submittal.

VII. NEUTRON SOURCE SAFETY ANALYSIS

"You should provide a safety analysis for the irradiation of the antimony-beryllium source to 100 curies of activity."

The safety analysis for the irradiation of the antimony-beryllium source to 100 Curies is presented in Appendix G.

APPENDIX A

INSTRUMENTATION

A. INSTRUMENTATION

A.1 Introduction

As part of the program to increase power (upgrade), and further reduce the probability of accidental release of radioactivity, a major effort lay in the design modifications to the reactor protective system. This system has been redesigned such that the system meets all applicable criteria of IEEE Standard 279: Criterion for Protection Systems for Nuclear Power Generating Stations. As the title of IEEE Standard 279-1971 implies, it is intended for application, especially in the design phase, to nuclear power generating station protection systems. Although the MURR is not a nuclear power generating station, much of the criteria of IEEE 279 can be applied to evaluate the adequacy of the MURR protection system with respect to present day standards for functional performance and reliability.

NUS Corporation of Rockville, Maryland, was engaged to provide the expertise necessary to competently evaluate the MURR conformance with IEEE 279-1971. This evaluation included a detailed study of all applicable instrumentation drawings, circuit prints, Hazards Summary Reports, and Technical Specifications, and an inspection of the facility to determine the physical arrangement of equipment and cables.

This appendix describes the proposed facility modifications resulting from the evaluation and comments on the instrumentation conformance with each section of IEEE 279.

A.2 Protective Systems

The protective system includes all sensing devices, circuits, signal conditioning equipment, electronic equipment, and electromechanical devices that serve to effect a reactor shutdown by removal of the holding current from the four control rod magnets or to activate engineered safeguards. The function of all reactor scram circuits is to reduce the four shim blade magnet currents to zero in the event of abnormal functioning of any essential reactor system. Secondary actions which may be initiated by reactor scram circuits are not considered part of the protective function. Although the control rod mechanisms are not within the scope of IEEE 279, they are an integral part of the protective system and therefore have been considered in the analysis of protective system reliability. The containment isolation system and the primary coolant siphon break system are considered engineered safety features and hence fall within the scope of IEEE 279.

A.3 Conformance with IEEE 279-1971

A.3.1 Design Basis (1)

"A specific protection system design basis shall be provided for each nuclear power generating station. The information thus provided shall be available, as needed, for making judgements on system functional adequacy.

The design basis shall document as a minimum, the following:

(1) the generating station conditions which require protective action;

(2) the generating station variables (for example, neutron flux, coolant flow, pressure, etc.) that are required to be monitored in order to provide protective actions;

(3) the minimum number and location of the sensors required to monitor adequately, for protective function purposes, those variables listed in Section 3(2) that have a spatial dependence;

(4) prudent operational limits for each variable listed in Section 3(2) in each applicable reactor operation mode;

(5) the margin, with appropriate interpretive information, between each operational limit and the level considered to mark the onset of unsafe conditions;

(6) the levels that, when reached, will require protective action;

(7) the range of transient and steady-state conditions of both the energy supply and the environment (for example, voltage, frequency, temperature, humidity, pressure vibration, etc.) during normal, abnormal, and accident circumstances throughout which the system must perform;

(8) the malfunctions, accidents, or other unusual events (for example, fire, explosion, missiles, lightning, flood, earthquake, wind, etc.) which could physically damage protection system components or could cause environmental changes leading to functional degradation of system performance, and for which provisions must be incorporated to retain necessary protective action;

(9) minimum performance requirements include the following:

(a) system response times;

(b) system accuracies;

(c) ranges (normal, abnormal, and accident conditions) of the magnitudes and rates of change of sensed variables to be accommodated until proper conclusion of the protective action is assured."

(1) References are contained at the end of each appendix.

(1) The reactor conditions that lead to protective action have already been defined (see Hazards Summary Report - July 1965).
(2) The reactor variables that are monitored in order to initiate protective action are also defined by the existing design. There are no plant variables to be added to the protection system.
(3) None of the monitored protection system variables have a spatial dependence of significance to the protection system.
(4), (5), and (6) The prudent operational limits on variables monitored by the safety system, the margin between those limits and levels that mark the onset of unsafe conditions (i.e., safety limits), and the levels at which protection action should be initiated have effectively been set by the overall system design and by established scram setpoints as discussed in Appendix H.
(7), (8), and (9) Design basis suggestions do not require further documentation. Although not explicitly documented, the ideas represented were adequately considered during MURR design and construction.

A.3.2 General Functional Requirement

"The nuclear power generating station protection system shall, with precision and reliability, automatically initiate appropriate protective action whenever a condition monitored by the system reaches a preset level. . . ." (1)

Since the MURR has been in service for more than six years, there seems little doubt that its protection system is functionally adequate. Expected extremes of power supply voltage and operating environment temperatures have already been experienced. Very extreme conditions such as fire, flood and earthquake have not been experienced; however, at a facility such as MURR these occurrences are such that more than sufficient time exists for prompt operator action. All electronics with the exception of remote sensors, shim blade magnets, and connecting wiring are located in the control room where a reactor operator is continuously present during operation.

A.3.3 Single Failure Criterion

"Any single failure within the protection system shall not prevent proper protective action at the system level when required." (1)

A.3.3.1 Control Rod Magnets and Current Control Circuits

The control rod magnets and current control circuits satisfy the single failure criterion.

A.3.3.2 Trip Actuator Amplifier

A failure in one trip actuator amplifier could result in the failure to turn off at most two magnet supplies. The other two rods would scram and successfully shut down the reactor (Figure A.1).

A.3.3.3 Noncoincidence Logic Units

In order to reduce the vulnerability of the noncoincidence logic units to single failure, a 1/N-1/2 logic has been proposed; any of N inputs to either of two logic units would initiate a scram (Figure A.1). Inputs to logic units A and B will be isolated from each other. The inputs to each unit, which are designated as E1A and E1B, etc., for corresponding terminals will be:

Logic Unit Input Terminal	Channel
E1A	Channel 2 - Period
E2A	Channel 4 - Power
E3A	Channel 6 - Power
E4A	Process String A
E1B	Channel 3 - Period
E2B	Channel 5 - Power
E3B	Process String B

The manual scram initiated by contact 5-6 of switch 1S10 will be relocated to interrupt input to both E4A and E3B. For redundancy contact 1-2 of the manual scram switch 1S10 also interrupts power to the Trip Actuator Amplifiers, thereby eliminating rod magnet current (Figure A.1).

Presently each logic unit contains inputs which are not used and therefore have 24 vdc continuously connected. The unused input resistors and diodes will be removed as this is preferable to supplying 24 vdc continuously.

The process string inputs to the two logic units have been changed due to the need for duplicate sensors, new auxiliary relays and switches to satisfy the single failure criterion. The adequacy of protection for each process variable is discussed in the succeeding paragraphs.

A.3.3.4 <u>Nuclear Instrumentation</u>

The redundant period scram channels (NI Channels 2 and 3) have separate detectors and separate electronics chassis. The redundant power level scram channels (NI Channels 4, 5 and 6) have separate detectors and separate electronics chassis. (Refer to Section 2.5.2 of Addendum 3 to Hazards Summary Report). This arrangement satisfies the single failure criterion except that the relative physical location of cables and electronics leaves them vulnerable to an external event such as a fire. However, resulting damage will cause a reactor scram. Further, an operator is always in the control room during operation where he is in the immediate vicinity of the cables and electronics.

A.3.3.5 <u>Reactor Loop Flow Scram</u>

Adequate heat removal from the core requires that the coolant flow be maintained above some minimum value determined by thermal-hydraulic analysis as presented in Appendix F of this report.

~~Reactor flow is the sum of flows through two heat exchangers for 10 MW operation ... ssure drop across orifice pl... (Figures A.2 and A.3). Aux... l E3B to the noncoinci... erated by differential pressure switches 928A and B monitoring the pressure drop across each heat exchanger (Figures A.2 and A.3). A low differential pressure sensed by switches 928A and B actuates relays which open contacts in the E3B and E4A input to the noncoincidence logic units respectively.~~

"Reactor flow is the sum of flows through two heat exchangers for 10 MW operation. Flow scram signals are developed by measuring the pressure drop across orifice plates with multiple taps and generating a trip from any one of the following Alarm Units: 920A, 920C, 920E or 920G. Auxiliary or internal relays from these alarm units open inputs E4A and E3B to the noncoincidence logic units (Figure A.1)."

The differential pressure across the reactor core is monitored by differential pressure switch 929 (Figures A.2 and A.3). A low core differential pressure results in actuation of relays 1K13 and 1K26 of the power level interlock circuit (Figure A.4). Relay 1K13 interrupts input E3B and relay 1K26 interrupts input E4A of the noncoincidence logic units, thereby causing a reactor scram. Additionally, should either reactor coolant isolation valve, 507A/B, leave its fully open position, limit switches would de-energize relay 2K11 which would result in a power level interlock scram via relays 1K13 and 1K26 (Figures A.5 and A.4).

The single failure criteria is satisfied for reactor loop flow scram.

A.3.3.6 <u>Reactor Coolant Temperature Scrams</u>

Resistance temperature detectors 980A and 980B (Figure A.6) monitor the reactor coolant outlet temperature of each heat exchanger. Exceeding a preset level opens a contact in the E3B process string input to the noncoincidence logic unit. As a backup, temperature element 901B (Figure A.6) actuates a reactor scram when the core coolant outlet temperature exceeds a preset valve. In addition, temperature element 901A (Figure A.6) which monitors core coolant inlet temperature actuates an alarm upon high temperature, thereby alerting the operator to take corrective action to prevent a high temperature scram.

The temperature protective equipment meets the single failure criteria.

A.3.3.7 <u>Reactor Coolant Pressure Scram</u>

Four sensors can independently cause a reactor scram should the reactor system coolant pressure decrease below a preset level. Pressure switches 944A and 944B are located on the core outlet stream (Figure A.2). Pressure switch 944A actuates relay 2K13 (Figure A.5) and relay 2K13 then opens the process input string to E4A of the noncoincidence logic units. Pressure switch 944B actuates auxiliary relay 2K28 which interrupts the input E3B of the noncoincidence logic unit, thereby causing a scram.

Pressure transmitter 943 monitors the reactor coolant pressure near the pressurizer connection to the coolant system (Figure A.2). Pressure transmitter 943, on sensing the low pressure, actuates the scram via alarm unit 942 (Figure A.3) to interrupt input E3B of the logic units. As an additional backup, pressure switch 938 (Figure A.2) monitors the pressurizer pressure and actuates a reactor low pressure scram via auxiliary relay K25 (Figures A.7 and A.1).

The single failure criteria is satisfied with respect to low pressure protection.

A.3.3.8 Pool Coolant Flow Scram

Protection against low flow operation is attained by the use of flow element 921, with multiple taps (Figure A.2). The flow element measures the differential pressure generated across the flow orifice by the total flow in the outlet to the heat exchanger. The reactor scram signal is generated by alarm unit 920D or 920B (Figure A.2) and actuates the scram via auxiliary relays K37 and K31 (Figures A.1 and A.7). As a backup the reflector differential pressure is monitored by pressure transmitter 917 (Figure A.3). On either a high or low differential pressure, a scram is initiated by opening contacts in the E3B process leg of the noncoincidence logic units. In the event the pool coolant isolation valve 509 leaves its fully open seat, a reactor scram is initiated by limit switches on the valve which actuate auxiliary relay 2K12 (Figure A.5) resulting in an interruption of noncoincidence logic unit input E4A.

The single failure criteria is satisfied with respect to pool flow.

A.3.3.9 Pressurizer Low Level Scram

A pressurizer low level is monitored by level controller switch 935 (Figure A.2). If a sufficiently low level is detected, an alarm is received in the reactor control room and if the level should continue to decrease, a reactor scram occurs as a result of level controller 935 actuating auxiliary relay K28 (Figures A.7 and A.1) which interrupts logic unit input E3B.

A.3.3.10 Reactor Loop High Pressure Scram

The reactor coolant system is protected from overpressure by two relief valves. As an additional precaution, a pressure switch 939 is utilized to cause a reactor scram upon sensing a high pressure. Pressure switch 939 (Figure A.2) will actuate relay K26 (Figure A.7) which in turn interrupts logic unit input E3B.

In addition to the scram function the reactor operator is made aware of an abnormally high pressure condition by an alarm actuated by pressure switch 946 (Figure A.2).

A.3.3.11 Containment Building Isolation and Reactor Scram

The detectors for the building air plenum and pool surface high radiation scrams and containment isolation channels are powered by independent low voltage power supplies and are therefore not subject to single failure event. Although the power supplies receive AC power from a common source, loss of the AC power will automatically scram the reactor and isolate the containment building due to the reactor isolation circuitry design.

A backup plenum exhaust detector and trip unit is incorporated in a separate rack unit from the primary plenum exhaust detector to provide a redundant plenum scram and isolation. All automatically initiated reactor isolations will trip the backup doors and close the fifth level motorized doors.

De-energizing the relays 2K1A or 2K1B actuates the containment building isolation system (Figure A.10) and causes a reactor scram (Figure A.1). Thus, no single failure can render the proposed containment building isolation/scram system inoperable.

At present, all potentially radioactive gases from the pool, beamports, thermal column, etc., pass to the MURR exhaust system (Figure A.11) through a 16" diameter pipe exiting the containment building west wall just below ceiling level. In the event of a containment building isolation, this line is sealed by two 16" air operated butterfly valves. The original valve is air-to-open, air-to-close and thus requires a backup air supply in addition to the facility air compressor. The backup air is provided by two separate systems: an emergency air compressor on the fifth level of the containment building and a local compressor and accumulator tank located at the valve. Thus, three possible sources of air are available to operate the valve. To ensure compliance with 10 CFR 50, Appendix A, Criterion 54 regarding redundancy, and IEEE-279 single failure criteria, a second 16" valve has been installed in the exhaust line. Control and operation are identical to the first valve, except the second valve is air-to-open, spring-loaded to close. It is located over the fifth level of the containment building about 12 feet from the east wall. Air is supplied to both valves from a common source. Control is by two electrical solenoid valves, either of which can vent and allow the valve to close (Figure A.12). These solenoids are wired into the MURR isolation/evacuation system (Figure A.10). This provides additional redundancy for the isolation of the reactor containment building.

The pneumatic tube exhaust is routed through particulate filters directly to the MURR off-gas stack in the west tower. The old 4 inch line in the containment building has been sealed.

A.3.3.12 Anti-Siphon System

The anti-siphon valve system as described in Appendix B meets the single failure criterion.

A.3.3.13 Core Natural Circulation System

The in-pool heat exchanger is utilized to remove decay heat from the core by establishing a natural circulation flow loop following a loss of forced flow. Parallel isolation valves 546A and 546B presently prevent flow through the in-pool heat exchanger during forced convection cooling (Figure A.2). Control circuits for these valves are shown in Figure A.5. As described, the natural circulation cooling loop isolation valve system meets the single failure criteria.

A.3.3.14 Pool Natural Circulation System

Upon shutdown of the pool forced circulation cooling system, valve 547 allows natural convection flow of pool water into the lower plenum and up through the reflector elements, control blade gaps, and flux trap. To comply with the single failure criterion, this valve is left open. Flow experiments conducted with valve 547 allows natural convection flow of pool water into the lower plenum and up through the reflector elements, control blade gaps, and flux trap. To comply with the single failure criterion, this valve is left open.

A.3.4 Quality of Components and Modules

"Components and modules shall be of a quality that is consistent with minimum maintenance requirements and low failure rates. Quality levels shall be achieved through the specification of requirements known to promote high quality, such as requirements for design, for the derating of components, for manufacturing, quality control, inspection, calibration, and test." (1)

The accrued MURR operating experience has demonstrated minimum maintenance requirements and low failure rates. Components that have exhibited high failure rates have by now been replaced by higher quality components. New components and modules shall be of a quality that is consistent with minimum maintenance requirements and low failure rates.

A.3.5 Equipment Qualification

"Type test data or reasonable engineering extrapolation based on test data shall be available to verify that protection system equipment shall meet, on a continuing basis, the performance requirements determined to be necessary for achieving the system requirements." (1)

This criterion is more applicable to components that may be exposed to extreme environments such as hot steam and very high radiation; therefore, it is not really applicable to the MURR. Operating experience at MURR for more than seven years, however, has shown that this requirement is satisfied.

A.3.6 Channel Integrity

"All protection system channels shall be designed to maintain necessary functional capability under extremes of conditions (as applicable) relating to environment, energy supply, malfunctions, and accidents." (1)

Like the preceding one, this criterion is more applicable to power reactors and is not especially relevant to the MURR. The reactor will not be operated unless the environmental conditions surrounding protective system channels are normal.

A.3.7 Channel Independence

"Channels that provide signals for the same protective functions shall be independent and physically separated to accomplish decoupling of the effects of unsafe environmental factors, electric transients, and physical accident consequences documented in the design basis, and to reduce the likelihood of interactions between channels during maintenance operations or in the event of channel malfunction." (1)

The channels used in the proposed protective system are redundant. The circuitry for the nuclear instruments and protective equipment located in the control room including the signal cables are not physically separated, however, these channels do not present a problem because virtually all of the parts are accessible to the reactor operators and hence, under continuous surveillance. Signal lines from the process sensors, transmitters, and switches located in the mechanical equipment room are presently mixed together in the same cable runs. As part of the modifications for upgrade to 10 MW, all cabling associated with redundant equipment shall be separated and identified. The complexity of circuitry at MURR is not comparable to that of a power system. A thorough check out of each system is performed following each maintenance operation.

A.3.8 Control and Protection System Interaction

"4.7.1 Classification of Equipment. Any equipment that is used for both protective and control functions shall be classified as part of the protection system and shall meet all the requirements of this document.

4.7.2 Isolation Devices. The transmission of signals from protection system equipment for control system use shall be through isolation devices which shall be classified as part of the protection system and shall meet all the requirements of this document. No credible failure at the output of an isolation device shall prevent the associated protection system channel from meeting the minimum performance requirements specified in the design bases.

Examples of credible failures include short circuits, open circuits, grounds, and the application of the maximum credible ac or dc potential. A failure in the isolation device is evaluated in the same manner as a failure of other equipment in the protection system.

4.7.3 Single Random Failure. Where a single random failure can cause a control system action that results in a generating station condition requiring protective action and can also prevent proper action of a protection system channel designed to protect against the condition, the remaining redundant protection channels shall be capable of providing the protective action even when degraded by a second random failure.

Provisions shall be included so that this requirement can still be met if a Channel is bypassed or removed from service for test or maintenance purposes. Acceptable provisions include reducing the required coincidence, defeating the control signals taken from the redundant channels, or initiating a protective action from the bypassed channel.

4.7.4 Multiple Failures Resulting from a Credible Single Event. Where a credible single event can cause a control system action that results in a condition requiring protective action and can concurrently prevent the protective action from those protection system channels designated to provide principal protection against the condition, one of the following must be met.

4.7.4.1 Alternate channels, not subject to failure resulting from the same single event, shall be provided to limit the consequences of this event to a value specified by the design bases. In the selection of alternate channels, consideration should be given to (1) channels that sense a set of variables different from the principal channels, (2) channels that use equipment different from that of the principal channels to sense the same variable, and (3) channels that sense a set of variables different from those of the principal protection channels using equipment different from that of the principal protection channels. Both the principal and alternate protection channels shall meet all the requirements of this document.

4.7.4.2 Equipment, not subject to failure caused by the same credible single event, shall be provided to detect the event and limit the consequences to a value specified by the design bases. Such equipment shall meet all the requirements of this document." (1)

Two parts of the MURR protection system need to be considered under this criterion; the wide range monitor channel and the core inlet temperature channel. Both of thes "The MURR protection system satisfies the intent of IEEE 279 with regard to control and protection system interaction. No instrument channel provides both safety and control functions." on system.

Neither between the protection system and the control system. The output from the wide range monitor to the power control servo-amplifier is in parallel with the output to the scram trip module. The dc temperature signal to TC 909 is presently in series with the signal to the dual alarm unit that initiates a high temperature scram.

If the wide range monitor fails downscale, it would not be able to initiate a scram and at the same time the servo-amplifier would call for increased power. However, overpower protection is still available even if one of the two remaining power range monitors should have also failed. Thus, the first paragraph under IEEE 279 Section 4.7.3 is satisfied (but not the second paragraph) by the power range nuclear instrumentation. Since no provision has been made to enable a power range channel to be removed from service during operation, the second paragraph of 4.7.3 does not apply. The reactor operating schedule permits shutdowns sufficiently frequent for adequate testing and maintenance. Two additional temperature channels, 980A and 980B, will be added to provide high core inlet temperature protection. The temperature signal to the controller, TC 909, will be removed from the protective system to provide compliance with criteria 4.7 (Figure A.6).

2001

A.3.9 <u>Derivation of System Inputs</u>

"To the extent feasible and practical, protection system inputs shall be derived from signals that are direct measures of the desired variables." (1)

In general this criterion is satisfied by the MURR protection system.

A.3.10 <u>Capability for Sensor Checks</u>

"Means shall be provided for checking, with a high degree of confidence, the operational availability of each system input during reactor operation.
This may be accomplished in various ways, for example:
(1) by perturbing the monitored variable; or

(2) within the constraints of paragraph 4.11, by introducing and varying, as appropriate, a substitute input to the sensor of the same nature as the measured variable; or

(3) by cross checking between channels that bear a known relationship to each other and that have readouts available." (1)

The following sensors' status cannot be checked during reactor operation because the sensor is a switch:

Valves 509 and 507A/B (limit switches),

Reactor outlet pressure (pressure switches 944A/B),

Pressurizer pressure (pressure switches 938 and 939), and

Pressurizer level (unmonitored level controller).

With the exception of these sensors, the measured variables are indicated in the control room and routinely recorded. Proper valve limit switch operation is checked weekly during plant startup and/or shutdown. Actuation of pressure switches 944A, 944B and 943 are verified weekly during plant startup and/or shutdown. Pressurizer high pressure and low level sensors are calibrated semiannually. The operating schedule for MURR does not call for long periods of continuous operation. Therefore, frequent shutdown checks are made on those components for which cross checking is not possible.

A.3.11 Capability for Test and Calibration

"Capability shall be provided for testing and calibrating channels and the devices used to derive the final system output signal from the various channel signals. For those parts of the system where the required interval between testing will be less than the normal time interval between generating station shutdowns, there shall be capability for testing during power operation." (1)

The capability exists to test and calibrate all channels and the devices used to derive the final system output with the reactor shutdown. Experience has shown that testing at more than on a weekly basis is not required for any channel.

A.3.12 Channel Bypass or Removal from Operation

"The system shall be designed to permit any one channel to be maintained, and when required, tested or calibrated during power operation without initiating a protective action at the systems level. During such operation the active parts of the system shall of themselves continue to meet the single failure criterion." (1)

Because of the flexibility of the reactor operating schedule, shutdowns for maintenance and testing can easily be accommodated. Criteria 4.11 need not be satisfied as no provision for on-line maintenance of protective equipment is required.

A.3.13 Operating Bypasses

"Where operating requirements necessitate automatic or manual bypass of a protective function, the design shall be such that the bypass will be removed automatically whenever permissive conditions are not met. Devices used to achieve automatic removal of the bypass of a protective function are part of the protection system and shall be designed in accordance with these criteria." (1)

The MURR protection system has none of the type bypasses to which this criterion is intended to apply.

A.3.14 Indication of Bypasses

"If the protective action of some part of the system has been bypassed or deliberately rendered inoperative for any purpose, this fact shall be continuously indicated in the control room." (1)

Bypass switches are utilized to change the protective system to correspond to the three modes of operation (50 kW, 5 MW, or 10 MW). All switches are located on the reactor console in the immediate view of the reactor operator and require a key for operation. Power Level Selector Switch 1S8, which is also immediately in front of the reactor operator is interlocked with the bypass switches to prevent reactor operation with a bypass function unless the reactor is lined up for the proper mode of operation.

The reactor mode of operation is clearly indicated by a series of lights located on the control cabinet (Figure A.4).

A.3.15 Access to Means for Bypassing

"The design shall permit the administrative control of the means for manually bypassing channels or protective functions." (1)

This criterion is satisfied.

A.3.16 Multiple Set Points

"Where it is necessary to change to a more restrictive setpoint to provide adequate protection for a particular mode of operation or set of operating conditions, the design shall provide positive means of assuring that the more restrictive setpoint is used. The devices used to prevent improper use of a less restrictive setpoint shall be considered a part of the protection system and shall be designed in accordance with the other provisions of these criteria regarding performance and reliability." (1)

The proposed 10 MW configuration has five multiple setpoint possibilities; reactor loop flow, pool loop flow, reactor power, core differential pressure, and heat exchanger differential pressure. The electrical interlocks associated with each setpoint ensure that the correct protective system lineup is utilized. The interlocking mechanism complies with this criterion. In December 2006, DPS 928A and DPS 928B were removed as part of the primary coolant heat exchanger replacement project; therefore, heat exchanger differential pressure is no longer used. 2006

A.3.17 Completion of Protective Action Once It Is Initiated

"The protection system shall be so designed that, once initiated, a protective action at the system level shall go to completion. Return to operation shall require subsequent deliberate operator action." (1)

This criterion is satisfied.

A.3.18 Manual Initiation

"The protection system shall include a means for manual initiation of each protective action at the system level (for example, reactor trip, containment isolation, safety injection, core spray, etc.). No single failure, as defined by the note following Section 4.2, within the manual, automatic, or common portions of the protection system shall prevent initiation of protective action by manual or automatic means. Manual initiation should depend upon the operation of a minimum of equipment." (1)

The reactor operator has a manual scram button, 1S10, on the control console which opens the input to the noncoincidence logic units as well as interrupts the power to the trip actuator amplifiers (Figure A.1). In addition, two switches (1S1 and 1S14) are available on the console which open the circuit supply power to the trip actuator amplifiers (Figure A.1). The facility evacuation which can be initiated by a switch on the reactor console, in the health physics office, or in the facility lobby, causes a containment building isolation and a reactor scram. A reactor isolation alone can also be initiated manually at the reactor console by switch 1S15 (Figure A.10).

A.3.19 Access to Setpoint Adjustments, Calibration, and Test Points

"The design shall permit the administrative control of access to all setpoint adjustments, module calibration adjustments, and test points." (1)

This criterion is satisfied.

A.3.20 Identification of Protective Actions

"Protective actions shall be indicated and identified down to the channel level." (1)

This criterion is not completely satisfied in that there are a few sensors which are combined to result in one annunciation at a system level (Figure A.7). This is the case, for example, for the four low pressure sensors which are combined to result in the "Reactor Loop Low Pressure Scram" annunciation and the three sensors which provide a "Reactor Loop High Temperature Scram" annunciation. A post trip review of available control room information would identify the initiating sensor.

A.3.21 Information Readout

"The protection system shall be designed to provide the operator with accurate, complete and timely information pertinent to its own status and to generating station safety. The design shall minimize the development of conditions which would cause meters, annunciators, recorders, alarms, etc., to give anomalous indications confusing to the operator." (1)

Strictly speaking, this criterion is not satisfied as discussed under the preceding section; however, it is felt that in view of its relative simplicity, the MURR protection system is adequate with respect to information readout.

A.3.22 System Repair

"The system shall be designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules." (1)

This criterion is adequately satisfied.

A.3.23 Identification

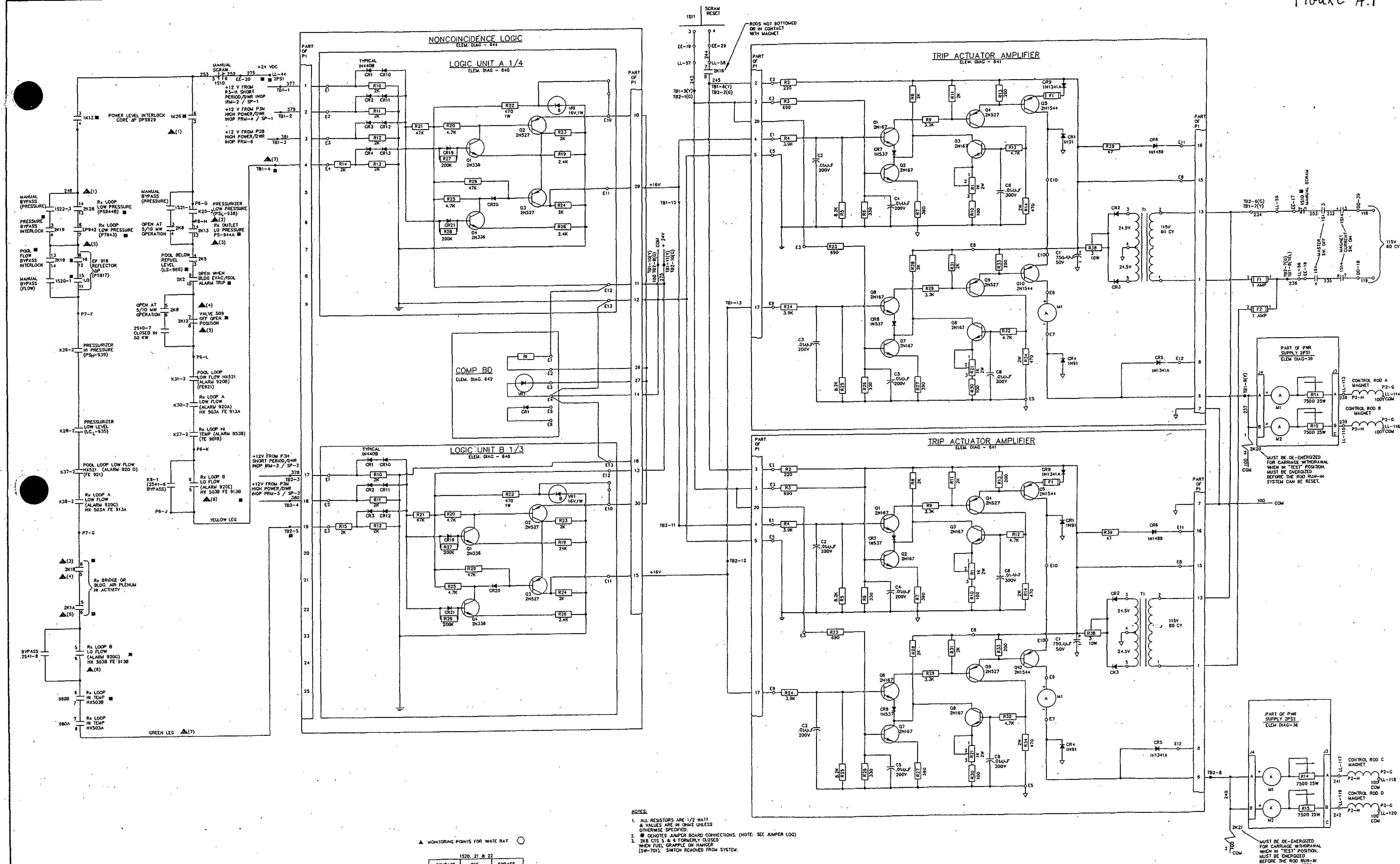
"In order to provide assurance that the requirements given in this document can be applied during the design, construction, maintenance and operation of the plant, the protection system equipment (for example, interconnecting wiring, components, modules, etc.) shall be identified distinctly as being in the protective system. This identification shall distinguish between redundant portions of the protection system. In the installed equipments, components, or modules mounted in assemblies that are clearly identified as being in the protective system do not themselves require identification." (1)

As part of the program to separate redundant portions of the protective system, system components, circuits, interconnecting wiring, etc., shall be clearly identified.

A.4 References

(1) IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations, 1971.

FIGURE A.1



▲ MONITORING POINTS FOR WHITE RAT ○

1S20, 21 & 22

CONTACT	OFF	BYPASS
1		X
2	X	
3		X
4	X	

CR2940

(KEY LOCK WITH KEY REMOVABLE IN OFF POSITION ONLY)

NOTES:

1. ALL RESISTORS ARE 1/2 WATT & VALUES ARE IN OHMS UNLESS OTHERWISE SPECIFIED.
2. ■ DENOTES JUMPER BOARD CONNECTIONS. (NOTE: SEE JUMPER LOG)
3. 2K8 CTS 5 & 6 FORMERLY CLOSED WHEN FUEL GRAPPLE ON HANGER (SW-701). SWITCH REMOVED FROM SYSTEM.

REFERENCE DRAWINGS

1. NEUTRON MONITORING SYSTEM ELEM. DIAG. - DWG. 40
2. REACTOR CONTROL ELEM. DIAG. - DWG. 42
3. ANNUNCIATOR CONTROL ELEM. DIAG. - DWG. 138
4. PROCESS INST. INTERLOCK ELEM. DIAG. - DWG. 41
5. RELAY FUNCTIONS - DWG. 229

REV. NO.	DESCRIPTION	DR. BY	DATE
26	MOD. PKG 04-1 ADDENDUM 1, 942 LFE TO BEEDE	THS	04/09/07
25	MOD. PKG 05-8 ADDENDUM 1	THS	12/26/06
24	MISCELLANEOUS CORRECTIONS	BJN	10/13/03
23	980 A/B MOD 01-3 ADDENDUM 1	JJL	10/10/02
22	MOD PKG. 01-3	JCA	4/5/01
21	REPLACED 928A, 928B, AS PER MOD 97-1	JCA	12/19/00
20	ADDED JUMPERS TO GRN & YELLOW LEGS AS PER MOD-75-1 ADD1	JCA	11/23/00

seal 2007

DATE: 3/3/98
DRAWN BY: J.RILEY
CHECKED BY:
ENGINEER: LDF
CODE:
REVISION NUMBER: 26
REVISION DATE: 04/09/07

FIGURE A.1

UNIVERSITY OF MISSOURI-COLUMBIA
FACILITY OPERATIONS
research reactor facility

SAFETY SYSTEM

MURR NUMBER: 139

SHEET 1 OF 1

[redacted]

REACTOR WATER

seal

2007

DATE: 2/22/95

DRAWN BY: JDR / DGN

CHECKED BY:
ENGINEER: CBE TONY S.

CODE: LPF 10-2-07

REVISION NUMBER:

REVISION DATE:

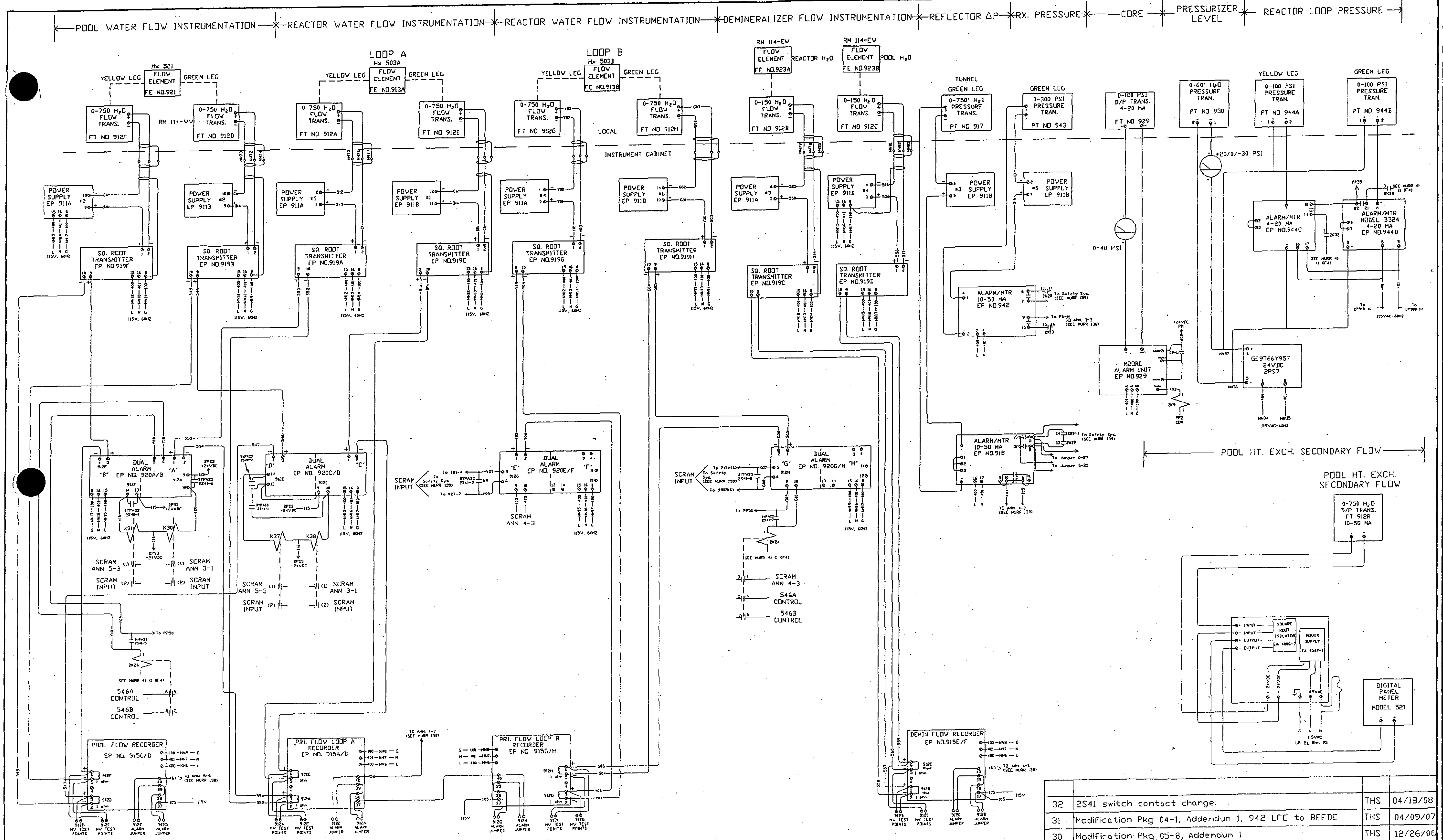
FIGURE A.2

UNIVERSITY OF MISSOURI–COLUMBIA
FACILITIES OPERATIONS
research reactor facility

PIPING & INSTRUMENT DIAGRAM

MURR NUMBER: 156

SHEET 1 OF 1



REV.	DESCRIPTION	BY	DATE
32	2S41 switch contact change.	THS	04/18/08
31	Modification Pkg 04-1, Addendum 1, 942 LFE to BEEDE	THS	04/09/07
30	Modification Pkg 05-8, Addendum 1	THS	12/26/06
29	Redrew to simplify Power Supply wiring. Addition of Modifications 04-1 add. 10 & 12	THS	8/09/06
28	Mod. Pkg. 06-01, Removed Bailey Inst. Rack, Resistor Unit and Alarm module.	THS	3/28/06
27	WO# 05-1990 Redesign of Pool Ht. Exch. Sec. Flow D/P Inst, plus misc. typo's	THS	7/13/05

2008

DATE:
DRAWN BY: TOM SEEGER
CHECKED BY:
ENGINEER: CPF
CODE:
REVISION NUMBER: 32
REVISION DATE: 04/18/08

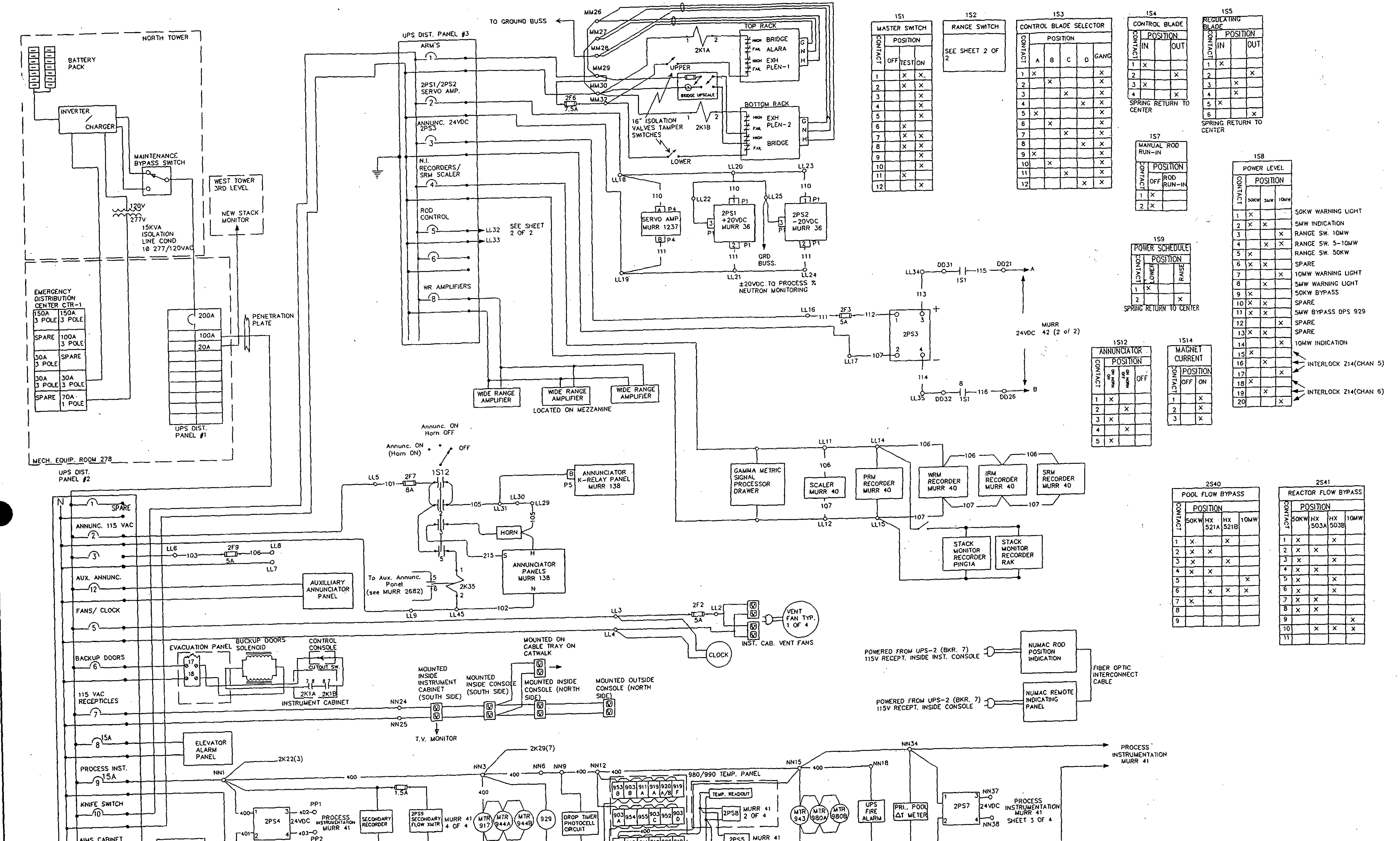
FIGURE A.3

UNIVERSITY OF MISSOURI-COLUMBIA
FACILITIES OPERATIONS
research reactor facility

PROCESS INSTRUMENTATION CONTROL & INTERLOCK

MURR NUMBER: 41
SHEET 3 of 4

FIGURE A.4.a



2007

DATE: 11/17/00

DRAWN BY: JCA

CHECKED BY:
ENGINEER: CPF
CODE:

REVISION NUMBER: 38

REVISION DATE: 05/08/07

FIGURE A.4.a

UNIVERSITY OF MISSOURI-COLUMBIA
FACILITIES OPERATIONS
research reactor facility

REACTOR CONTROL SYSTEM

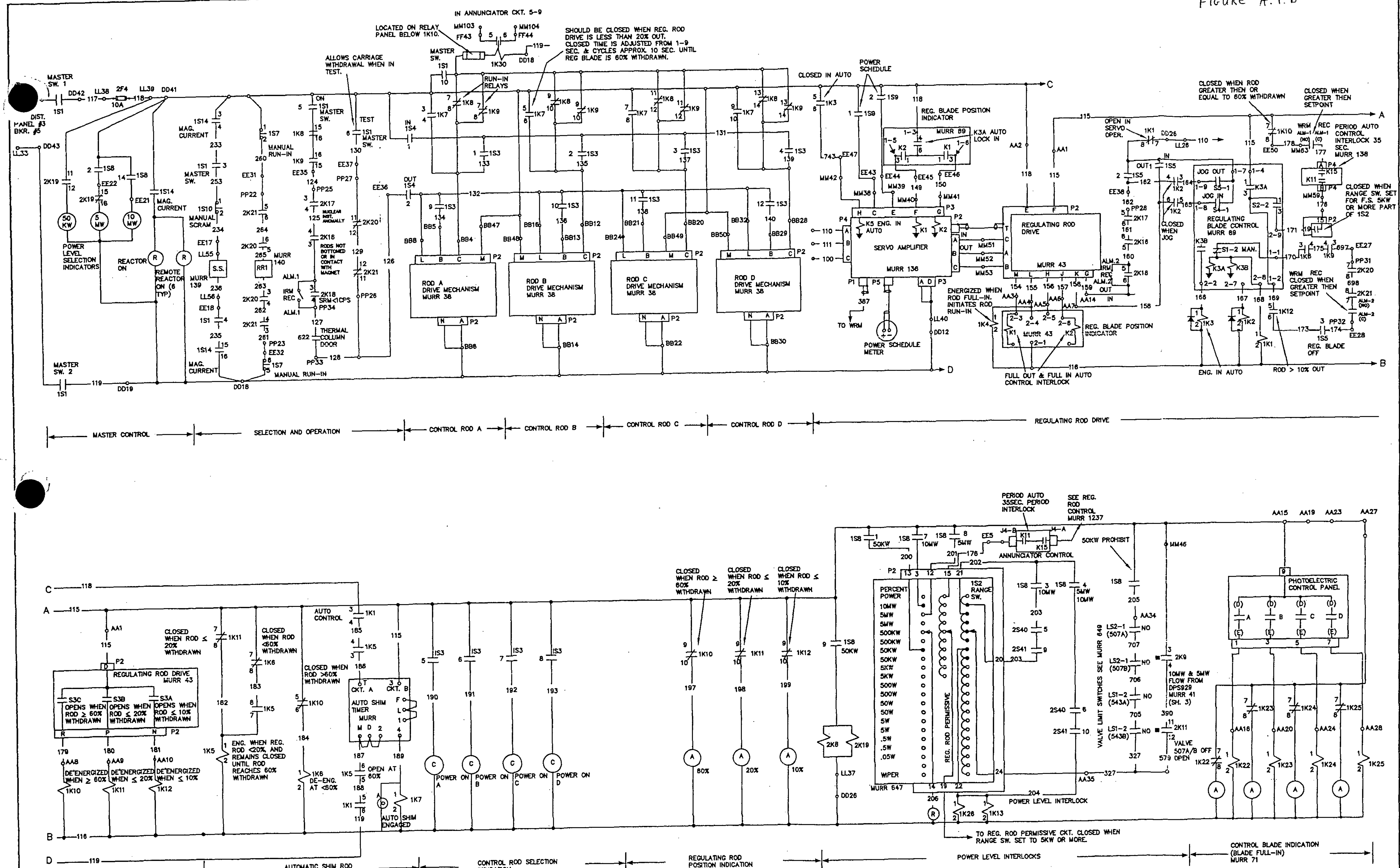
REVISIONS

NO.	DESCRIPTION	Init.	DATE
38	MOD. PKG. 07-2, ANNUNCIATOR LOSS OF POWER	THS	05/08/07
37	MOD. PKG. 05-8 ADDENDUM 1, AND GENERAL UPDATE	THS	12/26/06
36	INSTALLED NEW LINE CONDITIONER, REMOVED ELGAR AND KNIFE SWITCH MOD PKG 88-11 ADDEN #1	CHJ	10/13/04
35	ADDED ELVTR ALARM PANEL AND GENERAL UPDATE	BJN	01/15/03
34	CORRECTED UPS2 BREAKER 9 LOADS	JCA	10/17/01
33	CORRECTED SEVERAL INACCURACIES	JCA	5/30/01

MURR NUMBER: 42

SHEET 1 OF 2

FIGURE A.4.b



■ DENOTES JUMPER BOARD CONNECTIONS

NO.	DESCRIPTION	INIT.	DATE
32	MOD PKG 04-04 (AUTOSHIM), & TYPOS	BJN	7/14/04
31	GENERAL UPDATE	BJN	1/15/03
30	MOD PKG 95-1 ADD 1E	JCA	6/26/01
29	REDRAWN IN AUTOCAD	JCA	1/5/00
28	REMOVED FISSION COUNTER DRIVE	THS	6/27/97
27	ADDED TITLE BLOCK	THS	12/28/93
26	COMPLETELY REDRAWN	THS	7/19/90

NO.	DESCRIPTION	INIT.	DATE
33	RELOCATED AUTO SHIM ENGAGED LIGHT	CHJ	10/05/04

seal: 2004

DATE: 1/4/01
DRAWN BY: JCA
CHECKED BY:
ENGINEER: LPF
CODE:
REVISION NUMBER: 29
REVISION DATE: 1/4/01

UNIVERSITY OF MISSOURI–COLUMBIA
FACILITIES OPERATIONS
research reactor facility

REACTOR CONTROL SYSTEM

MURR NUMBER: 42
SHEET 2 OF 2

FIGURE A.4.b

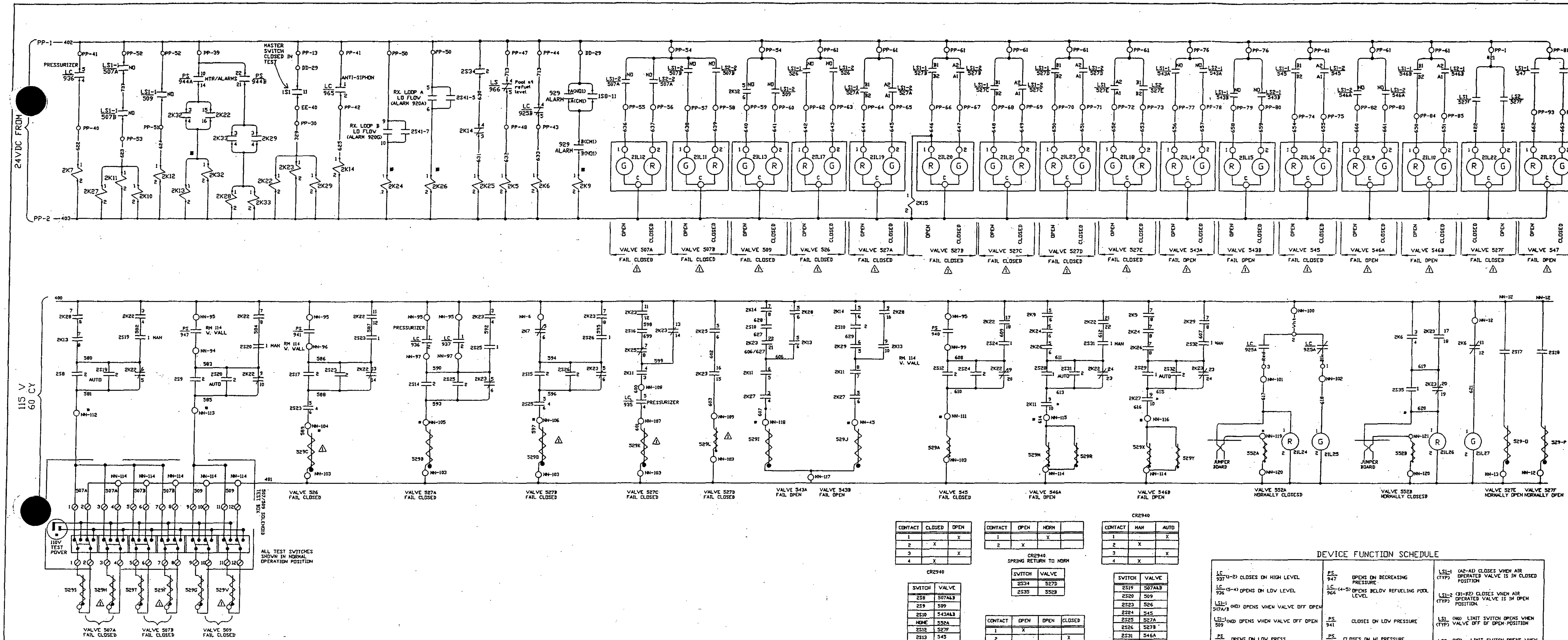


TABLE I

PUMP	PT. NO.	LIGHTS	SWITCH	RESISTOR
REACTOR LOOP CLEANUP	513A	2IL4	2S3	2R5&6
POOL LOOP CLEANUP	513B	2IL5	2S4	2R7&8
SKIMMER	532	2IL6	2S5	2R9&10

CONTACT	CLOSED	OPEN
1		X
2	X	
3		X
4	X	

CR2940

SWITCH	VALVE
2S8	507A&B
2S9	509
2S10	543A&B
NONE	552A
2S12	527F
2S13	545
2S14	527A
2S15	527B
2S16	527C
2S28	546A
2S29	546B
2S17	526
2S18	527F
2S33	547

CONTACT	OPEN	NORM	
1		X	
2	X		

CR2940
SPRING RETURN TO NORM

SWITCH	VALVE
2S34	527D
2S35	552B

CONTACT	OPEN	OPEN	CLOSED
1	X		
2			X
3	X		
4			X

CR2940
SPRING RETURN TO OFF

SWITCH	VALVE
2S30	530

2S1 THRU 2S7

CONTACT	OFF	ON
1		X
2	X	
3		X
4	X	

CR2940

CONTACT	MAN	AUTO
1		X
2	X	
3		X
4	X	

SWITCH	VALVE
2S19	507A&B
2S20	509
2S23	526
2S24	545
2S25	527A
2S26	527B
2S31	546A
2S32	546B
NONE	530
	543A/B
	527C
	527D
	552A
NONE	552B

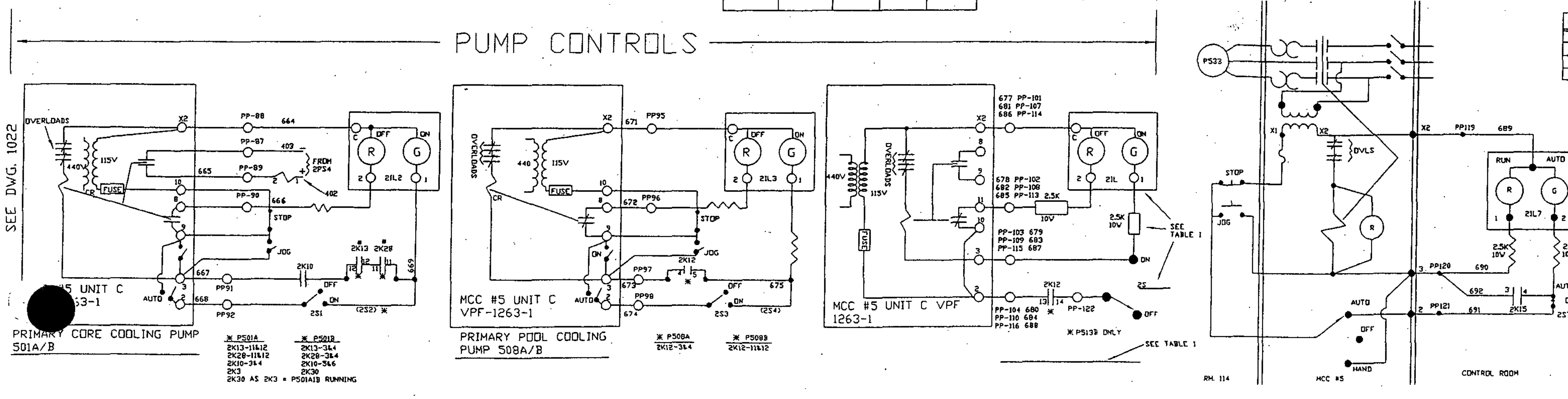
DEVICE FUNCTION SCHEDULE

LC/937 (1-2) CLOSES ON HIGH LEVEL
LC/936 (5-4) OPENS ON LOW LEVEL
LS1-1/507A/B (NO) OPENS WHEN VALVE OFF OPEN
LS1-1/509 (NO) OPENS WHEN VALVE OFF OPEN
PS/944A OPENS ON LOW PRESS
PS/944B OPENS ON LOW PRESS
DPS 929 OPENS ON LOW ΔP (LOW FLOW) ACROSS THE CORE.
DPS 928A (B) OPENS ON LOW ΔP (LOW FLOW) ACROSS HX 503A (B)
K9/271 OPENS ON LOW FLOW (AL920A)
LC/965 (1-2) OPENS ON RISING LEVEL

PS/947 OPENS ON DECREASING PRESSURE
LC/966 (4-5) OPENS BELOW REFUELING POOL LEVEL
PS/941 CLOSES ON LOW PRESSURE
PS/940 CLOSES ON HI PRESSURE
LC/925A (1-2) OPENS ON LOW LEVEL (2-3) CLOSES ON LOW LEVEL
LC/925B (2-3) OPENS ON LOW LEVEL
LC/936 (2-1) CLOSES ON HIGH LEVEL
LC/937 (2-1) CLOSES ON HIGH LEVEL
LC/935 (5-4) OPENS ON LOW LEVEL
LS/910 CLOSES ON LOW LEVEL

LS1-1 (TYP) (A2-A1) CLOSES WHEN AIR OPERATED VALVE IS IN CLOSED POSITION
LS1-2 (TYP) (B1-B2) CLOSES WHEN AIR OPERATED VALVE IS IN OPEN POSITION.
LS1 (TYP) (NO) LIMIT SWITCH OPENS WHEN VALVE OFF OF OPEN POSITION
LS2 (TYP) (NO) LIMIT SWITCH OPENS WHEN VALVE OFF OF CLOSED POSITION

PUMP CONTROLS



REFERENCE DRAWINGS

1. REACTOR CONTROL SYSTEM - DWG. NO. 42
2. PNEUMATIC CONTROL SYSTEM DIAGRAM - DWG. NO. 172
3. ELECTRIC CONTROL SYSTEM DIAGRAM - DWG. NO. 171
4. RELAY FUNCTIONS - DWG. NO. 229

----------- OLD G.E. DWG. NO. 104R784 -------

NOTES

△ VALVE RETURNS TO POSITION SHOWN IN EVENT OF AN ELECTRICAL OR PNEUMATIC FAILURE.

■ DENOTES JUMPER BOARD CONNECTIONS.

FIGURE A.5

24	Jumper Board Mod. 75-1 Addendum 2	THS	04/18/08
23	Modification Pkg. 05-8 Addendum 1	THS	12/26/06
22	CORRECTED MISC. TYPO'S	JCA	4/16/02
21	MOD PKG 01-6	JCA	9/21/01
20	REPLACED 928A&B, 929 AS PER MOD 97-1	JCA	11/21/00

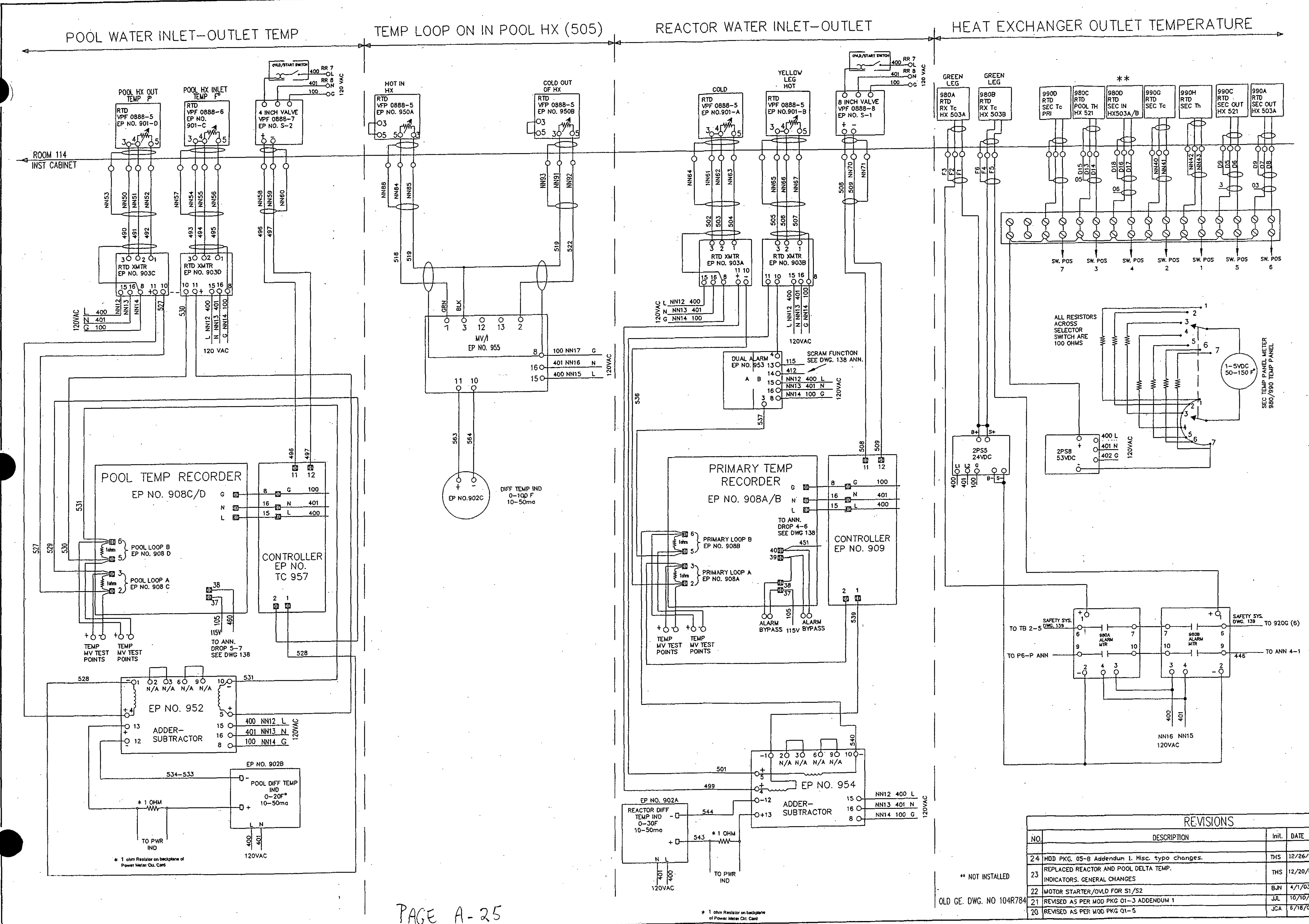
REVISIONS

TITLE
PROCESS INSTRUMENTATION CONTROL & INTERLOCK

RESEARCH REACTOR FACILITY
UNIVERSITY OF MISSOURI

DRAWN BY JCA | APPROVED CPF | DATE 12/5/00

SHEET NO. 1 OF 4 | DWG. NO. 41



REVISIONS

NO.	DESCRIPTION	Init.	DATE
24	MOD PKG. 05-8 Addendum 1. Misc. typo changes.	THS	12/26/06
23	REPLACED REACTOR AND POOL DELTA TEMP. INDICATORS. GENERAL CHANGES	THS	12/20/05
22	MOTOR STARTER/OVLD FOR S1/S2	BJN	4/1/03
21	REVISED AS PER MOD PKG 01-3 ADDENDUM 1	JJL	10/10/02
20	REVISED AS PER MOD PKG 01-5	JCA	6/18/01

seal 2006

DATE: 2/18/96
DRAWN BY: jca
CHECKED BY:
ENGINEER: CPF
CODE:
REVISION NUMBER: 24
REVISION DATE: 12/26/06

FIGURE A.6

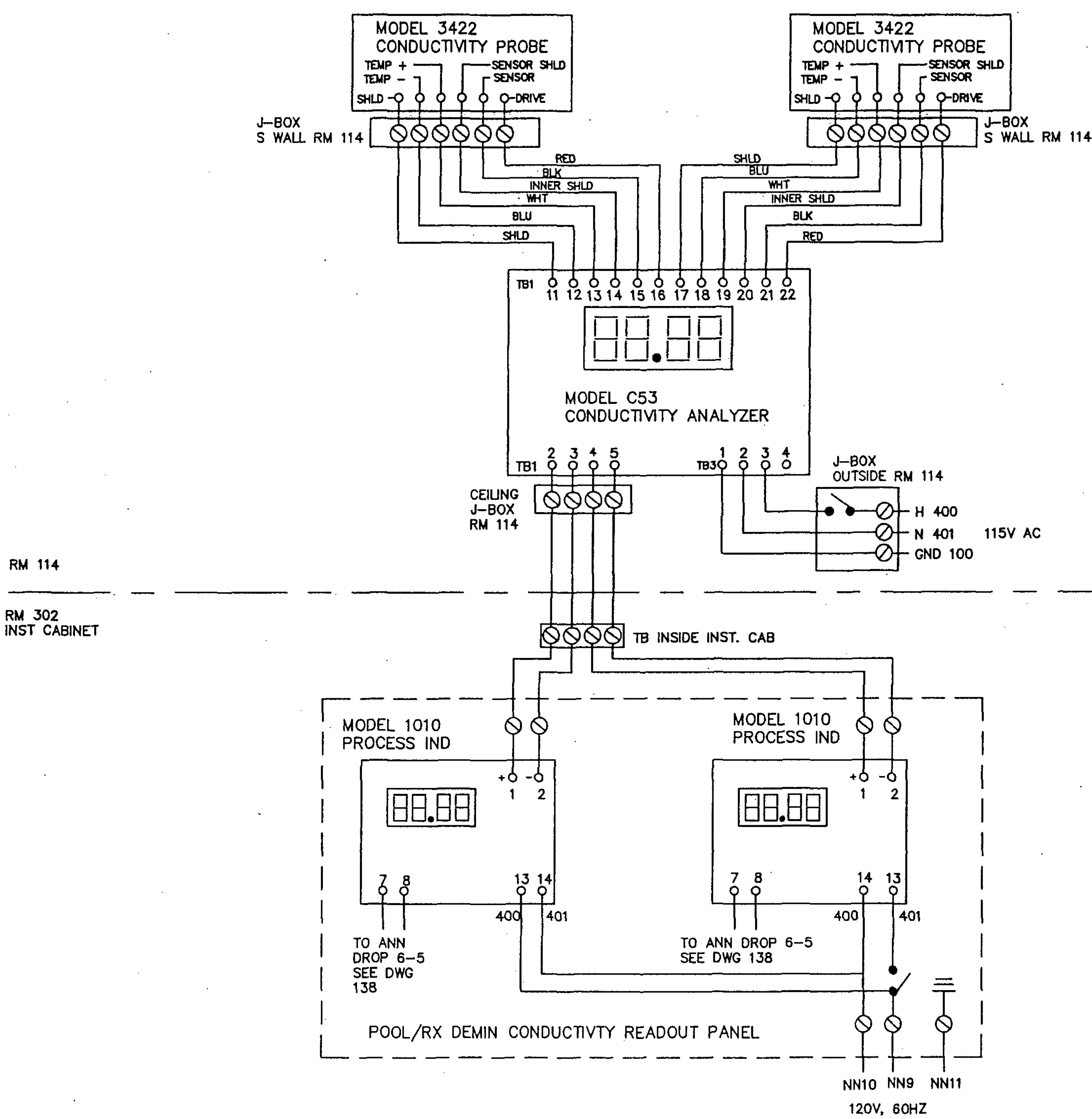
UNIVERSITY OF MISSOURI-COLUMBIA
FACILITIES OPERATIONS
research reactor facility

10MW PROCESS INSTRUMENTATION CONTROL & INTERLOCK (TEMPERATURE INST.)

MURR NUMBER: 41
SHEET 2 OF 4

CONDUCTIVITY INSTRUMENTATION

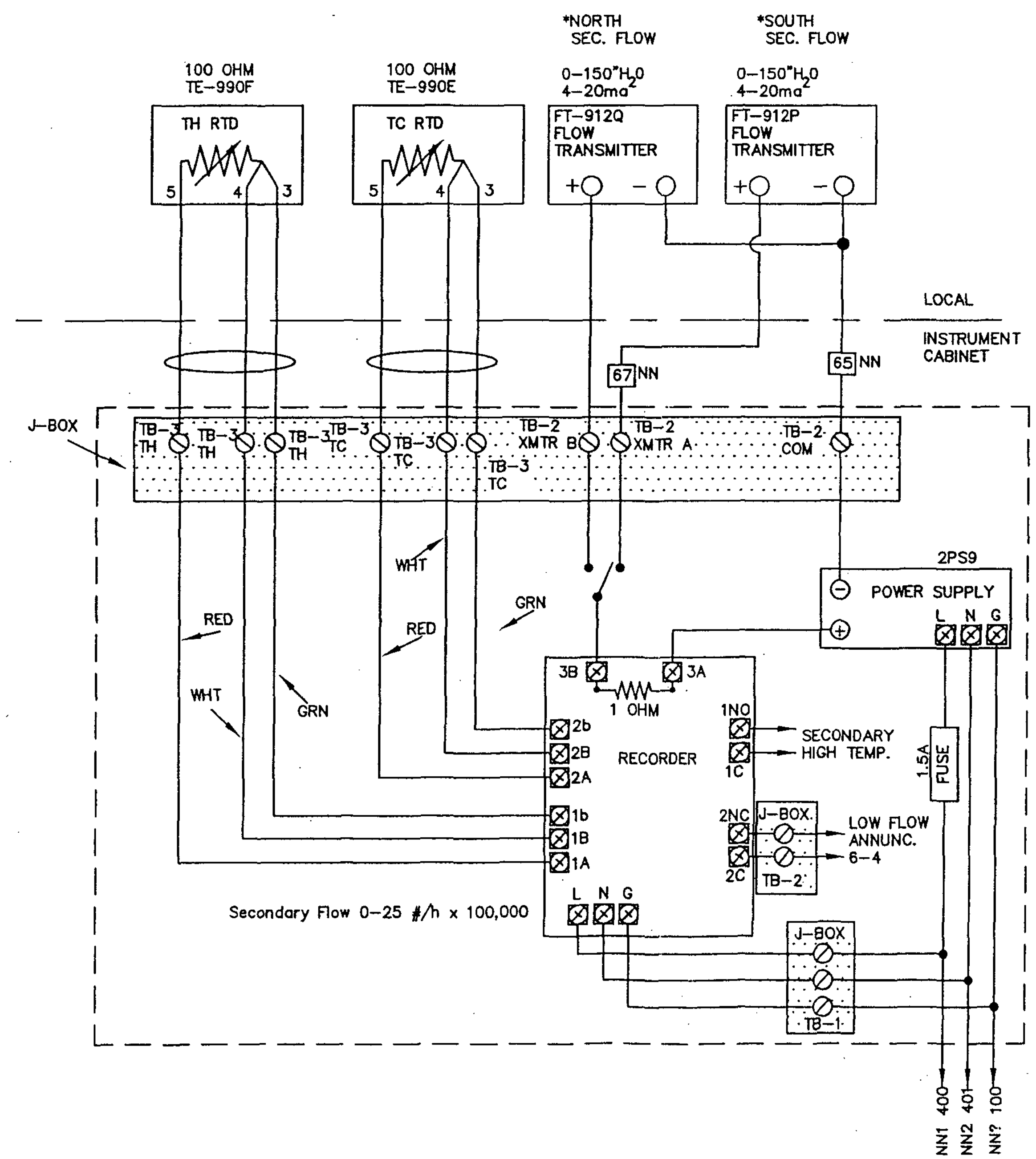
TYP OF 2



SECONDARY FLOW/ TEMPERATURE

FIGURE A.6.a

*Square Root function to derive flow from differential pressure is performed by channel 3 of secondary recorder.



PAGE A-25.a

REV.	DESCRIPTION	BY	DATE
2	MOD PKG 01-5 ADDENDUM 1	BJN	5/18/04
1	MOD PKG 99-4	JCA	11/7/01

seal 2004

DATE: 6/18/01
DRAWN BY: JCA
CHECKED BY:
ENGINEER:
CODE:
REVISION NUMBER: 2
REVISION DATE: 5/18/04

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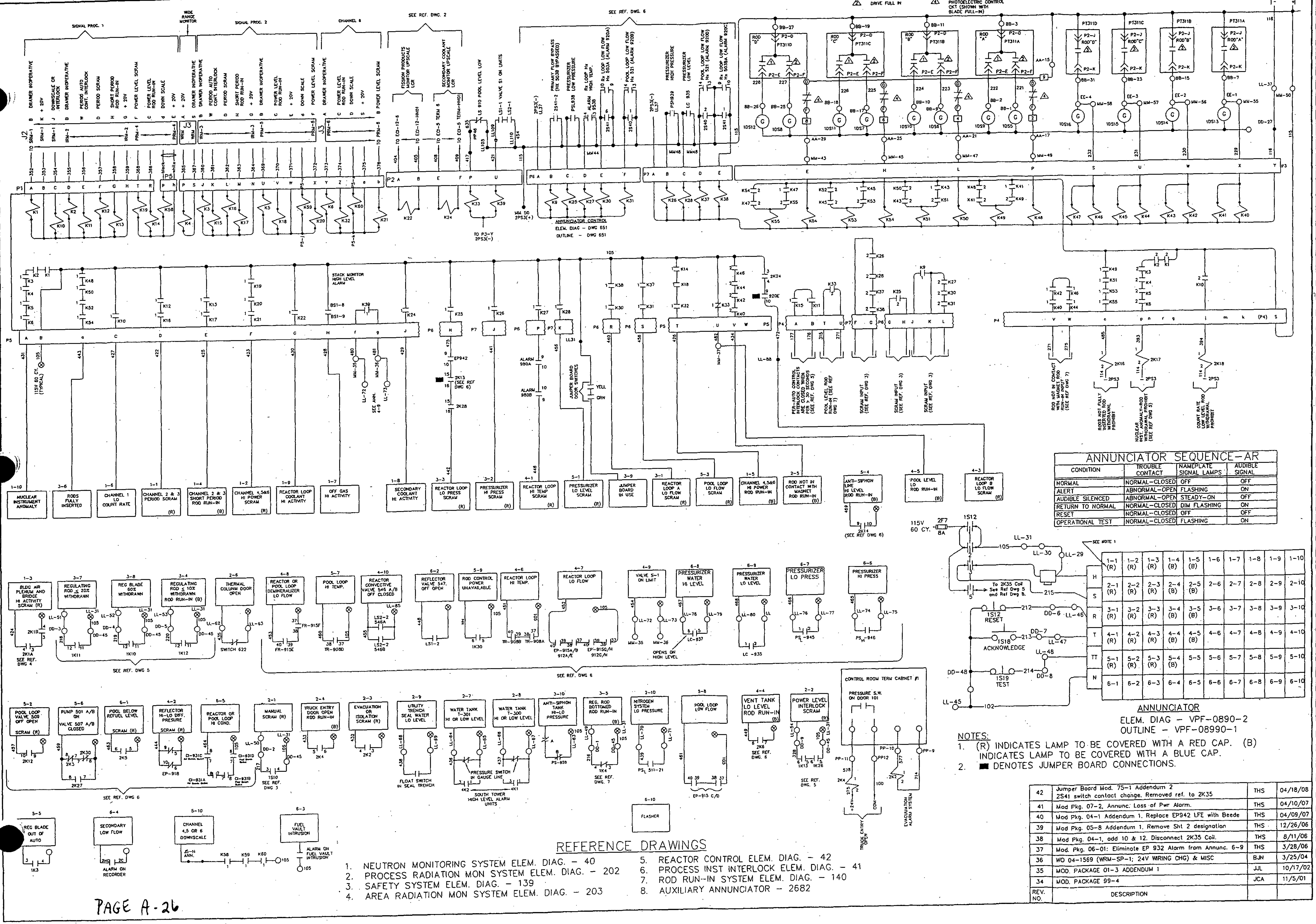
FACILITIES OPERATIONS
research reactor facility

FIGURE A.6.a

10 MW PROCESS INSTRUMENTATION CONTROL & INTERLOCK

MURR NUMBER: 41

SHEET 4 of 4



ANNUNCIATOR SEQUENCE–AR

CONDITION	TROUBLE CONTACT	NAMEPLATE SIGNAL LAMPS	AUDIBLE SIGNAL
NORMAL	NORMAL–CLOSED	OFF	OFF
ALERT	ABNORMAL–OPEN	FLASHING	ON
AUDIBLE SILENCED	ABNORMAL–OPEN	STEADY–ON	OFF
RETURN TO NORMAL	NORMAL–CLOSED	DIM FLASHING	ON
RESET	NORMAL–CLOSED	OFF	OFF
OPERATIONAL TEST	NORMAL–CLOSED	FLASHING	ON

H	1–1 (R)	1–2 (R)	1–3 (R)	1–4 (B)	1–5 (B)	1–6	1–7	1–8	1–9	1–10
S	2–1 (R)	2–2 (R)	2–3 (R)	2–4 (B)	2–5 (B)	2–6	2–7	2–8	2–9	2–10
R	3–1 (R)	3–2 (R)	3–3 (R)	3–4 (B)	3–5 (B)	3–6	3–7	3–8	3–9	3–10
T	4–1 (R)	4–2 (R)	4–3 (R)	4–4 (B)	4–5 (B)	4–6	4–7	4–8	4–9	4–10
TT	5–1 (R)	5–2 (R)	5–3 (R)	5–4 (B)	5–5	5–6	5–7	5–8	5–9	5–10
N	6–1	6–2	6–3	6–4	6–5	6–6	6–7	6–8	6–9	6–10

ANNUNCIATOR
ELEM. DIAG – VPF–0890–2
OUTLINE – VPF–08990–1

NOTES:
1. (R) INDICATES LAMP TO BE COVERED WITH A RED CAP. (B) INDICATES LAMP TO BE COVERED WITH A BLUE CAP.
2. ■ DENOTES JUMPER BOARD CONNECTIONS.

REFERENCE DRAWINGS

1. NEUTRON MONITORING SYSTEM ELEM. DIAG. – 40
2. PROCESS RADIATION MON SYSTEM ELEM. DIAG. – 202
3. SAFETY SYSTEM ELEM. DIAG. – 139
4. AREA RADIATION MON SYSTEM ELEM. DIAG. – 203
5. REACTOR CONTROL ELEM. DIAG. – 42
6. PROCESS INST INTERLOCK ELEM. DIAG. – 41
7. ROD RUN–IN SYSTEM ELEM. DIAG. – 140
8. AUXILIARY ANNUNCIATOR – 2682

REV. NO.	DESCRIPTION		
42	Jumper Board Mod. 75–1 Addendum 2 2S41 switch contact change. Removed ref. to 2K35	THS	04/18/08
41	Mod Pkg. 07–2, Annunc. Loss of Pwr Alarm.	THS	04/10/07
40	Mod Pkg. 04–1 Addendum 1. Replace EP942 LFE with Beede	THS	04/09/07
39	Mod Pkg. 05–8 Addendum 1. Remove Sht 2 designation	THS	12/26/06
38	Mod Pkg. 04–1, add 10 & 12. Disconnect 2K35 Coil.	THS	8/11/06
37	Mod. Pkg. 06–01: Eliminate EP 932 Alarm from Annunc. 6–9	THS	3/28/06
36	WO 04–1569 (WRM–SP–1; 24V WIRING CHG) & MISC	BJN	3/25/04
35	MOD. PACKAGE 01–3 ADDENDUM 1	JJL	10/17/02
34	MOD. PACKAGE 99–4	JCA	11/5/01

PAGE A-26

seal
2008

DATE: 5/18/95
DRAWN BY:
J RILEY
CHECKED BY:
ENGINEER: LPF
CODE:
REVISION NUMBER:
42
REVISION DATE:
04/18/08

FIGURE A.7

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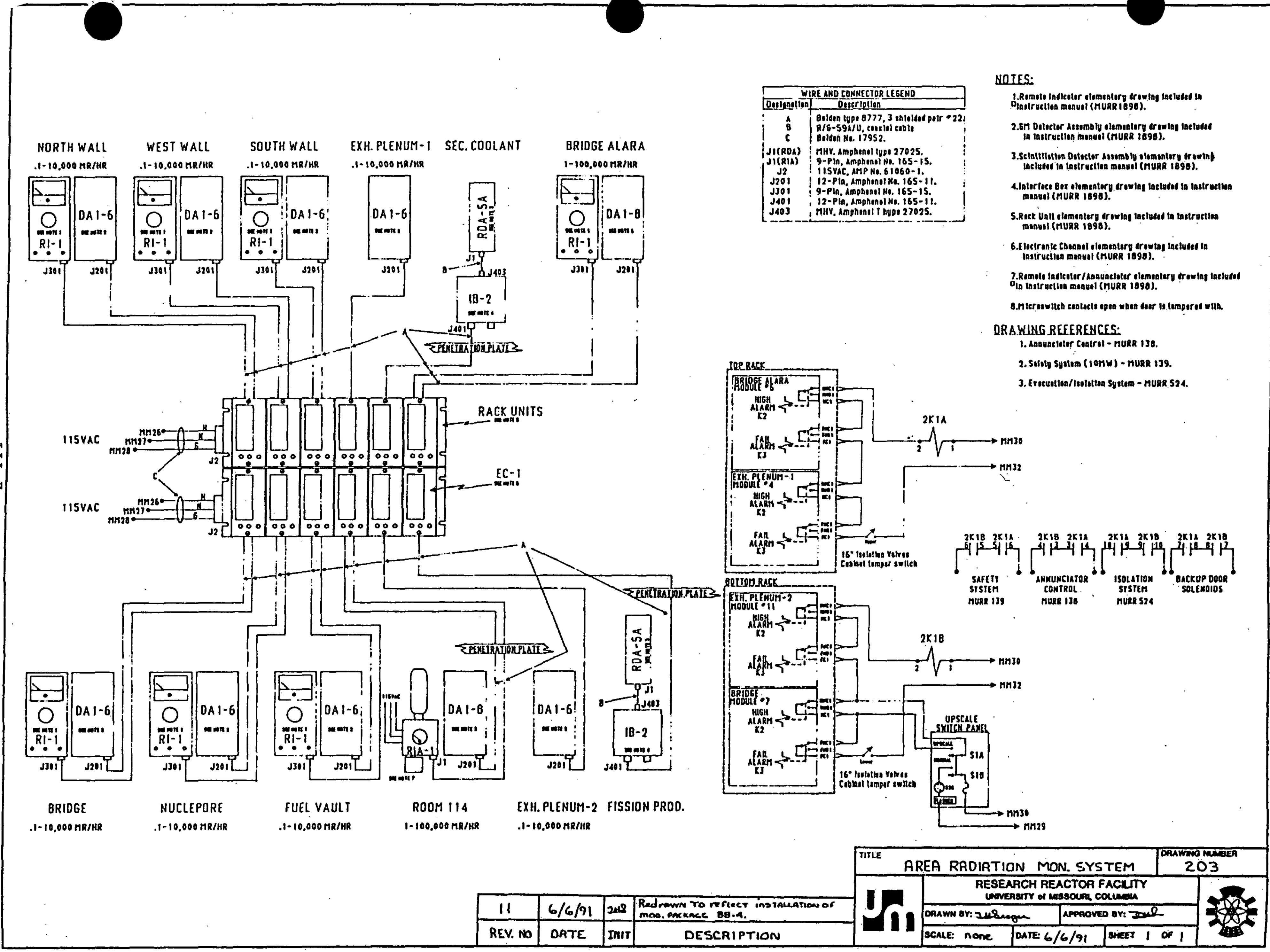
ANNUNCIATOR CONTROL 10 MW.

MURR NUMBER:
138

SHEET
1 OF 1



WIRE AND CONNECTOR LEGEND	
Designation	Description
A	Belden type 8777, 3 shielded pair #22
B	R/G-59A/U, coaxial cable
C	Belden No. 17952.
J1(RDA)	MHV, Amphenol type 27025.
J1(RIA)	9-Pin, Amphenol No. 165-15.
J2	115VAC, AMP No. 61060-1.
J201	12-Pin, Amphenol No. 165-11.
J301	9-Pin, Amphenol No. 165-15.
J401	12-Pin, Amphenol No. 165-11.
J403	MHV, Amphenol T type 27025.

NOTES:

1. Remote Indicator elementary drawing included in instruction manual (MURR 1898).
2. GM Detector Assembly elementary drawing included in instruction manual (MURR 1898).
3. Scintillation Detector Assembly elementary drawing included in instruction manual (MURR 1898).
4. Interface Box elementary drawing included in instruction manual (MURR 1898).
5. Rack Unit elementary drawing included in instruction manual (MURR 1898).
6. Electronic Channel elementary drawing included in instruction manual (MURR 1898).
7. Remote Indicator/Annunciator elementary drawing included in instruction manual (MURR 1898).
8. Microswitch contacts open when door is tampered with.

DRAWING REFERENCES:

1. Annunciator Control - MURR 138.
2. Safety System (10MW) - MURR 139.
3. Evacuation/Isolation System - MURR 524.

REV. NO	DATE	INIT	DESCRIPTION
11	6/6/91	JWB	Redrawn to reflect installation of mod. package 88-4.

TITLE: AREA RADIATION MON. SYSTEM

DRAWING NUMBER: 203

RESEARCH REACTOR FACILITY
UNIVERSITY of MISSOURI, COLUMBIA

DRAWN BY: J.W.Brager | APPROVED BY: JWB
SCALE: none | DATE: 6/6/91 | SHEET 1 OF 1

Figure A.8

1990-1991

FIGURE A.9

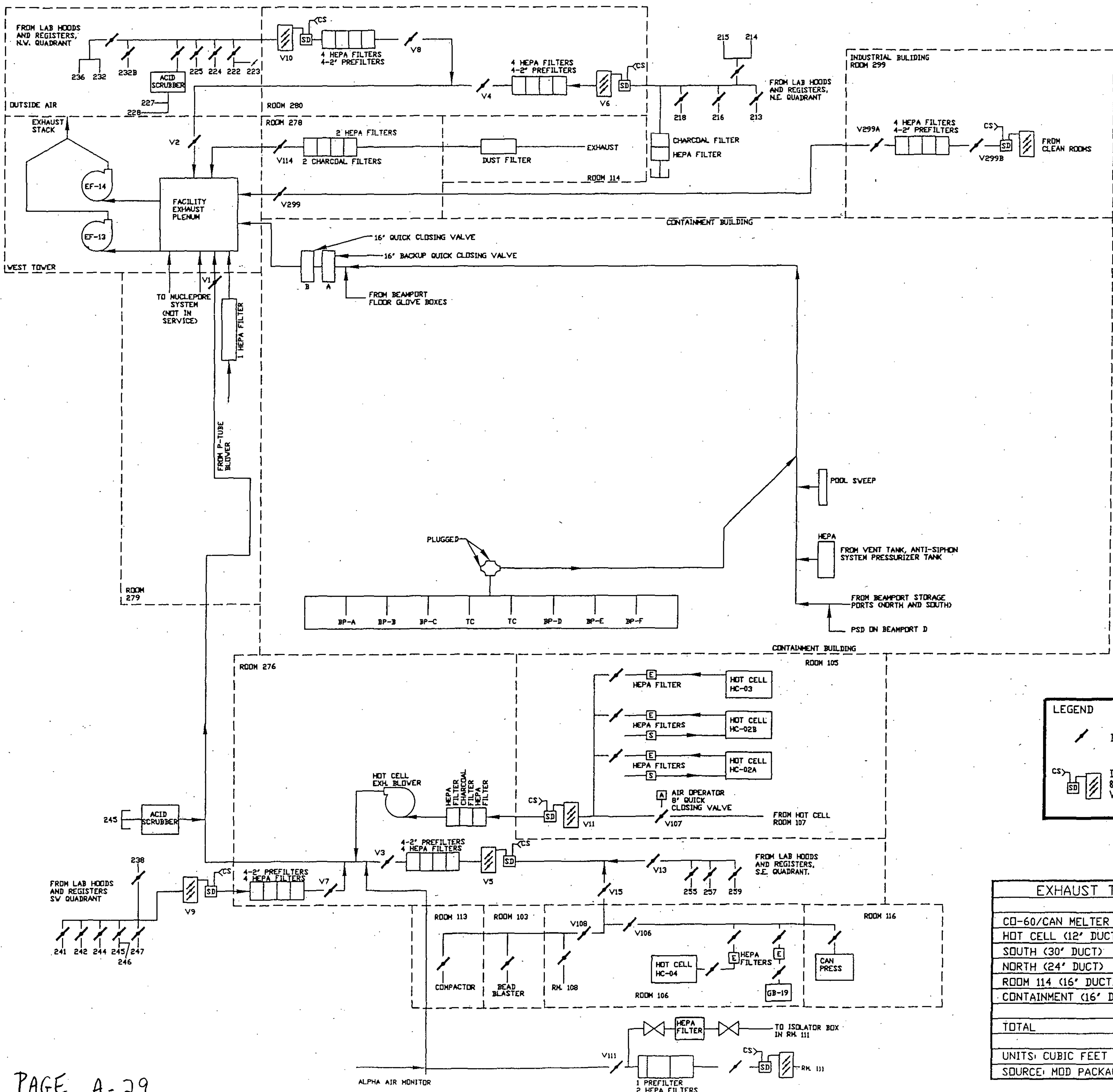
Deleted **(rev 1995)**

FIGURE A.10

Deleted Safeguards Information **(rev 1995)**

FIGURE A.11

NO. REV.	DESCRIPTION	DR. BY	DATE
4	UPDATED ON CAD	JDR	4/22/95
5	UPDATED PER CONTROL RM.	JDR	8/11/97
6	UPDATED PER CONTROL RM.	JCA	11/28/00
7	REMOVED EQUIPMENT	JCA	10/29/01
8	ADDED BP FLOOR GLOVE BOX	JCA	3/3/02
9	REMOVED GLOVE BOXES ADDED HD-166 H.C.	JJL	6/4/02
10	GENERAL UPDATE	JJL	8/19/02
11	ADD INDUSTRIAL BLDG	DN	12/19/02
12	GENERAL UPDATE	BJN	10/29/03
13	ADD SMOKE DETECTORS	DN	10/30/03
14	UPDATE SMOKE DETECTORS	BJN	12/22/03
15	REMOVED AIR SUPPLIES	DN	8/20/04
16	2004 GENERAL REVISION	DN	9/13/04
17	ADDED 246 ICP	CHJ	10/25/04
18	ADDED PSD ON BEAMPORT D	CHJ	12/15/04



EXHAUST TRUNK LINE AIR FLOW MEASUREMENT	EF 13 (FAST SPEED)	EF 14 (FAST SPEED)
CO-60/CAN MELTER (12" DUCT)	2322	2304
HOT CELL (12" DUCT)	828	1076
SOUTH (30" DUCT)	13125	13026
NORTH (24" DUCT)	7408	7352
ROOM 114 (16" DUCT)	2007	2209
CONTAINMENT (16" DUCT)	2550	2591
TOTAL	28240	28558
UNITS: CUBIC FEET / MINUTE		
SOURCE: MOD PACKAGE 88 - 7		

seal 2005

DATE:
DRAWN BY: JDR
CHECKED BY:
ENGINEER: CPF
CODE:
REVISION NUMBER:
REVISION DATE:

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research reactor facility

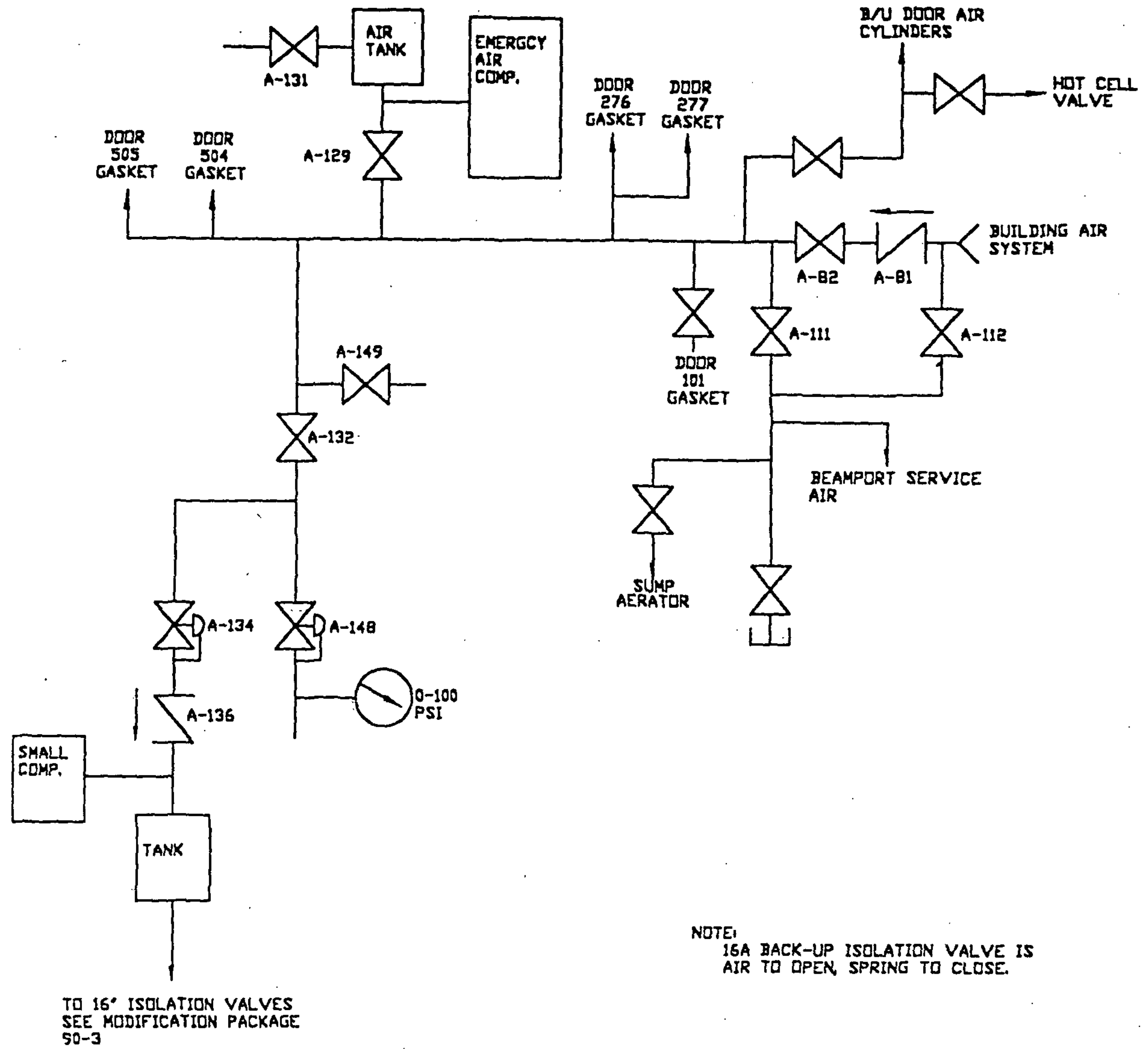
SCHEMATIC DIAGRAM OF LABORATORY AND CONTAINMENT BUILDINGS VENTILATION SYSTEM

MURR NUMBER: 1125

SHEET 1 of 3

FIGURE A.11

1998



NOTE:
16A BACK-UP ISOLATION VALVE IS AIR TO OPEN, SPRING TO CLOSE.

REV. NO.	DESCRIPTION	DR. BY	DATE
7	REMOVE MOD. PACK. 90-3, ADD MOD. PACK. 97-2	JDR	6/27/97
6	UPDATED ON CAD	JDR	5/29/97

VENTILATION AUTOMATIC VALVE AIR SUPPLY

FACILITY OPERATIONS
research reactor facility
UNIVERSITY OF MISSOURI-COLUMBIA

date 6/27/97
drawn by J.RILEY
checked by:
engineer
code
revision 7
rev.date 6/27/97

murr number 119

sheet 1 of 1

APPENDIX B
ANTI-SIPHON SYSTEM DESIGN ANALYSIS

B. ANTI-SIPHON SYSTEM DESIGN ANALYSIS

B.1 Introduction

As stated in the Loss of Coolant Analysis, Appendix E, the MURR anti-siphon system operates as a backup to various safety functions to insure that the core will not become uncovered. This appendix outlines the analysis of the present anti-siphon system. A proposed modification to retain any accident-produced gaseous fission products in a closed system is also discussed.

B.2 Analysis of the Present Anti-Siphon System

The following describes the model, mathematical analysis, and results of an engineering study (1) of the characteristics of the present siphon-break system for the Missouri University Research Reactor. A double rupture of both inlet and outlet coolant pipes was assumed to occur. The analysis demonstrates that a pipe opening to atmosphere of at least 3.5 inch diameter is required to insure that the core remains covered. A four inch pipe was installed for an added safety factor.

A double ended rupture on the inlet side has also been analyzed. The analysis indicates that the simultaneous double pipe rupture is the more severe case.

B.2.1 System Model

Figure B.1 shows the MURR primary coolant system. Dimensions are taken from reactor equipment as installed.

It is assumed that the pipe rupture will activate at least one of the two siphon-break valves making the pressure at the junction of the 12 inch riser and the siphon-break connection (point 2 of Figure B.1) equal to atmosphere instantaneously. It was assumed that the air-water interface entering the pipe at the upstream break remained intact.

Viscous resistance in both the siphon-break pipe and the main pipe was assumed to obey the Blasius equation for a smooth pipe. This is conservative in that the Blasius equation gives a low frictional resistance coefficient for Reynold's numbers above 10^6.

B.2.2 Mathematical Analysis

A piping system break causes unsteady flow immediately. The siphon-break prevents the upstream interface from moving through the pipe until the core becomes "dry." Viscous and turbulent flow losses resist siphoning but are not sufficient to stop flow. The siphon system in effect increases the pressure in the main-flow pipe at the high point of the downstream invert loop. This pressure increase is a function of the size of the siphon-break pipe. If the pressure at the high point is close to or greater than atmospheric, the air-water interface upstream of the core can be brought to rest before the core is uncovered.

Flow continuity must be maintained at the junction of the twelve-inch pipe and the vertical riser to which the siphon-break line is attached. Thus

$$V_3A_3 + V_1A_1 = V_2A_2 \tag{B.1}$$

where V_3, V_1 and V_2 are the velocities indicated in Figure B.1 and A_1, A_2 and A_3 are the corresponding areas. Also by continuity

$$V_AA_A = V_3A_3 \tag{B.2}$$

where V_A and A_A are the air velocity and the cross-sectional area of the siphon-break pipe.

The pressure P_3 inside the riser at the opening to the siphon-break pipe is given

$$P_3 = \frac{\rho_A V_A^{\ 2}}{2} + \Sigma K_A \frac{\rho_A V_A^{\ 2}}{2} + f_A \frac{L_A}{D_A} \frac{\rho_A V_A^{\ 2}}{2} \tag{B.3}$$

where:

ρ_A = Air density at ambient temperature

L_A = Length of siphon-break air line

D_A = Diameter of the air pipe

K_A = Energy loss coefficients for fittings

f_A = Friction coefficients for turbulent flow in a smooth pipe.

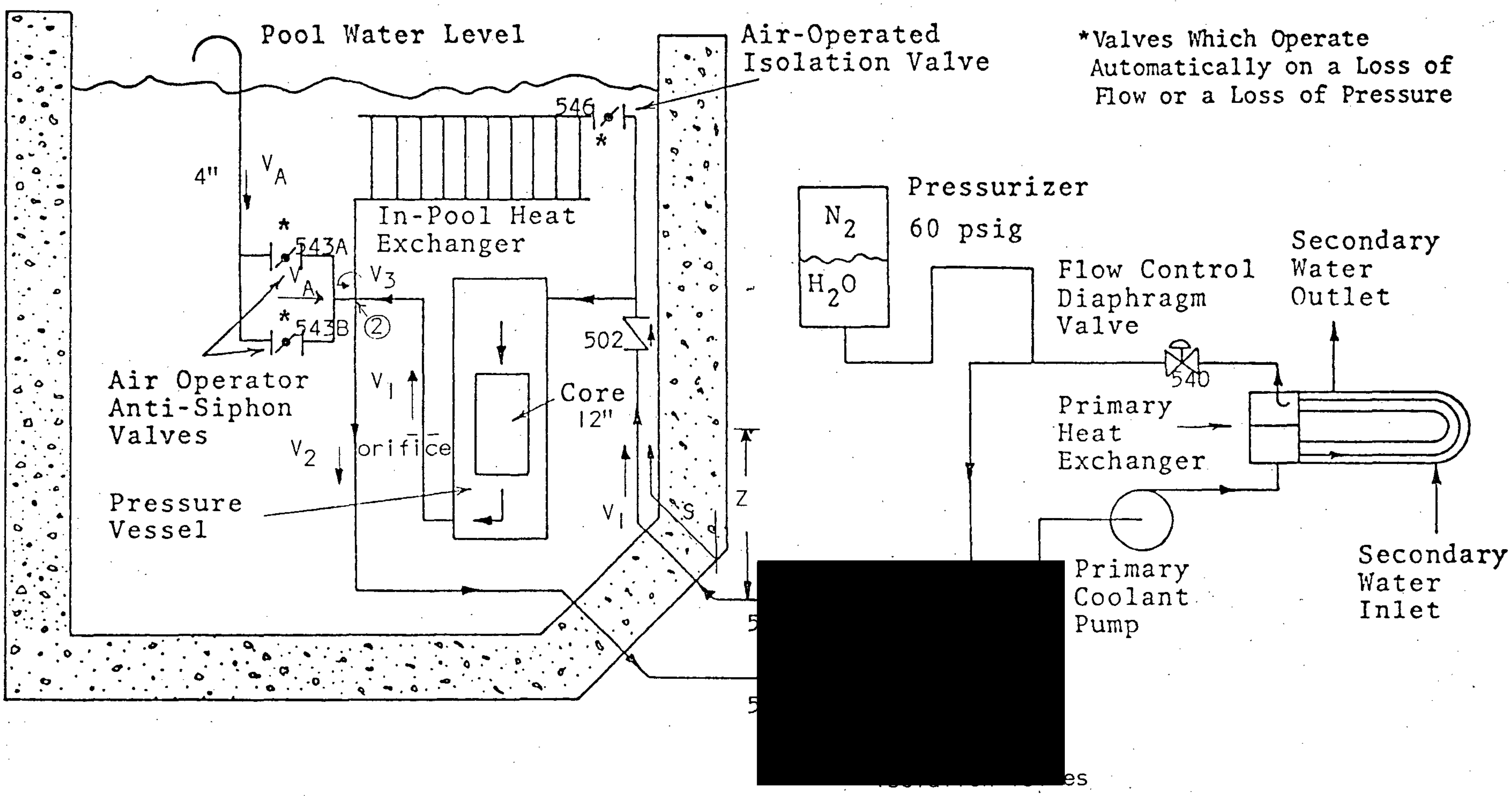


FIGURE B.1

SCHEMATIC DIAGRAM OF THE MURR PRIMARY COOLANT SYSTEM

Acceleration of the air column is neglected.

The unsteady state forms of the applicable Bernoulli equations for this configuration are

$$V_1 \rho (L_1 - Z)\frac{dV_1}{dS} = \frac{-\rho V_1^{\ 2}}{2}\left(\Sigma K_1 + \frac{f_1 L_1}{D_1}\right) + \rho g z + P_Z \tag{B.4}$$

$$V_1 \rho L_2 \frac{dV_2}{dS} = \frac{-\rho V_2^{\ 2}}{2}\left(\Sigma K_2 + \frac{f_2 L_2}{D_2}\right) - P_2 \tag{B.5}$$

Equation (B.4) refers to the portion of piping upstream from the siphon- break while equation (B.5) models the downstream section, and

- L_i = Pipe length
- D_i = Pipe diameter
- ρ = Water density
- Z = Elevation of the incoming air-water interface relative to datum level
- S = distance traveled by the incoming flow
- K_i = Energy loss coefficients for the i^{th} section of pipe.

The pressure at point 2 (see Figure B.1) can be written as

$$P_2 = P_3 + \gamma \Delta h_2 \tag{B.6}$$

where

- γ = Specific weight of water
- Δh_2 = Height of water in the riser leg to the air siphon line above point 2.

Equations B.1, B.2, B.3, and B.6 are combined with equations B.4 and B.5 to produce two nonlinear differential equations in V_1 and V_2.

$$\rho(L_1 - Z)\frac{dV_1}{dS} = \frac{-\rho V_1^{\ 2}}{2}\left(\Sigma K_1 + f_1 \frac{L_1}{D_1}\right) - \gamma(h_1 - Z) \tag{B.7}$$

$$+ \frac{\rho_A (V_2 - V_1)^2}{2}\left(\frac{A}{A_A}\right)^2 \left(1 + \frac{f_A L_A}{D_A} + \Sigma K_A\right)$$

$$\rho L_2 V_1 \frac{dV_2}{dS} = \rho \frac{-V_2^{\,2}}{2}\left(\Sigma K_2 + \frac{f_2 L_2}{D_2}\right) + \gamma \Delta h_2 \tag{B.8}$$

$$- \frac{\rho_A (V_2 - V_1)^2}{2}\left(\frac{A}{A_A}\right)^2\left(1 + \frac{f_A L_A}{D_A} + \Sigma K_A\right)$$

Numerical methods were utilized to solve these two equations simultaneously on a digital computer. The step method solution for V_1 and V_2 employed a truncated Taylor series.

$$V_{1_{n+1}} = V_{1_n} + \Delta S \frac{dV_1}{dS} + \frac{\Delta S^2}{2} \frac{d^2 V}{dS^2} \tag{B.9}$$

and

$$V_{2_{n+1}} = V_{2_n} + \Delta S \frac{dV_2}{dS} + \frac{\Delta S^2}{2} \frac{d^2 V_2}{dS_2} \tag{B.10}$$

The second derivative velocity terms were obtained by differentiating equations (B.7) and (B.8). The velocities V_1 and V_2 were initially set equal to the steady state value based on a nominal 3600 gpm flow. Equations (B.9) and (B.10) therefore describe the velocity change as the air-water interface advances a distance of ΔS.

B.2.3 Results

Results of the computer solution of equations B.9 and B.10 are shown in Figure B.2. Solutions were obtained for pipe sizes of 2 1/2, 3, and 3 1/2 inches. All velocities were normalized to the initial velocity. The vertical line $Z/L_0 = 0.193$ represents the top of the vertical pipe upstream of the core. Hence, if V_1 becomes zero before the interface reaches this line, the core will always remain covered by a minimum of six feet of water. The curves indicate clearly that a siphon-break air pipe diameter of 3 1/2 inches is the minimum required to insure that the core remains covered.

In order to determine if a single rupture occurring at the inlet would present a greater hazard, an analysis was run holding V_2 constant. The dashed curve for this case is also shown in Figure B.2. It is obvious that the double rupture is more severe.

B.3 Proposed Modification to the Anti-Siphon and Vent Tank System

In order to reduce the exposure that would result from the meltdown of four fuel plates (design basis accident), the present anti-siphon system will be modified such that any gaseous fission products will be retained in a closed volume. The modified system is illustrated in Figure B.3.

The primary component of the system is the 25 gallon gas retention tank. This tank will be a cylindrical aluminum tank with nominal dimensions of 16 inches in diameter and 30 inches in length. The tank will be located on the refueling level of the pool adjacent to the existing vent tank. The existing anti-siphon vent line will be cut off below the refuel level and reduced down to a 3/4" aluminum line which will terminate in the bottom of the gas retention tank. The vent line from the existing vent tank will be extended and discharge through filters to the containment exhaust system.

The existing level controller 965 which initiates an alarm and rod run-in for a water level greater than six inches above the anti-siphon valves is to be removed and replaced with a level controller operated by a float suspended from a long cable in an aluminum dry well. This modification puts the controller switch housing out of the pool and thus facilitates maintenance and eliminates the possibility of flooding the switch with pool water. The level controller dry well will be located against the pool wall away from the core and coolant piping to prevent radiation streaming.

The pressure within the system will be maintained by manually introducing air from the facility air supply and by manually venting the system through particulate and charcoal filters. The retention tank is also equipped with a relief valve (set at 100 psig) to provide over-pressure protection for the system. The relief valve exhausts through the particulate and charcoal filters.

As indicated above, the system pressure is maintained manually with air and vent valves located on the reactor bridge area. The pressure on the system is indicated on a pressure gauge located near these valves. The system will also have two pressure switches which will activate an alarm on the Annunciator Panel indicating a high (45 psig) or low (30 psig) pressure.

The construction of this system will be subject to the same design criteria and test requirements as the primary coolant system.

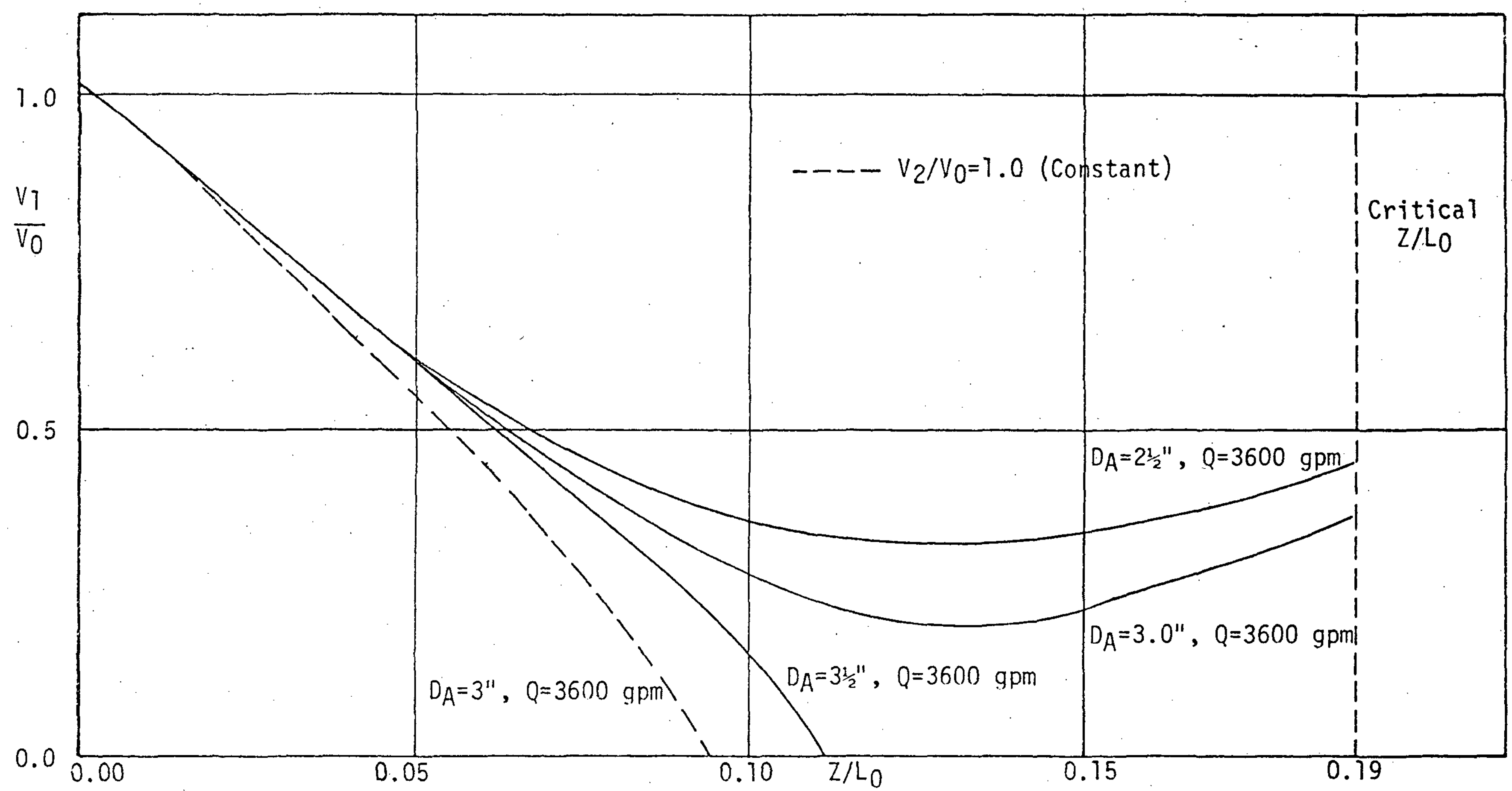
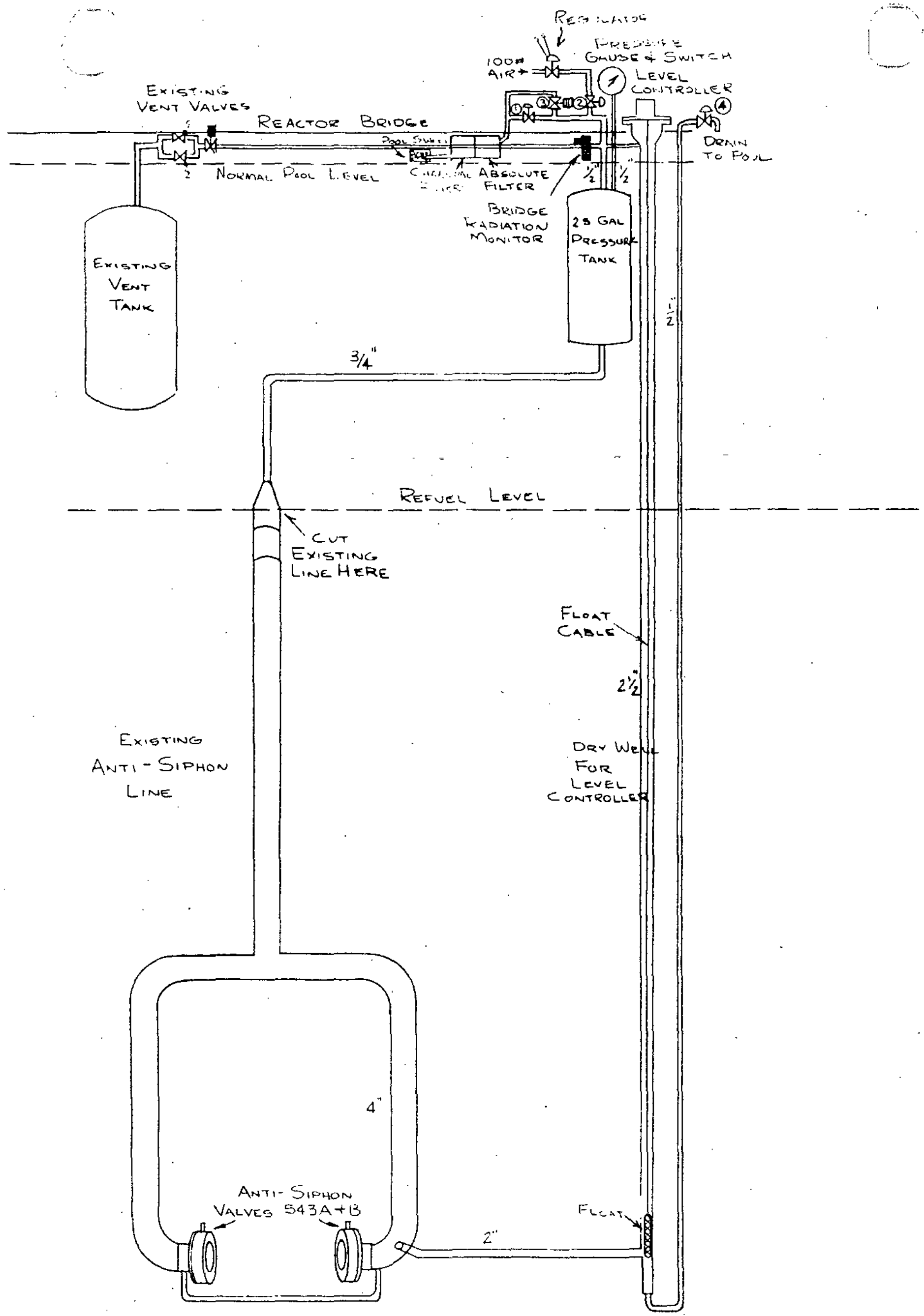


FIGURE B.2
INFLOW VELOCITY VERSUS ELEVATION OF THE INCOMING INTERFACE.
PIPING CONFIGURATION AS GIVEN BY THE SOLID LINES IN THE PREVIOUS FIGURE.



Valve Legend
① Manual Operated Vent Valve
② Manual Operated Air Supply Valve
③ Relief Valve (Set Pressure 100 psig)
④ Manual Operated Drain Valve

Notes:
1. All New Piping Is To Be Sch. 40 Alum (6061-T6) And Will Be Welded.
2. Operating Pressure Of The System Will Be 27-45 psig
3. System Will Be Hydrostatically Tested At 100 psig

Changed Capacity Of Press. Tank
1 Vickie Nichols 9-27-78
Revisions

Title
Pressurized Vent Line For Primary Coolant System

RESEARCH REACTOR FACILITY
UNIVERSITY OF MISSOURI

DRAWN BY Vickie Nichols
APPROVED
SHEET NO. 1 OF 1
DWG. NO. 402

Figure B-3

Two main criteria were considered in the design of the system. First, the entire system must contain sufficient air to break the siphon should a double-ended pipe rupture occur. Second, the maximum pressure in the system will be maintained below the operating core pressure to minimize the possibility of air leakage into the primary coolant due to leaking anti-siphon valves. Analysis based on these criteria imposed the following limits on the anti-siphon system pressure.

Maximum System Pressure = 45 psig
Minimum System Pressure = 27 psig

The method used to determine the minimum required pressure will now be outlined. As discussed earlier in this Appendix, if atmospheric pressure can be maintained at the junction of the riser leg (leading to the natural convection flange) and the anti-siphon valves, the siphon can be broken in the event a double-ended pipe rupture. In other words, if the initial anti-siphon system pressure is such that all water can be displaced in the outlet leg to the level of the outlet primary isolation valve (507A) and atmospheric pressure maintained, the core will remain covered with a minimum of six feet of water.

The outlet leg consists of 14 1/2 feet of twelve inch pipe. The volume of water displaced is therefore 81.3 gallons. The proposed anti-siphon system volume is 44.3 gallons. Therefore, atmospheric pressure can be maintained with an initial system pressure of 27 psig.

B.4 <u>System Operation</u>

During normal operation the system will be maintained dry and pressurized to a pressure of 30 to 45 psig. The system pressure will be checked and recorded every four hours as part of the facility routine patrol. The operator making the check will add or vent air as necessary to keep the system near the middle of the control band. In the event of a high or low pressure alarm immediate action will be taken to add or vent air to clear the alarm. If a system leak or other malfunction prohibits the maintenance of at least 27 psig on the system, the reactor will be shutdown until the malfunction has been corrected.

The low point drain line and valve (#4 in Figure B.30) will be used to blow out excess water during plant starting.

B.5 References

1. Cassidy, John J., Engineering Report - Characteristics of a Siphon Break for the Research Reactor at the University of Missouri, Internal Communication Report, November, 1964.

APPENDIX C
DESIGN BASIS ACCIDENT

C. DESIGN BASIS ACCIDENT

C.1 Introduction

Many conceivable accidents have been considered that might occur in conjunction with the operation of the University of Missouri Research Reactor (MURR). In all cases, safety systems have been designed in such a manner that the likelihood of an accident occurring in which a significant amount of fission products is released has been essentially eliminated. The safety systems take the form of automatic reactor shutdown circuits and process system design to ensure with redundancy that the reactor will shutdown upon a significant deviation from normal operating conditions. In addition, the reactor is housed in a containment building providing further protection against a significant release of radioactive byproducts to the environment.

The design basis accident (DBA) is selected to postulate conditions which lead to consequences worse than those resulting from any other anticipated accident.

C.2 The Design Basis Accident

As a design basis accident it is assumed that an accident condition has led to the melting of four fuel plates in the reactor core. Because this accident is considered worse than any anticipated accident, the conditions that lead to this circumstance are immaterial to the analysis. The DBA may be postulated to result from partial flow blockage to the fuel. However, the coolant strainer, the fuel end fitting and the preoperational inspection of the pressure vessel and core following any fuel handling, all prevent an accident of this type. In addition, it has been shown that a 75% blockage of flow to the hot channel is insufficient to cause a clad failure (1).

It is assumed that the fuel plate melting occurs in the peak power region of the core; that is, four plates are chosen which represent the area of peak fission product inventory.

C.3 Consequences of the Design Basis Accident

The DBA postulates a partial fuel melt and release of fission products to the primary coolant system. If the DBA occurs, the primary coolant system will continue in operation and the fission products will quickly be dispersed in the system.

With the redesign of the anti-siphon system there is no longer a path for the fission products to escape to the containment (Appendix B). Particulate activity will remain in the coolant. Gaseous activity as it comes out of solution will collect in the vent tank system and be held there. In the past it was assumed that any gaseous activity in the primary system could escape to the containment building through the anti-siphon valves. With the modification proposed, the anti-siphon valves will perform their safety function but the gaseous activity cannot escape.

With the proposed modification to the anti-siphon system the primary relief valves and the pressurizer remain as the only avenues for the release of significant quantities of fission products to the environment.

The DBA will result in a negligible release of energy to the primary coolant system and thus the introduction of pressure surges which would lift the relief valves is not considered credible. The pressurizer is an isolated system and the water in this system is not subject to mixing with the primary coolant system because the DBA will not cause significant pressure surges.

Any significant gaseous activity which is entrapped in either the vent tank or the anti-siphon pressure tank will cause a reactor scram and building isolation by action of the pool surface radiation monitor. The location of these tanks under the pool surface precludes significant dose to reactor staff, visitors or researchers since shielding is provided by the water and the biological shield.

Fission products entrapped in the primary coolant system can be removed by the reactor demineralizer system. The cleanup procedure would be undertaken under closely monitored and controlled conditions.

Although a closed loop, the primary system does experience leakage. This leakage has been determined to be not more than eight gallons per week. Based upon this leakage, a radiation exposure can be calculated as follows.

C.4 Analysis

C.4.1 Fission Product Release

The four fuel plates at the peak flux position contain 78.58 grams of U-235. Considering the total core inventory of 6.2 Kg of U-235, 1.27% of the core melts. For purposes of calculation, 1.3% meltdown shall be assumed.

The release of iodine constitutes the major fission product hazard. For operation at 10 MW for 80 days with 6.2 Kg of U-235 the following radioiodine activities will be present in the core (2).

^{131}I - 0.25 x 10^6 Curies
^{132}I - 0.38 x 10^6 Curies
^{133}I - 0.54 x 10^6 Curies
^{134}I - 0.63 x 10^6 Curies
^{135}I - 0.55 x 10^6 Curies

A power peaking factor of 1.6 is taken into consideration by increasing the fuel meltdown from 1.3 to 2.08%. A value of 70% release of the fission products from the fuel is assumed in calculating the fission product inventory in the primary coolant system.

Realistically it is difficult to postulate even a small fraction of the gaseous activity escaping due primarily to the properties of iodine, since it is readily absorbed in solution or deposited on materials.

Proceeding on a worst case basis, we assume that during a normal weeks operation eight gallons (1.5 x 10^{-3} gallons per minute) escapes from the primary system. With 2000 gallons of water in the primary coolant system the activity released in two minutes is as follows:

^{131}I - 5.5 mCi
^{132}I - 8.3 mCi
^{133}I - 11.8 mCi
^{134}I - 13.8 mCi
^{135}I - 12.1 mCi

This activity uniformly distributed to the reactor containment would result in the following concentrations in the building:

^{131}I - 0.86 $\mu Ci/m^3$
^{132}I - 1.30 $\mu Ci/m^3$
^{133}I - 1.85 $\mu Ci/m^3$
^{134}I - 2.17 $\mu Ci/m^3$
^{135}I - 1.90 $\mu Ci/m^3$

where the minimum building volume is 225,000 cubic feet.

The object of this calculation is to present a worst case condition for a person who remains two minutes in the containment building following the accident. Tests have shown that two minutes provides ample time for complete evacuation of the building.

The above activity is assumed to be released instantaneously. Based on a uniform release over a two minute period the concentration can be further reduced by a factor of two.

C.4.2 Exposure Calculations

From the fission product radioiodine released it is possible to calculate the exposure to an individual in the containment building immediately following the accident.

The internal exposure to a body organ of mass m (grams) in which a fraction (f_a) of the inhaled material (I (μCi) resides for an effective half life of T (days) is (3):

$$D = 73.8\, f_a\, I\, E_i\, (RBE)\, NT/m \text{ rem}$$

where
E_i = effective energy (Mev/dis)
$\underline{I}$ = 5×10^{-4} Qt (μCi)
Q = concentration ($\mu Ci/m^3$)
t = time in cloud (seconds)
T = effective half-life (days)
f_a = 0.15
m = 20 grams for the thyroid
N = deposition factor

i	EiN	T
^{131}I	0.23	7.5
^{132}I	0.66	0.1
^{133}I	0.48	0.85
^{134}I	0.80	0.04
^{135}I	0.41	0.28

The calculated internal dose to the thyroid from a two minute exposure to the iodine concentrations is therefore:

$D\ (^{131}I) = 24.6$ mrem

$D\ (^{132}I) = 1.4$ mrem

$D\ (^{133}I) = 12.5$ mrem

$D\ (^{134}I) = 1.6$ mrem

$D\ (^{135}I) = 3.6$ mrem

D (total) = 43.7 mrem

The external exposure from a large cloud may be estimated not to exceed (3):

$$D = 2.6 \times 10^{-7} E_s\, Q\, t \text{ (rem)}$$

$\overline{Q}$ = average concentration ($\mu Ci/m^3$)

E_s = effective particle energy (Mev)/dis)

t = time (seconds)

On the basis of radioiodine the total external dose obtained in two minutes in the containment is:

$D\ (^{131}I) = 0.007$ mrem

$D\ (^{132}I) = 0.010$ mrem

$D\ (^{133}I) = 0.027$ mrem

$D\ (^{134}I) = 0.082$ mrem

$D\ (^{135}I) = 0.067$ mrem

D (total) = 0.188 mrem

The external dose obtained in two minutes in the containment from all other radionuclides is also negligible.

The direct exposure due to the fission products leaked to the pool water would be in actuality the prime source of radiation exposure. However, this activity is further diluted by the 20,000 gallons of water in the pool. The staff, visitors, and researchers are not normally in the immediate vicinity of the pool surface and if so would be there only a few seconds following the isolation alarm set off by a dose rate of only 10 mr/hr as seen by the pool surface radiation monitor. The exposure would therefore be very limited.

C.4.3 Internal Dose from Radioiodine Outside of Containment

It is to be shown that a release of gaseous activity in the containment building will not produce a hazard to the health and safety of the public.

To consider the worst situation and for ease of calculation it shall be assumed that all the radioiodine produced in the DBA is released to containment. No attempt is made to explain how the release may occur, as no mechanism exists that reasonably leads to this situation. This analysis shows therefore, even if the worst conceivable situation occurs, what the circumstances are outside the building.

The analysis is made at a distance of 500 feet from the containment. The dose from radioiodine is calculated as this source of radiation has been shown to be the controlling factor over the external whole body dose from fission products (2).

The total core inventory at time of release is:

^{131}I - 0.25×10^6 Ci
^{132}I - 0.38×10^6 Ci
^{133}I - 0.54×10^6 Ci
^{134}I - 0.63×10^6 Ci
^{135}I - 0.55×10^6 Ci

Iodine concentration in the building as a result of a 2.08% meltdown, a 70% release and dilution in the building volume of 225,000 cubic feet:

^{131}I - 1.62×10^4 $\mu Ci/ft^3$

^{132}I - 2.46×10^4 $\mu Ci/ft^3$

^{133}I - 3.50×10^4 $\mu Ci/ft^3$

^{134}I - 4.08×10^4 $\mu Ci/ft^3$

^{135}I - 3.56×10^4 $\mu Ci/ft^3$

The behavior of radioiodine was studied in the containment mockup facility at Oak Ridge National Laboratory. From these experiments it was shown that up to 75% of iodine released will be deposited in the containment vessel.

The design basis accident will not cause an increase in pressure inside the reactor containment vessel (C.4.4). Any leakage from the building would occur as a result of normal changes in atmospheric pressure and pressure equilibrium between the containment and the atmosphere. Over a period of hours it would be reasonable to expect a possible barometric pressure change of 0.7 inches of mercury. Although the pressure in the building would not lag the change outside the building, over a long period of time it is conceivable that a 0.7" Hg (.33 psi) pressure differential could exist between the building and outside. The air leakage from the building can be expressed as (4):

$$L = 160\, e^{-0.031t}\ (ft^3/hr)$$

under conditions of a .33 psi overpressure.

Considering each cubic foot of air released as containing the same amount of radioactivity as a cubic foot of containment building air, the activity leaking from the building in microcuries per second is (75% disposition):

Iodine Isotope	Release Rate (μCi/sec)
131	$180 \exp(-8.6 \times 10^{-6}\, t)$
132	$272 \exp(-8.6 \times 10^{-6}\, t)$
133	$388 \exp(-8.6 \times 10^{-6}\, t)$
134	$452 \exp(-8.6 \times 10^{-6}\, t)$
135	$395 \exp(-8.6 \times 10^{-6}\, t)$

where t is now in seconds.

By diffusion formulas the concentration of the radioactive cloud downwind may be calculated and the dose to the thyroid determined. Corrections are made to account for the building not being a point source and for wind variability.

The continually changing atmospheric conditions make it impossible to determine a dose to the thyroid that could be expected on any particular day. It is through the dispersion coefficients that atmospheric conditions are taken into account. These dispersion coefficients vary according to a rather general description of the type of weather. For the purpose of obtaining the dose under the worst condition, stable atmospheric conditions are chosen.

The generalized Gaussian plume model shall be used to calculate the diffusion and correction factors applied to correct for building effect, wind variability and decay.

The generalized Gaussian plume formula is:

$$X = \frac{Q}{\pi\, \sigma_y \sigma_z}\, \bar{\mu}\, \exp\left[-\frac{1}{2}\left(\frac{y^2}{\sigma_y^2}\right) + \frac{h^2}{\sigma_z^2}\right]$$

where X = concentration in curies per cubic meter

Q = source strength (curies/sec)

$\bar{\mu}$ = mean wind speed in meters per second

y = crosswind distance from the plume axis in meters

h = height of the source above ground in meters = 15m

σ_y, σ_z = dispersion coefficients in square meters

The application of this formula is made by relating the dispersion coefficients to Sutton's dispersion model.

$$\sigma_y^2 = \frac{1}{2} C_y^2\, x^{(2-n)}$$

$$\sigma_z^2 = \frac{1}{2} C_z^2\, x^{(2-n)}$$

where C_y, C_z = diffusion coefficients (meters$^{n/2}$)

n = atmospheric stability parameter

x = downwind distance from the source in meters

The values of the diffusion coefficients, stability parameters and wind speed are selected for the neutral condition:

$\bar{\mu}$ = 2 m/sec

C_y = 0.45

C_z = 0.80

n = 0.15

To correct for building effect the following formula is used (6):

$$C_B = \frac{\pi C_z C_y x^{2-n}}{\pi C_z C_y x^{2-n} + A/2}$$

where A = projected cross-sectional area = 465 m^2.

The formula for wind variability is (6):

$$C_W = \left[1 + \frac{x^n \tan^2(\pi/16)}{C_y^2 \ln 1000}\right]^{-1/2}$$

The source term Q is time dependent both on the basis of release rate from the building and radioactive decay.

The value of Q is found by considering release rate and decay according to

$$\begin{aligned} Q(t) &= A_i \exp(-8.6 \times 10^{-6}\, t) \bullet \exp(-\lambda_i t) \\ &= A_i \exp[-(8.6 \times 10^{-6} + \lambda_i)t] \end{aligned}$$

where A_i is the release rate constant of the i^{th} isotope of iodine and λ_i is the appropriate decay constant in sec^{-1}.

The formula for calculating the radiation dose to the thyroid is (5):

$$D = C\, T_{eff}\, (73.8\, \overline{E} + 33.1 \times 10^{-3} K\overline{g}) \text{ rads}$$

With an RBE factor of 1 for β and γ the above dose may be stated in rem.

C = concentration in mCi per gram

T_{eff} = effective half-life (days)

$\overline{E}$ = mean beta energy

K = a factor giving the exposure dose rate from a gamma emitting point source of 1 millicurie activity

$\overline{g}$ = average geometrical factor for the organ

$$C = \frac{(X')(BR)(T)}{m} \cdot R.F$$

where BR = breathing rate = 20 liters/min = 3.3×10^{-4} m^3/sec
T = time in cloud
m = thyroid weight = 20 grams
R.F. = retention factor = 0.23

X' is found by integrating Q over the period of time in the cloud and time averaging.

The value of C in the thyroid dose formula may now be evaluated for each iodine isotope i.

$$C_i = \frac{(B.R)(R.F)A_i}{m(8.6 \times 10^{-6} + \lambda_i)} F\, C_B\, C_W \left[1 - \exp - (8.6 \times 10^{-6} + l_i)\, t\right]$$

$$F = \frac{Q}{X} \text{ in plume formula.}$$

Values for the dose equation are (5):

i	E	T_{eff}	λ	K	g
131	0.919	7.52	9.93×10^{-7}	2.3	17.5
132	0.398	0.10	8.52×10^{-5}	6.0	17.5
133	0.407	0.85	9.25×10^{-6}	3.2	17.5
134	0.833	0.04	2.20×10^{-4}	11.15	17.5
135	0.317	0.28	2.88×10^{-5}	9.0	17.5

The above formula is obtained by integrating over an anticipated time of occupancy in the radioactive cloud. In order to obtain the dose assuming constant occupancy in the cloud the integration of the equation is made from limits 0 to ∞. The exponential term on the right will go to zero and the time integrated concentration then will be:

$$C_i = \frac{(B.R)(R.F)\, A_i F\, C_B\, C_W}{m(8.6 \times 10^{-6} + \lambda_i)}$$

The dose is now calculated for each iodine isotope:

Isotope		Infinite Dose
131		300 mrem
132		< 1
133		19
134		< 1
135		3
	Total	322 mrem

C.4.4 Energy Release

The potential energy release from the melting of four fuel plates could occur as a possible metal water reaction. While hydrogen would be formed it is unlikely that in the water environment a hydrogen deflagration reaction would occur.

The amount of material which would be involved in a metal water reaction under the conditions of four fuel plates melting is not predictable as the amount is dependent upon many conditions. For purposes of calculation it shall be assumed that all cladding exposed in the area is involved in the reaction.

The reactor core contains a total of 33.56 kg of aluminum. Of this, 1.3% or 0.436 kg is assumed to react according to the equation:

$$Al + n\,H_2O \rightarrow Al\,O_n + n\,H_2 + \text{heat}$$

The energy release is 18 MW sec/kg Al or a total energy of

$$7.9\ \text{MW sec} = 7.5 \times 10^3\ \text{BTU}$$

This amount of heat would be easily carried away by transfer to adjacent fuel elements and to the coolant water in the core. The only conditions in which the DBA would be expected to occur is with the reactor operating at 10 MW and coolant flow to the balance of the core. It is to be expected that steam would be formed in the vicinity of the molten area which would serve to assist in carrying the heat away.

C.5 Discussion

Generally, the most severe condition which is analyzed with regard to a reactor accident is either the loss of coolant or loss of flow during reactor operation. Both of these accidents have been analyzed with respect to the operation of MURR and the results show no core damage. There are no accidents other than the DBA which would result in a release of fission products from the reactor fuel. Even if such an event would occur, the primary coolant anti-siphon and vent system is to be modified such that any activity released would be contained in the primary system.

System design and operational procedures prevent the likelihood of any foreign material in the core of the reactor which would cause a partial flow blockage. Calculations have been performed which indicate that even partial flow blockage to a fuel element will not result in clad melting (1). Considerable margin of safety has been designed into the system in this regard. The selection of the melting of four fuel plates in the reactor as the design basis accident thus represents a condition worse than any anticipated accident. It is not expected that any fission products would reach the containment building. Considering the results of analysis which show no core damage in the event of an anticipated accident and the modification of the anti-siphon and vent system, there is no radiation hazard to personnel in the reactor containment.

In the application for license to operate the reactor at 10 MW, it is proposed to move the exclusion area boundary to the outside wall of the facility building. The calculations of the hazard associated with the release of fission products in the containment building show that no hazard will exist in containment. Calculations have been made which show no significant radiation problem at 500 feet from the containment building. A model is not available which can predict radiation dose closer to the building. However, it can be stated that the direct radiation at the new exclusion boundary will be no greater than that calculated for the containment building.

With the radiation monitoring system and health physics staff available at the facility to take appropriate action, it is reasonable to move the exclusion area to the outer walls of the facility. If circumstances so warrant, facility staff and the University police can be provided on short notice to move personnel farther from the building.

C.6 References

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APPENDIX D

LOSS OF FLOW ACCIDENT

D. LOSS OF FLOW ACCIDENT

D.1 Introduction

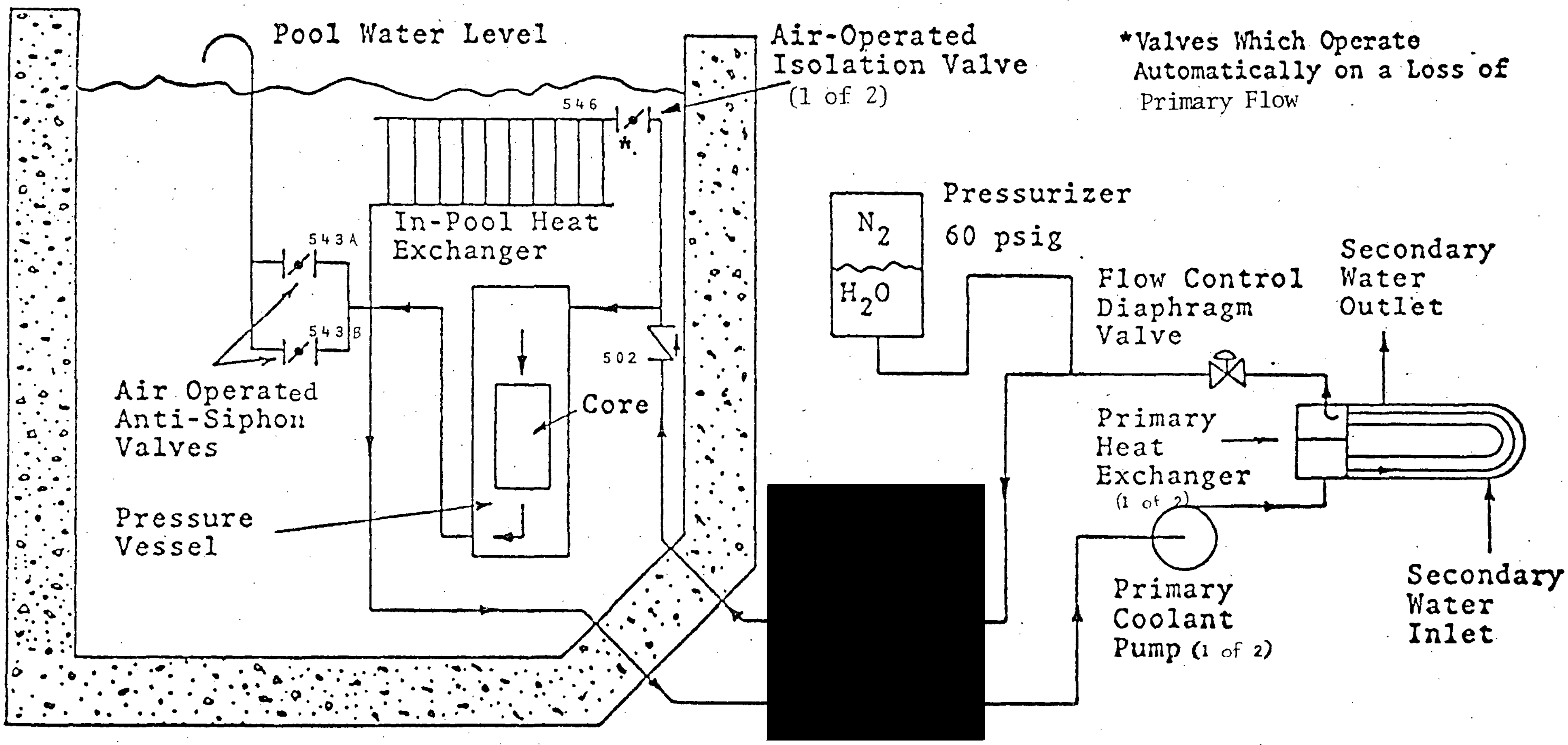
A loss of flow (LOF) accident with no scram was assumed to be the maximum credible accident for MURR in the initial submittal of the Hazards Summary (1). A loss of flow accident can be initiated by one or a combination of many anomalies, such as:

(a) Loss of facility power (or pump power).
(b) Inadvertent closure of loop isolation valve/s.
(c) Locked rotor in a pump.
(d) Failure of a pump coupling.

Any of these anomalies is considered to be possible for the MURR plant, but a loss of flow (by any means) without an accompanying scram is no longer considered credible because of the redundancy in the protective circuit. Since the MURR is a downflow reactor, the LOF accident presents a potential hazard because of the flow stagnation and reversal following the LOF accident.

A schematic diagram of the MURR primary coolant system is shown in Figure D.1. It is noted that for 10 MW operation the system will contain two pumps and two heat exchangers operating in parallel. For simplicity only one pump and one heat exchanger are shown in the figure. There is also another parallel air-operated in-pool heat exchanger isolation valve.

The four types of LOF accidents listed above were analyzed in this study, but only the results of the worst case accident (b) will be discussed in this report. The LOF accident (a) due to a loss of pumping power was also analyzed in detail, but will not be discussed in this report because it was found to be a less serious accident from the point of reactor safety. Accidents (c) and (d) above will not result in a total loss of flow. These accidents affect only one of the pumps and the final flow will be approximately one-half of the initial flow. This reduced flow will result in a scram from all five detectors and no hazard exists.



D-3

FIGURE D.1

1994 SIMPLIFIED SCHEMATIC DIAGRAM OF THE MURR PRIMARY COOLANT SYSTEM

"(NOTE: This schematic does not include system changes made since 1974.)" 1994

1994

D.2 Description of the Accidents Analyzed

The LOF accident which is initiated by a failure of the main loop isolation valves can begin by some anomaly such as a break in the air line to the isolation valve air operators, a failure of the control solenoid valves, or a loss of power to the control circuit. The air pressure bleeds off the valve operator in about three seconds and the valves begin to close. This is effectively the beginning of the accident because the system operates normally until the valves begin to move.

As the isolation valve(s) begin to close, the pumps automatically stop and primary flow begins to drop rapidly. When the flow rate has decreased to 85% of its initial value the reactor scrams, and the isolation valves to the in-pool heat exchanger open. After 6.5 seconds, the isolation valves are fully closed reducing the primary coolant system to the in-pool loop. The momentum of the coolant maintains the downward flow through the core until the buoyancy and friction forces stop and reverse the flow.

A scram signal is initiated by either of the isolation valves leaving their full open position. This scram function was assumed to have failed for this accident and the reactor did not scram until the low flow scram was initiated.

D.3 Method of Analysis

The above accident was evaluated using a model developed by Gagliardo (2) for predicting the flow coastdown in a downflow reactor loop. This model predicts the flow through the core as a function of time following the accident. The code determines the flow coastdown to stagnation and the subsequent reversed flow which is established and maintained by the buoyancy forces. The model also calculates the bulk coolant temperatures throughout the loop.

This model is the downflow extension of that developed by Burgreen (3) for a reactor loop with upward flow through the core. The model is sufficiently general to include: temperature dependent coolant parameters, time dependent flow restrictions, heat input from an operating reactor with temperature feedback (or shutdown decay heat), parallel pumps and heat exchangers, and multiple modes of accident initiation.

The output of the above flow coastdown model was used as input data to the thermal analysis code PARET (4). This code was set up to calculate the heatup in the hot channel and in the average channel for the input conditions. Use of the PARET code is outlined in more detail in a subsequent section of this appendix.

D.4 Results of Coastdown Analysis

The LOF accident which is initiated by the failure of the loop isolation valve was analyzed by the computer codes DAMOKE and PARET. In the description of the accident above, it was noted that so far as the reactor is concerned the accident does not begin until the valve begins to move. The analysis has shown, however, that the valve closure has a negligible effect on the flow except during the final 2.5 seconds of closure. This fact was shown analytically and confirmed experimentally with the reactor shutdown and normal operating flow conditions established. The above discovery resulted in a shifting of the zero time for the accident from the onset of valve movement to the beginning of the flow reduction. With this new definition of zero time, it is found that the flow rapidly decreases to stagnation in 2.5 seconds as shown in Figure D.2. Flow reversal begins almost immediately and is shown by the curve in Figure D.3.

During the flow stagnation period and early in the reverse flow period there is essentially no movement of coolant through the core and thus a slug of relatively hot water is developed in the core. As this hot slug of water moves up to the in-pool heat exchanger piping, the flow rate is accelerated when the hot water is in the vertical pipe leading to the heat exchanger and then rapidly decelerated when the hot water flows down through the heat exchanger tubes. This fact accounts for the large peak and valley in the flow curve.

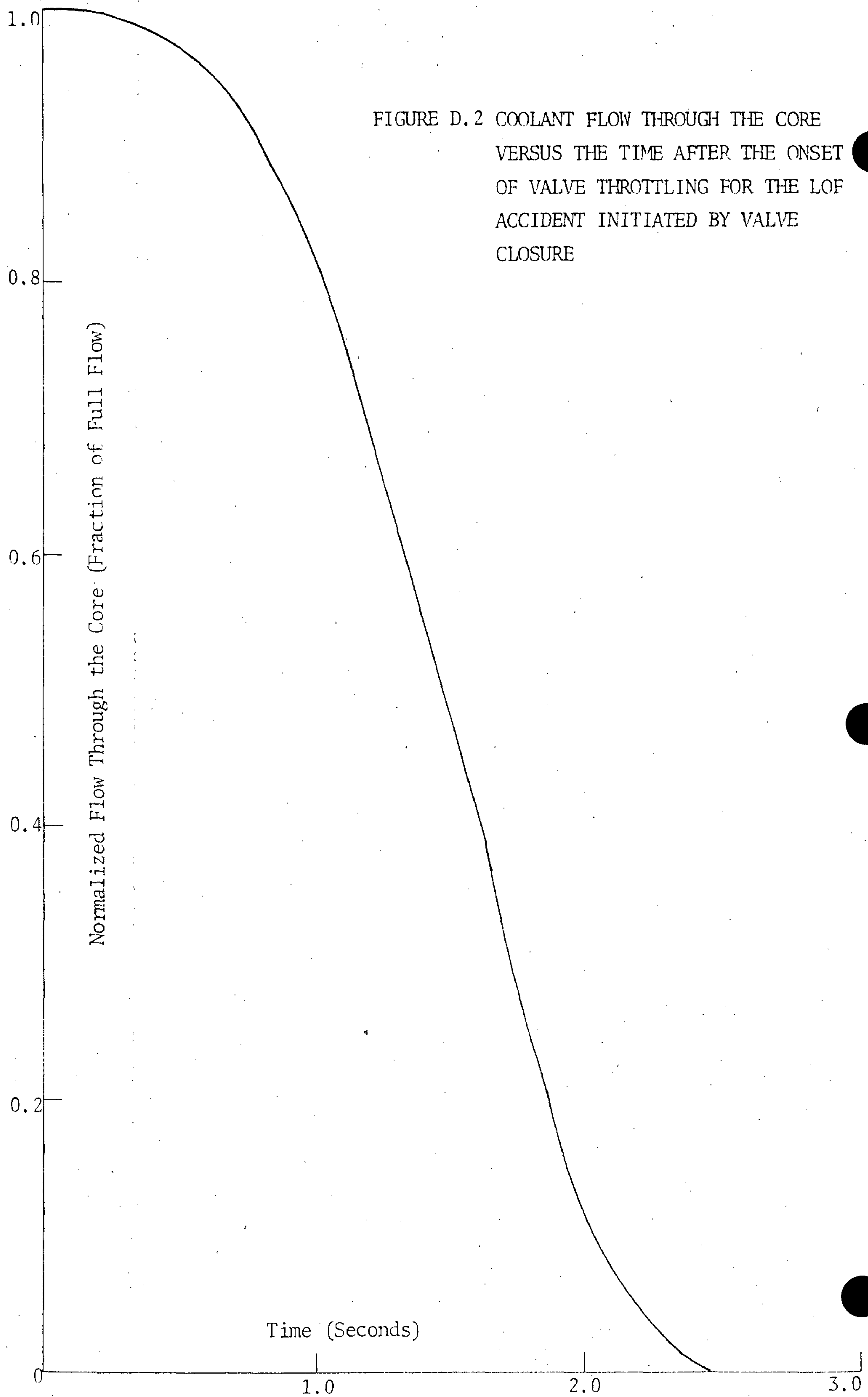


FIGURE D.2 COOLANT FLOW THROUGH THE CORE VERSUS THE TIME AFTER THE ONSET OF VALVE THROTTLING FOR THE LOF ACCIDENT INITIATED BY VALVE CLOSURE

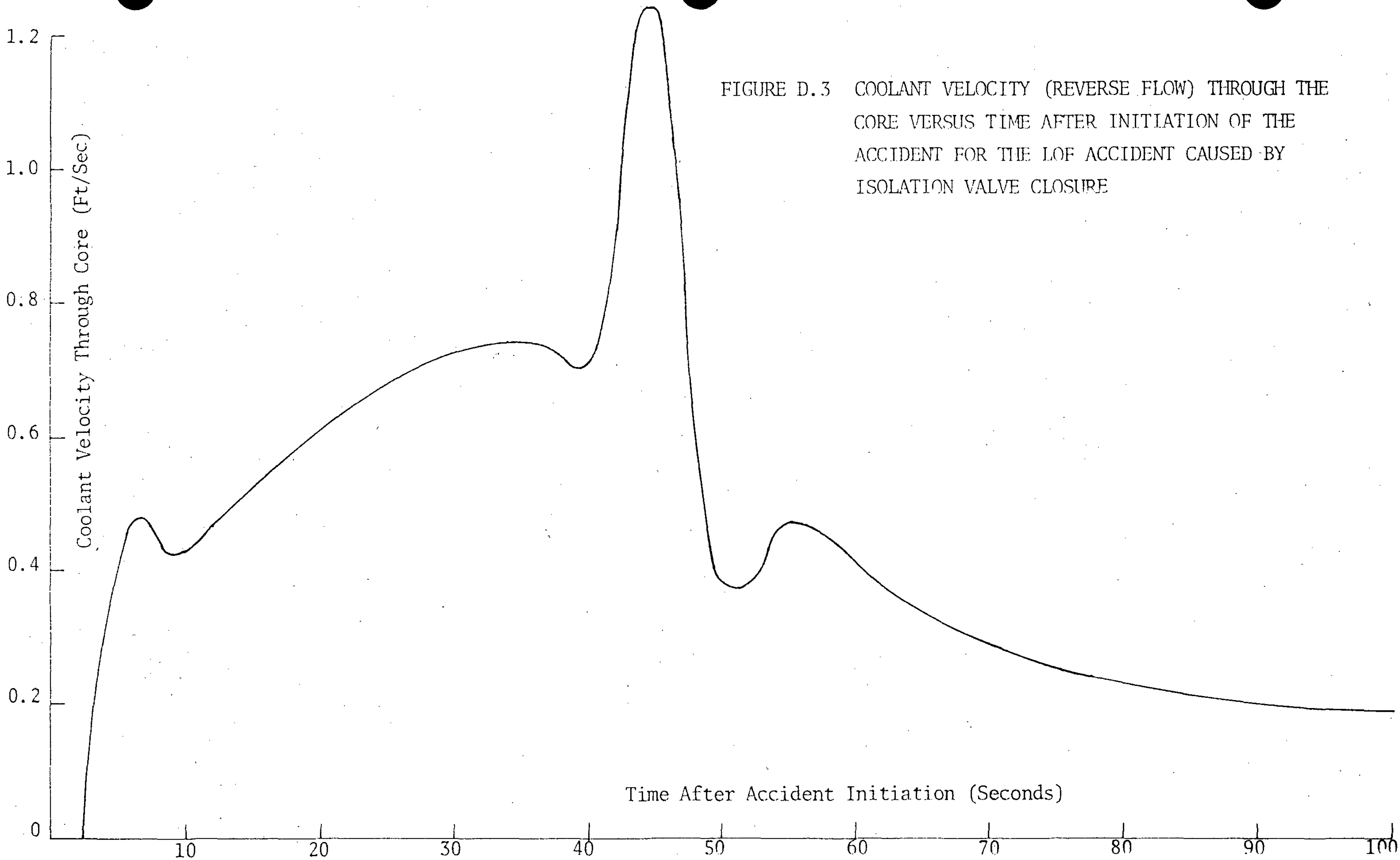


FIGURE D.3 COOLANT VELOCITY (REVERSE FLOW) THROUGH THE CORE VERSUS TIME AFTER INITIATION OF THE ACCIDENT FOR THE LOF ACCIDENT CAUSED BY ISOLATION VALVE CLOSURE

D.5 Input to PARET

Table D.1 lists the conservative assumptions made for the reactor system when the transient starts.

Table D.1

	Conservative Assumption	Normal Condition
Reactor Power	11 MW	10 MW
Coolant Inlet Temperature	155°F	140°F
Coolant Inlet Flow Rate	3,000 gpm	3,600 gpm
Pool Temperature	120°F	95°F
Pressurizer Pressure	50 psig	70 psig

The PARET model and general description of MURR system is presented in Appendix E dealing with the loss of coolant accident. Presented here is additional description needed for the case analyzed in this appendix.

Two forced flow channels were chosen. Channel one is designated to be the average channel and channel two is the hot channel as determined from the radial flux profile of the core (Figure D.4). Eleven axial nodes and three radial nodes were utilized. Slab geometry was specified to describe the plate type of fuel elements.

It was assumed that following the loss of flow accident, the core was depressurized through the siphon break system to a pressure of 24.2 psia (atmospheric pressure + 22 feet 9 inches of water). The above pressure was input as system pressure during the transient to the PARET code. The coolant inlet temperature to the core was chosen to be 178°F, which is the inlet temperature of the reversed flow due to natural circulation and was assumed as inlet temperature through the whole transient period.

Approximately one second after initiation of the transient, coolant flow rate decreases to 80% of normal. This would initiate a reactor scram. The reactor power and flow rate are shown in Tables D.2 and D.3.

D.6 PARET Results and Discussion

The output from PARET includes as a function of time and at each axial node point: coolant temperature, surface heat flux, heat transfer coefficient, burnout ratio, coolant regime, DNB surface temperature and radial temperature profile. The reactor power, coolant flow rate, and maximum temperature of the fuel centerline in the hot channel are shown in Figure D.5. Maximum clad surface temperature is shown in Figure D.6. The maximum clad surface temperature of 282°F is substantially below the DNB surface temperature of 295°F.

The highest temperature of the fuel centerline at the hot spot was found to occur during the first second. The fuel centerline temperature then decreases due to the shutdown of the reactor.

The useful results from PARET are summarized as below

(1) Maximum fuel centerline temperature290°F
(2) Cladding surface maximum temperature282°F
(3) DNB surface temperature295°F
(4) Maximum void fraction at exit of hot channel 0.87

The accident discussed above represents the worst case LOF accident which could be realized at MURR. In each case that was analyzed, the initial conditions used were worst case conditions with all coolant parameters at the safety system setpoint and reactor power at 11 MW. It was also assumed that in all cases the primary protective function did not operate and the reactor had to be shutdown by the backup protective function. These conditions were used to maintain a maximum of conservatism. Since there was no cladding failure predicted for any of the accidents under these very conservative conditions, it is concluded that the MURR reactor safety is not jeopardized by any LOF accident.

FIGURE D.4
RADIAL IN-CORE NEUTRON FLUX PROFILE

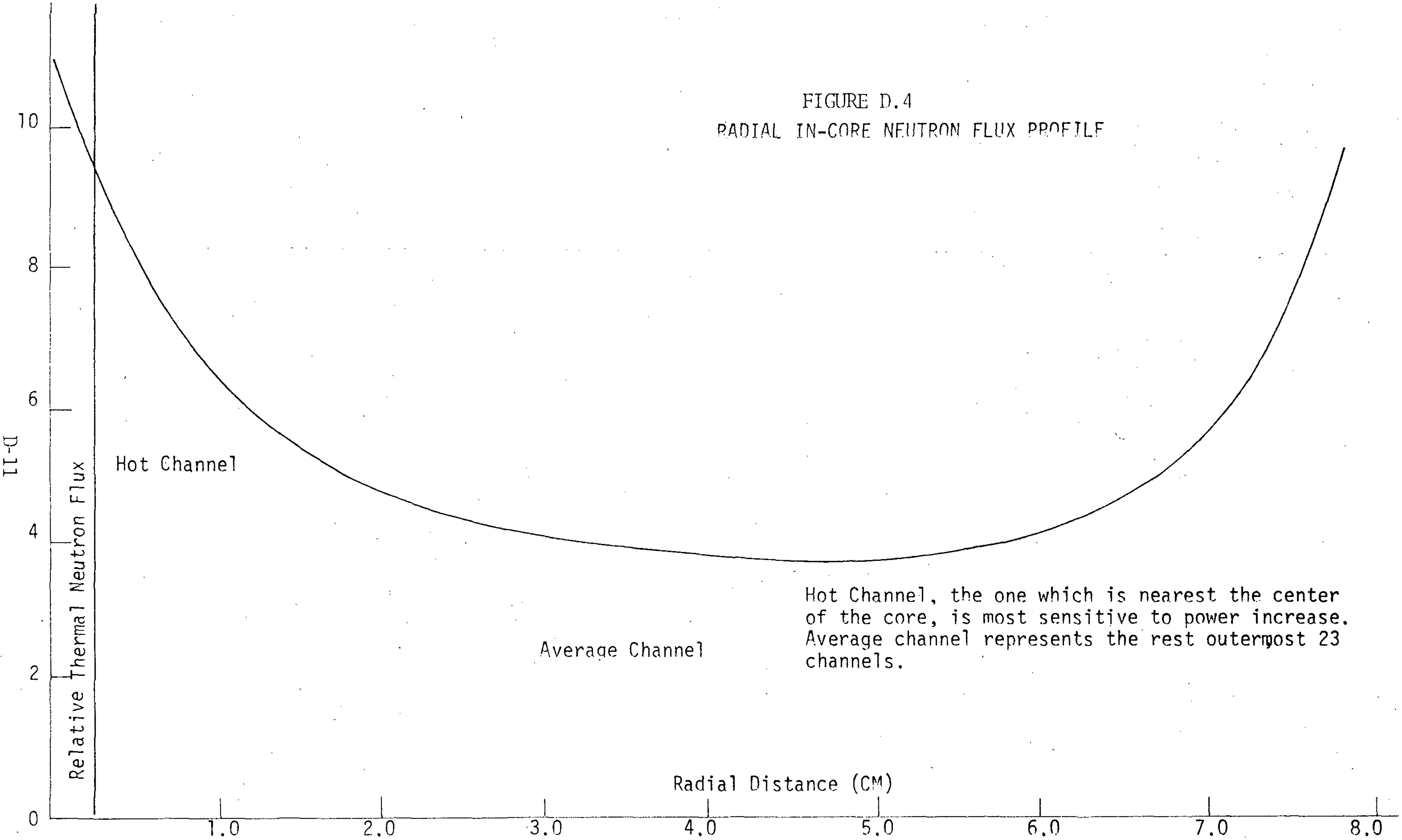


TABLE D.2. REACTOR POWER VS. TIME (VALVE CLOSURE)

TIME (SEC)	REACTOR POWER (MW)
0.0	11.00000
1.200000	10.70000
1.250000	8.92999
1.299999	5.610000
1.349999	3.440000
1.400000	2.589999
1.500000	1.790000
1.599999	1.410000
1.700000	1.110000
1.799999	0.9000000
3.000000	0.6180000
10.00000	0.5400000
20.00000	0.4930000
30.00000	0.4630000
40.00000	0.4380000
50.00000	0.4209999
60.00000	0.4069999

TABLE D.3. COOLANT MASS FLOW RATE VS. TIME (VALVE CLOSURE)

	TIME (SEC)	MASS FLOW RATE (LB/HR/FT**2)	
	0.0	4200000.	
	0.3250000	4157000.	
	0.7500000	3880000.	
	1.075000	3317000.	
	1.629999	1530000.	
	2.000000	422099.0	
	2.174999	209999.0	
	2.500000	29500.0	
Flow reversal	6.599999	105697.0	Flow reversal
	10.70000	93953.00	
	18.70000	136232.0	
	22.29999	138581.0	
	37.00000	176162.0	
	41.00000	162069.0	
	43.00000	201999.0	
	46.00000	293604.0	
	48.70000	209046.0	
	52.00000	11744.00	
	54.00000	140930.0	

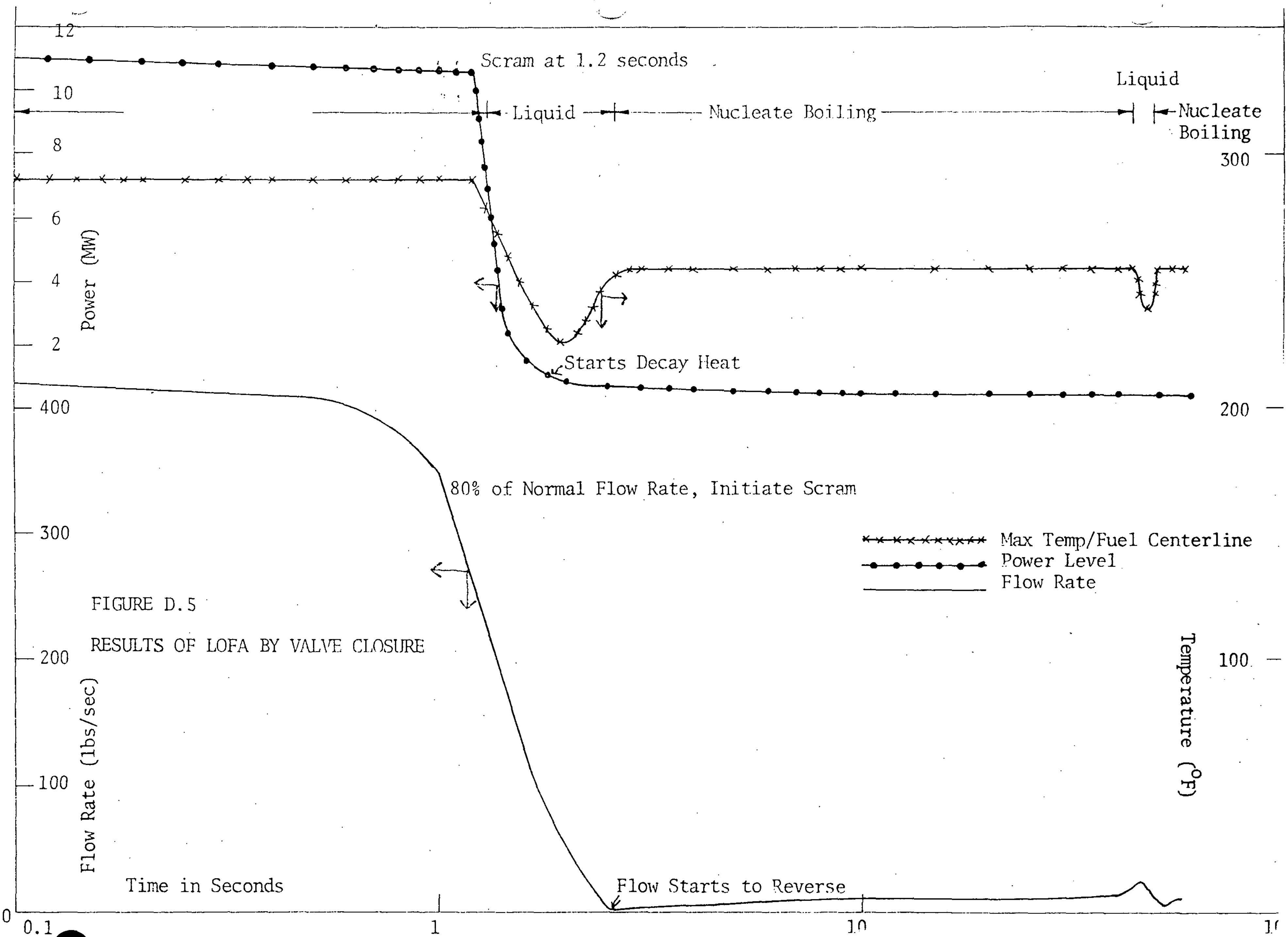
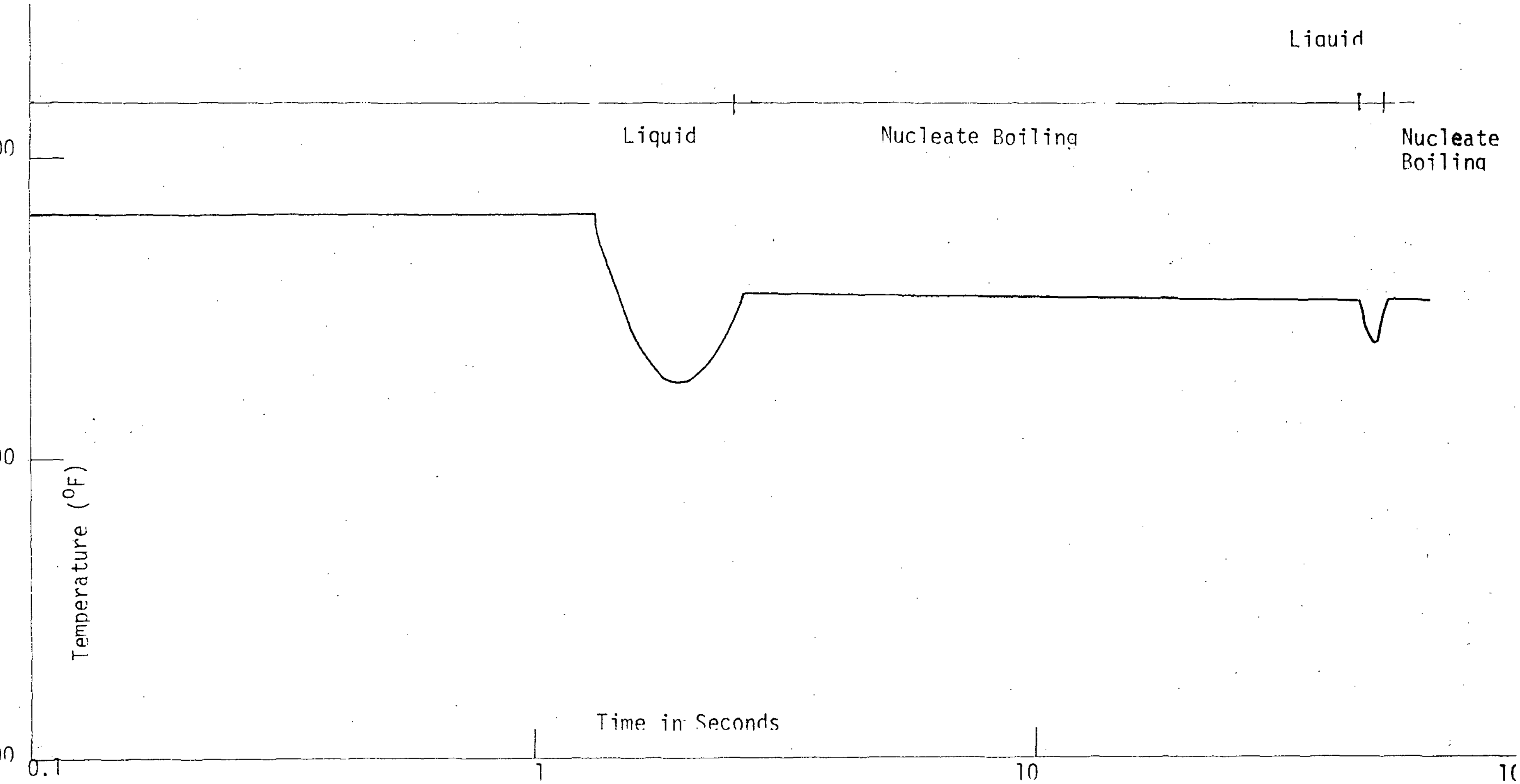


FIGURE D.5

RESULTS OF LOFA BY VALVE CLOSURE

FIGURE D.6

CLADDING SURFACE TEMPERATURE (VALVE CLOSURE ACCIDENT)



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APPENDIX E
LOSS OF COOLANT ANALYSIS

E. LOSS OF COOLANT ANALYSIS

E.1 Introduction

The most serious accident considered in the safety analyses is the hypothetical loss of coolant accident (LOCA) occasioned by the double-ended rupture of a main coolant pipe. Engineered safeguards provide a very low probability for the loss of coolant accident. However, the consequences of such an accident should be considered. It is required that in the event of such an accident the appropriate off-site radioactivity dosage limitations of the Code of Federal Regulations are not exceeded.

This report considers the sequence of events and expected consequences of a double-ended rupture of the MURR's largest cooling pipe. Section E.2 is based on actual flow experiments conducted at the MURR with the reactor shutdown. Other analyses discussed are based on mathematical models which reflect the phenomena occurring within the reactor coolant system and the reactor core during and after a loss of coolant accident. Computer codes, RELAP 3 (1) and WHAM (2), were employed for blowdown analysis. The PARET (3) code was used for the reactor core heat-up analysis. Conservative conditions and parameter values have been assumed for these mathematical models.

E.2 Consequences of Pipe Ruptures

Figure E.1 is a schematic of the in-pool portion of the MURR primary coolant system. Upon a rupture of the 12" primary coolant piping, the loss of pressure will be immediately detected by four pressure sensors: pressure switches 944A and B, PT 942, and PS 938. Each will actuate a reactor scram. Low pressure trips from PS 944A and 944B will drop relays 2K13 and 2K28, respectively, each of which will stop the primary coolant pumps, close the isolation valves 507A and B, and open the anti-siphon valves 543A and 543B. If the rupture is at considerable distance upstream or downstream from the core, the closure of the isolation valves adjacent to the pool penetrations will prevent the core from being uncovered. A rupture of the in-pool primary piping will simply admit pool water to the system until flow is stopped by the isolation valves and the core will remain covered. In either accident, the decay heat will be removed by the in-pool heat exchanger. The accident of greatest consequence is the rupture of the primary coolant piping between either isolation valve and the pool liner. The automatic protective actions mentioned above will occur.

E.2.1 Double-Ended Rupture of the Reactor Inlet Pipe Between Valve 507B and the Pool Liner

Several independent analyses have concluded this to be the most severe accident for the MURR. Upon rupture and loss of pressure the redundant pressure sensors PS 944A and 944B will cause the primary coolant pumps to stop and the reactor outlet isolation valves 507A/B to close. Figure E.2 and E.3 present the results of flow coastdown test performed on the MURR. Figure E.2 is a graph of the flow coastdown after an intentional shutdown of the primary coolant pumps with the isolation valves remaining open. Figure E.3 demonstrates the effect of closing the isolation valves with the pump continuing to run. Both curves present percent of nominal flow versus time in seconds after the electrical actuation of the pump and valve controls.

Momentum of the moving water will tend to empty the pressure vessel. It is the objective of this analysis to show that stopping the pumps and closing valve 507A will stop the flow before the core can become uncovered. Figure E.2 demonstrates that the flow can be reduced to less than 10% of nominal in six seconds. The isolation valve closure occurs somewhat slower, however throttling becomes significant at six seconds after electrical actuation and two seconds later the flow is stopped. The assumption will be made, based on this data, that both effects together stop the flow in six seconds. The flow coastdown curve of Figure E.2 may be conservatively approximated by the equation

$$F = F_O e^{-0.4t} \qquad \text{(E.1)}$$

where F = instantaneous flow rate (ft^3/sec)

F_O = initial nominal flow rate (ft^3/sec)

t = time in seconds after electrical actuation to stop the pumps

Making the conservative assumption that one second is required for the pressure sensors to initiate electrical actuation of the pumps and valves, one finds that the loss of coolant to the in-pool primary piping in this accident will be

$$V = F_O\left[1 + \frac{1}{0.4}\left(1 - e^{-0.4t}\right)\right] \quad (E.2)$$

where V = total coolant volume loss (ft^3)

τ = coastdown time = 6 seconds

For a nominal coolant flow rate of 3600 gpm F_O is 8.022 ft^3/sec. For a coastdown time of $\tau = 6$ sec

$V = 26.26\ ft^3$

The calculated volume of the reactor inlet piping from the outlet of primary isolation valve 507B to the top of the fuel elements is 28 cubic feet, thus the closing of outlet isolation valve 507A and stopping of the primary coolant pumps will prevent the core from becoming uncovered in the postulated accident. Once flow is stopped, the remaining water will settle to the low point of the in-pool piping and fill the pressure vessel to a level of at least five feet above the core.

In the case of failure of the previously described safety functions to operate on a break in the inlet pipe, the MURR anti-siphon system operates as a backup to ensure that the core will not become uncovered. Figure E.1 illustrates the system. Two redundant 4" butterfly valves, 543A and B, connect the primary system to a pressurized tank via a standpipe reaching to the top of the pool. These valves are air to close and spring loaded open. For reactor operation under forced convection, the anti-siphon standpipe must be drained down such that there is less than 6" of water above the 543 valves. A rupture of the primary system followed by loss of pressure causes these valves to open admitting air to the high point of the reactor outlet piping and thus breaking the siphon and preventing the core from becoming uncovered. Appendix B of this report presents an analysis of the effectiveness of the anti-siphon system.

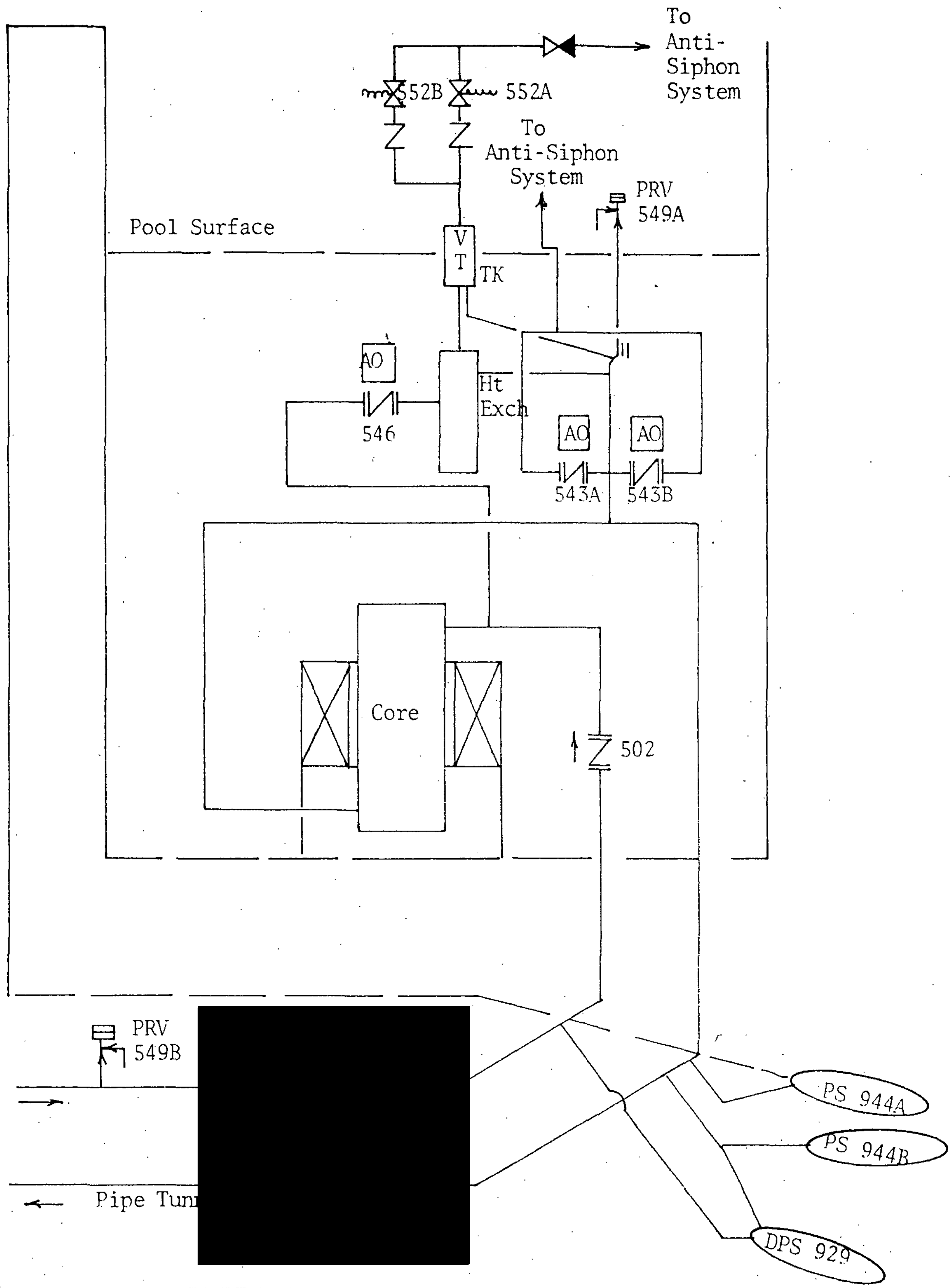


FIGURE E.1
IN POOL PORTION OF MURR PRIMARY COOLANT SYSTEM

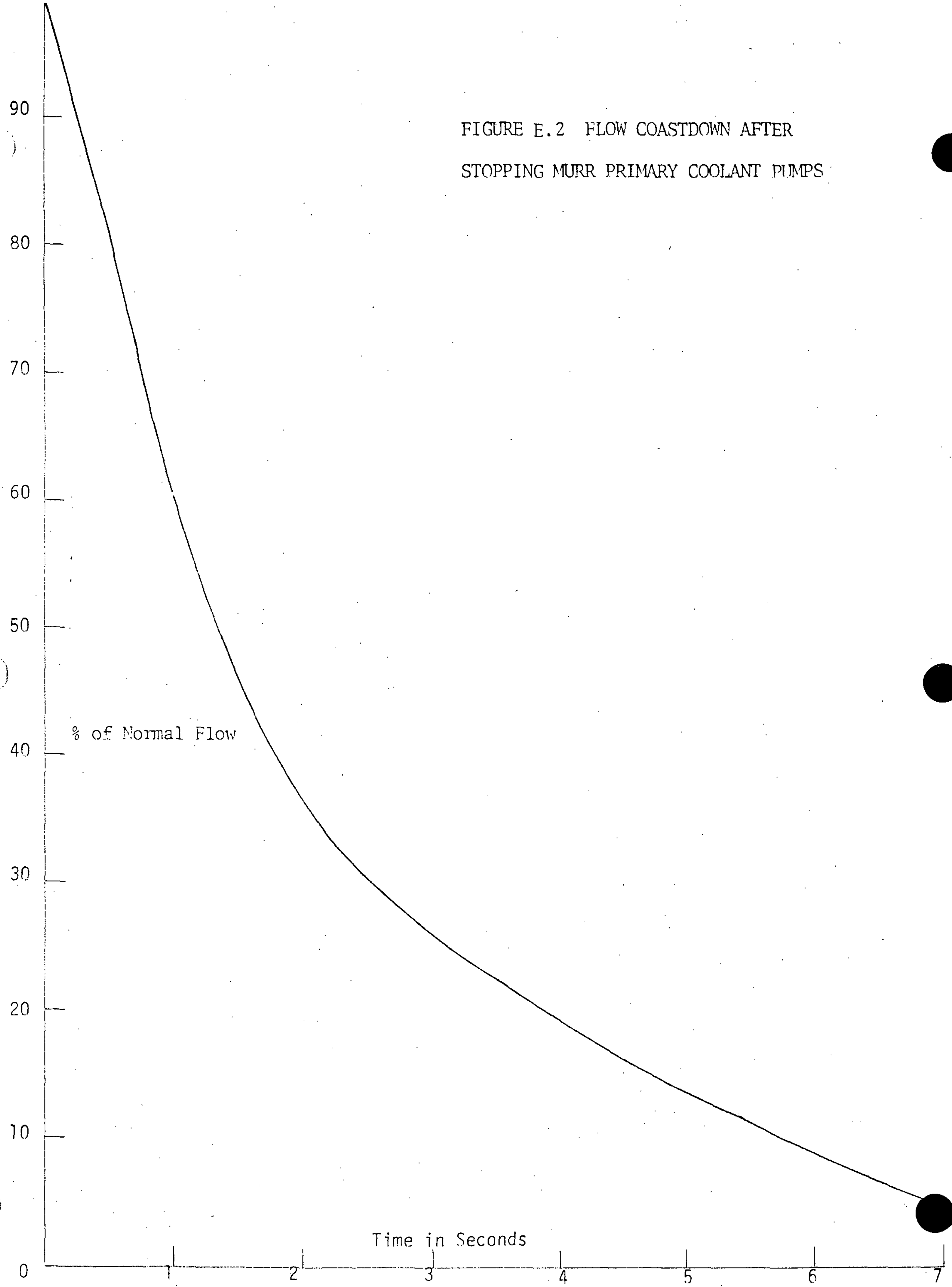


FIGURE E.2 FLOW COASTDOWN AFTER
STOPPING MURR PRIMARY COOLANT PUMPS
% of Normal Flow
Time in Seconds
90
80
70
60
50
40
30
20
10
0
1
2
3
4
5
6
7

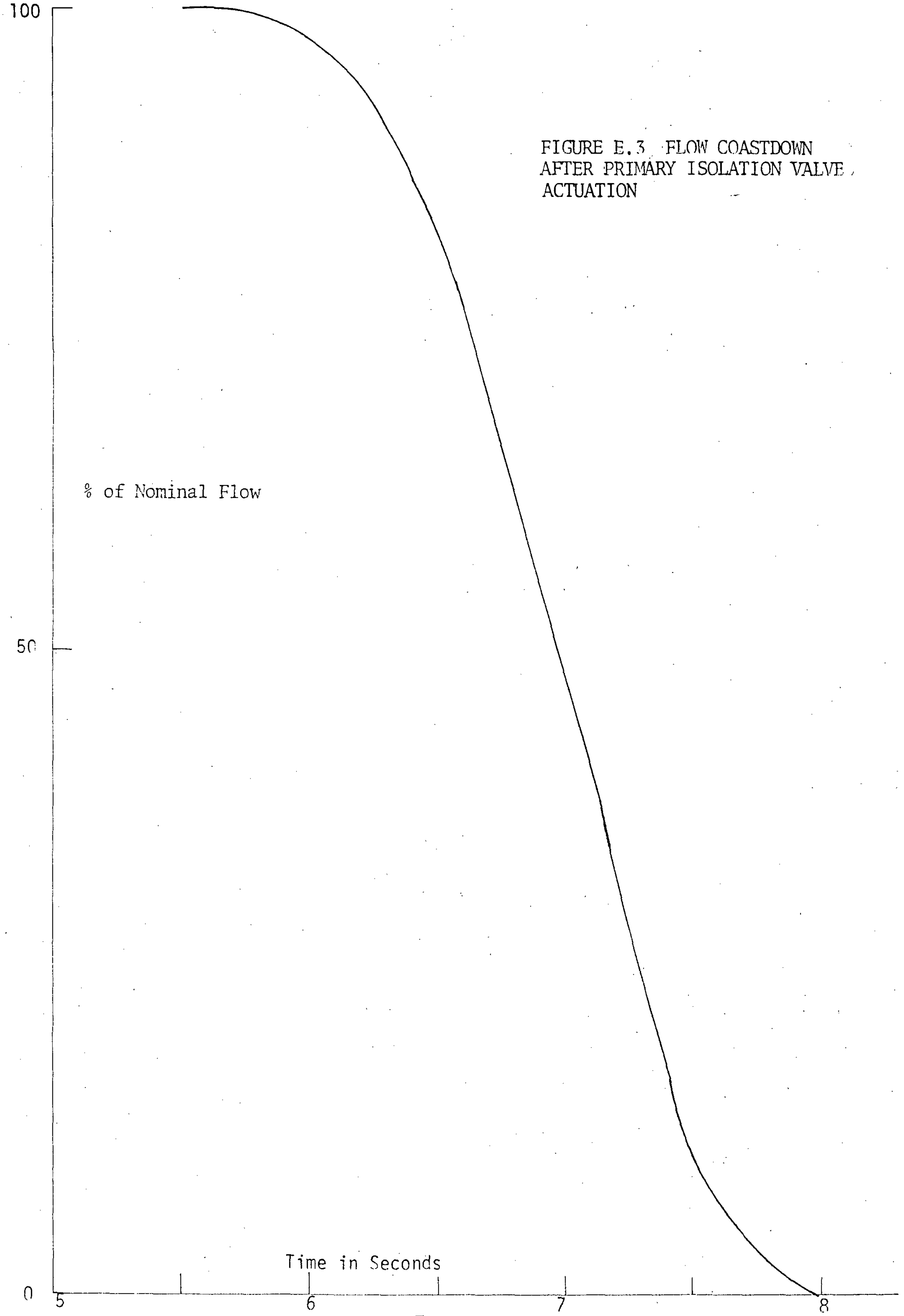


FIGURE E.3 FLOW COASTDOWN AFTER PRIMARY ISOLATION VALVE ACTUATION
% of Nominal Flow
100
50
0
Time in Seconds
5
6
7
8

E.2.2 Double-Ended Rupture of the Reactor Outlet Pipe Between Valve 507A and the Pool Liner

Again on rupture of the primary coolant line, pressure switches 944A and B will sense the loss of pressure and cause the pumps to stop and isolation valves 507A/B to close. This accident does not present a problem as the air-water interface moves away from the core.

E.2.3 Decay Heat Transfer to the Pool

Next one must consider the sequence of events once the flow has been brought to a stop and the decay heat is being dissipated by the remaining water in the partially drained pressure vessel.

The reactor is assumed to be shutdown and flow has been halted with the in-pool piping partially drained such that five feet of water remains above the core. The volume of water in and above the core was calculated to be 6.44 cubic feet or 392.3 pounds, assuming an average temperature of 165°F. The situation is similar to that considered in Addendum 3 to the MURR Hazards Summary Report (4) describing the consequences of a failure of valve 546 to open after a loss of primary coolant flow and a reactor scram. As in the previous work, one may look at the heat transfer capabilities from the pressure vessel to the surrounding 120°F pool. The water in the pressure vessel will rapidly increase in temperature until the mixed temperature is sufficiently above the 120°F pool temperature to achieve steady state. McAdams (7) cites the following equation for the natural convection heat transfer coefficient between the pool and pressure vessel.

$$h = 0.13\,k\left[\frac{\rho^2\,g\,\beta\,\Delta t}{\mu^2}\left(\frac{c_p\mu}{k}\right)\right]^{1/3} \qquad \text{(E.3)}$$

where all physical properties are evaluated at the film temperature, i.e., the arithmetic average of the pool and wall temperatures, and

k = water thermal conductivity (BTU/hr ft °F)

ρ = water density (lb/ft^3)

g = 32.2 ft/sec^2

μ = water viscosity (lb/ft sec)

β = water coefficient of volumetric expansion (1/°F)

c_p = water specific heat (BTU/lb °F)

Δt = temperature difference, wall to pool (°F)

Application of this equation to the present system yields Figure E.4; a plot of the product $h\Delta t$ versus pressure vessel wall temperature. The residual decay heat level from the reactor core decreases rapidly with time after a reactor shutdown and the ratio of decay heat to full operating power may be expressed as (6)

$$\frac{P}{P_O} = 0.1 \begin{bmatrix} (\tau - T_O + 10)^{-0.2} - 0.87\,(\tau - T_O + 2\text{x}10^7)^{-0.2} \\ - (\tau + 10)^{-0.2} - 0.87\,(\tau + 2\text{x}10^7)^{-0.2} \end{bmatrix} \qquad \text{(E.4)}$$

where

P = decay heat load

P_O = reactor operating power

T_O = past reactor operating time (sec)

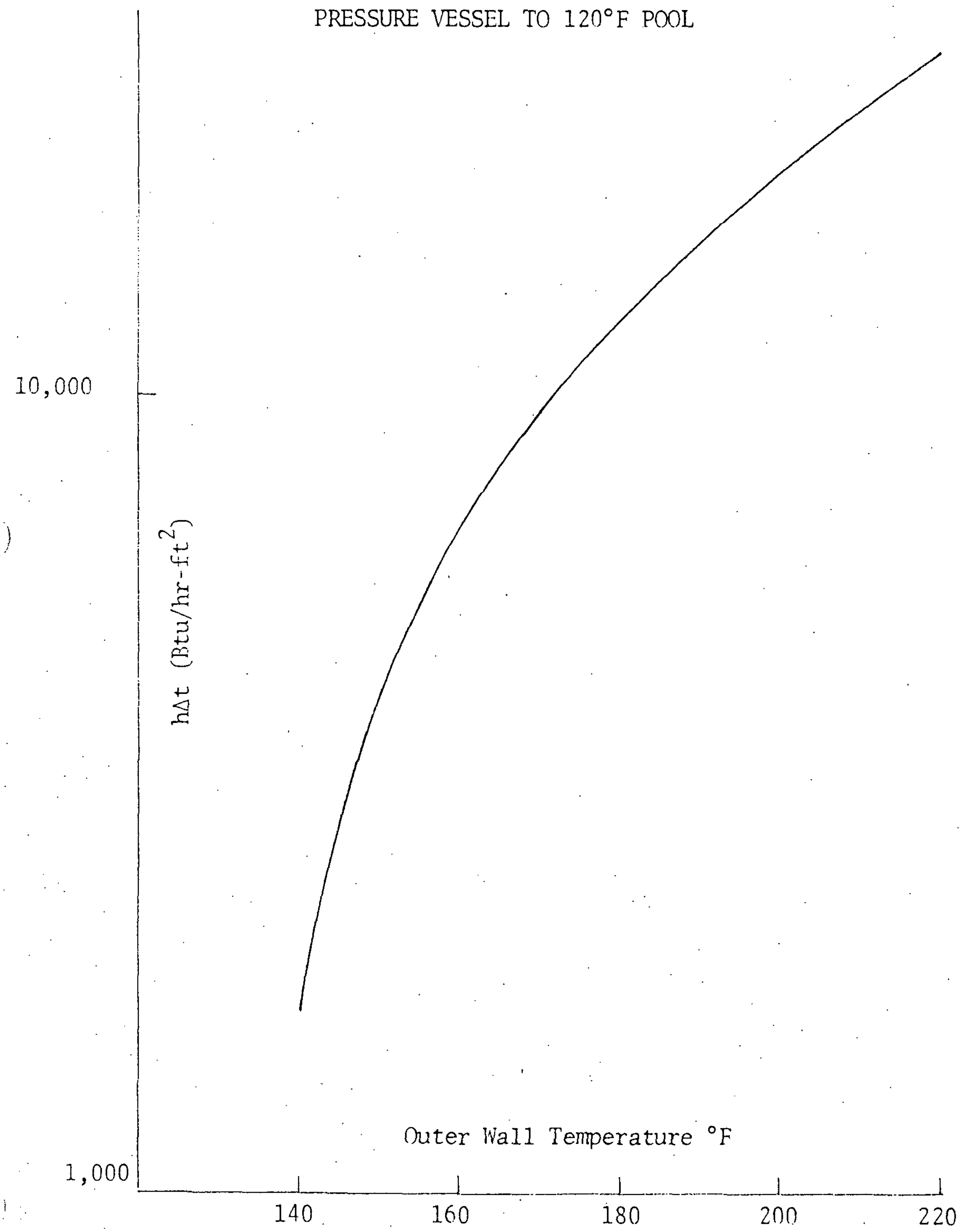
$\approx 1.369 \times 10^7$ for the MURR licensed burnup limit

τ = T_O + time after shutdown

This equation may be integrated for the time period $\tau = T_O$ to an arbitrary decay time to obtain the total decay heat generated after the shutdown. Figure E.5 presents the relationship between this integral and the decay time, i.e., time after reactor shutdown.

FIGURE E.4

HEAT REJECTION RATE
PRESSURE VESSEL TO 120°F POOL



One may approach the problem from the following very conservative method. Assume that there is no heat transfer between the 392.3 lb. of water in and above the core with its surroundings. The heat input required to raise this water to the saturation temperature of 220°F from an initial average temperature of 165°F is found to be 2.16×10^4 BTU. Taking a value of 966 BTU/lb for the latent heat of vaporization, it will require an additional 3.33×10^5 BTU to boil away the five feet of water covering the core for a total required heat input of 3.55×10^5 BTU required to begin to uncover the reactor core. The time required for this heat input is found from the equation

$$Q = 0.627\, P_O \int_{\tau=T_O}^{\tau=T_O+x} \frac{P}{P_O}\, d\tau = 3.55 \times 10^5 \text{ BTU} \tag{E.5}$$

where Q = accumulated decay heat (BTU)

P_O = Full operating reactor power

= 9478.33 BTU/sec

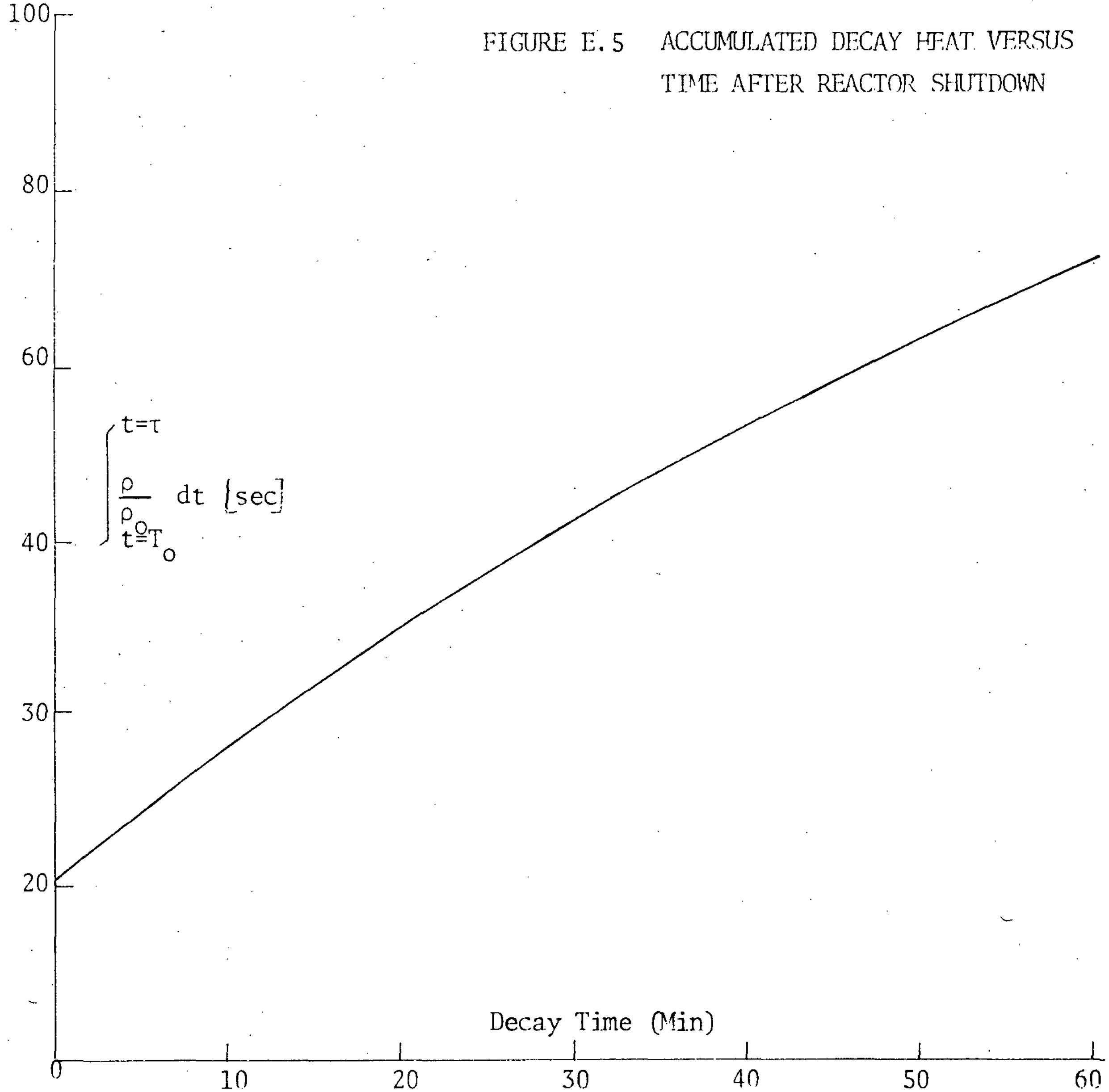
0.627 = fraction of decay heat absorbed in the core region (4)

Consulting Figure E.5 one finds that the value of the integral required to balance Eq. (E.5) corresponds to a decay time of 46 minutes after reactor shutdown. Evaluating Eq. (E.4) for a decay time of 46 minutes, one finds that the decay heat fraction will be 0.0165 or 1.65% of full operating power, corresponding to a decay heat rate of 98.06 BTU/sec being absorbed in the core region.

Next consider the capabilities of the system for rejecting this decay heat load. Make the conservative assumption that heat is transferred only in the radial direction to the 120°F pool water. The heated portion of the pressure vessel around and above the core is about 7 feet in length corresponding to an external surface area of 22.54 ft^2. Thus

$$\frac{Q}{A} = \frac{98.06 \text{ Btu/sec} \times 3600 \text{ sec/hr}}{22.54 \text{ft}^2} = h\Delta t = 1.566 \times 10^4 \text{ Btu/hr ft}^2 \tag{E.6}$$

FIGURE E.5 ACCUMULATED DECAY HEAT VERSUS TIME AFTER REACTOR SHUTDOWN



One may now consult Figure E.4 to find that the outer pressure vessel wall need only be 193°F to reject this amount of heat. The inner pressure vessel wall will be 197°F, well below the saturation temperature of the core water. Thus, boiling will have ceased before the core could be uncovered.

This extremely conservative approach has shown that after a reactor shutdown from full power with a loss of coolant accident resulting in 5 feet of water above the flooded core, the decay heat can be safely dissipated to the pool without core damage. It should also be noted that the steam postulated from boiloff will very quickly condense in the empty piping and in-pool heat exchanger and drain back into the pressure vessel. Therefore, a much longer time would be required to uncover the core than the conservative value calculated above. The MURR reactor at present possesses sufficient redundant safety features to prevent core damage as a result of the double-ended rupture of the largest primary coolant pipe and requires no additional emergency core cooling system for core protection in the event of a loss of coolant accident. Computer codes were used to support this conclusion and the results are presented in the following section.

E.3 Introduction to Mathematical Models Used in LOCA

In the LOCA the cold-leg break is the most severe because the reactor would lose its coolant most rapidly from that break.

In the case of the cold leg break, the saturation pressure is first reached in the hot leg at the pressure vessel exit. If a flow reversal occurs in the core, cooling of the fuel may momentarily stop until natural circulation is established.

For the MURR the operating pressures and temperatures are low. Any loss of coolant accident or loss of flow accident will be accompanied by flow reversal because the forced convection flow is in opposite direction to natural convection flow.

The complex nature of the coolant blowdown and core heat-up phenomena has led to the development of the Loss-of-Fluid Test (LOFT) at Phillips Petroleum Company in Idaho and to the development of a number of calculational models. Waage (7) provides a good review of the techniques that were available in 1967. The information shows that RELAPSE, the Phillips Petroleum modified version of FLASH, which originated at Westinghouse-Bettis, is the most detailed of these computer codes since it has provisions for core heat transfer. Two modifications of the original RELAPSE, RELAP-2 (March 1968)(8) and RELAP-3 (June 1970)(1), are available. The RELAP series of computer programs is being developed in support of the LOFT safety analysis effort. The basic purpose of the RELAP codes is to determine hydrodynamic conditions inside a reactor primary system (blowdown analysis). RELAP-3 retains most of the calculational methods used in previous versions but provides greater freedom in describing the system geometry. Another code, WHAM (2), has been written by Kaiser Engineers for subcooled blowdown analysis.

In view of the simplified heat transfer and power-transient approaches in RELAP, the code is generally not considered to provide a sufficiently detailed view of core behavior. The significant results of RELAP-type calculations (i.e., core inventory, pressure, quality, flow rates through the core, etc.) are commonly used as input data for a detailed analysis of the power transient and core thermal transient.

The basic tools used for the reactor power transient calculation during blowdown are: The CHIC-KIN (9) program developed by Westinghouse- Bettis Company and the PARET (3) program developed by the Phillips Petroleum Company in support of the Spert project conducted for the USAEC. Basically, the PARET program is an extension of CHIC-KIN. The fuel elements in PARET may be either cylinders or plates. The core power transient analyzed by PARET is used to calculate the clad temperature for the thermal transient.

E.4 Blowdown Analysis

E.4.1 Blowdown Analysis Using the RELAP 3 Computer Code

The two-phase blowdown code used in this study is RELAP 3, Mod 36, prepared by the Aerojet Nuclear Corporation. This code has been accepted for use in similar analysis.

RELAP 3 represents the PWR system as a set of nodal volumes interconnected by flow junctions. Mass and energy balance calculations are made for each volume in small time increments, analyzing for the system inflow, outflow, and production terms. Momentum calculations for flow junctions are governed by pressure, inertia, flow area and friction. There are many features and options to model the volume-junction network as closely as possible to the physical system. These include provisions for bubble rise and phase separation in each volume, gravity head, choked flow, energy addition or removal, pump head and coastdown.

The RELAP 3 code is written in FORTRAN IV, double precision. It contains some 50 subroutines which can be overlayed to render the computation more efficient. The code can handle up to 200 volumes and 250 junctions. As the number of volumes and junctions increase, the time increment for calculations must be reduced. The computation time consequently must be increased to cover the whole LOCA period. In this study a time step of 0.005 seconds was found to be the most suitable for the ten volume reactor model.

The basic mass, energy, and momentum balance equations are employed by RELAP 3 to calculate the primary quantities for the system during the blowdown period. The reactor core is always treated as a heat source during normal operation or during blowdown. One-dimensional heat conduction in cylindrical rods is assumed to describe the heat transfer from inside the fuel rod to the coolant. The geometric transformation from rod to plate type geometry is outlined in another report (10).

RELAP 3 uses a series of heat transfer correlations to cover the wide spectrum of system conditions during blowdown. The basic parameter influencing the choice of correlation is the quality of coolant in the channels. Next considered for correlation application are factors such as fuel surface temperature, surface heat flux, and bulk coolant temperature.

In order to establish whether or not film boiling takes place, the critical heat flux is computed and compared with the actual heat flux. The critical heat flux is also obtained from a series of correlations, each of which is applicable to a certain regime of system conditions. The parameters which define the critical heat flux are the coolant pressure and mass flow.

Power generation is determined by either a table of power versus time or a kinetics calculation using point reactor kinetics equations. Data for one prompt neutron group, 6 delayed neutron groups and 11 gamma groups are stored in the program. Options available to the user include the input of power distribution between various portions of the core plus Doppler, void and temperature coefficients. The two-phase separation model used in RELAP 3 is a semiempirical fit to a number of experimental results. RELAP 3 was developed in support of the LOFT project. It has been tested in a series of semi-scale blowdown simulations. The code's modular structure facilitates improvements as more experimental data, better numerical techniques, and different methods of problem solution are obtained.

Like most blowdown codes in current use, RELAP 3 has the following deficiencies:

(a) RELAP 3 does not account for flow blockage due to clad swelling and rupture. For high temperature-high pressure reactors, calculations and experiments have indicated that quite early in the blowdown, when pressure, flow and therefore heat transfer coefficients drop drastically, the cladding in certain hot spots fails under the influence of the steep temperature transient and internal gas pressure. Flow channel blockage may be considerable. MURR operating pressures and temperatures are very low and therefore this phenomena is not important.

(b) The forward difference numerical technique employed in the solution of balance equations requires that a very small time step be used. This often results in excessive computation time. Furthermore, flow through the core may fluctuate unrealistically in the late stages of the blowdown.

(c) RELAP 3 does not take into account the air-intake to the system. The double-ended rupture in the cold leg was originally modeled by assuming that there is a valve at the rupture point that closes at the same moment as the rupture occurs, and that there are two leaks just before and after the valve with the maximum flow areas equal to the pipe cross-sectional area. After several runs of the code, it was found that the system pressure was unreasonable due to the air-intake from the one leak. Thus, in this report, the double-ended rupture was modeled by specifying a leak with maximum flow area twice the pipe cross-sectional area.

Figure E.6 is a diagram of the model for the analysis of the loss of coolant accident caused by a double-ended pipe rupture in the cold leg. The double- ended rupture in the cold leg was considered as the most serious condition for the MURR primary cooling system. Table E.1 lists details of the nodes and junctions.

Ten volumes were chosen to represent the MURR primary system. The volumes included 3 for the cold leg piping, 1 for the core, 1 for the in-pool heat exchanger, 2 for the hot leg piping, 1 for the primary heat exchanger, 1 for the pressurizer and 1 for the pipe tunnel. The physical dimensions and properties of these volumes have been extracted from the actual design of the MURR. Thermodynamic properties of the coolant in these volumes are those present when the plant is operating at a steady state power of 10 MW. The characteristics of the core were averaged to obtain proper values for the calculation.

Volume 8 is connected to the broken volume 2 through a valve which would open at the beginning of the LOCA. After many tests of the code, the results showed that the behavior of the system pressure was most reliable with the pipe tunnel represented by a large volume.

Ten junctions were used to connect the volumes together in a closed loop. The length and friction coefficients of a junction are set equal to the combined length and friction coefficients of the two connected volumes. The combining was done in a manner such that, at steady state, the computed flow agrees with the actual flow in the system.

As a further check on the validity of the model, the code was used to model flow coastdown after loss of pumping power with no pipe rupture. The MURR has conducted coastdown experiments with the reactor shutdown and Figure E.7 demonstrates the excellent agreement between calculation and experiment.

Seven neutron groups (one prompt, six delayed) and eleven delayed gamma groups have been used to describe the core kinetics and heat production rates following the start of the LOCA. Negative reactivity contributions include void formation, fuel Doppler coefficients (negligible for MURR) and coolant temperature coefficients. Void formation is almost solely responsible for the negative feedback reactivity.

Selection of the time step is very important for RELAP 3. A large step usually results in numerical instability. A time step approximately equal to the time required for an acoustic wave to travel the length of the shortest pipe in the system is usually suggested.

For the model under consideration, a time step of 0.0005 seconds was chosen and tested for up to 20 seconds without any numerical fluctuations.

The principal parameters calculated as a function of time after the break are: reactor power history, coolant flow rates, coolant quality, heat transfer coefficients, and pressure in the core. All quantities are averaged for the system or the portion of the system considered, and are stepwise calculated.

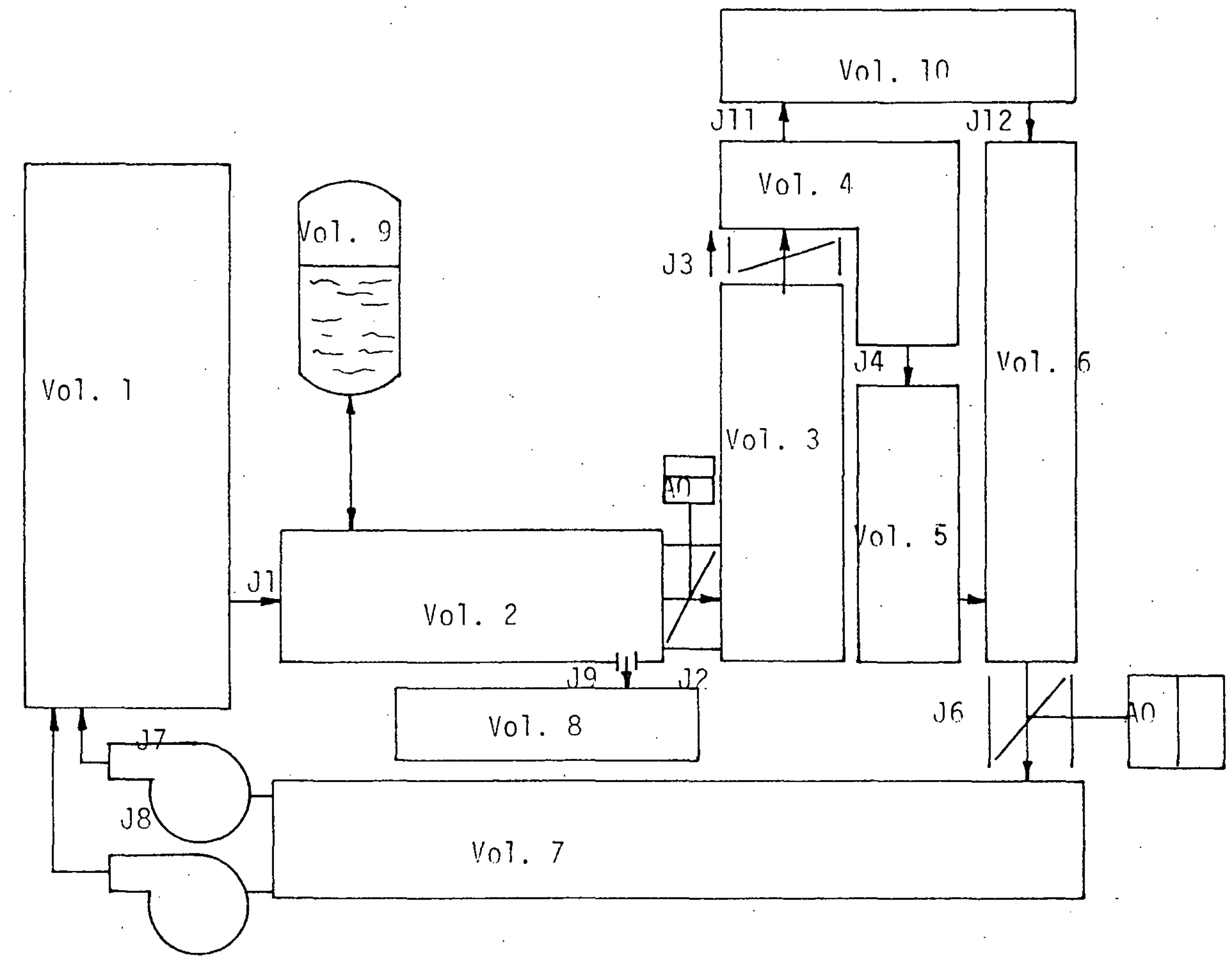


FIGURE E.6 RELAP 3 MODEL FOR LOC ACCIDENT

TABLE E.1

Description of Model Nodes and Junctions

Node	Description	Junction	Connects
V1	Heat Exchanger and Pipings	J1	Vol 1 to Vol 2
V2	Pipings	J2	Vol 2 to Vol 3
V3	Cold Legs	J3	Vol 3 to Vol 4
V4	Pressure Vessel	J4	Vol 4 to Vol 5
V5	Reactor Core	J5	Vol 5 to Vol 6
V6	Hot Legs	J6	Vol 6 to Vol 7
V7	Pipings	J7	Vol 7 to Vol 1
V8	Pipe Tunnel	J8	Vol 7 to Vol 1
V9	Pressurizer	J9	Vol 2 to Vol 8
V10	In Pool Heat Exchanger	J10	Vol 9 to Vol 2
		J11	Vol 4 to Vol 10
		J12	Vol 10 to Vol 6

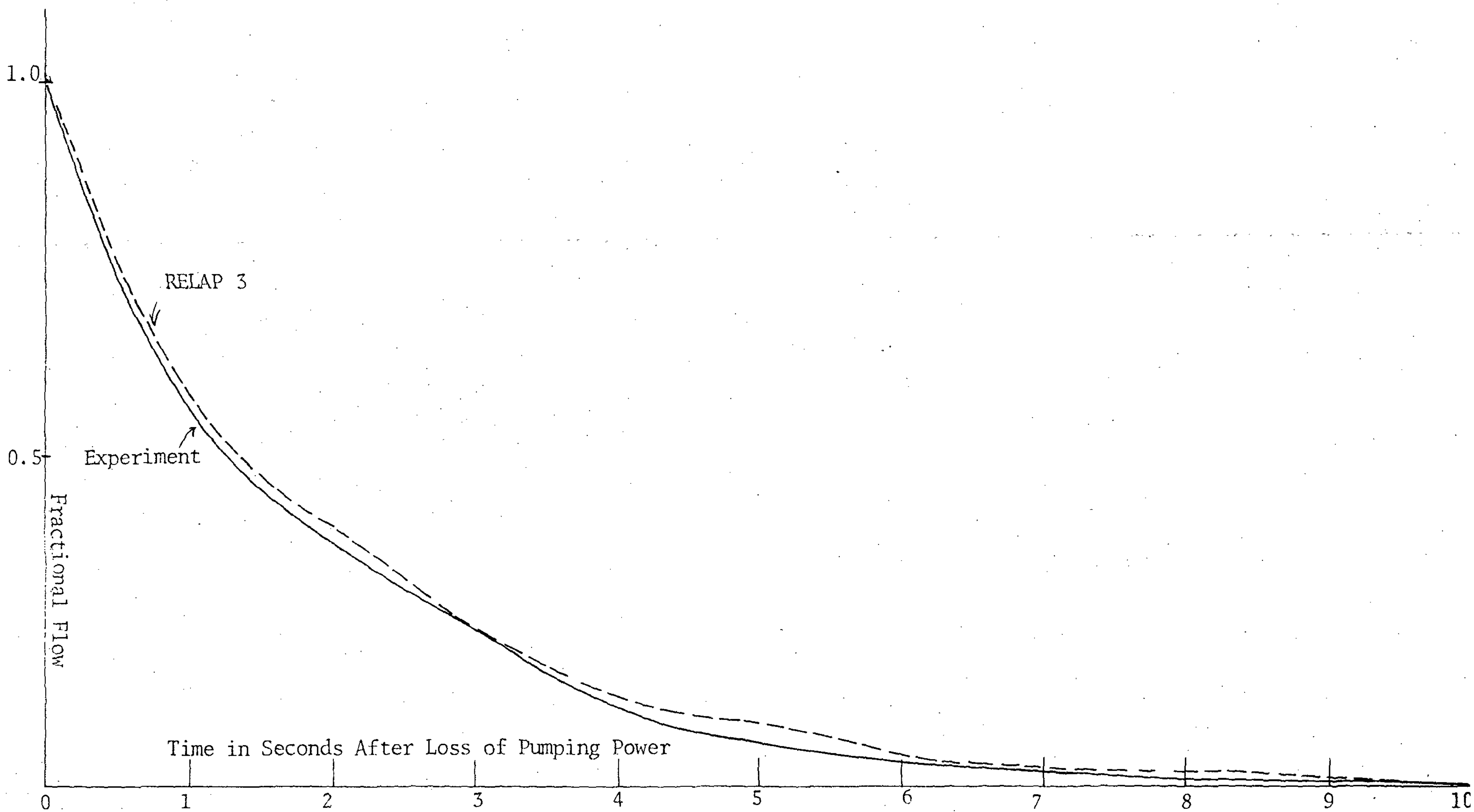


FIGURE E.7 COMPARISON OF EXPERIMENTAL AND RELAP-3 CALCULATION OF FLOW COASTDOWN

The input power history of the core is shown on Figure E.8.

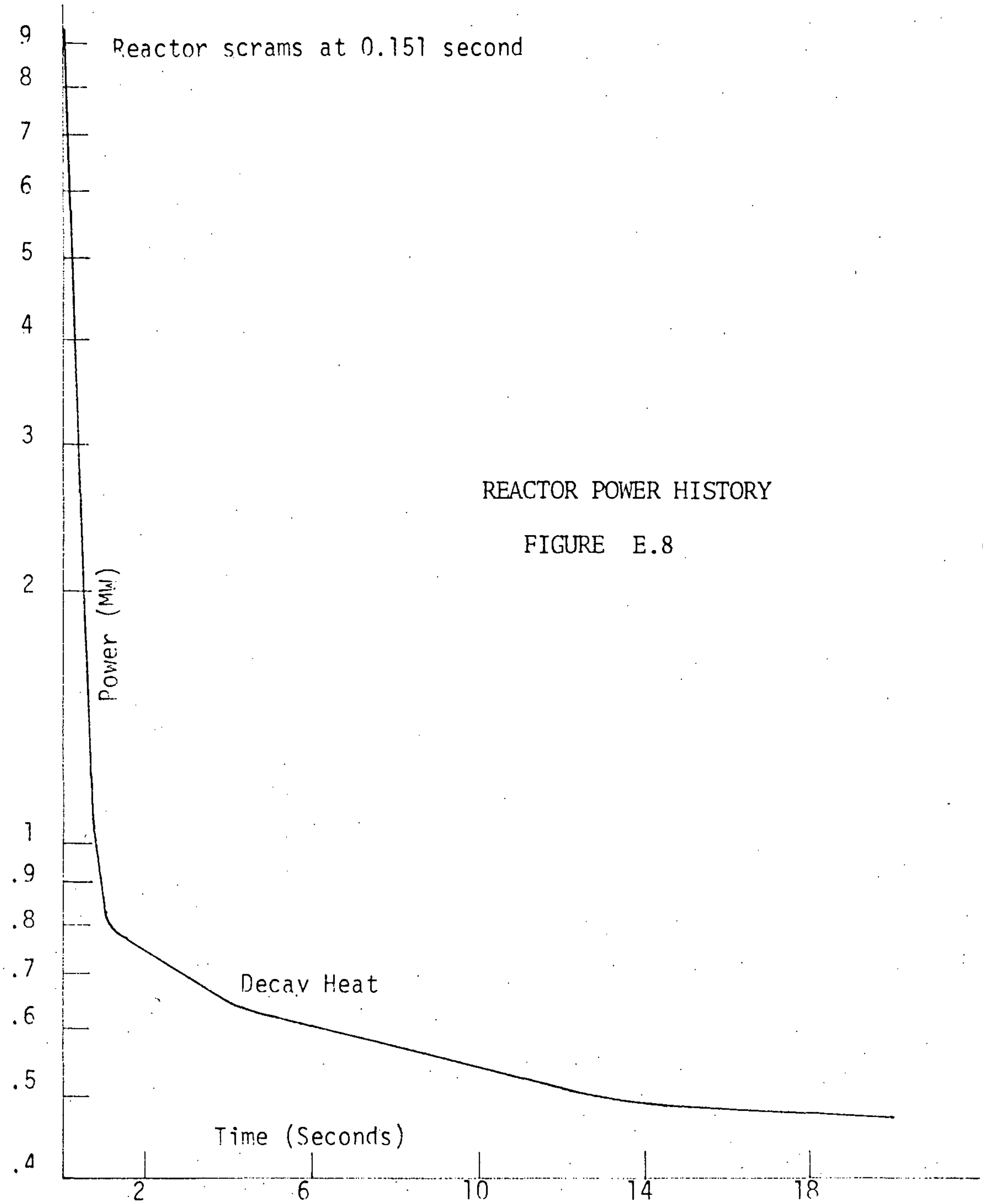
The reactor scram system is activated due to loss of pressure 0.001 seconds after the rupture of the pipe. Control rods start to move to scram the reactor at 0.151 seconds (0.15 second electronic time delay) and the reactor is shutdown after 0.85 seconds. The reactor scram overrides the effect of temperature feedback, Doppler reactivity feedback or void reactivity feedback.

Figure E.9 shows the calculated flow rate at the inlet of the core. Following the rupture of the cold leg, the coolant inlet flow rate decreases rapidly to half of its normal value in 0.2 seconds. Addition of water from the in-pool heat exchanger to the primary loop will increase the flow rate slightly as valves 546A and B open, further protecting the reactor core from thermal effects. Flow stagnation and reversal occurs at 8.8 seconds. The flow reversal is very short due to the loop discontinuity caused by the draining of the in-pool heat exchanger.

Figure E.10 illustrates the coolant flow rate through the rupture. This coolant loss is acceptable. There is no choking or flashing at the break and the core remains covered.

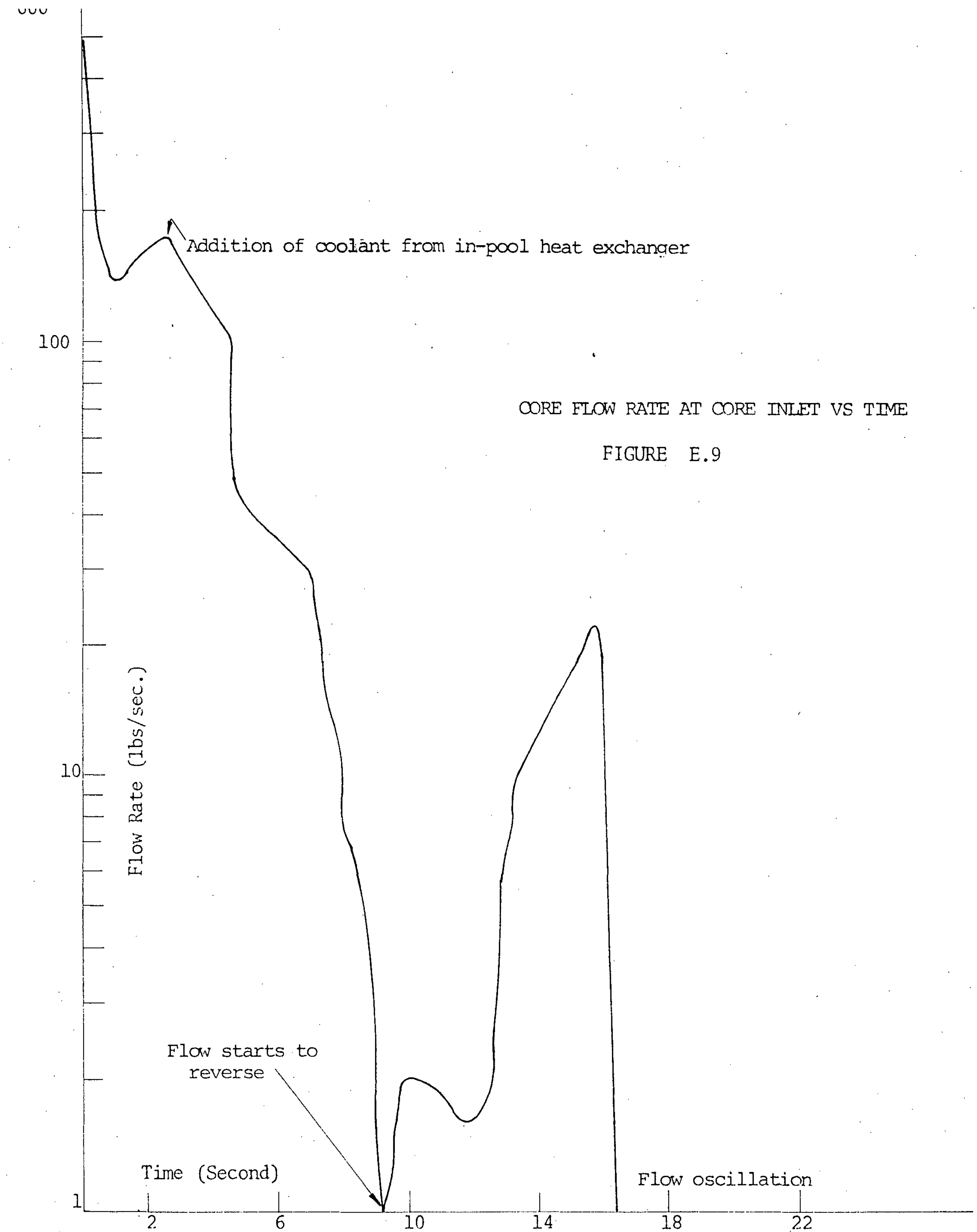
Figure E.11 illustrates the pressure history in the reactor core. It is obvious that depressurization is very fast leading to a steady state pressure of about 5 psig in 0.6 seconds. The steady pressure is above the saturation pressure because of the low initial system temperature.

Figure E.12 shows the time dependent behavior of the hot-leg coolant quality. The hot leg is most sensitive to pressure transients because of its lower operating pressure and higher operating temperature. Usually the saturation pressure is first reached at the hot leg. For MURR the pressure transient is not significant. The highest quality in the hot leg is only 5.35×10^{-5}. Thus, the system can be assumed to stay subcooled during blowdown.



REACTOR POWER HISTORY

FIGURE E.8

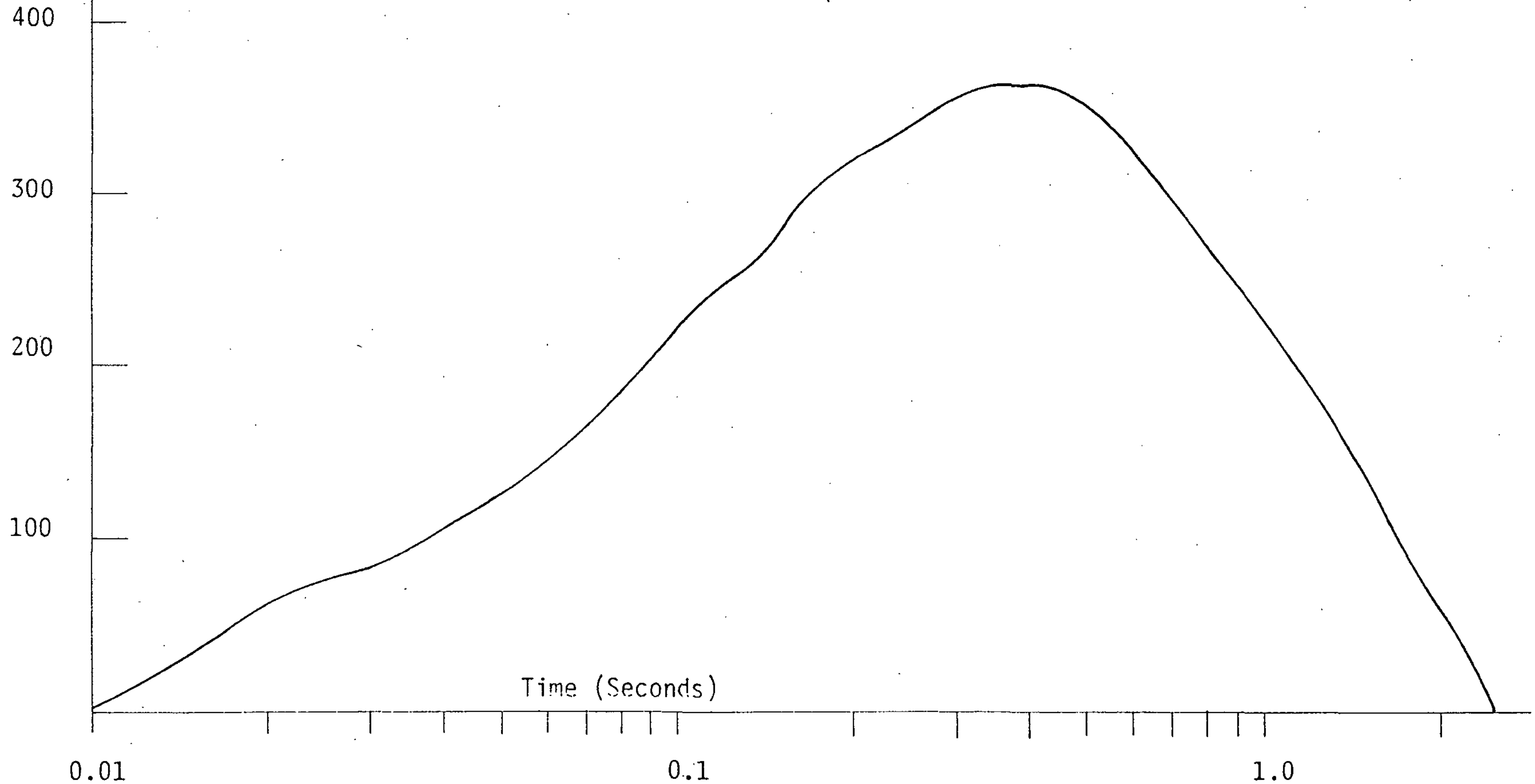


000
Addition of coolant from in-pool heat exchanger
100
CORE FLOW RATE AT CORE INLET VS TIME
FIGURE E.9
Flow Rate (lbs/sec.)
10
Flow starts to reverse
Time (Second)
Flow oscillation
1
2
6
10
14
18
22

COOLANT FLOW RATE THROUGH LEAK VS TIME

FIGURE E.10

(Total amount of coolant loss = 566.5 lbs)



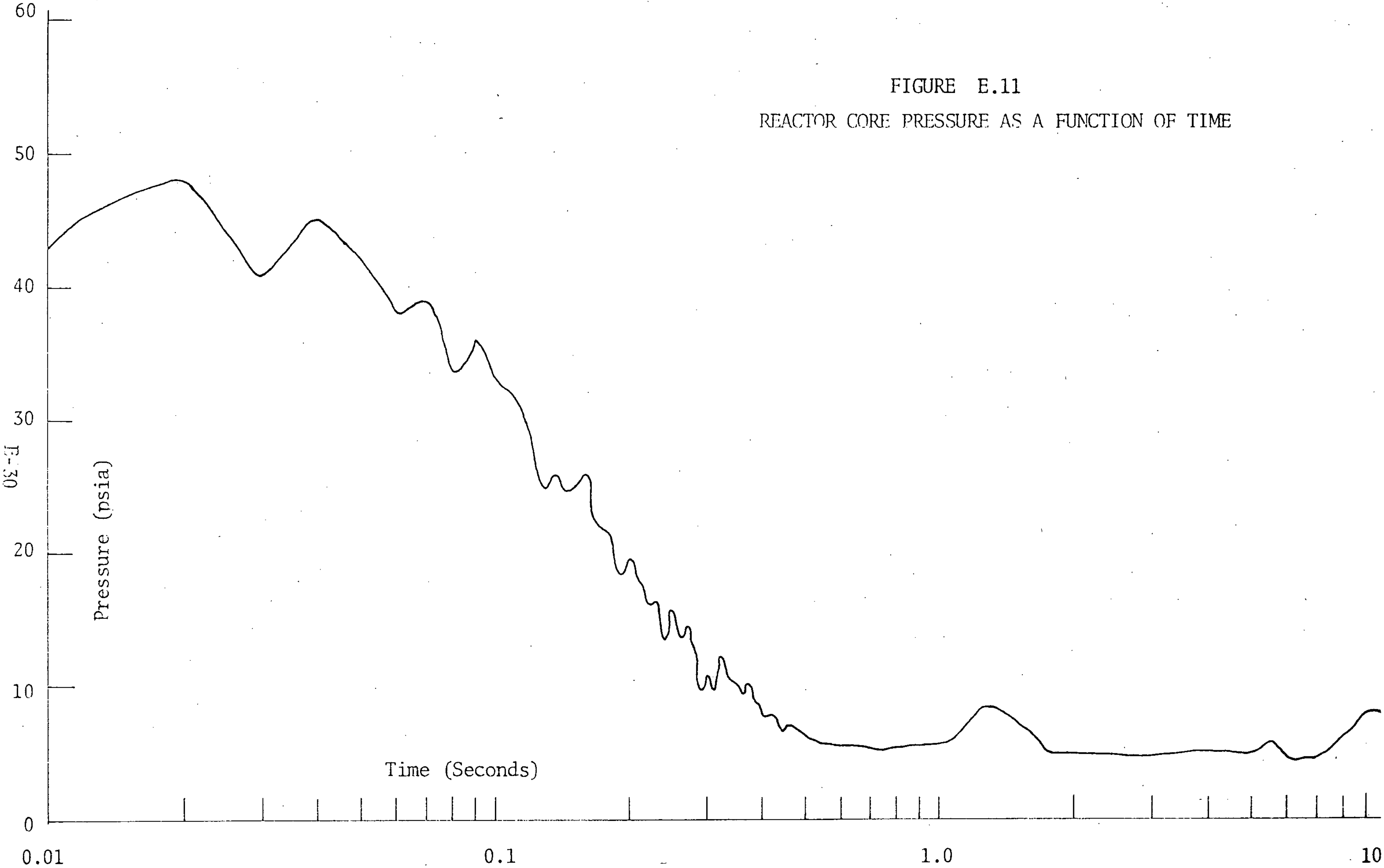


FIGURE E.11
REACTOR CORE PRESSURE AS A FUNCTION OF TIME

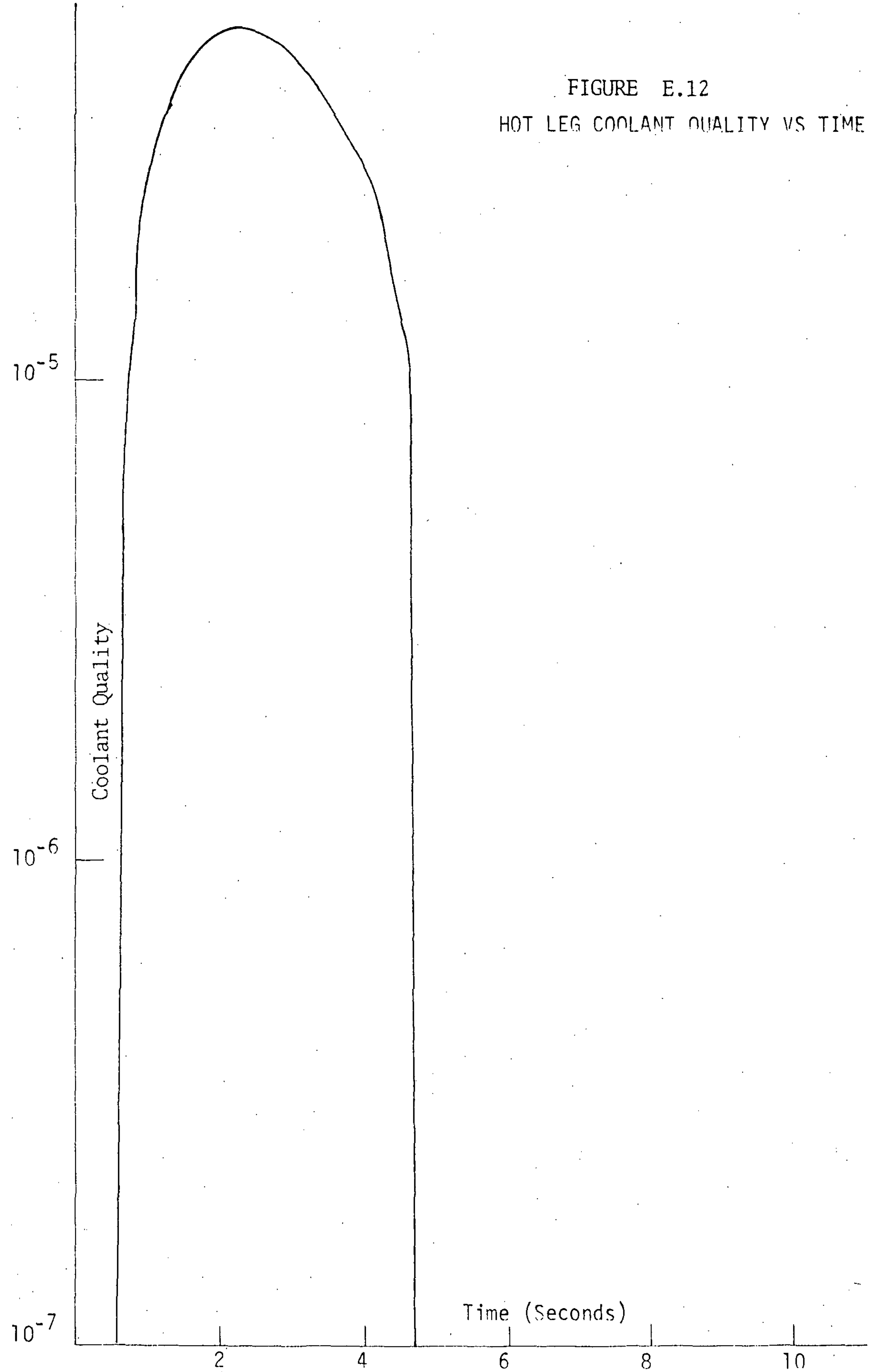


FIGURE E.12
HOT LEG COOLANT QUALITY VS TIME
Coolant Quality
10^{-5}
10^{-6}
10^{-7}
Time (Seconds)
2
4
6
8
10

E.4.2 Blowdown Analysis Using the WHAM Computer Code

Results of the two-phase blowdown code RELAP-3 indicate that the qualities of coolant in the pipings and core are less than 10^{-4} during the depressurization and blowdown of the system. So, it can be assumed that the reactor system stays subcooled during the loss of coolant accident without introducing any significant errors. The subcooled blowdown code, WHAM, is employed here to simulate the behavior of the primary coolant transient of MURR. WHAM has been experimentally verified and is suitable for MURR.

The basic equations to be solved in order to obtain the force and pressure loadings as a function of time are the conservation equations of mass, momentum and energy. WHAM employs the wave superposition technique to account for the pressure and velocity. In order to account for space dependence, it assumes the loop to consist of one-dimensional segments. Segment length, sonic velocity, and unit time interval for the wave to travel through the segment are all input so that wave transmission, reflection and superposition can be accounted for as time progresses. The assumption of constant sonic velocity in each segment limits the validity of the code to the subcooled portion of the blowdown where sound velocity is not drastically altered by the two-phase mixture.

Advantages of WHAM are its versatility in modeling a complex three-dimensional network with multiple junctions and parallel flow paths, ease of data preparation, and low computation cost. Gruen and Hansen (11) have used WHAM for the prediction of the IDAHO 700 and 800 series semiscale blowdown tests and have concluded that the calculated pressure histories match the measured values sufficiently well.

A disadvantage of WHAM is the assumption of plane wave propagation in one-dimensional tubes. This assumption would work well for the core channels, the hot segments, the cold segments, and steam generator tubes, but would fare poorly at the upper plenum, lower plenum and particularly at downcomer annuli. For MURR, the above disadvantage is not important.

WHAM considers compressible liquid flow in one-dimensional, elastic pipes. The physical laws obeyed are:

(a) conservation of mass,

(b) conservation of momentum,

(c) linear relationship between compressed liquid density and pressure,

(d) linear relationships between internal pressure and pipe strain (Young modulus of elasticity),

(e) linear superposition of plane acoustic waves.

A sudden double-ended break in the cold leg will send disturbance waves through the system. WHAM has been reasonably successful in accounting for these waves. The degree of success depends to a great extent on accurate modeling of the primary loop.

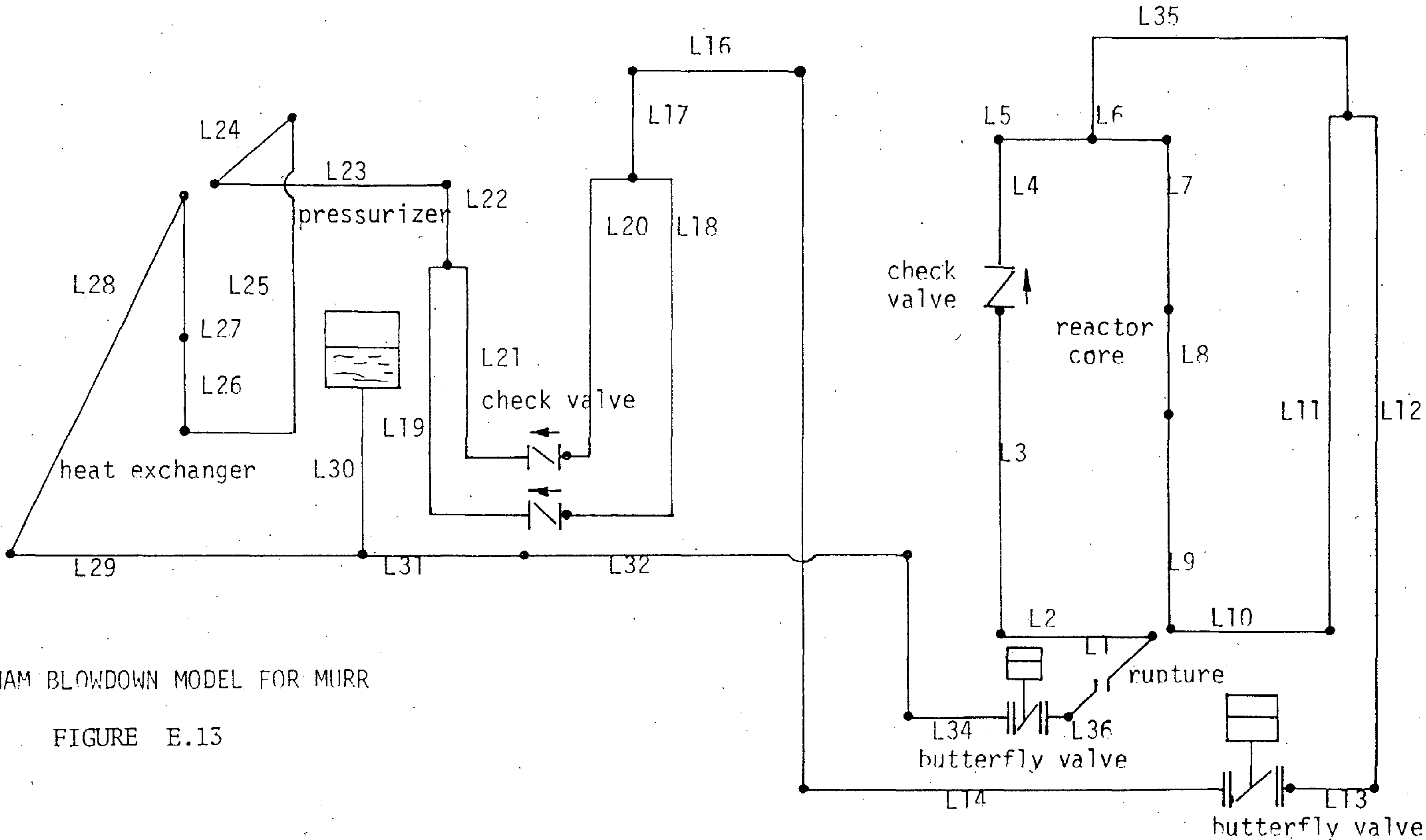
Figure E.13 illustrates the model developed for WHAM calculation of subcooled loadings subsequent to the cold leg break. It is noted that the location of leg 1 is the artifice employed by WHAM to represent the break. Starting with leg 1, the breaks in all other legs are numbered sequentially in such a manner that their order must increase in the direction away from the break (the direction that the decompression wave travels). The last leg also ends with a break to simulate the double-ended rupture. Table E.2 indicates the system legs modeled for WHAM.

TABLE E.2

Leg Number	Description
leg 1 to leg 7	cold leg piping
leg 8	reactor core
leg 9 to leg 26	hot leg piping
leg 27 to leg 29	system piping
leg 30	pipings from primary coolant loop to pressurizer
leg 31 to leg 36	coolant loop piping
leg 35	in-pool heat exchanger

Some special features of the model are:

(a) The presence of the pump has been neglected. The pump will be shutdown in the very early stage of blowdown. Due to the experimentally proven property of low running momentum, the pump can be ignored without causing significant error. The user's guide to the WHAM computer program shows that the variation of results between cases where the pump is included and the pump is neglected is very small (11).



WHAM BLOWDOWN MODEL FOR MURR

FIGURE E.13

(b) The influence of the in-pool heat exchanger has been neglected due to a limitation of the WHAM program. The program cannot use the time dependent specification of valve opening. The butterfly valves between the inlet piping and the in-pool heat exchanger would open in one second after the blowdown transient.

The time dependent reactor inlet flow rate by both WHAM and RELAP-3 codes is shown on Figure E.14. The curve derived from WHAM output drops quickly to zero in 0.86 seconds. RELAP-3 still indicates a flow rate of 140 lbs/sec one second after the rupture. The different ways of modeling the double-ended rupture by these programs are the major reasons for the deviations in Figure E.14. RELAP-3 specified a rupture of double pipe cross-sectional area on the surface of the pipe to approximate the double-ended rupture. Only part of the flow is lost from the system. The loss of momentum due to the rupture is not as severe and it would take a longer time for the flow to drop to zero. WHAM simulated the double-ended rupture more realistically. The system flow will be discontinued through the rupture and almost all the system momentum will be lost through the rupture. This is why the transient flow rate predicted by WHAM drops to zero in 0.86 seconds, and the air-water interface does not reach the check valve. Thus, WHAM predicts that the core coolant will start reverse circulation through the in-pool heat exchanger at 1 second after the transient and the rupture presents no hazard to the reactor.

E.5 Core Heat Up Analysis Using the PARET Computer Code

The time-dependent reactor power and core inlet coolant flow rate supplied by RELAP-3 were input to the PARET reactor transient computer code. The output from PARET includes all the important time-dependent information for each specified axial node point of the fuel coolant channel, i.e., coolant temperature, surface heat flux, heat transfer coefficient, burnout ratio, coolant quality, DNB surface temperature and radial temperature profile in the coolant and fuel plates. The above information supplied by PARET is sufficient to predict the behavior of the fuel following the loss of coolant accident.

The transient is forced by specifying an average core power versus time and an inlet coolant mass flow rate as a function of time. PARET has the ability to calculate the power and channel coolant flow rate by its point kinetics equation and one-dimensional hydrodynamic equation (3).

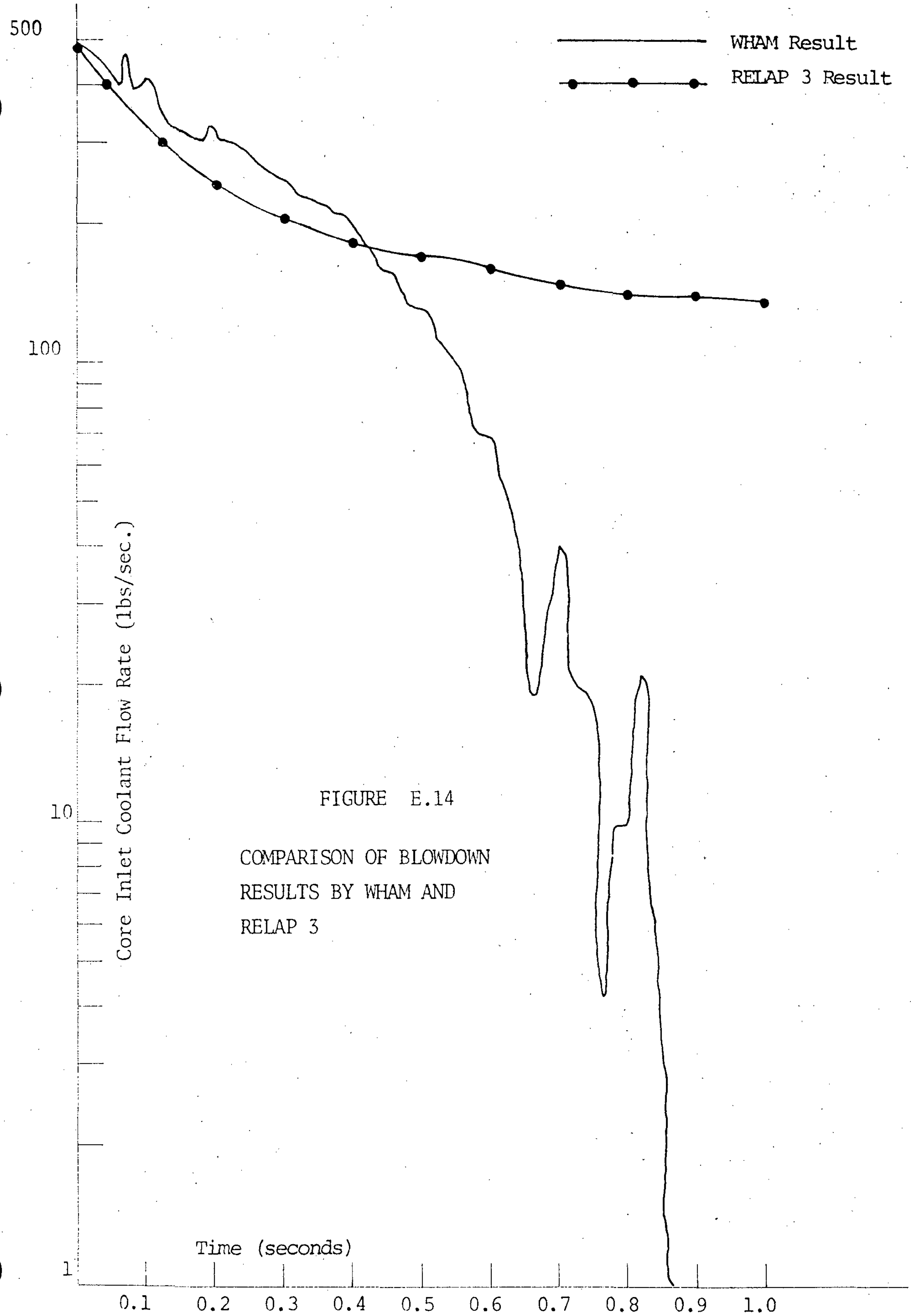


FIGURE E.14

COMPARISON OF BLOWDOWN RESULTS BY WHAM AND RELAP 3

Hydrodynamics equations are solved to determine coolant pressure, enthalpy, temperature, density and void fraction. Reactivity feedback due to coolant density change is also calculated. The various reactivity feedback effects are summed and furnished to the kinetics equations to determine the time dependence of reactor power. For problems in which the power is specified, as in this case, the reactivity feedback effects are neglected.

The PARET code represents a reactor core with up to four coolant channels. Each channel has its own power generation, coolant flow rate, hydraulic parameters, and each is separately weighted for reactivity feedback. It is possible to describe the overall performance of the reactor with up to four regions. Each channel may be divided into 20 axial sections (21 nodes) and up to 43 radial sections (44 nodes). A detailed description of the code is available in reference 3.

A time dependent inlet coolant temperature specification is not allowed in PARET. Therefore, the inlet temperature of the reverse flow due to natural circulation (152°F) was assumed as inlet temperature from the beginning of transient to meet the PARET requirement.

The control system will scram the reactor almost immediately after the rupture of the pipe due to the low pressure. The reactor power and coolant flow rate from RELAP-3 supplied to PARET are shown in Table E.3 and Table E.4, respectively.

The output from PARET as a function of the time includes all the important information for each node point such as coolant temperature, surface heat flux, heat transfer coefficient, burnout ratio, coolant quality, DNB surface temperature and radial temperature profile. The reactor power, coolant flow rate and maximum temperature of fuel centerline in the hot channel are shown in Figure E.15. Coolant flow rate, heat transfer coefficient, DNB surface temperature, and clad surface temperature (highest point in hot channel) are shown on Figure E.16.

RELAP-3 predicts that primary coolant flow at the inlet of the reactor core stops at 8.8 seconds and then reverses its direction due to natural convection forces. This reversed flow stops at 16.2 seconds due to the discontinuity of flow loop because there is no water left in in-pool heat exchanger. After 16.2 seconds the core coolant would be evaporating and condensing on the walls of the empty in-pool heat exchanger, however, it has been shown that the condensation capability is larger than the steam generation rate, therefore, the core will not become uncovered. The evaporation of coolant is accompanied by nucleate boiling in the core without exceeding DNB.

It can be concluded that the reactor core would maintain its integrity during the loss of coolant accident.

Useful results from PARET based on RELAP-3 input data are summarized below:

1.	Maximum fuel centerline temperature	283°F
2.	Cladding surface maximum temperature	277°F
3.	DNB surface temperature	288°F
4.	Maximum void fraction at exit of hot channel	0.98

In a comparative analysis the coolant blowdown results from the WHAM code and coolant reversed flow rate from the DAMOKE code (12) were used as input data to PARET. The time dependent reactor power was supplied by RELAP-3. The reactor is scrammed at 0.2 seconds due to the loss of system pressure. With the exception of the coolant flow rate and reactor power level, the input data to PARET of this study is the same as previous section, because the system condition before the accident is considered the same. Coolant flow rate and reactor power are shown in Table E.5 and Table E.6.

WHAM predicts that primary coolant flow at the inlet of reactor core stops at 0.86 seconds. Then, the flow would reverse its direction due to natural convection forces. This reversed flow rate is large enough to assure that no DNB would be present following the LOCA.

The reactor power, coolant flow rate and maximum temperature of the fuel centerline in the hot channel are shown in Figure E.17. Coolant flow rate, heat transfer coefficient, DNB surface temperature and maximum clad surface temperature are shown in Figure E.18.

The maximum clad surface temperature does not exceed 281°F.

According to the results from WHAM and PARET it can be concluded that no core damage will result. Only nucleate boiling will be present and the core will be completely protected by reversed flow through the in-pool heat exchanger.

The useful results from PARET based on WHAM input data are summarized as follows:

1.	Maximum clad surface temperature	281°F
2.	Fuel centerline temperature (maximum)	287°F
3.	DNB surface temperature	288°F
4.	Maximum void fraction at exit of hot channel	0.66

In summary, the results of the two blowdown computer codes input to the PARET code yield similar results and both studies conclude that although nucleate boiling occurs in the fuel, DNB and fuel damage will not result from a double-ended rupture of the reactor inlet primary coolant pipe.

TABLE E.3

Reactor Power Vs. Time (RELAP-3)

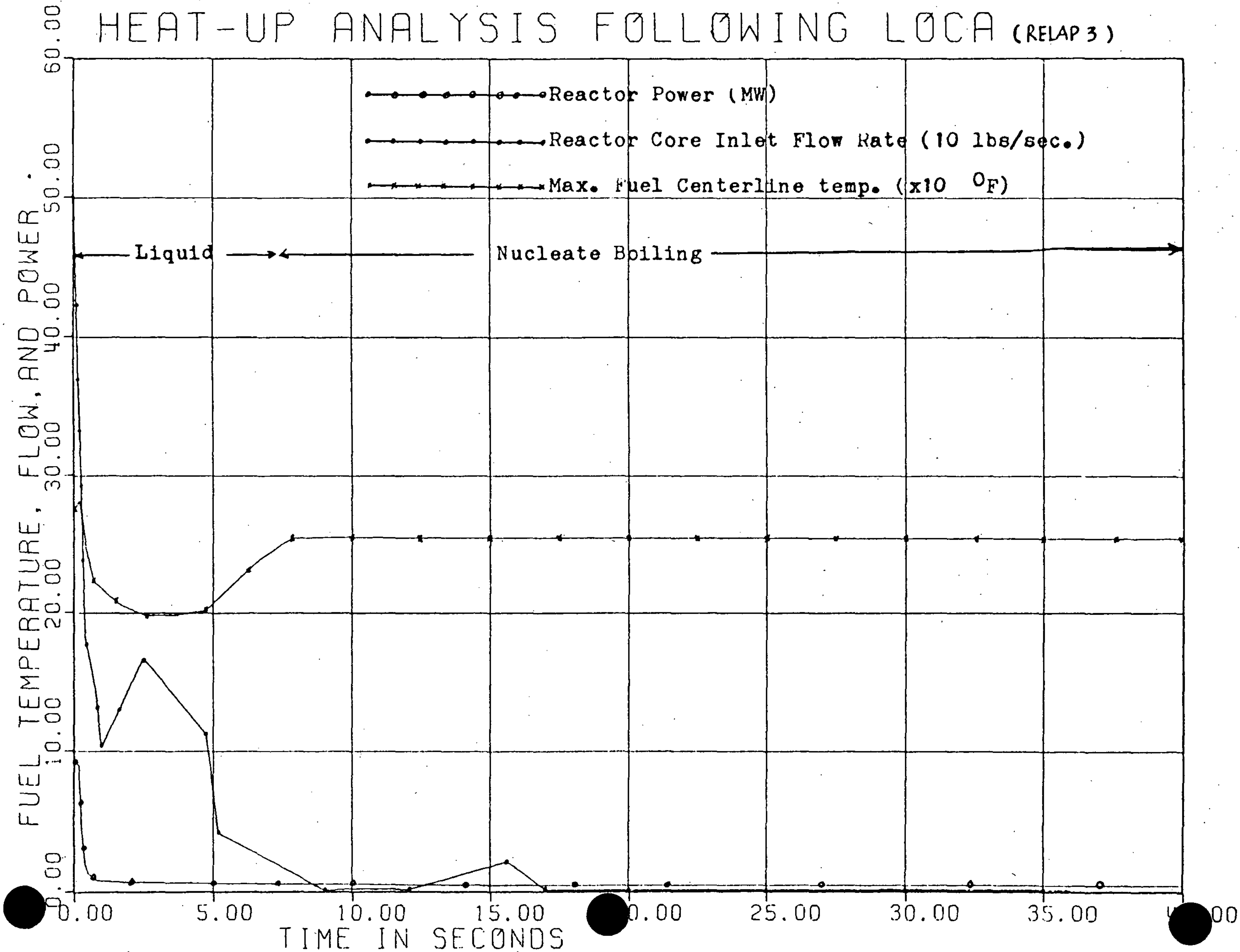
Time (sec)	Reactor Power (MW)
0.0	9.299999
0.1500000	9.049999
0.2500000	4.750000
0.3500000	2.190000
0.5000000	1.360000
0.8000000	0.850000
1.0000000	0.800000
1.5000000	0.760000
2.0000000	0.730000
3.0000000	0.679999
4.0000000	0.640000
5.0000000	0.619999
14.0000000	0.489999
20.0000000	0.460000
40.0000000	0.450000

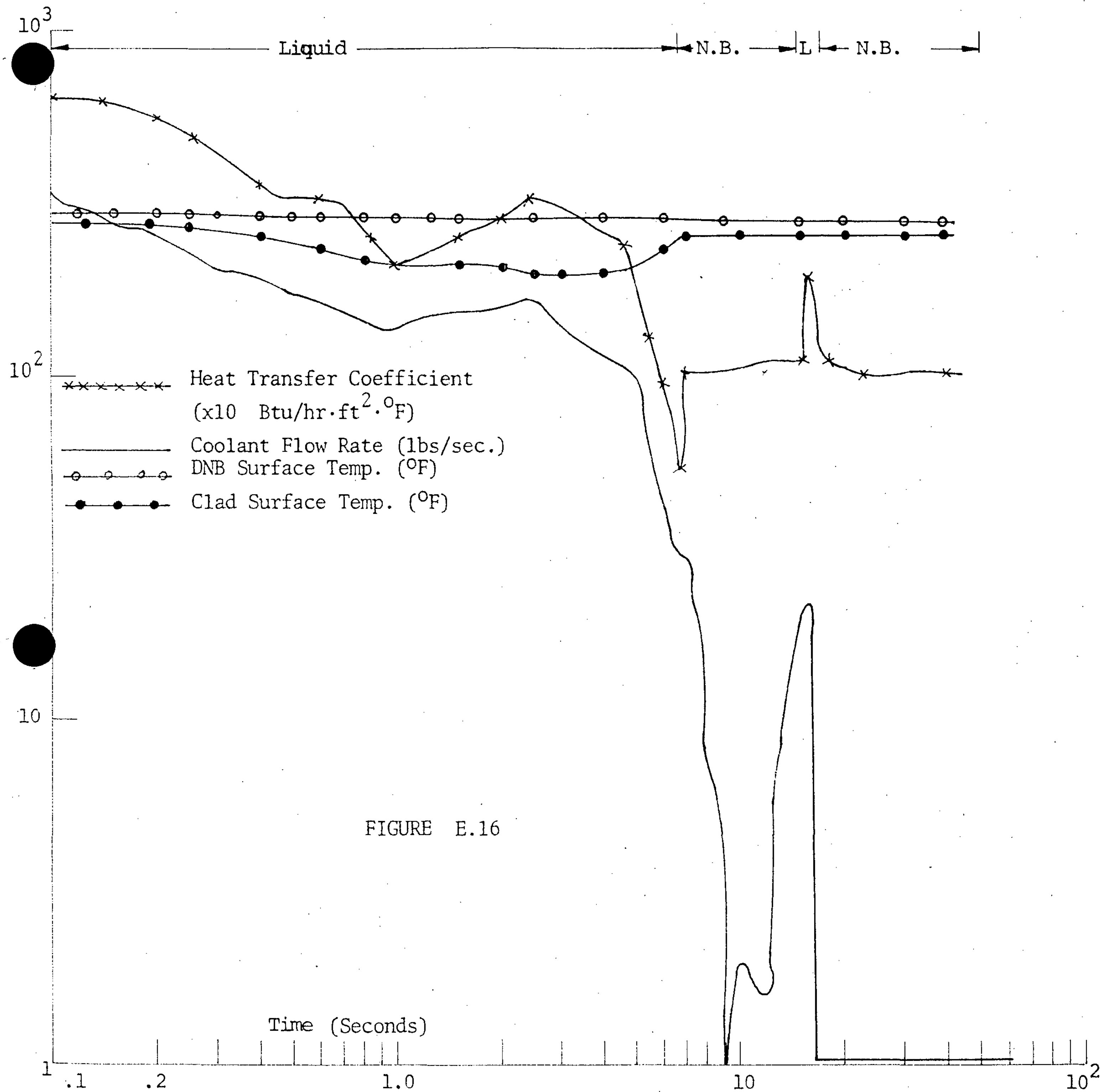
TABLE E.4

Coolant Flow Rate Vs. Time (RELAP 3)

Time (sec)	Mass Flow Rate (lb/hr/ft**2)	
0.0000000	5050000.00	
0.4000000	1842504.00	
0.8000000	1433059.00	
1.0000000	1023613.00	
2.5000000	1678726.00	
4.7500000	1125975.00	
5.2000000	409445.00	
8.5999999	51180.00	
9.0000000	10236.00	Flow
9.7999999	20472.00	Reverses
12.0000000	16377.00	
14.0000000	122833.00	
15.6000000	214958.00	
17.0000000	10236.00	
36.0000000	10236.00	
95.0000000	20472.00	

FIGURE E.15


HEAT-UP ANALYSIS FØLLØWING LØCA (RELAP 3)
Reactor Power (MW)
Reactor Core Inlet Flow Rate (10 lbs/sec.)
Max. Fuel Centerline temp. (x10 ^{O}F)
Liquid
Nucleate Boiling
FUEL TEMPERATURE, FLOW, AND POWER
60.00
50.00
40.00
30.00
20.00
10.00
0.00
5.00
10.00
15.00
25.00
30.00
35.00
TIME IN SECONDS



REACTOR CORE INLET FLOW RATE, HEAT TRANSFER COEFFICIENT, DNB SURFACE TEMP, AND CLAD SURFACE TEMP FOLLOWING LOCA BY RELAP 3

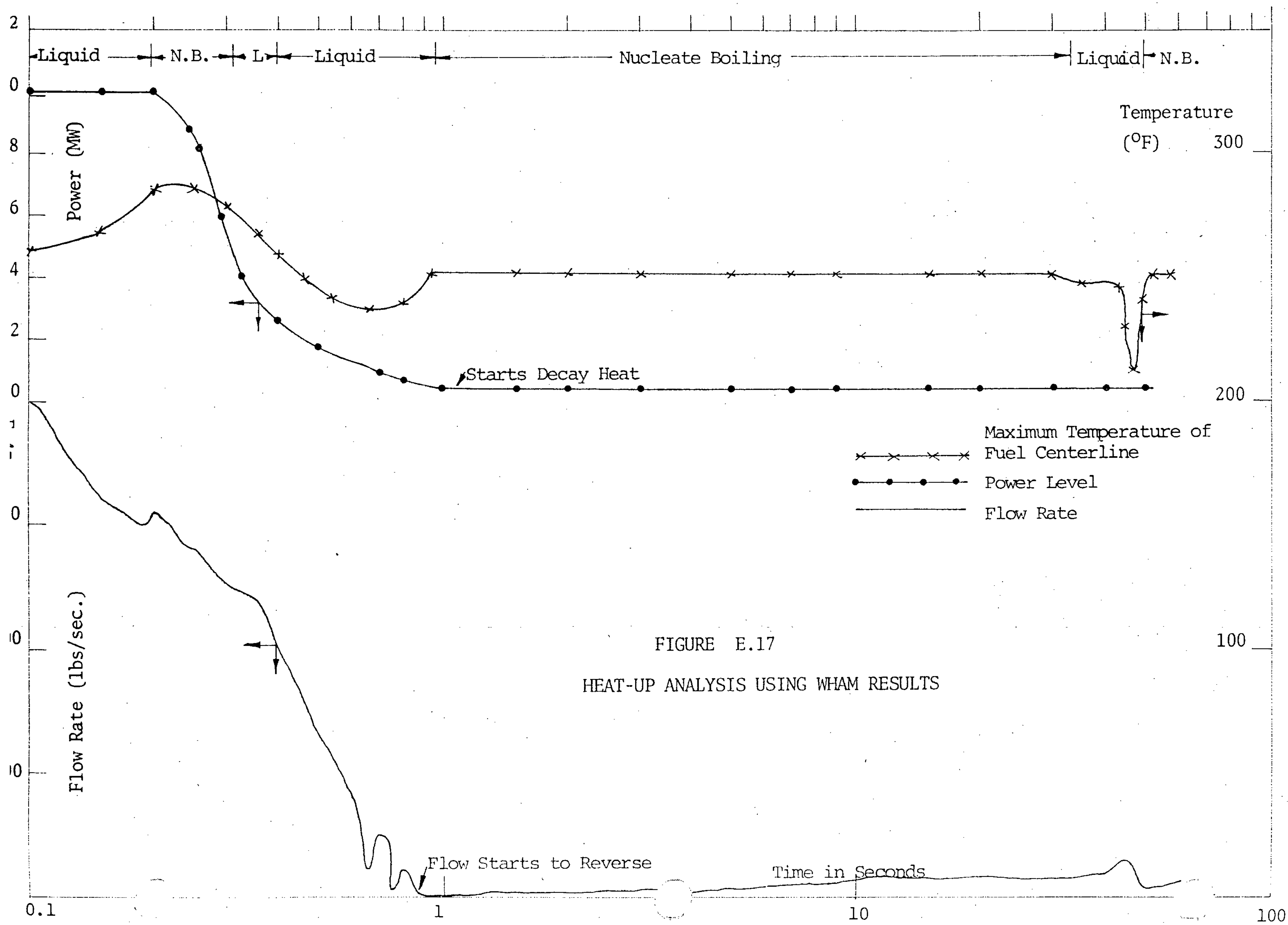
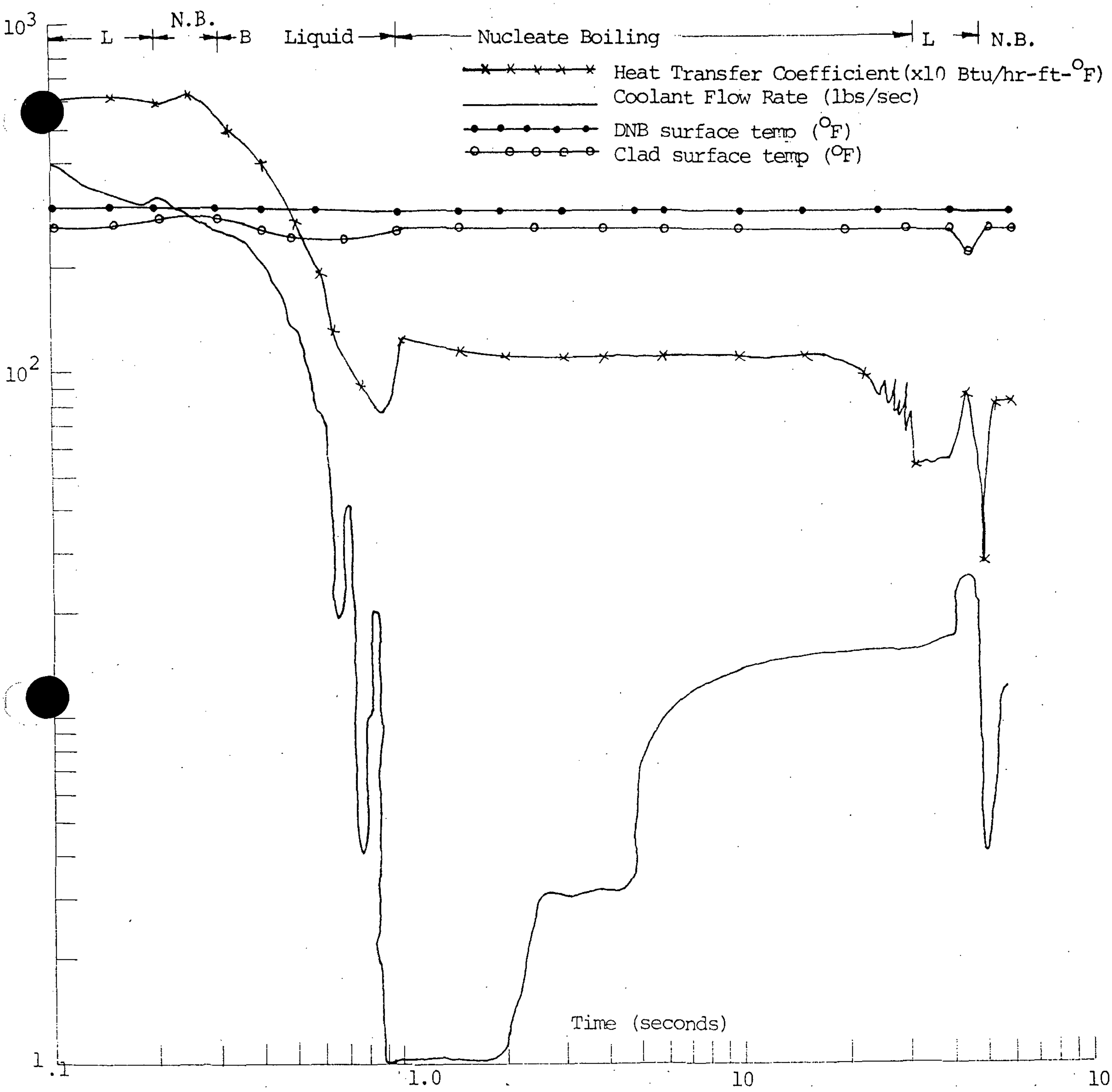


FIGURE E.17

HEAT-UP ANALYSIS USING WHAM RESULTS



Reactor core inlet flow rate, heat transfer coefficient, DNB surface temp and clad surface temp following LOCA by WHAM

FIGURE E.18

TABLE E.5
Coolant Flow Rate Vs. Time (WHAM)

Time (sec)	Mass Flow Rate (lb/hr/ft**2)	
0.0	5050000.00	
0.5000000E-01	4061885.00	
0.6999999E-01	4852247.00	
0.7999998E-01	4083840.00	
0.1100000	4303385.00	
0.1500000	3425198.00	
0.1799999	3139765.00	
0.1900000	3447154.00	
0.2000000	3161721.00	
0.2100000	3425198.00	
0.2200000	3139765.00	
0.3500000	2261509.00	
0.4800000	1405210.00	
0.5800000	724562.00	
0.6600000	197608.00	
0.7300000	329347.00	
0.7600000	43913.00	
0.8099999	109782.00	
0.8200000	219565.00	
0.8500000	21000.00	
1.0499999	8690.00	Flow
1.6500000	13388.00	Reverses
2.0499999	23254.00	
2.2500000	28186.00	
2.6500000	37581.00	
5.0499999	98651.00	
6.0499999	112744.00	
6.4500000	112744.00	
6.8499999	110395.00	
8.6500000	96302.00	
10.2500000	100999.00	
14.2500000	129186.00	
15.0500000	131534.00	
19.8499999	138581.00	
28.2500000	162069.00	
30.6499999	166767.00	
35.4500000	178511.00	
36.2500000	176162.00	
40.2500000	162069.00	
41.0499999	169116.00	
41.8499999	199651.00	
42.6499999	246627.00	
43.4500000	284209.00	
44.2500000	289320.00	
45.0499999	310046.00	
45.8499999	286558.00	
46.6499999	237232.00	
47.4500000	162069.00	
48.5000000	136232.00	
50.0000000	45976.00	
52.0000000	82209.00	
55.0000000	145627.00	

TABLE E.6

Reactor Power Vs. Time (WHAM)

Time (sec)	Reactor Power (MW)
0.0	10.000000
0.200000	10.000000
0.250000	8.900000
0.300000	5.599990
0.350000	3.400000
0.400000	2.500000
0.500000	1.700000
0.600000	1.400000
0.700000	1.099999
0.800000	00.900000
0.900000	0.700000
1.049999	0.605999
6.049999	0.568999
19.849999	0.498000
28.250000	0.4680000
36.250000	0.448999
48.500000	0.438000
95.000000	0.369999

E.6 References - LOCA

1. RELAP-3, W. H. Rettig, et al., A Computer Program for Reactor Blowdown Analysis, TID-4500, June 1970, Argonne Code Center #369.
2. WHAM, A Computer Program for Calculation of Pressure, Velocity, and Force Transients in Liquid Filled Piping Networks, Stanislav Fabic of Kaiser Engineers, Report Number 67-49-R, November 1967, Argonne Code Center #278.
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5. McAdams, W. H., Heat Transmission, McGraw-Hill, 1954, p. 172.
6. Glasstone, S. and A. Sesonske, Nuclear Reactor Engineering, Van Nostrand Reinhold Co., 1967, p. 101.
7. Waage, J. M., "Description of Calculational Methods and Digital Computer Codes for Analyzing Coolant Blowdown and Core Heat-Up Phenomena," Nuclear Safety, Vol. 8, No. 6, Nov-Dec 1967.
8. RELAP-2, W. H. Rettig, et al., IDO-17263, 1968.
9. CHIC-KIN, J. A. Redfield, A Fortran Program for Intermediate and Fast Transients in a Water-Moderated Reactor, WAPD-TM-479 (1965), Argonne Code Center #473.
10. Chung, J., "Geometrical Transformation from Pin to Plate Geometry," MURR Internal Report, June 1973.
11. Gruen, G. E., WHAM Prediction of Semiscale Test Results, IN 1431, October 1970.
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APPENDIX F

SAFETY LIMIT ANALYSIS FOR THE MURR

F. SAFETY LIMIT ANALYSIS FOR THE MURR

F.1 Introduction

The University of Missouri authorized the NUS Corporation to develop safety limit curves for MURR operation. These curves establish the maximum allowable power limits for safe operation for different combinations of measurable reactor operating variables. The measurable operating variables or process variables used in this study include reactor power, pressurizer pressure, and coolant temperature and flow rate. The safety limits presented herein provide the basis for determining the limiting safety system set points and operating limits required in submission of a Safety Analysis Report pursuant to a license for proposed MURR operation at 10 MW.

For any combination of the process variables, safe reactor operation is achieved by limiting the reactor power to a level which avoids either (1) subcooled boiling burnout (or departure from nucleate boiling) or (2) flow instabilities which can lead to premature burnout. Operation above this power limit can cause unpredictably high fuel and clad temperatures and consequential permanent fuel damage and fission product release to reactor coolant. This condition must be avoided for every core region and for every reactor operating condition.

All data used in the determination of the MURR safety limits were obtained from the MURR Hazards Summary Reports (1,2,3),* the MURR Design Data report (4), and the MURR hydraulic analysis (5).

F.2 Conclusions and Results

The results of the MURR safety limit analysis are summarized in Table F.1 and are plotted on Figures F.1 and F.2. The data presented are reactor thermal power limits for a range of measurable coolant conditions at the core inlet and at two pressurizer operating pressures. The criterion used to establish the safety limit on reactor power depends on the combination of the independent process variables. This can be seen by referring to Table F.1.

---

*Numbers in parenthesis refer to References in Section F.5.

TABLE F.1

SAFETY LIMITS FOR MURR OPERATION

Maximum Allowable Core Power Level, MW with Pressurizer at 60 psia

INLET WATER CONDITIONS

Temp. °F	400.	800.	1200.	1600.	2000.	2400.	2800.	3200.	3600.	4000.
	Flow Rate, GPM									
120.	3.011	5.870	7.980	9.843	11.574	13.099	14.426	15.450	16.217	16.654
140.	2.650	5.262	7.299	9.035	10.582	11.960	13.155	14.071	14.729	15.075
160.	2.292	4.546	6.675	8.202	9.600	10.822	11.877	12.669	13.228	13.501
180.	1.935	3.834	5.667	7.409	8.612	9.685	10.603	11.267	11.715	11.906
200.	1.583	3.131	4.615	6.009	7.282	8.400	9.301	9.863	10.204	10.267

Maximum Allowable Core Power Level, MW with Pressurizer at 75 psia

INLET WATER CONDITIONS

Temp. °F	400.	800.	1200.	1600.	2000.	2400.	2800.	3200.	3600.	4000.
	Flow Rate, GPM									
120.	3.278	6.334	8.647	10.742	12.668	14.435	16.050	17.394	18.532	19.438
140.	2.916	5.798	7.939	9.906	11.667	13.282	14.746	15.967	16.993	17.787
160.	2.556	5.080	7.317	9.067	10.676	12.138	13.458	14.534	15.437	16.139
180.	2.197	4.363	6.474	8.236	9.680	10.988	12.152	13.104	13.892	14.467
200.	1.843	3.656	5.415	7.099	8.686	9.845	10.868	11.689	12.339	12.810

NOTE: Underlined power levels are limited by bulk boiling.

TABLE F.2

SUMMARY OF MURR HOT CHANNEL FACTORS

<u>On Enthalpy Rise</u>

Power-related Factors		
Nuclear Peaking Factors		
Radial	2.220	
Local (Circumferential)	1.040	
Nonuniform Burn-up	1.112	
Axial	1.000	
Engineering Hot Channel Factors on Enthalpy Rise		
Fuel Content Variation	1.030	
Fuel Thickness/Width Variation	1.030	
Overall Product		2.72
Flow-Related Factors		
Core/Loop Flow Fraction	1.000	
Assembly Minimum/Average Flow Fraction	1.000	
Channel Minimum/Average Flow Fraction		
Inlet Variation	1.000	
Width Variation	1.000	
Thickness Variation	1./1.080	
Within Channel Minimum/Average Flow Fraction		
Thickness Variation	1./1.050	
Effective Flow Area	0.3231/0.3505	
Overall Product		0.81

<u>On Heat Flux</u>

Power-Related Factors		
Nuclear Peaking Factors		
Radial	2.220	
Local (Circumferential)	1.040	
Nonuniform Burn-up	1.112	
Axial	1.432	
Engineering Hot Channel Factors on Flux		
Fuel Content Variation	1.030	
Fuel Thickness/Width Variation	1.150	
Overall Product		4.35
Energy Fraction Generated in Fuel Plate	0.930	

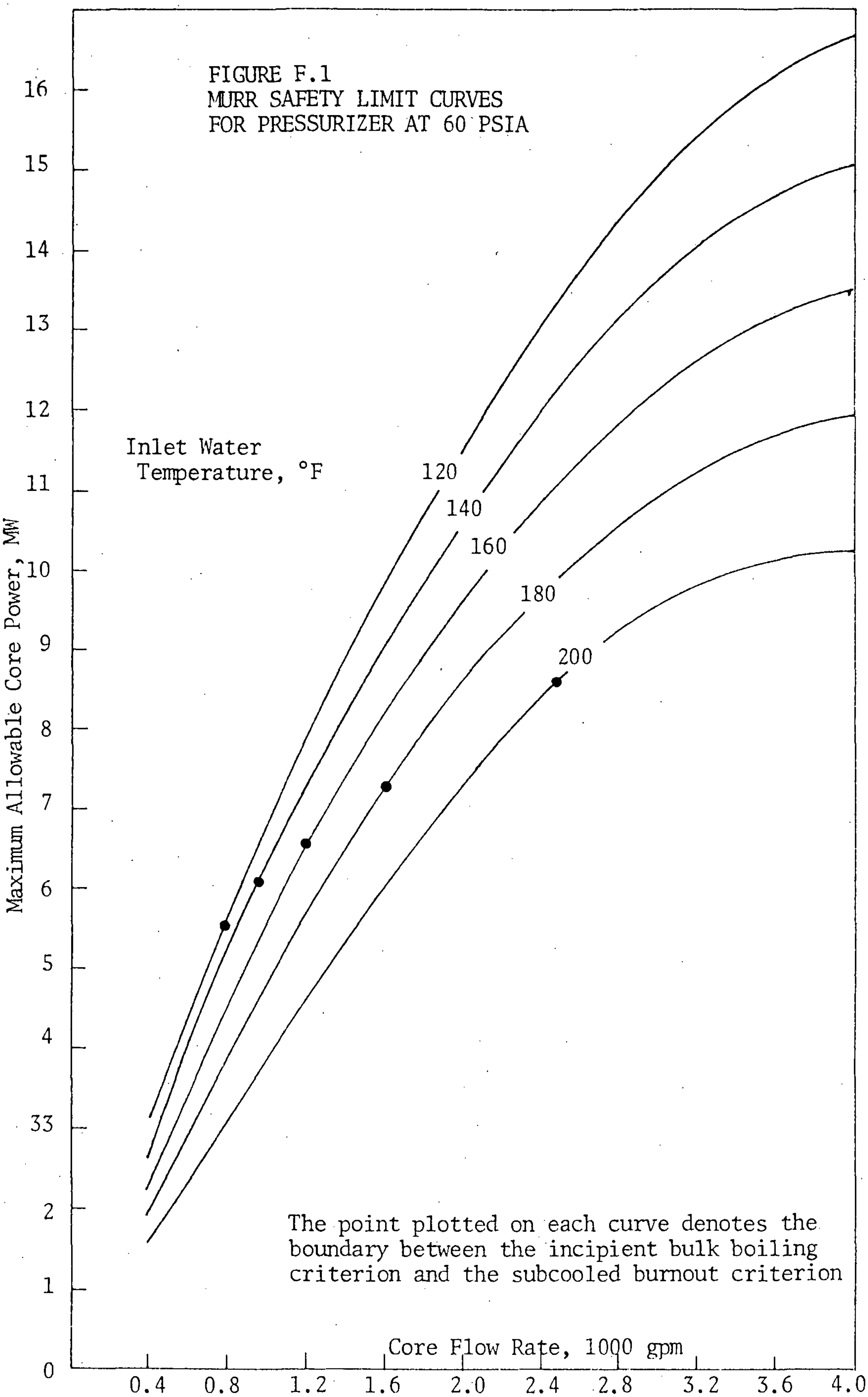
FIGURE F.1
MURR SAFETY LIMIT CURVES
FOR PRESSURIZER AT 60 PSIA
Inlet Water
Temperature, °F
120
140
160
180
200
Maximum Allowable Core Power, MW
16
15
14
13
12
11
10
9
8
7
6
5
4
33
2
1
0
The point plotted on each curve denotes the boundary between the incipient bulk boiling criterion and the subcooled burnout criterion
Core Flow Rate, 1000 gpm
0.4
0.8
1.2
1.6
2.0
2.4
2.8
3.2
3.6
4.0

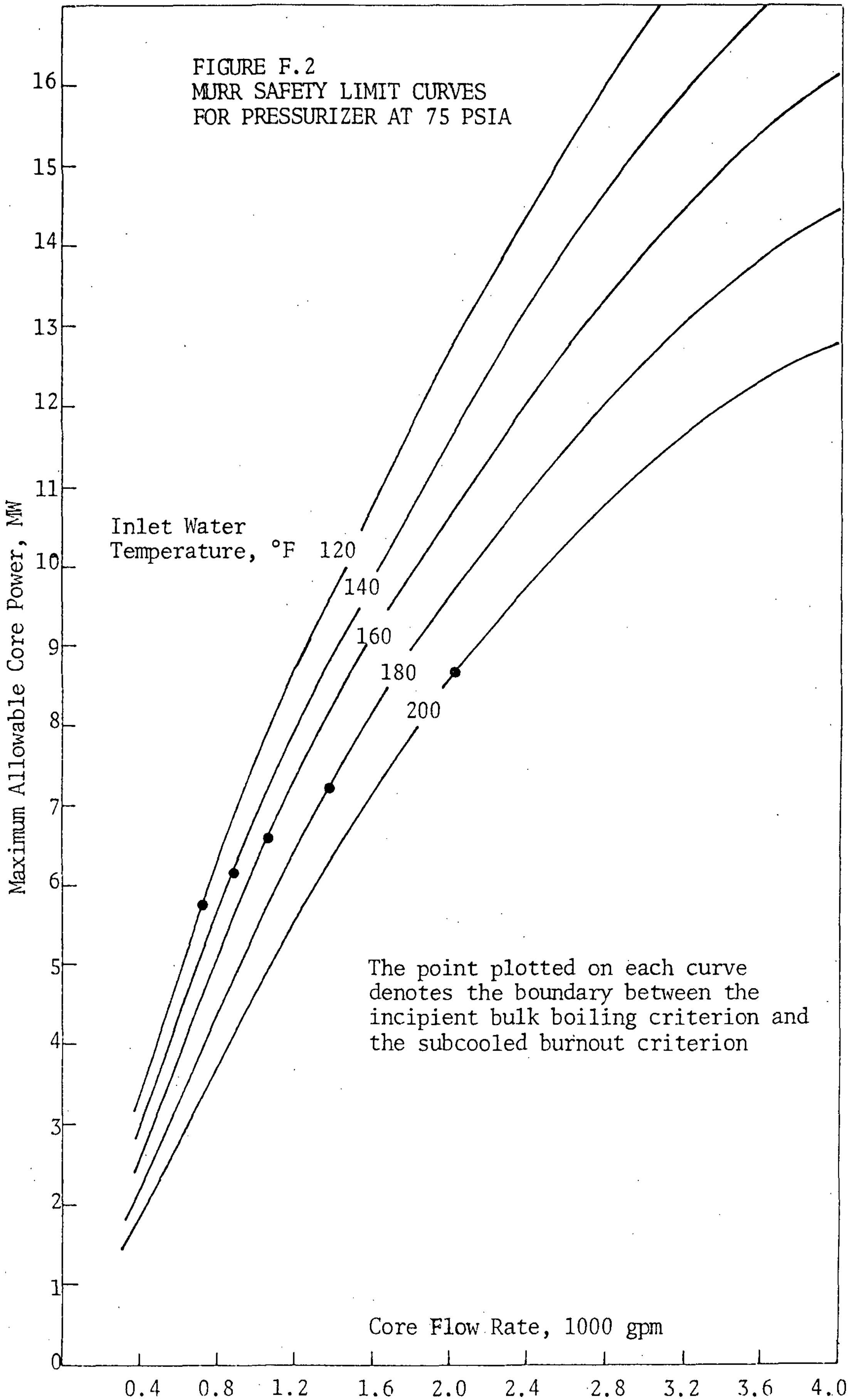
FIGURE F.2
MURR SAFETY LIMIT CURVES
FOR PRESSURIZER AT 75 PSIA
Maximum Allowable Core Power, MW
Inlet Water
Temperature, °F 120
140
160
180
200
The point plotted on each curve denotes the boundary between the incipient bulk boiling criterion and the subcooled burnout criterion
Core Flow Rate, 1000 gpm
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
0.4
0.8
1.2
1.6
2.0
2.4
2.8
3.2
3.6
4.0

The underscored table entries are the power limits as established by the criterion of avoiding any bulk boiling of the coolant, whereas the remaining entries reflect the thermal limits established by the subcooled burnout criterion. The safety limit criterion on incipient bulk boiling of the coolant is associated with experimentally observed premature burnout caused by hydraulic instabilities. In the present study, the power limits for coolant flow rates greater than 2800 gpm are always dictated by the burnout criterion, while for flow rates less than 800 gpm the incipient bulk boiling criterion dictates the safe power level.

Table F.2 presents a summary of hot channel factors used in the analysis. The limiting channel (or hot channel) used as the basis for the safety limit analysis has a power level 2.72 times the average and a flow rate of 0.81 times the average. The safety limits given in Table F.1 and Figures F.1 and F.2 implicitly depend on these power and flow-related factors. Any future changes in these factors will require a corresponding change to the power limit results of this study. Changes to power-related factors can be treated in a straightforward manner; namely, by maintaining the product of the limiting power and the affected power factor equal for both the new and referenced condition. Corresponding changes to flow-related factors are more difficult to accommodate, because of the nonlinear dependence of the limiting power on the core flow rate. The effect of this nonlinearity is to introduce a proportionately greater change in the limiting power level than the change in the flow-related factor. For small changes in flow (not to exceed 5%), it is possible to estimate the new limiting power from the slope of the power-flow curve (Figure F.1 or F.2) for the desired operating conditions. Larger changes in the flow-related factors will require a reevaluation of the safety limits.

F.3 Method of Analysis

The method for evaluating the core power limits for Table F.1 are discussed below. The details for selecting the safety limit criteria, and for using the BOLERO (6) computer program are included.

F.3.1 Safety Limit Criteria

The study objective was to determine core power limits for safe operation at specified combinations of possible core operating conditions. Safe operation here is defined to mean avoiding burnout (or DNB) where excessive fuel or clad temperatures could cause clad failure and thereby release fission products into the primary coolant. To avoid DNB, the heat flux at each local section in the core is maintained at a value less than the locally-evaluated DNB heat flux. It is also necessary to avoid any core operating conditions (such as hydraulic instability) that could prematurely reduce the DNB heat flux. The following discussion presents the basis for specifying criteria to include both possibilities.

The MURR fuel geometry (near rectangular channels in a closed matrix) and the MURR operating conditions (subcooled water near atmospheric pressure) are outside the normal range of interest for today's commercial reactors. Consequently, only a limited amount of experience is available for establishing safety limit criteria. Fortunately, however, the MURR fuel assembly geometry is similar to the Advanced Test Reactor (ATR) fuel element so that ATR experience (8,9) can be applied to MURR. Since the MURR fuel channel length (~ 24") is about one-half that of ATR, the use of ATR test results can, in fact, provide conservatism for MURR because investigators (10) have shown higher or equal burnout heat flux levels for shorter channel length. Similarly, the shorter channel lengths are less susceptible to the hydraulic instabilities related to incipient bulk boiling.

Other test reactors (HFIR, ETR) have design and operating conditions that depart further from the MURR conditions, and their test results were not directly useful in developing the MURR safety criteria.

Preliminary ATR testing (8) indicated that both subcooled burnout and bulk boiling burnout can occur for the range of channel thicknesses then under design consideration. Tests were performed at Argonne in 1963 on three channel thicknesses (0.054", 0.072", 0.094"), and it was found that for the two thinnest channels (0.054", 0.072") the burnouts were due to hydraulic instability (or autocatalytic vapor binding) when the coolant reached saturation at the channel exit. Presumably, the hydraulic instabilities led to subnormal flow conditions and a lower burnout heat flux. Subcooled burnout occurred for the 0.094 inch channel before the coolant reached saturation conditions at channel exit. The subcooled burnout heat flux data obtained in these tests were 0.6 of the burnout heat flux predicted by the Bernath correlation (7):

$$\phi DNB = h_{bo} (T_{bo} - T_{sat} + \Delta T_{sub})$$

where:

$$h_{bo} = 10890. \left[\frac{D_e}{D_e + D_i}\right] + \frac{48}{D_e^{0.6}} \cdot V$$

$$T_{bo} = 1.8 \left[57 \ln P - 54 \left(\frac{P}{P + 15}\right) - \frac{V}{4}\right] + 32.$$

T_{sat} = saturation temperature at P, °F

ΔT_{sub} = bulk water temperature, degrees subcooling, °F

D_e = wetted hydraulic diameter, ft

D_i = heated hydraulic diameter, ft

V = coolant velocity, fps

P = system pressure, psia

Subsequent full-scale ATR testing (9) at Battelle Northwest with a channel thickness of 0.070" confirmed the earlier test results; namely, that burnout induced by hydraulic instability was the limiting factor for ATR. In addition, it was established that the hydraulic instability condition did not correspond to initiation of local boiling, but to the beginning of bulk boiling at the channel exit in the region where the coolant enthalpy was highest. Test results also indicated that lateral mixing (in the channel) was quite small.

In view of the ATR experience, and in absence of burnout test results for MURR fuel and at MURR operating conditions, the following safety limit criteria were adopted for this study:

> The coolant exit temperature from the hot channel shall be less than the saturation temperature at the core exit pressure.
>
> The local heat flux at any point in the core shall be less than 0.5 of the burnout heat flux as given by the Bernath correlation at that point.

The bulk boiling limitation is adopted to exclude occurrence of the in-core hydraulic instabilities related to incipient bulk boiling. The above burnout heat flux limitation is adopted to provide some additional design safety margin by a reduction of the correlated ATR test data by the factor 0.5/0.6 relative to the original Bernath correlation. The above criteria are sufficient to preclude the possibility of fuel failure and attendant fission product release due to excessive temperatures.

F.3.2 Calculational Method

The BOLERO program was used to perform the calculations which determine local conditions of enthalpy, heat flux, and DNB heat flux for the core hot channel. Since the Bernath burnout heat flux depends on absolute pressure, it was necessary to calculate the absolute pressure at the core exit for each set of inlet water conditions and core power. Since most BOLERO input is dependent on absolute pressure and on either flow rate or power, a special computer program MURRPGM, was written to generate consistent input for all the cases needed for the study. A description of the MURRPGM program, the basis for BOLERO input, and the treatment of BOLERO results are presented below.

F.3.2.1 MURRPGM Program

The MURRPGM program was developed to calculate the absolute pressure (psia) at the core outlet for every combination of operating conditions in this study. Since the core outlet pressure calculation required the same data as BOLERO, the program was expanded further to generate input cards for the BOLERO program.

The pressure drop from the pressurizer to the core outlet was calculated by correcting individual Δp components as given in reference (5) to new flow, temperature, and core power conditions (see Table F.3). The new Δp components were then totaled and the result was subtracted from the desired pressurizer operating pressure (60 psia or 75 psia) to obtain the absolute pressure at the core outlet.

The method for correcting the reference Δp components depended on the type of pressure drop involved. For non-frictional components, pressure drop is proportional to density and flow,

$$\Delta P = \Delta P_o \left(\frac{\rho\,(T)}{\rho\,(T_o)}\right)^{1.0} \left(\frac{Q}{Q_o}\right)^{2.0}$$

where the subscript o denotes the reference conditions as given in Table F.3.

TABLE F.3

REFERENCE PRESSURE DROP DATA*

COMPONENT**	ΔP_0(PSI)	Q_0(GPM)	T_0(F)	FRICTIONAL	IN CORE
1,2,3	3.259	1800	155	yes	no
4	0.2689	1800	155	no	no
5,6,...10	4.08	3600	155	yes	no
11	0.1977	3600	155	yes	no
12	0.8980	3600	155	no	no
13	12.35	3600	165	yes	yes

* Data from Reference (5)

** Component description using notation of reference (5)

1. Across pressurizer surge line to pressurizer outlet
2. Across 5 feet of 8 inch pipe
3. Across 8 inch Y strainer
4. Across 8 inch/12 inch expansion
5. Across 80 feet of 12 inch pipe
6. Across four 12 inch 90 degree elbows
7. Across three 12 inch 45 degree elbows
8. Across one 12 inch butterfly valve (507B)
9. Across one 12 inch swing check valve (502)
10. Across entrance to annular pressure vessel
11. Across 6 feet of annular pressure vessel
12. Across entrance to fuel element plates
13. Across Core ...25.5 inches of fuel element plates..to core exit

For the frictional loss components, the pressure drop was assumed to be given by the Blasius equation and,

$$\Delta P = \Delta P_o \cdot \left(\frac{\rho\,(T)}{\rho\,(T_o)}\right)^{0.8} \left(\frac{Q}{Q_o}\right)^{1.8} \left(\frac{\mu\,(T)}{\mu\,(T_o)}\right)^{0.2}$$

If the core pressure drop component was involved, then the temperature T in the above equation was taken as the average core temperature calculated from the core power and flow. Otherwise the value for T was the core inlet water temperature.

MURRPGM also includes:

An iterative scheme to determine the core power level that would cause incipient bulk boiling at the hot channel exit.

Interpolation routines to evaluate intermediate fluid property values from tabulated input values using absolute pressure as the independent variable.

Simple transformation to generate BOLERO input from non-standard BOLERO flow and power units.

F.3.2.2 BOLERO Input

The BOLERO program performs all necessary thermal-hydraulic calculations required to establish the minimum ratio of the local burnout heat flux to the local surface heat flux (DNBR) for a single coolant channel. BOLERO input specifies the single channel dimensions, operating conditions, and the Bernath DNB correlation and its parameters.

The single channel analyzed in BOLERO is a representation of the thermally limiting channel (or hot channel). The channel power is 2.72 times average channel power, and the channel flow rate is 0.81 times average channel flow rate. The basis for these data and for the local heat flux multipliers are given in Table F.2. The normalized axial power distribution used for the channel is given in Figure 1 of TM-WRP-62-10 contained in reference (4). This power distribution occurs at beginning core life when the control rods are partially inserted and represents the most limiting condition during core life due to the high flux level at the channel exit. Channel dimensions are developed from nominal core dimensions such as flow area

($0.3505\ ft^2$), heat transfer surface area ($184.28\ ft^2$) and core length (2.0 ft). The effects of worst-case dimensions are included in the corresponding hot channel factors.

BOLERO input data for the Bernath DNB correlation include a DNB heat flux multiplier (0.5), a heated-to-wetted perimeter ratio (0.924) and a saturation temperature corresponding to the absolute pressure at the core exit (available from the MURRPGM program results) for each core power, pressurizer pressure, and core inlet condition. This approach ensures the correct Bernath DNB heat flux when the minimum DNBR occurs at the channel exit, and produces a conservative result when the minimum DNBR occurs elsewhere in the channel.

F.3.2.3 BOLERO Output

The maximum core power levels summarized in Table F.1 were limited by either the bulk boiling or DNB heat flux criterion. Those values limited by bulk boiling (underscored values in Table F.1) were immediately evident because BOLERO results indicated that

$$DNBR = \frac{\Phi DNB}{\Phi LOCAL} > 1.0$$

for the initial core power estimate evaluated by the MURRPGM program at the threshold of bulk boiling. No further iterative procedure was required because any core power increase to reach the DNB flux limit would also violate the bulk boiling criterion.

The core power levels limited by the DNB criterion were the result of an iterative procedure. The procedure included the sequential use of the MURRPGM program to calculate the absolute pressure at core exit and the BOLERO program to calculate the DNBR. The DNB-limited power levels in Table F.1 were determined by terminating the iteration procedure when the DNBR = 1.0000 + 0.01.

F.4 Discussion of Results

Figures F.1 and F.2 illustrate the effects of core operating conditions on the maximum allowable core power for safe MURR operation. The trends noted here generally represent the behavior of the two design criterion for various core operating conditions.

The variable most strongly affecting safe core operation is core flow rate. The higher the core flow rate, the higher the maximum allowable core power level. The effect is essentially linear at low core flow rates where the bulk boiling criterion is controlling and becomes nonlinear as the flow rate is increased into the DNB controlled regions. The nonlinearity in the safety limit is more pronounced for higher inlet water temperatures. Two competitive coolant flow related phenomena are responsible for this observed behavior. An increase in the coolant flow rate results in (1) lower absolute pressures at core exit which, in turn, decreases the water saturation temperature and thereby decreases the Bernath burnout heat flux limit; and (2) higher predictions of the Bernath burnout heat flux limit with increasing coolant velocity.

The allowable core power limit is inversely related to the core inlet water temperatures. This is readily understood in terms of a higher permissible core power level for an increased inlet subcooling; that is, the channel power to achieve incipient bulk boiling or local burnout increases as the inlet subcooling increases (coolant inlet temperature decreases) with all other variables held constant.

The effect of pressurizer pressure is available from a comparison of corresponding curves on Figures F.1 and F.2. Clearly, higher pressurizer pressure results in an increase in the safety limits on core power due to the increase in the coolant saturation temperature and the pronounced absolute pressure dependence of Bernath correlation at low absolute pressure. As already noted, the influence of the coolant flow rate on the channel exit pressure and the dependence of the Bernath correlation on absolute pressure is responsible for the slope change observed in the safety limit curves of Figures F.1 and F.2.

F.5 References

(1) "MURR Hazards Summary Report," University of Missouri Research Reactor Facility (1965).

(2) "MURR Hazards Summary Report," Addendum 1, University of Missouri Research Reactor Facility (1966).

(3) "MURR Hazards Summary Report," Addendum 3, University of Missouri Research Reactor Facility (1972).

(4) "MURR Design Data, Volume I" (copy 86) by Internuclear Company (1962).

(5) "Hydraulic Analysis of the MURR Primary Cooling System - 10 MW Operation," 1973.

(6) Schmidt, E. R., Couchman, M. L., and Edwards, D. R., BOLERO-II, Burnout Limit Evaluation Routine-II, NUS-TM-ENG-119 (Rev. 3, modified).

(7) Bernath, L., "A Theory of Local-Boiling Burnout and Its Application to Existing Data," Chem Eng. Progm. Symp. Ser., 56, No. 30, 95-116 (1960).

(8) Croft, M. W., "Advanced Test Reactor Burnout Heat Transfer Tests," USAEC Report IDO-24475, Babcock and Wilcox Co., January 1964.

(9) Waters, E. D., "Heat Transfer Experiments for the Advanced Test Reactor," USAEC Report BNWL-216, Battelle-Northwest, May 1966.

(10) Tong, L. S., Boiling Crisis and Critical Heat Flux, USAEC Office of Information Services (1970) page 26.

APPENDIX G

SAFETY ANALYSIS OF INCREASED SOURCE STRENGTH OF THE ANTIMONY-BERYLLIUM NEUTRON SOURCE

G. SAFETY ANALYSIS OF INCREASED SOURCE STRENGTH OF THE ANTIMONY-BERYLLIUM NEUTRON SOURCE

G.1 Introduction

The antimony-beryllium neutron source was originally that of a startup source. However, due to the long period of reactor operation and the resultant inventory of activation products (i.e., in structural material), the photoneutron reaction in the beryllium reflector dominates the Sb-Be source in total neutron production by several orders of magnitude. At the present time the source is stored in the deep pool. It is used primarily for subcritical multiplication measurements during the loading of depleted fuel for shipment. The increase in source strength is requested to improve counting statistics for subcritical measurements. Safety considerations with respect to the increased source strength will be analyzed in subsequent paragraphs.

G.2 Source Leakage

The present neutron source was manufactured by Monsanto Company. It consists of 138.4 grams of antimony and 39.03 grams of beryllium. The compressed powder mixture of these materials is doubly encapsulated in 304 stainless steel. All seams, etc. are T.I.G. fusion welded. The inner container was leak tested using the standard bubble test. The outer capsule was placed under 500 psi pressure and then leak tested by helium mass spectrograph methods. Sensitivity of the latter test is 10^{-8} cc/sec of leakage.

If a leak developed, it would be detected during the weekly pool water analysis. Detection limits for Sb-122 and Sb-124 are 5×10^{-6} µCi/ml and 10^{-6} µCi/ml, respectively. Thus, a leaking source would be detected well before the allowable restricted area water concentrations of appendix B of 10CFR 20 are reached. The pool volume is approximately 9.8×10^{7} ml. Therefore, a 10^{-6} µCi/ml concentration of Sb-12 would imply an inventory of only 98 µCi in the total pool volume.

G.3 Inadvertent Removal from Deep Pool Storage

Administrative controls exist to prevent inadvertent removal of the source from deep pool storage. The source location is described in a sample log located on the control console. The handling line connected to the source is tagged. The tag indicates the approximate source activity at the time of removal from its last irradiation. Health Physics surveys are conducted during movement of the source out of deep pool storage. Health Physics surveys are also required any time that the pool level is lowered. These controls serve to limit adequately the possibility of personnel exposure.

G.4 Sudden Capsule Rupture

The previous discussion showed that a small leak can be readily detected before any hazardous condition arises. A sudden capsule rupture which could release large amounts of activity will now be examined. It will be shown that both external and internal forces cannot credibly cause a large activity release.

The first source of capsule damage to be examined will be a rupture due to an external force, i.e., capsule damage due to impact with a large object. During manufacture of the source, the containment was subjected to a 500 psi overpressure with no loss of integrity. While the source is being irradiated in the beryllium reflector source hole, it is relatively isolated. It is highly improbable that any object large enough to cause damage would inadvertently fall on the source. When in use, careful sample handling techniques ensure that source will not be damaged.

The second cause of containment loss to be examined will be a rupture due to a buildup of internal pressure. Analysis supplied with the source implies a purity of greater than 99.5% in antimony and beryllium. The remaining 0.5% impurities are all metallic. Thus, material decomposition is not a problem. The doubly-encapsulated compressed powder mixture is 80% of theoretical density. Thus, there are a large number of internal vacancies where any gaseous irradiation products could collect before any internal pressure increase would be observed.

To get an estimate on the magnitude of pressures that could be produced, assume that for each disintegration of Sb-124 there is one atom of gas produced in the beryllium. For each 100 Curies of Sb-124 activity this implies that 3.7×10^{12} atoms are born. This is equivalent to 6.14×10^{-2} moles of gas. The 80% theoretical density implies that there is a free volume 3.3 cc. At atmospheric pressure 6.14 x 10^{-2} moles of gas would occupy 1.38 x 10^{-7} cc. Thus, internal pressure buildup due to gaseous release is not a credible problem.

Tipson (1) points out that beryllium has been irradiated to neutron fluences of 1.8 x 10^{20} nvt with no dimensional changes. To achieve this nvt, the source would have to be irradiated for approximately 13 weeks. This length of irradiation far exceeds that required to produce the proposed level of activity. No specific data was found for antimony. However, since there are no direct gaseous products, its expansion should be no more than that of beryllium. Therefore, pressure buildup due to volumetric expansion is also negligible.

G.5 References

(1) Tipton, C. R., Reactor Handbook - Materials, Interscience, 1960, p. 911.

APPENDIX H

BASES FOR LIMITING SAFETY SYSTEM SETTINGS FOR MODES I AND II OPERATION

H. BASES FOR LIMITING SAFETY SYSTEM SETTINGS FOR MODES I AND II OPERATION

The limiting safety system settings (LSSS) proposed for Modes I and II, i.e., 10 MW and 5 MW operation respectively, of the MURR are as follows. For reactor power level the LSSS is 125% of full power for both modes thus the highest powers obtainable before a reactor scram would be 12.5 MW (1.25 X 10 MW) in Mode I and 6.25 MW (1.25 X 5 MW) in Mode II. For both modes, the LSSS on pressure is a minimum of 75 psia in the pressurizer, and the LSSS on primary coolant core inlet temperature is a maximum of 155°F. The LSSS on primary coolant flow for Mode I operation is a minimum of 1625 gpm in either of the parallel coolant loops. The same LSSS of 1625 gpm applies for the single operating loop in Mode II operation. Since 50 gpm of the primary coolant flow is diverted to the cleanup system, the actual core flow rates at the LSSS are 3200 gpm and 1575 gpm in Modes I and II, respectively.

Appendix F of this report presents parametric curves for the conditions which would lead to departure from nucleate boiling (DNB) and resulting fuel damage. From that analysis, Figure H.1 depicts the DNB conditions for the LSSS on pressurizer pressure of 75 psia. From this curve one may predict the safety margin for several anticipated transients.

Case one postulates a severe power transient with primary flow and pressure already reduced to their LSSS value in Mode I operation. Figure H.1 predicts that the temperature LSSS of 155°F could not be reached until the reactor power has risen to 14.75 MW, or 2.25 MW above the reactor power scram point, thus an ample safety margin exists for safety reaction time required to prevent reaching the DNB threshold.

Case two postulates steady state Mode I operation of the reactor with flow and pressure again reduced to their LSSS and reactor power at the LSSS of 12.5 MW. Figure H.1 predicts that DNB would not occur until a core inlet temperature of approximately 185°F was obtained. The safety margin is thus 30°F above the LSSS of 155°F on core inlet temperature. Primary coolant temperature increase would be slow, so little or no margin is required for safety system reaction time. Frequent compliance checks and past operating history provide confidence that the primary coolant temperature measurement error is no greater than ± 5°F. Therefore there is excess safety margin for a temperature transient of this type.

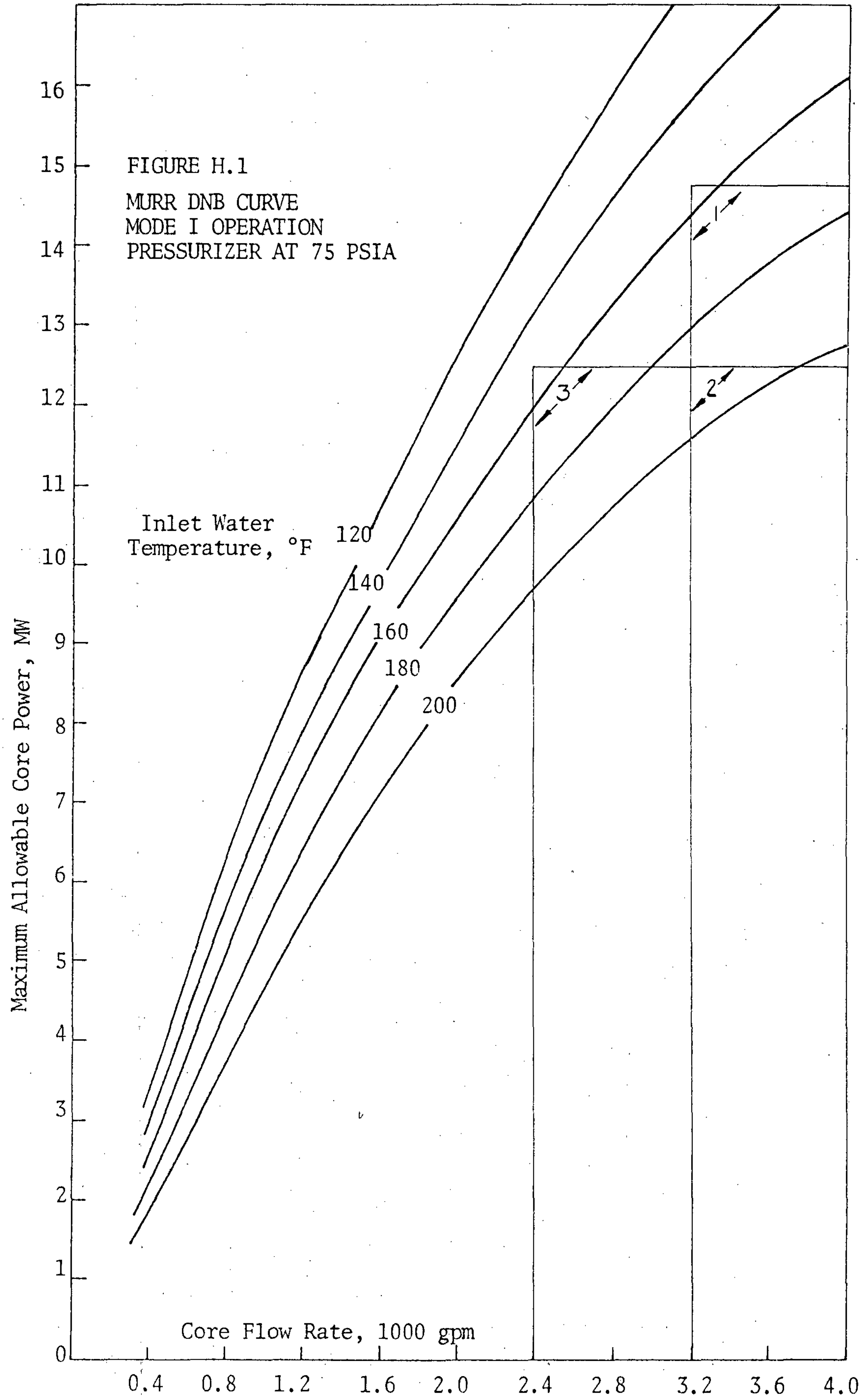
FIGURE H.1
MURR DNB CURVE
MODE I OPERATION
PRESSURIZER AT 75 PSIA
Inlet Water
Temperature, °F
120
140
160
180
200
1
2
3
Maximum Allowable Core Power, MW
Core Flow Rate, 1000 gpm
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
0.4
0.8
1.2
1.6
2.0
2.4
2.8
3.2
3.6
4.0

Case three postulates Mode I operation with pressurizer pressure reduced to the LSSS of 75 psia, reactor power and coolant inlet temperature raised to their LSSS of 12.5 MW and 155°F, respectively. Figure H.1 predicts that the primary coolant flow rate could be reduced to approximately 2400 gpm before DNB would occur, implying a safety margin of 800 gpm below the LSSS of 3200 gpm on coolant flow through the core. Operating history has shown that the true value of primary coolant flow does not vary from the measured value by more than ± 50 gpm, thus there is excess margin for safety system reaction time to scram the reactor before DNB occurs. Appendix D provides a detailed analysis of the results of the most severe loss of flow accident for the MURR.

Consideration of the same transients for Mode II (5 MW) operation yields even greater safety margins. Figure H.2 presents the results, for Mode II operation, of the same transients discussed above. Case one predicts DNB at 9.25 MW, i.e., 3 MW above the LSSS of 6.25 MW for Mode II. Case two results indicate that the reactor could be operated with a coolant inlet temperature in excess of 200°F for Mode II without reaching DNB. Case three shows DNB occurring only with flow reduced to 1000 gpm or 575 gpm below the LSSS. Thus the safety margin is 36% of the LSSS flow value for Mode II operation as compared to 25% for Mode I.

The LSSS for pressurizer pressure is 75 psia; a margin of 15 psi above the safety limit of 60 psia. Past operating experience has shown the pressurizer pressure sensors to be accurate within ± 2 psi. Additionally, there are four independent sensors capable of causing a reactor scram in the event of a loss of pressure transient, thus there is sufficient margin to ensure that the low pressure safety limit will not be violated.

Therefore the proposed limiting safety system settings on the four important parameters of reactor power, pressurizer pressure, primary coolant flow rate and primary coolant inlet core temperature are easily capable of causing the reactor to scram and preventing the violation of the safety limit envelope.

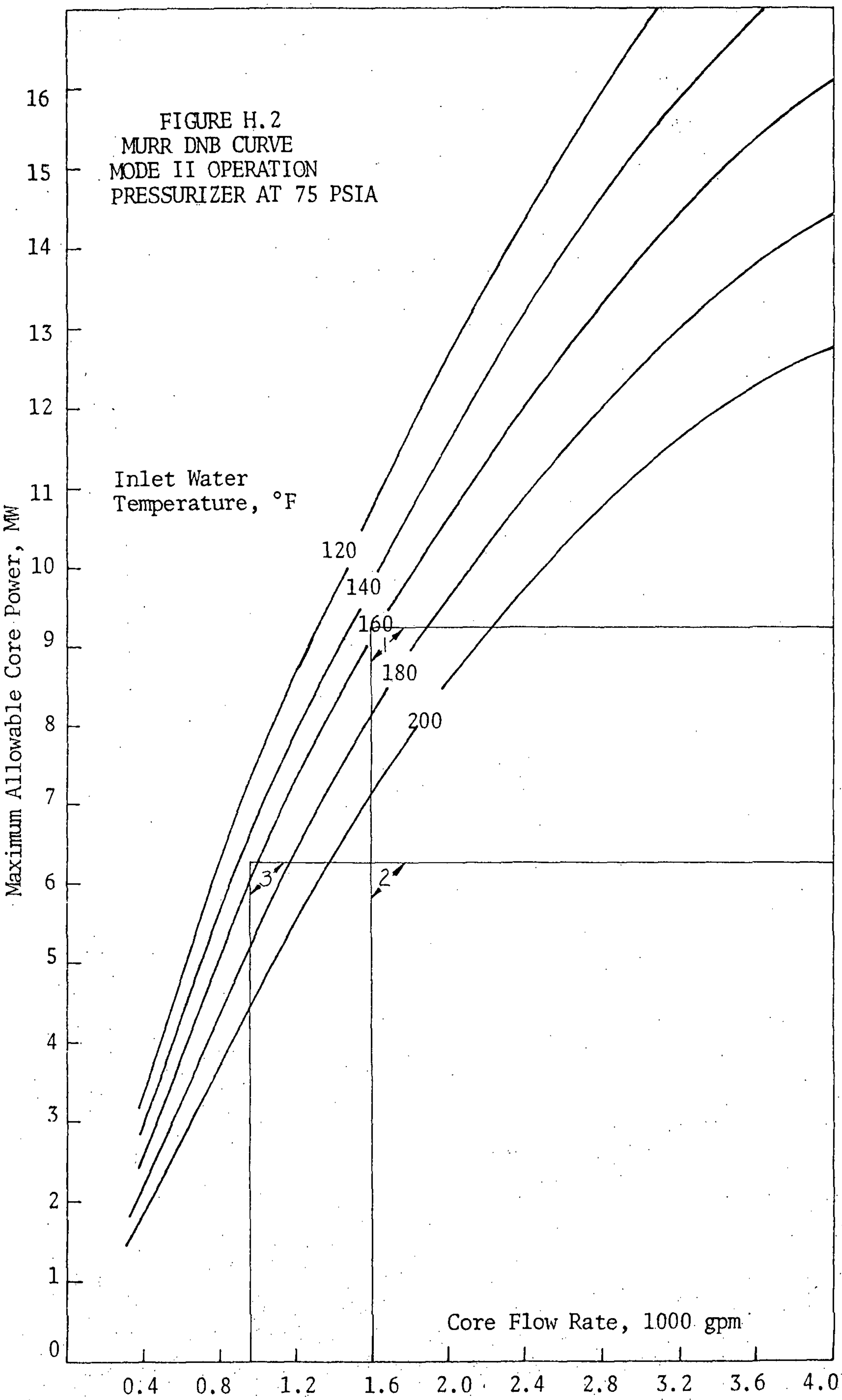
FIGURE H.2
MURR DNB CURVE
MODE II OPERATION
PRESSURIZER AT 75 PSIA
Inlet Water
Temperature, °F
120
140
160
180
200
1
2
3
Maximum Allowable Core Power, MW
Core Flow Rate, 1000 gpm
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
0.4
0.8
1.2
1.6
2.0
2.4
2.8
3.2
3.6
4.0

ADDENDUM NO. 5

HAZARDS SUMMARY REPORT

University of Missouri
Research Reactor Facility

Compiled and Edited by
the Staff
Research Reactor Facility

Submitted by
The University of Missouri
Columbia, Missouri

January 1974

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1.0 INTRODUCTION

On August 25, 1972, the University of Missouri requested changes to License R-103 with the most significant change being an increase in the maximum operating power level from 5 MW(t) to 10 MW(t). The Commission responded by letter, dated February 23, 1973, and identified a number of areas of concern. These areas of concern were addressed in the University's submittal of Addendum 4 to the Hazard Summary Report (HSR) on October 5, 1973. Proposed new Technical Specifications were also submitted as an accompanying document.

Subsequent to the submittal of Addendum 4, considerable informal communications (verbal and written) have been exchanged between the University and the Commission. This document which shall be referred to as Addendum 5 to the HSR is submitted to present analyses which will resolve question over additional areas of concern.

Appendix A to this submittal contains Change 1 to the proposed Technical Specifications of October 1973. The major portion of the Technical Specification change is to comply with the Commission's position on experiment reviews and limitations outlined in Regulatory Guide 2.2 issued in November 1973.

2.0 ANALYSIS OF A LOSS OF ELECTRICAL POWER TO THE MURR

2.1 Introduction

This report contains an analysis of a complete loss of power at the MURR. This implies a loss of commercial power followed by a failure of the emergency generator system. The emergency generator system is described and the routine surveillance tests are outlined. Accident analyses will then be presented for a complete loss of electrical power during a period when the reactor is shutdown.

2.2 Description

Upon loss of normal electrical power to the facility, the emergency generator assumes the desired electrical loads. Drive power to the generator is provided by a Cummins, six cylinder, turbocharged diesel engine. The engine is provided with a 270 gallon fuel storage tank and a mechanically driven fuel injection system. The emergency generator is capable of assuming full load from a cold start in seven seconds. A 24 volt nickel-cadium storage battery is used to start the emergency generator. A static type dual rate float/equalizer charger automatically maintains the startup battery fully charged.

A four pole generator equipped with a brushless permanent magnet exciter produces 60 cycle, 277/480 volts, 3-phase service, and has a standby continuous load capacity of 275 kW. The design of the exciter and regulator provides voltage regulation of better than plus or minus 2%. Stable generator output voltage and frequency are established within two seconds after a transition from no load to full load conditions.

An automatic transfer switch (ATS) selects the power source for the emergency electrical loads from one of two inputs:

(1) Commercial Power
  (i) City Power Plant, or
  (ii) University Power Plant
(2) Emergency Generator Power

2004

During normal operation, all loads are supplied from commercial power. Whenever a commercial power failure occurs for greater than one second duration the engine starts, the automatic transfer switch functions, and the emergency generator assumes the load. Commercial power must be restored for a full ten minutes before the transfer switch functions to transfer the load to commercial power.

The emergency generator will continue to run five minutes after the load is transferred back to commercial power in order to cool down the engine.

The emergency generator and engine are located in a building addition on the southwest corner of the laboratory building. The diesel generator room has local temperature controllers to maintain room temperature above 55°F. The emergency generator starting system is designed to start the emergency generator at temperatures as low as 32°F. The operation of the temperature controllers is checked every four hours by the operating staff during reactor operator.

The emergency bus is routed through the automatic transfer switch to an emergency distribution panel located on the wall in the north inner corridor of the laboratory building. This feed panel distributes power to the following circuits (see Figure 2.1):

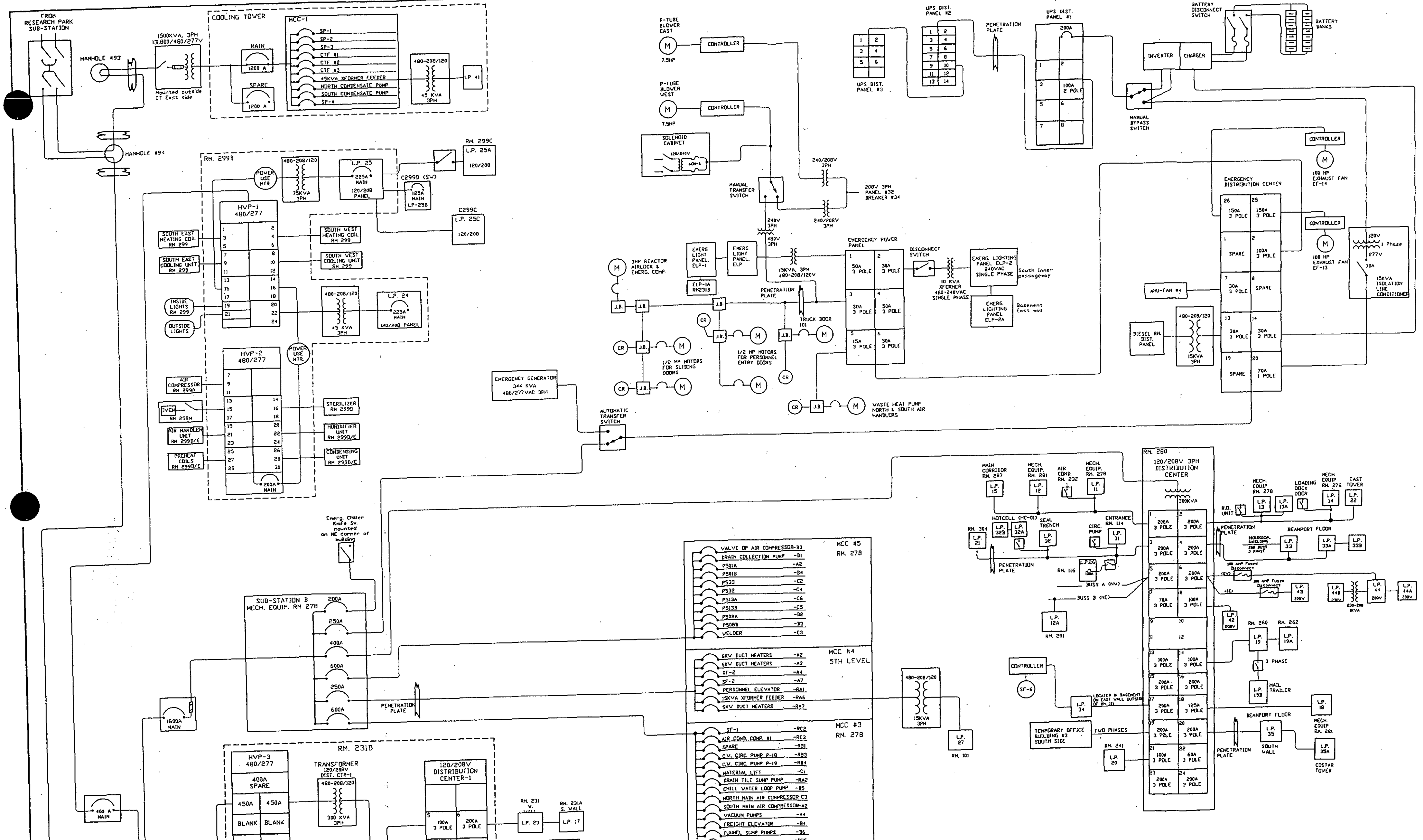
(1) Two circuits service reactor and laboratory exhaust fans EF-13 and EF-14 located in the west tower.

(2) One circuit services a 120 VAC distribution panel providing power for exit lights, stairway lights, fan failure alarm, intercommunication system, and the reactor evacuation and isolation alarms. Nitrogen station, fire protection system, 2004

(3) One circuit services a 120 VAC distribution panel via either an uninterruptible power supply or line conditioner. This distribution panel provides power to the area radiation monitoring system, the annunciator control system, control room clock, all nuclear and process instrumentation in the control room, including control relays, solenoids, indications of primary, pool and other valve positions, control rod drives, rod run-in system, safety system, and servo amplifier system.

(4) One circuit provides power for the operation of the containment ventilation system isolation doors, emergency compressor, truck entry door, and personnel airlock doors.

FIGURE 2.1



REV. NO	DESCRIPTION	DR. BY	DATE
30	Added drawings for North Office Addition (NOA). Redesignated sheet numbers. Added locations to title block.	THS	11/28/07
29	Modification Record 01-14, Adden 1	THS	07/16/07
28	Added LP-25C, Chill Wtr Loop Pump, WP-2/PWMP. Misc. editorial changes.	THS	08/08/06
27	Cooling Tower Electrical Upgrade, WO# 04-2743. Removed Labyrinth & Containment Sump ckts., Misc. changes to B.O.B. Added LP-27	THS	08/04/05
26	Changed TO 522 1 of 3. Added LP-26, 27, 43, 44A and 44B. Misc. editing changes. Removed old Rev. info.	THS	11/10/04

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DATE: 12/15/00
DRAWN BY: TOM SEEGER
CHECKED BY: ENGINEER: CPF
CODE:
REVISION NUMBER: 30
REVISION DATE: 11/28/07

FIGURE 2.1

UNIVERSITY OF MISSOURI-COLUMBIA
FACILITIES OPERATIONS
research reactor facility

ELECTRICAL DISTRIBUTION
REACTOR/LABORATORY

MURR NUMBER: 522
SHEET 1 of 5

PAGE 4

2.3 Surveillance Tests of Emergency Generator

The emergency generator and the Cummins diesel engine are tested routinely on the following basis:

(1) At least once a week the Cummins diesel engine which powers the generator is started and allowed to run for a period of 30 minutes without load.

(2) In addition, the Cummins diesel engine is started and run for 30 minutes prior to each reactor startup following a shutdown greater than 24 hours.

(3) The ability of the emergency electrical generator to assume the emergency load is verified on at least a semi-annual basis. Commercial power to the reactor facility is interrupted at the transfer switch, simulating a complete loss of commercial power to the reactor facility. This requires the emergency generator to automatically start and assume full emergency electrical load.

(4) The entire unit is serviced routinely as part of a planned preventative maintenance program.

2.4 Accident Analysis

2.4.1 Loss of Commercial Power with the Reactor Operating at 10 MW and the Emergency Generator Fails to Start

Each system that is affected by a complete loss of electrical power is listed and commented upon in the following paragraphs.

(1) Reactor Control System

At the time of loss of commercial power while operating at 10 MW, the reactor would scram as a result of loss of power to the electromagnets holding the blades in position. The blades would drop into the core by gravitational force and the reactor would be shut down.

(2) Reactor Process System

All process systems (e.g., primary cooling, pool cooling, etc.) will be placed in the shutdown condition due to the failsafe design of these systems. Loss of electrical power would cause a cessation of coolant flow and a closing of the isolation valves. In the primary system, redundant valves 546A and B open by spring actuation placing the in-pool heat exchanger in service. The failsafe design of the system permits shutdown decay heat removal with no electrical power (reference: Appendix D of Addendum 4 to Hazards Summary Report).

(3) Containment Building Ventilation Isolation Doors

Power would be lost to the motor operated doors and they would fail to close (or open) in response to any electrical signal. Also, the gasket seals would not inflate since the inflating mechanism responds only when the doors are closed against their stops. The backup isolation doors, however, fail closed upon loss of solenoid power and hence would automatically close upon loss of building power.

(4) Emergency Air Compressor

The emergency compressor motor would fail to operate in response to a falling pressure in the reserve tank. The reserve tank holds a volume of 10.5 ft^3 at a nominal pressure of 100 psi, which would be sufficient to inflate all gasket seals on all isolation doors if this were required. But the ability to recharge the tank to nominal operating pressure would be lost in the event of a complete power failure. The primary function of this compressor is to provide air to the seal gaskets of all isolation doors. Since the doors would not be operable with no power there is little demand for the emergency air supply.

(5) Truck Entry Door: Door 101 Beamhole Floor

During reactor operation and during periods when the reactor is left unattended, the door is closed and the seal is inflated. Loss of commercial power would prevent one from being able to open this door or deflate the seal. Hence, loss of commercial power during normal reactor operation would leave the status of this door unaffected.

(6) Personnel Airlock Door: Doors 275/276 Grade Level

During reactor operation and during periods when the reactor is left unattended, one of these doors remains closed and the gasket inflated. Loss of commercial power without the ability of the emergency generator to provide emergency power, would prevent one from operating these doors electrically. There is in existence, however, a procedure by which the gaskets can be deflated manually and the doors manually opened or closed. Even though the doors cannot be operated electrically, it is possible for one to leave the building through these doors in the event of a power failure. However, the ability to maintain at least one door in the closed position with its seal gaskets inflated is lost if power to the doors is not available. The containment integrity of the building, therefore, cannot be guaranteed if emergency and electrical power were not available to operate both doors. However, the reactor will be shutdown and containment is not a vital requirement.

(7) Laboratory and Reactor Exhaust Fans: EF-13 and EF-14, Fifth Level, West Tower

Upon isolation of the reactor building, the operation or inoperation of these fans would have no consequence on the status of the reactor building. Upon loss of commercial building power without the availability of emergency power, both of these fans would cease to function.

(8) Reactor and Laboratory Corridor and Exit Lights

The 120-volt corridor and exit lights in both the reactor building and laboratory building depend upon commercial power or emergency backup. In most areas, emergency wall battery pack lights which operate upon loss of commercial building power provide sufficient lighting for all personnel to leave the reactor building and laboratory corridor areas safely. Visibility in all critical areas and emergency escape route is assured by the strategic placement of these battery pack emergency lights.

(9) Fan Failure Alarm System

The exhaust fans, EF-13 and EF-14, "failure to operate" alarm system would not function. Loss of building commercial power and loss of emergency backup power would prevent the operation of these fans as previously discussed. Such a situation precludes the need for these alarms.

(10) Intercommunication System

The laboratory area and the reactor building are provided with a multiple station intercommunication system. The loss of this system results in the inability to transmit messages rapidly to the entire facility. However, telephone communication to each laboratory area and various areas inside the reactor building would not be interrupted by a loss of commercial power. There is also provided portable battery-powered transmitter-receiver packs which can be used to maintain communication between the Emergency Director in the laboratory lobby and investigation parties sent out from that area.

(11) Reactor Building Isolation and General Evacuation Alarm

Loss of reactor building power without emergency backup power would result in the loss of all audible and visual evacuation and isolation alarms.

(12) Diesel Room Distribution Panel

Power to the emergency generator control panel, emergency generator room lighting, and emergency generator room temperature controls are provided by this panel. Loss of commercial power as well as emergency power would leave these loads de-energized. However, failure of the generator engine to operate preempts the need for these loads.

(13) Reactor Controls and Instrumentation

Power to all reactor instrumentation and controls, both process and nuclear, is provided through a 120 VAC distribution panel located in the control room. Loss of commercial power without emergency backup would not effect the control and instrumentation power for 20 minutes because of the capacity of the uninterruptible power supply (UPS). Once the reactor is confirmed to be shutdown and before the UPS batteries reach a low voltage condition, the control and instrumentation systems would be secured. The UPS unit would be secured prior to the batteries reaching a low voltage condition to prevent a low voltage transient on the system. The reactor operators would then have no control console information relating to changes in the status of reactor system different than that gathered when the UPS was operating.

All subsequent information regarding valve positions and the status of the reactor would have to be obtained visually by the reactor operator. He can visually observe that the control blades are fully inserted and that the reactor power had been reduced by the intensity of the Cerenkov glow in the region of the core. Visual examination of the valve operators in the pool area would indicate the position of

2002
2004

(15) Fire Protection System

Normal supply power would be lost to the fire detection system. However, the system is equipped with a battery backup that would provide power for the entire system for a period of twenty-four (24) hours. Additionally, fire protection is not required to accomplish a safe shut down of the reactor or to maintain a safe shutdown condition. 2004

these valves and the fact that they had functioned properly. Operation of the emergency pool fill system would be unaffected.

"(14) Nitrogen Station

The loss of electrical power to the solenoid-operated valves of the nitrogen station would prevent the nitrogen station from being able to supply nitrogen gas to the pressurizer. Since the reactor is shutdown due to the loss of electrical power and decay heat is being removed by the in-pool heat exchanger, the loss of the nitrogen station would have no effect on the status of the reactor."

Periods

iscussed in paragraph 2.4.1 2002

(1) Reactor Control System

In this case it can be assumed that the reactor is in the shutdown mode with all systems secured. This would be assured by the fact that prior to the loss of commercial power a complete shutdown checksheet for the reactor and systems had been completed. Therefore, there would be no need for the reactor operator to determine the status of the reactor or reactor systems after the loss of commercial power.

(2) Persons within Containment at the Time of Loss of Commercial Power

Research and other non-reactor staff personnel may be within the containment building at the time of the loss of commercial power. All personnel allowed unescorted access to containment have a knowledge of how to operate the personnel airlock door manually without assistance at a time when electrical power to these doors is unavailable. Simple directions are posted next to the airlock doors. (rev 1990)

(3) Containment of the Reactor Building

Containment integrity of the reactor building would be assured by virtue of the status of the reactor during a normal shutdown period. Truck entry door 101 on the beamhole floor would be closed and sealed, the 16" building exhaust isolation valves would have failed closed, the supply and return fan on the fifth level would have stopped operating, the primary isolation doors would remain open, however, the backup isolation doors would have failed closed. (rev 1990)

2.5 Conclusions

Under the postulated failures, the reactor will shutdown and the core will be cooled indefinitely by natural convection circulation through the in-pool heat exchanger. It may be necessary to violate containment briefly to allow personnel to enter and exit containment, however, the reactor is in a safe configuration. Health Physics monitoring with portable instruments would preclude the accidental exposure of personnel to radiation.

3.0 STEP REACTIVITY INSERTION ANALYSIS FOR DETERMINATION OF EXPERIMENT REACTIVITY LIMITS

3.1 Introduction

This section outlines the basis for the reactivity limits placed on each unsecured experiment and each movable experiment. The guidelines followed are those presented in Regulatory Guide 2.2 (1).

3.2 Reactivity Limit - Unsecured Experiments

Section C.1.a[3] of Regulatory Guide 2.2 requires that the magnitude of the potential reactivity worth of each unsecured experiment be less than the value of reactivity which would cause a violation of a safety limit.

In order to determine this limit on reactivity, the PARET code (2) was used to analyze the reactor transient behavior following step insertions of various amounts of reactivity. This work is essentially an expansion (to lower values of reactivity insertions) of the study presented in Addendum 3 to the Hazard Summary Report (3). The PARET code is an improved version of the CHIC-KIN code used in the previous study.

The following conditions were assumed at the initiation of the accident. Nominal conditions of operation are indicated for comparison.

Parameter	Assumed Condition	Nominal Condition
Power	11 MW	10 MW
Primary Flow	3000 gpm	3600 gpm
Pressurizer Pressure	75 psia*	85 psia
Inlet Temperature	155°F*	120°F

*These values correspond to the scram setpoint for that parameter.

Results of this study are illustrated in Figure 3.1. Employing these curves and the safety limit curves for a pressurizer pressure of 75 psia (4), the following conclusions can be drawn. First, for an insertion of 0.003 ΔK, the peak power reached is 14.48 MW which is only slightly less than the safety limit of 14.5 MW. An insertion of 0.0025 ΔK results in a peak power of 13.78 MW which is substantially less than the safety limit. Finally, the safety limit curves presented in Figure 6.3 of this report imply that for nominal operating conditions, an insertion of even 0.004 ΔK can be tolerated without exceeding a safety limit. The 150 msec scram response time implies that the transient will be terminated before any safety limit is exceeded.

Thus, the chosen limit of 0.0025 ΔK placed on each unsecured experiment ensures that a safety limit will not be violated.

3.3 <u>Reactivity Limit - Movable Experiments</u>

Section C.1.a[4] of the Regulatory Guide requires that the rate of change of any movable experiment be such that when the experiment is intentionally set in motion, the capacity of the control system to provide compensation is not exceeded. For the purposes of this analysis, this criteria was interpreted to mean (1) in manual control the operator and/or rod run-in circuit would have sufficient time to shim control rods and control the transient before a high power scram was initiated, and (2) in automatic control the capacity of the regulating rod will be sufficient to compensate for the reactivity inserted.

A step reactivity insertion of + 0.001 will result in a prompt jump of 15.67% followed by a stable reactor period of 63.8 seconds. If the initial power level is 10 MW, the prompt jump will take the power level to 11.5 MW which will initiate a rod run-in. If no rod run-in were to occur, the power would increase on the 63.8 second period to the high power scram trip point of 12.5 MW in 5 seconds. This is sufficient time for the operator to take control of the transient because in manual control the operator would be continuously monitoring power level.

If the reactor is in automatic control and the regulating blade is at its minimum operating position of 10.00 inches, the step reactivity insertion of + 0.001 will cause the regulating rod to drive in to a position of 4.5 inches. It is noted that at a regulating rod position of 5.2 inches the automatic shim circuit described on page 9-19 of the Hazards Summary Report will be activated and insert the shim rods to compensate for the reactivity addition.

A negative reactivity insertion of .001 would require operator action in both automatic and manual control. In automatic control the regulating blade would be driven out to its fully withdrawn position but the reactor would still be subcritical. The "regulating blade 60% withdrawn" alarm would actuate and alert the operator to the transient. If the operator did not assume manual control, the NI Channel 4 interlock would cause the control system to shift to manual control when the power level decreases 25%.

In conclusion, the control system has sufficient capacity to compensate for a step reactivity insertion of 0.001 ΔK.

References

1. Regulatory Guide 2.2, Development of Technical Specifications for Experiments in Research Reactors, November 1973.
2. PARET, A Program for the Analysis of Reactor Transients, IDO-17282, Argonne Code Center Abstract 555, Version 5, March 1973.
3. Hazards Summary Report, Addendum 3, Missouri University Research Reactor, August 1972, pp 62-67.
4. Hazards Summary Report, Addendum 4, Missouri University Research Reactor, October 1973, p F-7.

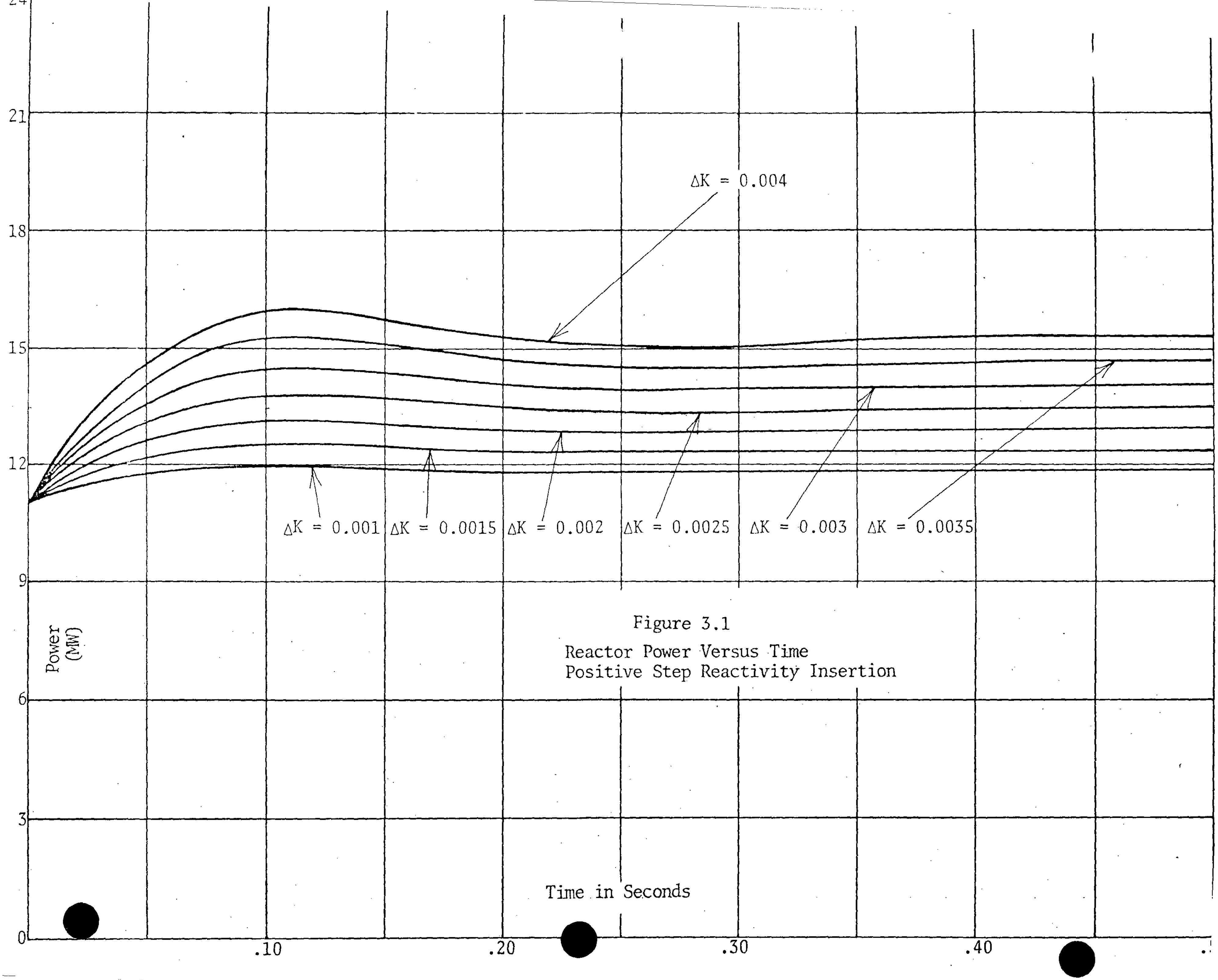


Figure 3.1

Reactor Power Versus Time
Positive Step Reactivity Insertion

4.0 HIGH PRESSURE TRANSIENT ANALYSIS

4.1 Introduction

The pressurizer system is described very briefly in pages 5.2 and 5.3 of the Hazards Summary Report. The analysis of a high pressure accident has not been submitted to the Commission. In this report the pressurizer system will be described in more detail and then an analysis will be made of the consequences of a high pressure transient initiated by any of the following means:

a. Nitrogen addition valve 526 sticks open.
b. Maximum plant heat-up with vent valve 545 sticking closed.
c. Continuous charging of water with charging pump 533.

4.2 The Pressurizer System

A schematic diagram of the pressurizer system is shown in Figure 4.1. The major components of the system are:

a. The pressurizer (300 gallon, 3/4" thick aluminum tank).
b. Nitrogen addition valve 526 (air-operated to open, spring to close).
c. Nitrogen vent valve 545 (air-operated to open, spring to close).
d. Water addition valve 527B (air-operated to open, spring to close).
e. Water drain valve 527A (air-operated to open, spring to close).
f. Primary system drain valve 527D (air-operated to open, spring to close).
g. Surge line isolation valve 527C (air-operated to open, spring to close).
h. Charging pump P-533 (positive displacement, capacity 50 gpm).
i. Relief valve 537 (to be set at 100 psig).

As noted in the HSR, the pressurizer pressure will be maintained at a pressure sufficient to ensure a reactor inlet pressure of at least 65 psia. The normal operating pressure of the pressurizer has thus been set at 60 psig (75 psia). The increased primary coolant flow with 10 MW operation will increase the pressure drop between the pressurizer and the reactor inlet. Subsequently, the normal operating pressure for the pressurizer will be increased to 85 psia for all modes of reactor operation utilizing forced circulation flow in the primary coolant system.

The pressurizer relief valve 537 was originally designed to operate at a set pressure of 150-175 psig. For 10 MW operation the set pressure of this relief valve will be reduced to 100 psig to provide additional protection from overpressurization of the primary coolant system.

The nitrogen (N_2) used in the pressurizer is supplied from one of two banks of nitrogen bottles. Three bottles are connected to each bank. The nitrogen pressure from the bottles is reduced to a pressure of 140 psig and is then piped to the equipment room where it is used for the pressurizer. The control system for the nitrogen banks will switch the nitrogen supply to the standby bank when the pressure falls below 140 psig in the operating bank. In the equipment room the 140 psig nitrogen is further reduced to 70 psig (80 psig at 10 MW) by a regulator upstream of valve 526.

4.3 System Operation

Pressure is maintained in the system by nitrogen gas in the top of the pressurizer. If pressure decreases sufficiently, pressure switch 941 sends a signal to open valve 526 to emit regulated nitrogen and thus increase the pressure. As the pressure increases above normal, pressure switch 940 sends a signal to open valve 545 to vent the pressurizer to the exhaust system. The water level in the pressurizer is maintained by a 50 gpm charging pump (P-533). If the level decreases sufficiently, level controller 936 sends a signal to open valve 527B and start pump P-533 which adds water to the pressurizer. If the level increases above normal, level controller 936 sends a signal to open valve 527A to drain water to the drain collection system.

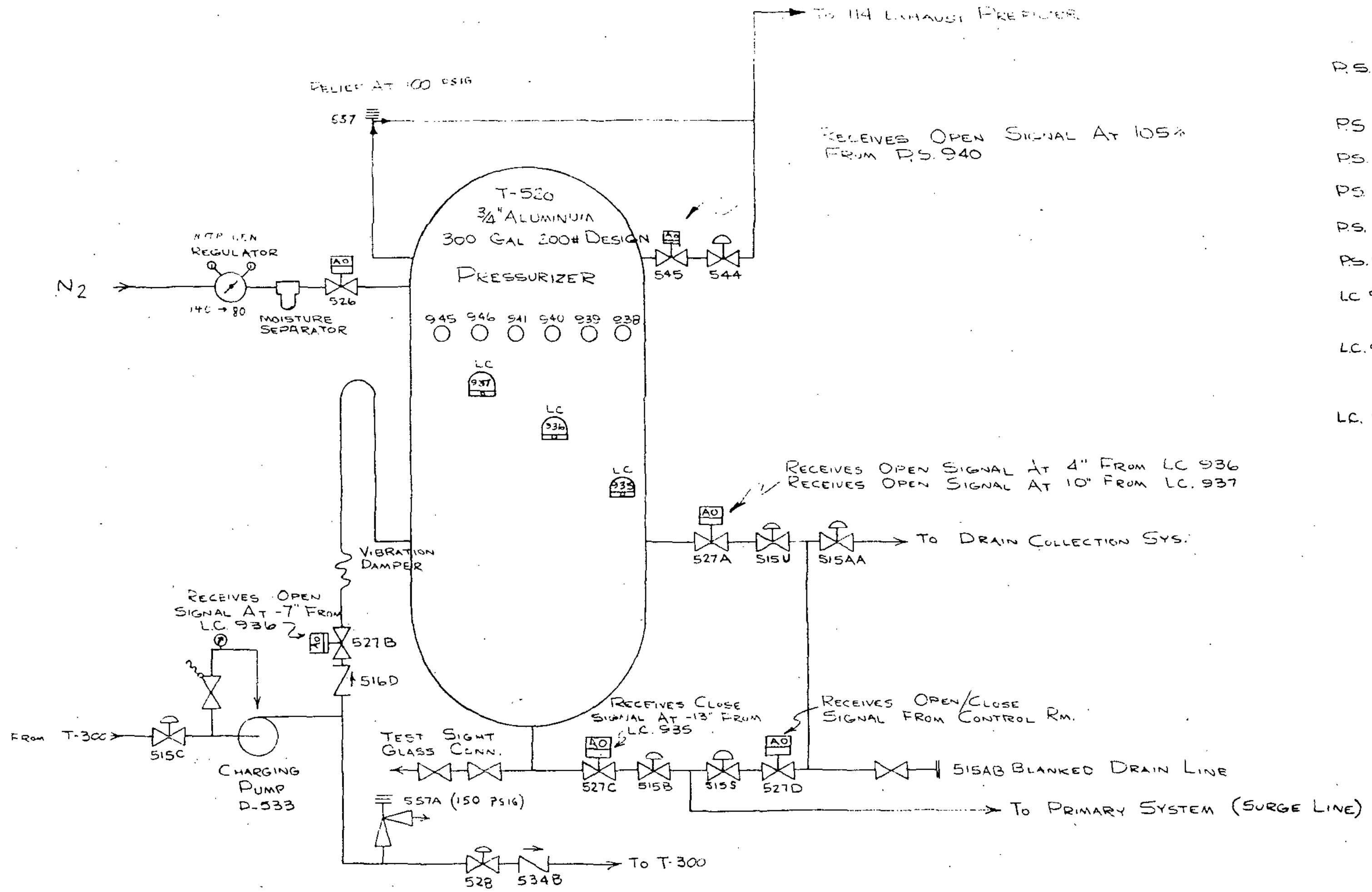
4.4 High Pressure Transient

There are three ways to increase system pressure: (1) from the nitrogen system through valve 526, (2) plant heatup, and (3) charging water with pump P-533.

If valve 526 should fail open, the maximum pressure in the system (assuming that valve 545 should fail) would be 80 psig (the high pressure scram setpoint) because the nitrogen supply regulator will be set at that pressure. If the regulator should fail and supply 140 psig nitrogen, it has been shown by test that valve 545 will vent sufficient nitrogen to prevent the system from exceeding 73.5 psig.

The maximum pressure transient caused by plant heatup would be as follows. The primary temperature is at 70°F and the pressurizer is half full of water. The plant is started up and heated to 160°F. The heatup would cause 43 gallons of water to expand into the pressurizer and compress the nitrogen bubble to a pressure of 103 psig (assuming that valve 545 should fail to open and vent off the pressure). The transient would be terminated, however, by operator action when the high pressure alarm is received at 77 psig, or if there were no operator action the heatup would be stopped at a pressure of 80 psig by a high pressure scram. The system is further protected from exceeding the Technical Specification limit of 110 psig by the pressurizer relief valve and the two primary relief valves, all set at no greater than 110 psig.

The most severe pressurizer pressure transient would be caused by continuous operation of charging pump P-533. Assume that the charging pump starts with an initial pressure of 73.5 psig (upper level of control band). If valve 545 fails to open, the pressure could reach no greater than 110 psig at which time the pressurizer or one of the two primary coolant reliefs would prevent further increase. This transient, however, would be terminated by operator action at 77 psig when the high pressure alarm sounds, or if there were no operator action the plant would be shutdown by a high pressure scram at 80 psig.



P.S. 938	Reactor Scram At 85% Of 70 psi. Lo Pressure Scram. Pressure Going Down
P.S. 939	→ Reactor Hi Pressure Scram At 115% Of 70 psi Going Up
P.S. 940	→ Pressurizer Vent V545 Opens At 105% Of 70 psi Going Up.
P.S. 941	→ Pressurizer N2 Makeup 526 Opens At 95% Of 70psi Going Down
P.S. 946	Reactor Hi Pressure Alarm At 110% Of 70psi Going Up.
P.S. 945	Reactor Low Pressure Alarm At 90% Of 70psi Going Down.
LC 937	Hi Level Alarm And Drain Valve 527A Opens (Backup) At 10" Above ℄ Going Up
L.C. 936	527A Opens On Hi Level At 2 to 6 Going Up. 527B Opens And 533 Starts On Lo Level At -6" To -8" Going Down
L.C. 935	Lo Level Alarm At -9" To -12" Going Down. Lo Level Scram And 527C Shuts At -12" To -15" Going Down

Refer Also To P&ID Print And Process Instrumentation Control & Interlock (MURR #'s 156 and 41)

1	D WILLIAMS	1-4-74		TITLE: PRESSURIZER SYSTEM 10 MW
CHANGE PSI LEVEL				RESEARCH REACTOR FACILITY, UNIVERSITY OF MISSOURI
				DRAWN BY Vicki Nichols; APPROVED [signature]
				SHEET NO. 1 OF 1; DWG. NO. 105 R1

Figure 4.1

5.0 CONTINUOUS ROD WITHDRAWAL ACCIDENT ANALYSIS

5.1 Introduction

This analysis develops the power history of an excursion resulting from the continuous withdrawal of the four shim blades. The results of this study differ only slightly from the data presented in HSR Addendum 1 (1) despite the fact that the rod worth curves for the existing 6.2 Kg (U-235) core are not identical to the curves used for the 5.2 Kg (U-235) cores. Another factor that introduced some variation between this and the earlier analysis is the fact that the "initial shutdown" power is higher than that in the original report due to the increased neutron source contribution from the beryllium reflector. The approach to this analysis is first to calculate the new value for the period of the reactor when it passes through critical and then calculate the new value for the power level at a reactor period of 8 seconds (scram setpoint).

5.2 Assumptions

The original analysis concluded that a continuous rod withdrawal from source power without a scram would cause significant damage to the reactor core. This conclusion is unchanged for this analysis, but a failure to scram on high power or short period is not considered credible, because the 10 MW safety system has been designed in conformance with the IEEE 279 criteria. Thus, it is assumed that upon reaching a scram setpoint for high power or short period that the reactor is shutdown. It is further noted that no credit was taken for the high power or short period rod run-in which will in actuality be activated before the scram setpoints are reached.

5.3 Calculation of Reactor Period

The following approximate calculation was used to determine the reactor period at criticality. The subcritical and prompt supercritical approximations given by Schultz (2) were employed:

$$\tau > \frac{-\Delta K}{\gamma} \qquad \text{for the subcritical case}$$

$$\tau < \frac{\ell^*}{\Delta K - \beta}$$

where τ = period

ΔK = reactivity

γ = rate of reactivity change

β = effective delayed neutron fraction

ℓ^* = neutron lifetime

The reactivity insertion rate was evaluated by using straight line approximations to the rod worth curve for the 6.2 Kg core (3). Employing the above equations and the straight line approximation, the reactor period was plotted as a function of reactivity (Figure 5.1). The region of discontinuity $0 \leq \Delta K < \beta$ requires some interpolation. Based on this interpolation it is estimated that the reactor period would be approximately four seconds at the time the control rods pass through critical. Thus, the reactor should experience a rod run-in on a 10 second period or a scram on an 8 second period trip before criticality is reached. Figure 5.1 indicates that an 8 second period would be obtained when the reactivity reaches -0.002 ΔK. The rod worth curve of reference 3 implies that this would correspond to a shim bank position of 10.1 inches withdrawn.

5.4 Power Level Calculations (Accident Initiated from Shutdown Condition)

In order to calculate the power level at the time of the scram due to short period one must begin with the definition of the reactor period, namely:

$$\tau = \frac{n}{dn/dt} \text{ or } \frac{dn}{dt} = \frac{n}{\tau}$$

Using the computed values for reactor period from the straight line rod worth approximation and an initial power level from the beryllium (γ,n) source of 0.1 watt, a power level at any time in the accident can be calculated by numerical integration of the above equation. The power level at the time an 8 second period is reached was calculated by numerical integration of the above equation. The results of this calculation indicate that a power level of 225 watts will be reached before the reactor scrams on an 8 second period and thus no core damage will be done from this accident. It is noted that no credit was taken for the short period rod run-in which occurs at a period of 10 seconds. The rod run-in circuit has been designed to comply with the IEEE-279 criteria and thus its failure is not credible.

5.5 <u>Accident Initiated From Full Power</u>

A continuous rod withdrawal accident from a reactor power level of 10 MW will also be terminated before the core is damaged. The maximum shim blade insertion rate permitted by the Technical Specifications is 3 x 10^{-4} ΔK/sec. At this insertion rate, 3.3 seconds would elapse before an insertion of even 0.001 ΔK has been achieved. Section 3.0 of this addendum demonstrated that a step insertion of 0.001 ΔK would not exceed the capacity of the control system and would result in a rod run-in and/or automatic shim insertion. Either of these actions would terminate the accident and prevent core damage.

If the reactor were operating at power but at less than 10 MW, the withdrawal would continue for 7.3 seconds until a reactor period of 10 seconds was reached and a rod run-in initiated to terminate the accident. The reactivity inserted by a 7.3 second shim is 0.00219 ΔK. If the reactor power level was just below the high power rod run-in setpoint of 115% when the short period rod run-in occurred, the resulting power overshoot would not exceed a safety limit. Section 3.0 of this addendum presents the results of a step reactivity insertion from 11 MW and demonstrates that a step insertion of 0.003 ΔK would not cause a safety limit to be exceeded.

It is further noted that if one assumes that the rod run-in fails, the scram setpoint of an 8 second period would be reached after withdrawing rods 9.2 seconds. The reactivity inserted after 9.2 seconds is only 0.00277 which provides adequate margin below the 0.003 value above.

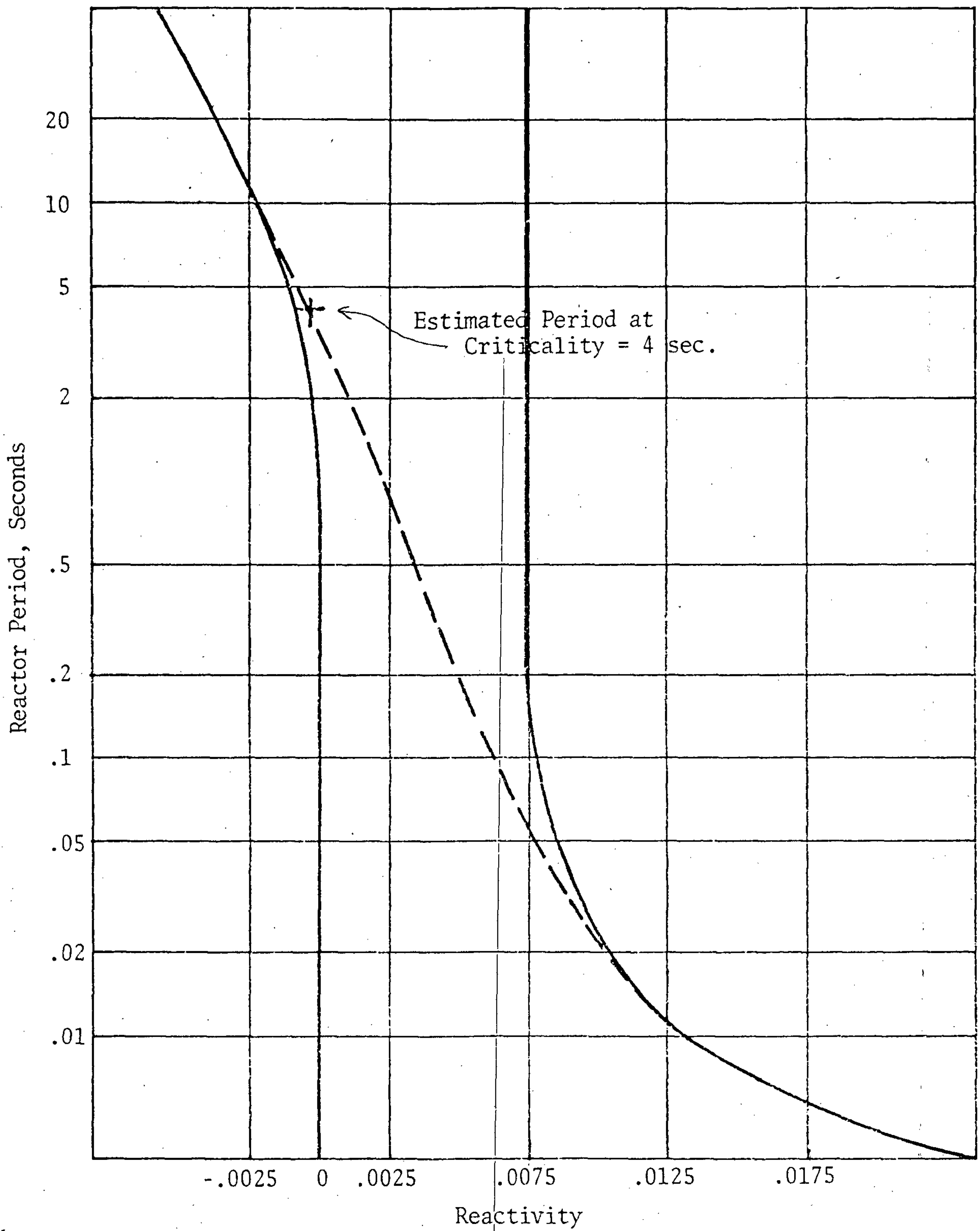


Figure 5.1

Reactor Period vs. Reactivity
Continuous Rod Withdrawal

5.6 Conclusions

The major difference between this analysis and that presented in Addendum 1 to the HSR is that the initial indicated power level for the startup accident on cores subsequent to Core I will be about 1-5 watts. This is considerable above the 1.95 x 10^{-4} watt power level calculated for Core I. With this very low power level the operator cannot monitor the transient on the installed nuclear instruments and thus, the reactor passes through criticality undetected at a power level of about 10^{-3} watts. By the time the power level reaches the detector sensitivity levels the reactor is on a severe power transient and the power overshoot is significant.

The strong neutron source from the (γ,n) reaction on the beryllium reflector has now eliminated the possibility of a startup with a neutron level below the minimum sensitivity of the installed instruments. This fact means that the transient will be detected earlier by the nuclear instruments and the accident will be terminated at a much lower power level. It is further noted that with the continuous indication on the nuclear instrumentation the probability that the operator will recognize the accident and take corrective action is considerably increased.

In conclusion, the consequences of a continuous rod withdrawal accident at MURR will be a rod run-in or a reactor scram on short period or high power with no resulting fuel damage.

References

1. Hazards Summary Report, Addendum 1, Missouri University Research Reactor, February 1966, p. 43 ff.
2. Schultz, M. A., Control of Nuclear Reactors and Power Plants, McGraw-Hill, N.Y., 1961, pp. 381 ff.
3. Lower Power Testing Report for the 6.2 Kg MURR Core, October 1971, p. 49.
4. Startup Nuclear Data for the Initial Criticality Study of MURR Core 10, November 1973.

6.0 ADDENDUM TO THE SAFETY LIMIT ANALYSIS FOR THE MURR

6.1 Introduction

In October 1973 the MURR submitted, as Appendix F of Addendum 4 to the Hazards Summary Report, a safety limit analysis performed by the NUS Corporation on the MURR reactor (1). In response to AEC inquiry, the University of Missouri authorized NUS Corporation to extend that analysis. The results are discussed below.

6.2 Safety Limit Curves

The original work generated two safety limit curves corresponding to pressurizer pressures of 60 to 75 psia. These curves are reproduced here as Figures 6.1 and 6.2, respectively. Additional work, employing techniques identical to the original study, has produced Figure 6.3 which depicts a safety limit curve for a pressurizer pressure of 85 psia, i.e., the nominal operating pressurizer pressure. Table 6.1 presents the numerical results of this work. These three curves together define a four-dimensional safety limit envelope prescribing limiting combinations of values for reactor power, pressurizer pressure, primary coolant inlet temperature and core flow rate. Operation of the MURR within this safety envelope will prohibit fuel meltdown or cladding damage as a result of departure from nucleate boiling (DNB). To evaluate safety limits for pressurizer pressures intermediate to the three cases considered, interpolation will be used. For example, the true values of core flow and inlet temperature in a particular case may be applied to the three curves to obtain a three point relationship between pressurizer pressure and the limiting reactor power. The safety limit on reactor power level will then be fixed by interpolation. For pressurizer pressures below 60 psia extrapolation will be used to determine the safety limits.

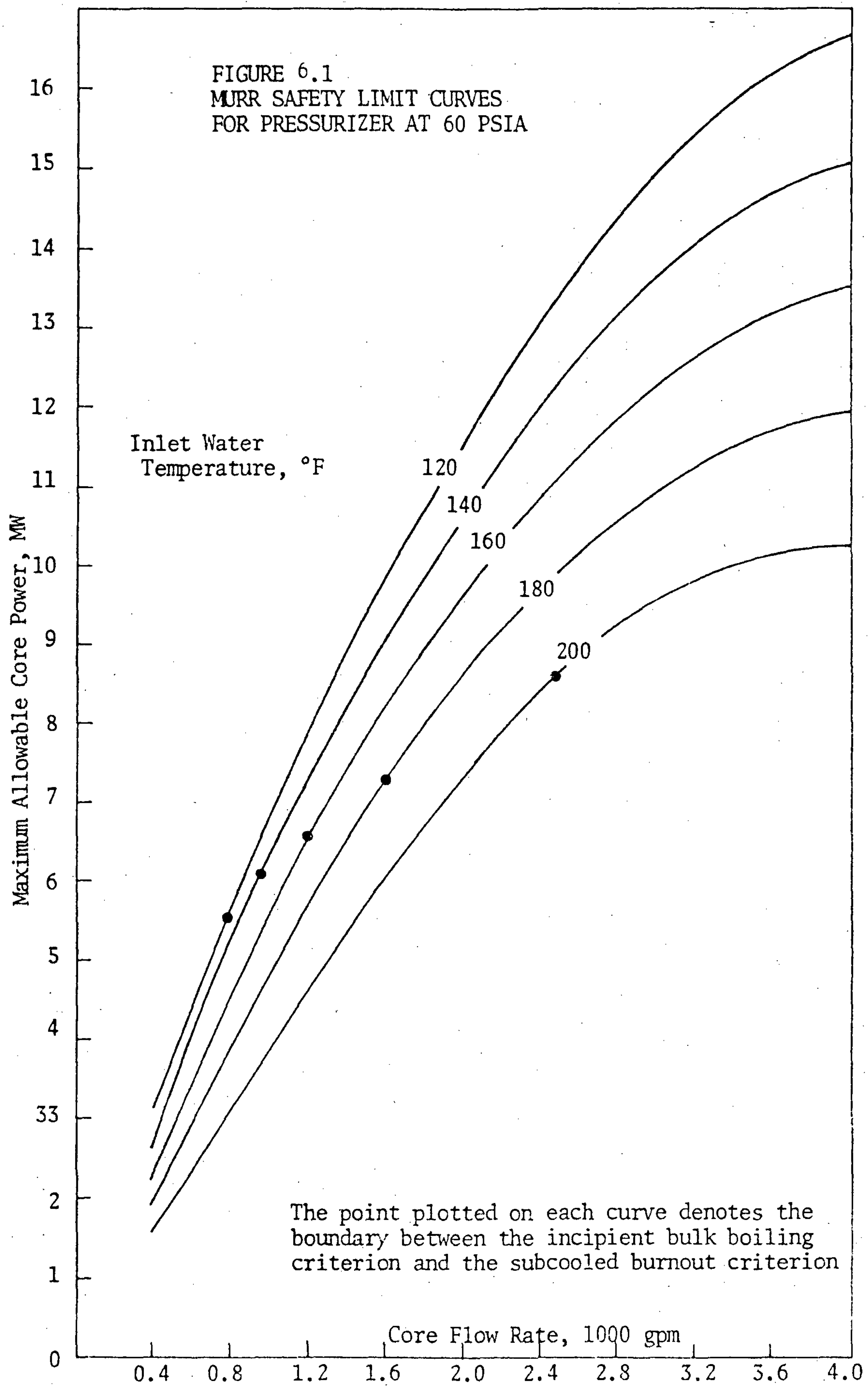


FIGURE 6.1
MURR SAFETY LIMIT CURVES
FOR PRESSURIZER AT 60 PSIA
Inlet Water
Temperature, °F
120
140
160
180
200
Maximum Allowable Core Power, MW
16
15
14
13
12
11
10
9
8
7
6
5
4
33
2
1
0
The point plotted on each curve denotes the
boundary between the incipient bulk boiling
criterion and the subcooled burnout criterion
Core Flow Rate, 1000 gpm
0.4
0.8
1.2
1.6
2.0
2.4
2.8
3.2
3.6
4.0

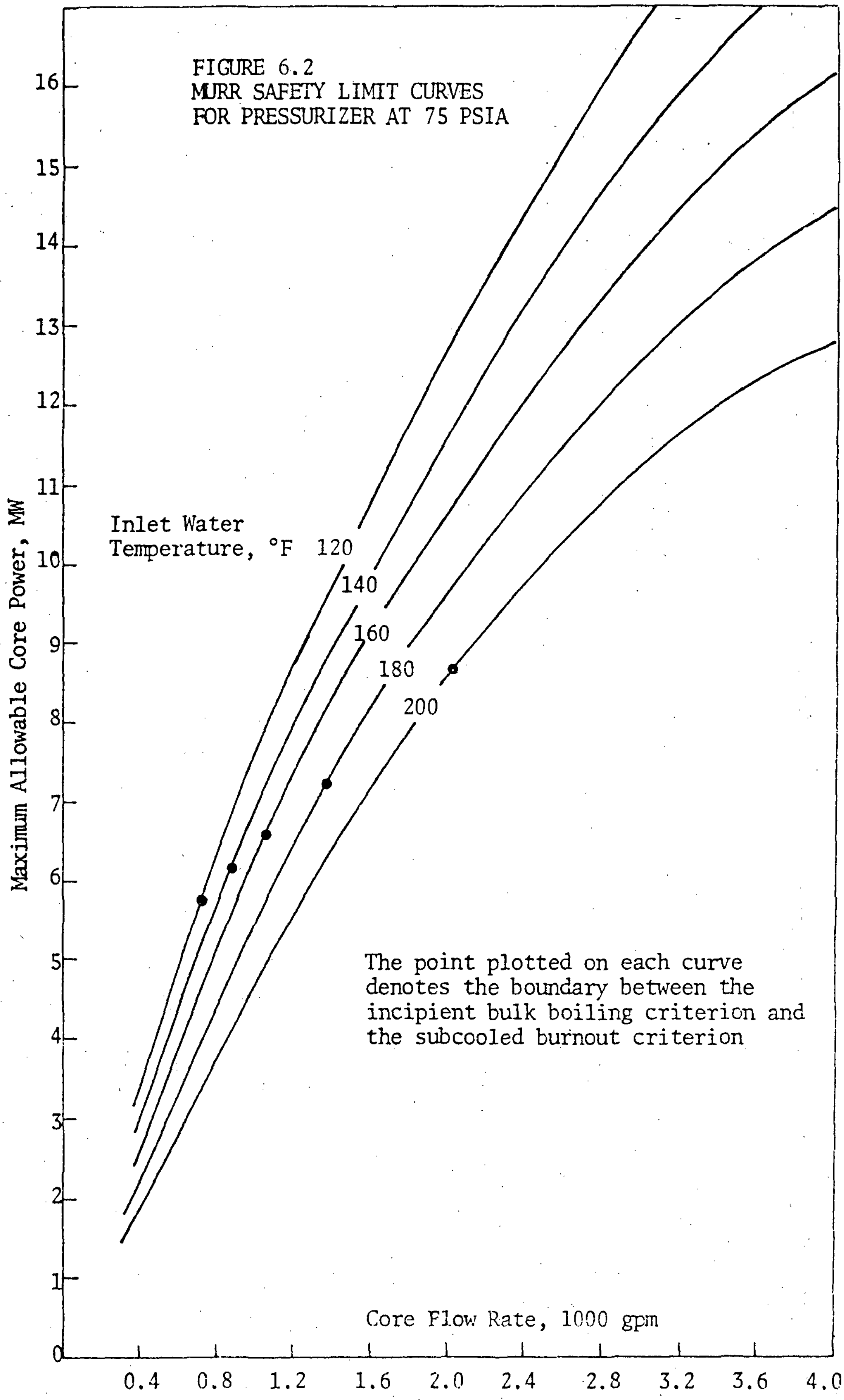


FIGURE 6.2
MURR SAFETY LIMIT CURVES
FOR PRESSURIZER AT 75 PSIA
Maximum Allowable Core Power, MW
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
Inlet Water
Temperature, °F 120
140
160
180
200
The point plotted on each curve denotes the boundary between the incipient bulk boiling criterion and the subcooled burnout criterion
Core Flow Rate, 1000 gpm
0.4
0.8
1.2
1.6
2.0
2.4
2.8
3.2
3.6
4.0

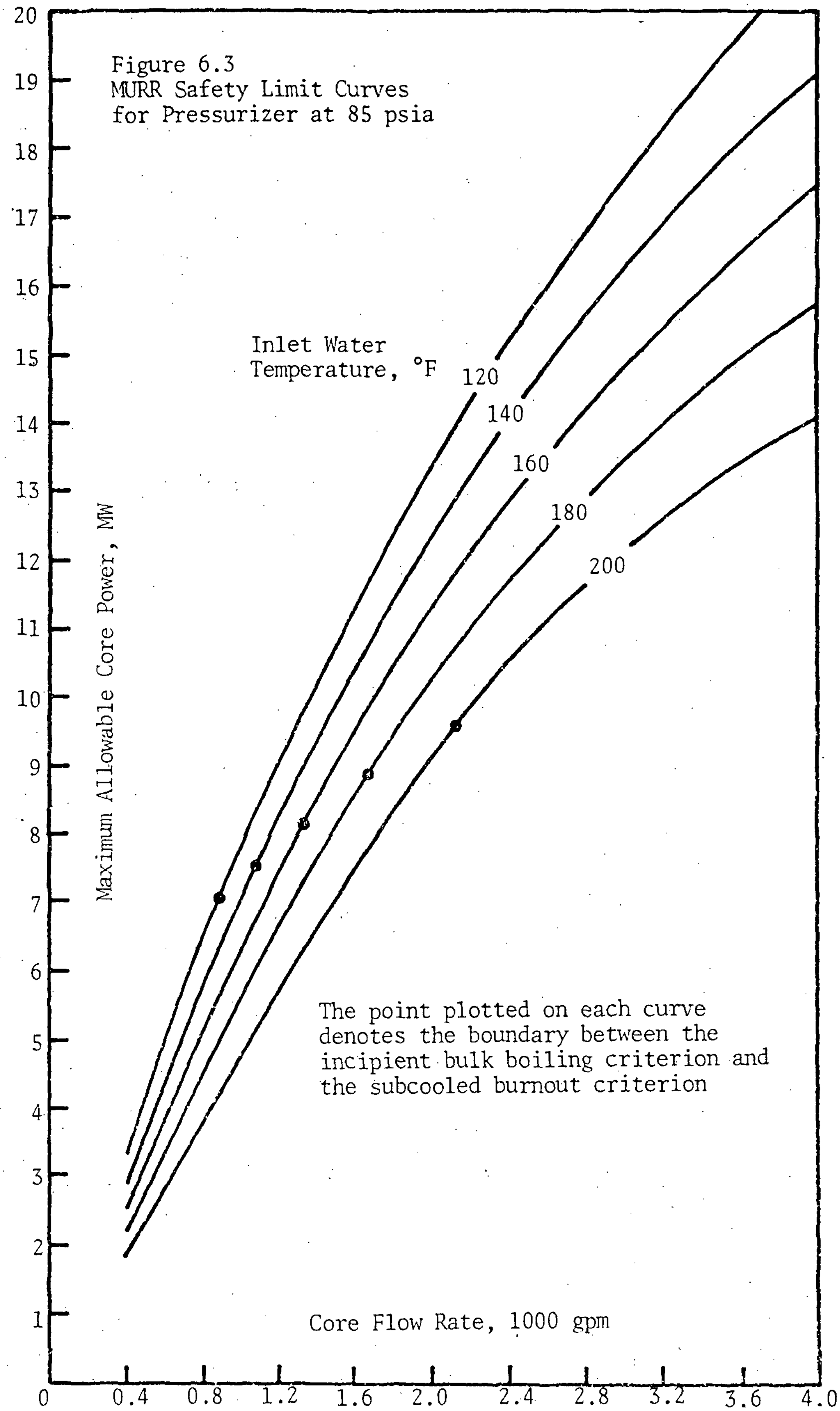


Figure 6.3
MURR Safety Limit Curves
for Pressurizer at 85 psia
Inlet Water
Temperature, °F
120
140
160
180
200
Maximum Allowable Core Power, MW
The point plotted on each curve
denotes the boundary between the
incipient bulk boiling criterion and
the subcooled burnout criterion
Core Flow Rate, 1000 gpm
0
0.4
0.8
1.2
1.6
2.0
2.4
2.8
3.2
3.6
4.0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20

TABLE 6.1
SAFETY LIMITS FOR MURR OPERATION

Maximum Allowable Core Power Level, MW with Pressurizer at 60 psia

INLET WATER CONDITIONS

Temperature °F	Flow Rate, gpm 400.	800.	1200.	1600.	2000.	2400.	2800.	3200.	3600.	4000.
120.	3.011	5.870	7.980	9.843	11.574	13.099	14.426	15.450	16.217	16.654
140.	2.650	5.262	7.299	9.035	10.582	11.960	13.155	14.071	14.729	15.075
160.	2.292	4.546	6.675	8.202	9.600	10.822	11.877	12.669	13.228	13.501
180.	1.935	3.834	5.667	7.409	8.612	9.685	10.603	11.267	11.715	11.906
200.	1.583	3.131	4.615	6.009	7.282	8.400	9.301	9.863	10.204	10.267

Maximum Allowable Core Power Level, MW with Pressurizer at 75 psia

INLET WATER CONDITIONS

Temperature °F	Flow Rate, gpm 400.	800.	1200.	1600.	2000.	2400.	2800.	3200.	3600.	4000.
120.	3.278	6.334	8.647	10.742	12.668	14.435	16.050	17.394	18.532	19.438
140.	2.916	5.798	7.939	9.906	11.667	13.282	14.746	15.967	16.993	17.782
160.	2.556	5.080	7.317	9.067	10.676	12.138	13.458	14.534	15.437	16.139
180.	2.197	4.363	6.474	8.236	9.680	10.988	12.152	13.104	13.892	14.467
200.	1.843	3.656	5.415	7.099	8.686	9.845	10.868	11.689	12.339	12.810

Maximum Allowable Core Power Level, MW with Pressurizer at 85 psia

INLET WATER CONDITIONS

Temperature °F	Flow Rate, gpm 400.	800.	1200.	1600.	2000.	2400.	2800.	3200.	3600.	4000.
120.	3.292	6.584	9. 097	11.299	13.336	15.227	16.878	18.356	19.662	20.776
140.	2.930	5.860	8.421	10.452	12.326	14.062	15.568	16.920	18.108	19.102
160.	2.570	5.139	7.709	9.607	11.326	12.908	14.271	15.488	16.549	17.434
180.	2.211	4.421	6.632	8.766	10.319	11.749	12.969	14.051	14.980	15.741
200.	1.856	3.712	5.568	7.424	9.280	10.593	11.673	12.617	13.418	14.069

NOTE: Underlined power levels are limited by bulk boiling.

6.3 Calculational Methods

As described in the previous analysis (1) the BOLERO and MURRPGM computer codes were used to model the steady state thermodynamic and hydrodynamic operation of the MURR reactor. The singularity of design makes experimental determination of an exact DNB correlation for every individual research reactor prohibitively expensive. A literature review demonstrates that the Advanced Test Reactor (ATR) closely compares to the MURR in fuel design, although the ATR core is twice as long as that of the MURR. Extensive experimental tests (2,3) were made on mockups of ATR coolant channels to determine the most accurate of the numerous DNB burnout correlations available. It was observed that the limiting conditions for ATR operation were set by subcooled burnout due to hydraulic instabilities in the hot channel. This was found to occur at 60% of the DNB heat flux predicted by the widely used Bernath correlation (4). To provide a reasonable margin between predicted DNB conditions and the MURR safety limits, the safety limit criterion was established that the local heat flux at any point in the core shall be less than 50% of the burnout heat flux given by the Bernath correlation at that point.

A parameter of safety significance for nuclear reactors is the DNB ratio (DNBR) defined as the ratio of the anticipated DNB heat flux to the actual peak reactor core heat flux. Thus, DNB and associated fuel damage will not occur as long as the DNBR is greater than 1.0. For the study in question, safety limits were derived based on a DNBR of 2.0 relative to the Bernath correlation, i.e., conditions were limited to 50% of the predicted burnout heat flux. However, in relation to the experimentally observed burnout at 60% of the Bernath prediction, one can say with assurance that the DNBR for the MURR safety limit curves is not less than 60%/50% or 1.2. Thus, the derived curves allow sufficient margin between the safety limits and actual predicted DNB. The usual conservatism of worst case power peaking and nonuniform fuel loading and appropriate hot channel factors were used, lending greater assurance that the MURR reactor will not approach DNB under the most severe anticipated transient from the proposed 10 MW power level.

The safety limit criteria for Mode I and II operation with core flow rates below 400 gpm has been changed to a fuel plate cladding surface temperature of 366°F. This number is derived from additional calculations made with the PARET code for the Loss of Flow Accident. The input data for these calculations was identical to that used in the analysis summarized on pages D-9 through D-13 of Addendum 4 (1), with the exception of the core pressure. In the previous analysis it was assumed that the core was completely depressurized through the anti-siphon system. In the proposed design of the process control system (Figure A.5 of Add 4) the anti-siphon valves (543A/B) do not open on a loss of flow and thus core depressurization is not credible for this accident.

The latest PARET results for a core pressure of 75 psia (scram set point) and the worst case loss of flow accident yield a maximum cladding surface temperature of 327°F and a DNB surface temperature of 366°F. The DNB ratio for this maximum surface temperature is 2.84 and thus there is sufficient margin to ensure that this accident will not result in core damage.

References

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