Armed Forces Radiobiology Research Institute

AFRRI R-84 Annual Report
October 1980

Part A - Changes in the Facilities and Procedures

This section specifies actions taken during the report period that reflect changes to the FSR not previously reported to USNRC. AFFRI is currently preparing a complete update of the FSR, portions of which have been submitted to the USNRC as part of our relicensing effort. The actions taken are categorized on a chapter basis as listed in the FSR. When complete, a copy of the updated FSR will be forwarded to the USNRC.

Chapter I - Site

A new animal facility is almost complete on the eastern side of the AFRRI. No permanent changes have, as yet, been required for the emergency evacuation plan or the environmental monitoring plan.

Chapter II - Facility

No changes have been made that could reflect on reactor operation.

Chapter III - Reactor

No changes to the reactor or reactor systems were made which affected the use or intent of any system and no changes to the Technical Specifications were made. A new set of Technical Specifications were submitted to the USNRC shortly after this reporting period as part of AFRRI's relicensing effort.

A change in the transient rod was made and reported to the commission in February of 1980. A copy of the replacement report is enclosed.

The following is a listing of malfunctions that occurred and action taken.

(1) Malfunction: Pin hole leak in transient rod (see attachment).

Action taken: Transient rod replaced.

(2) Malfunction: During a scheduled check of fuel temperature measuring channels and installed thermocouples, a defective thermocouple (earlier identified) was placed on fuel temperature safety channel two. The defect was such that upon reaching a temperature of about 250°C the junction would open thereby causing a scram. At lower temperatures the junction read correct. After testing, this thermocouple was left on safety channel two. As the reactor power was increasing and reached a fuel level of 250°C a scram occurred.

Action taken: The proper thermocouple was placed on fuel temperature safety channel two. During this event all safety systems performed their intended functions.

(3) Power failure resulting in damage to safety channel one (see attached report previously submitted).

(4) Two minor malfunctions occurred; (a) a failure of the steady state timer resulting in an early scram and (b) a failure of a final amplifier in the safety channel two during a weekly instrumentation checklist training session. During these events all safety systems performed their intended functions.

Chapter IV - Experimental Facilities

The IN CORE experiment tube (CET) previously used on a intermittent basis was set up in core position F-28 for continuous use. No other changes have been made in the experimental facilities.

Chapter V - Hazard Analysis

Although no changes have been made that would affect the hazard analysis, a new hazard analysis has been prepared and submitted to USNRC as a part of relicensing. A copy of that analysis is enclosed.

Chapter VI - Nuclear Analysis

No changes have been made which affect the nuclear analysis.

Chapter VII - Organization

No change to the AFRRI administrative structure have been made. A list of current personnel follows.

Director - CAPT Paul E. Tyler, MC, USN

Deputy Director - COL Bobby R. Adcock, MSC, USA

Chairman, Scientific Support Department - LTC Felix Owen, USA

Chief, Radiation Sources Division - MAJ Ronald R. Smoker, USA

The Reactor Staff has a current staffing as follows:

Physicist-In-Charge - MAJ Ronald R. Smoker, SRO

Chief Supervisory Officer - Mr. Mark Moore, SRO

Reactor Operator - Capt Joe Sholtis, RO

Reactor Operator - SFC Charles Harmon, RO

Reactor Operator Trainee - HMC Virgil Pryor

Reactor Operator Trainee - CPT Len Alt

Reactor Operator Trainee - Angela Melcher

Reactor Operator Trainee - SP6 Larry Vernon

The current membership of the Reactor and Radiation Facility Safety Committee is as follows:

COL Bobby R. Adcock, Chairman - AFRRI

LTC Felix Owen - AFRRI

CDR L.J. Beuchler - NNMC

MAJ H. Johnson - AFRRI

MAJ R. Smoker - AFRRI

Mr. W.L. Gieseler - Consultant (HDL)

Mr. Mark Moore - AFRRI

Mr. T.G. Hobbs - NBS

Mr. Lester Slaback - AFRRI

Mr. J. Stone - NRL

LT Kenneth Ferlic - AFRRI

Mrs. B. Markovich - Recorder

Chapter VIII - Procedures

There have been no significant procedure change since the last report.

Part B - Tests and Experiments

1. No test or experiments were performed during this reporting period that exceeded the limits stated in the Technical Specifications.

2. All experiments performed during the reporting period were in accordance with Technical Specifications and authorized as either "Routine" or "Special".

3. Changes in experimental procedures have been made such that operation and performance have been uneffected; however, the amounts of effluent releases have been decreased.

4. The reactor was utilized at the various experimental facilities such that the total power production was:

   a. in Steady State - 57991 Kw-hr.
   b. in Pulse - 141 Kw-hr.

5. Experimental workload for FY 81 is anticipated to be the same or slightly greater than FY 80.

REPLACEMENT OF THE AFRRI-TRIGA REACTOR TRANSIENT ROD

During the AFRRI Reactor Annual Shutdown an inspection of the control rods revealed a pin hole leak in the upper weld of the poison followed rod used in core position A1. Position A1 is the transient rod used to provide pulsing capabilities to the TRIGA Mark F Reactor. Only two reactors, both Mark F types, were installed by General Atomic using a poison followed transient rod, one at AFRRI, and one at the Army's now decommissioned Diamond Ordinance Radiation Facility.

AFRRI is currently the only facility using a poison followed rod in position A1. The replacement cost of the control rod is high and delay in manufacturing can run as long as 5 months. The alternative is to replace the poison followed rod with a standard TRIGA transient rod using an aluminum follower. To insure compliance with 10 CFR 50, paragraph 59, the following reviews safety and operational aspects of such a change

The AFRRI-TRIGA Mark F Reactor Final Safeguards Report describes nine hazard analyses. Of these nine, only two are concerned with reactivity worth changes such as would result from a change in the construction or materials used in a control rod. These two concern predicted reactivity changes and a failure of the fuel element cladding.

The only difference in a poison followed control rod and an aluminum followed control rod is the addition of a 5/16" diameter core of poison material in the core of the aluminum cylinder that comprises the lower section of the rod (see Figure 1).

The effect of a control rod is to remove or insert negative reactivity to the reactor core. A control rod follower is any material that enters the core as the main poison section exits the core. The poison followed control rod replaces the main poison section with a smaller poison section thereby reducing to net effect (addition positive reactivity) to the core. An aluminum followed control rod replaces the main poison section with a solid aluminum section which has lesser value thereby increasing the net (addition positive reactivity) effect to the core. When determining the value of control rods, their worth can be measured against any known reactivity value of other materials, such as fuel or other control rods in the core. In comparing the two types of transient rods the worths given will be stated in terms of the known reactivity values against which they were measured.

The main poison section of the poison followed control rod was measured using the standard period method whereby small withdrawals of the rod are made and, from the resultant positive period, calculations of the dollar worths are made. Using such measurements, the main section of the poison followed rod measured \$3.04 from full down to full up. This is with the poison follower section replacing the main poison section in core. The poison follower of the rod was measured relative to the worths of the safe, shim, and reg. rods. The poison follower section measured \$1.22. The total worth of the rod is \$3.04 when followed by the poison follower and \$4.26 when followed by a water void.

The aluminum followed control rod was measured using the same procedure. Its worth was \$3.86 when followed by the aluminum follower. The aluminum follower was measured as worth \$0.23 against a water void. This gives a total value of the main poison section against a water void of \$4.09. The data used to tabulate these results are contained in Table 1.

There is no difference in the two rods from a mechanical standpoint, each has the same probability of malfunction. Therefore, the primary safety consideration is the result of a malfunction whereby the highest possible positive reactivity could be introduced into the reactor core. This would occur if the main poison section of the rod were to be instantaneously removed from the core followed by a water void.

Section 4 of the Hazard Summary Report describes this event as "THE INCREDIBLE ACCIDENT" of a complete transient rod removal from the core due to the failure of the connection between the pneumatic piston and the shock absorber of the transient drive assembly (core critical below 1 KW). The Final Safeguard Report states that if such an accident did occur, the "peak measured fuel temperature would be between 650°C and 700°C." This fuel temperature is well below the 900°C integrity of aluminum clad fuel. The current AFRRI fuel is stainless steel clad which has a greater integrity level. Using the aluminum clad fuel temperature limits as a conservative maximum fuel temperature level, Fuch's Model calculations were made using the aluminum and poison followed control rod values. The result with the \$4.09 maximum available insertion from the aluminum followed transient rod was that the fuel temperature would reach a maximum of 578°C. This level is still below the 700°C in the already conservative original FSR.

From a safety analysis we see that total ejection of the transient rod would result in a prompt critical excursion that would not exceed the safety limit of the fuel element cladding. There are several aspects to consider in converting to an aluminum followed transient rod.

The technical specifications contained in AFRRI License R-84 states (Section II, part A-2) "The maximum step insertion of reactivity shall be \$3.28 (2.3% delta K/K) in the pulse mode." With a transient rod worth \$4.09, it would be possible for an operator to inadvertently insert reactivity above that in the specification. To prevent such an occurrence, a physical stop has been placed on the transient drive such that the limit of \$3.28 cannot be exceeded. This is accomplished by stopping the rod drive at approximately 11 inches of travel rather than the full 15 inches previously available (See Figure 2 for stop location).

Another aspect of converting to an aluminum followed transient rod is the inability of the core to achieve a critical configuration in either exposure room without using a small portion of the transient rod to overcome the reactivity losses to the rooms. Several pulsing type TRIGA reactors use, as a standard operating procedure, subcritical pulsing. Subcritical pulsing occurs when the transient rod is used in a withdrawn position equal to the total travel to be experienced in the intended pulse; the standard rods (safe, shim, and reg.) are then used to bring the reactor critical; the transient rod is dropped to the full down position; the value of the reactivity insertion desired is then placed by lifting the standard rods to the desired amount. The reactor is still in a subcritical state. Pulsing the transient rod from this position results in an induced transient. This type of pulsing is not prohibited by the reactor license, and operational procedures will be placed in appropriate documents.

A review of operational requirements contained in the reactor license, RSD instructions, AFRRI instructions, and operational instruction fail to disclose any prohibitions on the use of an aluminum followed transient rod or proposed operational changes.

The use of an aluminum followed control rod and the drive stop limit was accomplished at DORF on a twin reactor for several years without incident. Attached are the integral rod worth curves for both the aluminum and poison followed control rods. Symmetry indicates correct placement in core relative to fuel location of both rods.

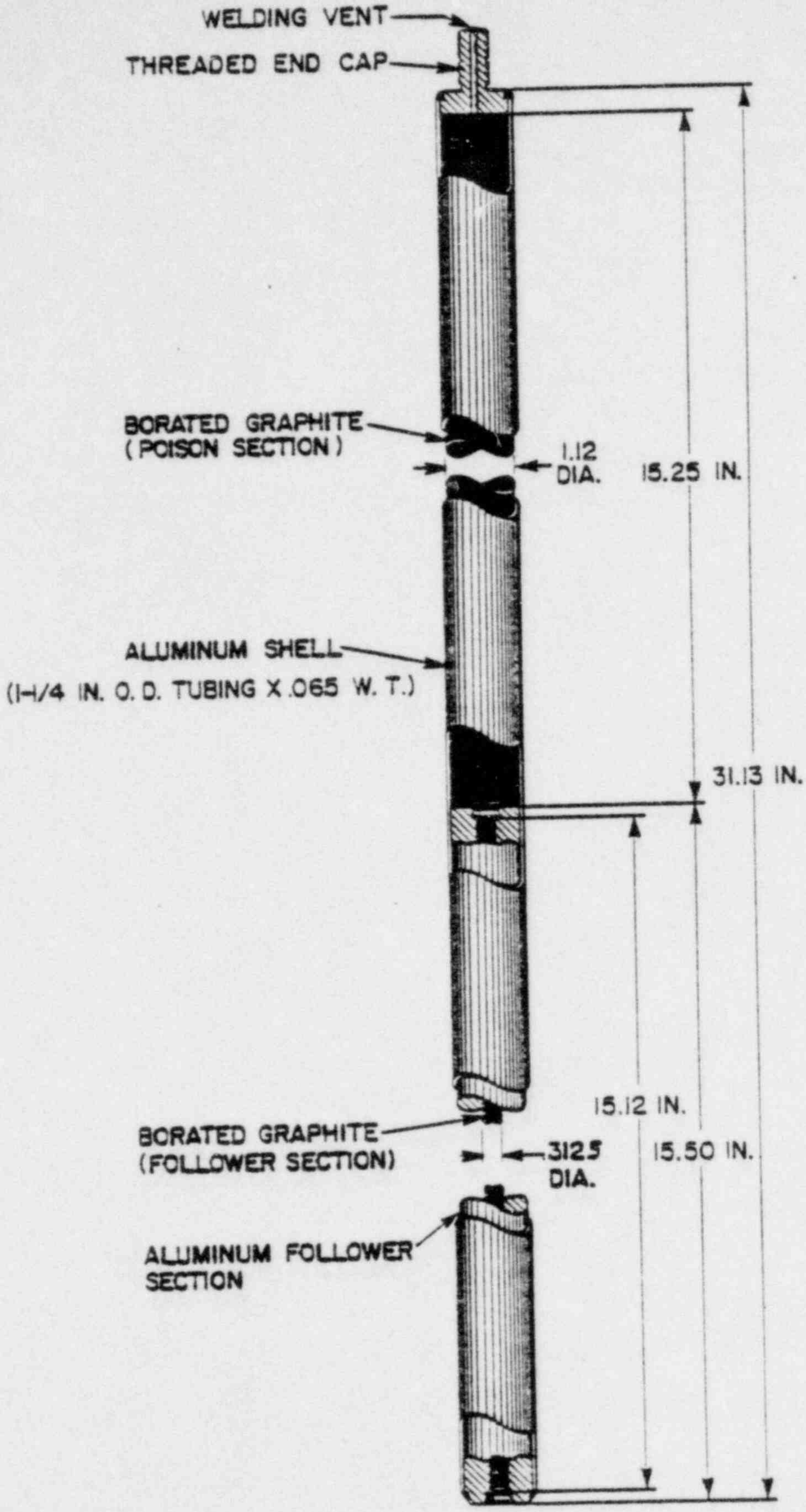
TABLE 1 - ROD COMPARISON

	POISON FOLLOWER	ALUMINUM FOLLOWER
Main poison worth with follower measured by period method.	\$3.04	\$3.86
Follower worth relative to water void.	\$1.22 (1)	\$0.23 (2)
Total rod worth relative to water void.	\$4.26	\$4.09
Fuch's Model maximum fuel (3) temperature on total rod loss	607°C	578°C
Maximum steady state rate of insertion.	\$0.073/sec	\$0.093/sec

(1) Measured by comparison to other rods.

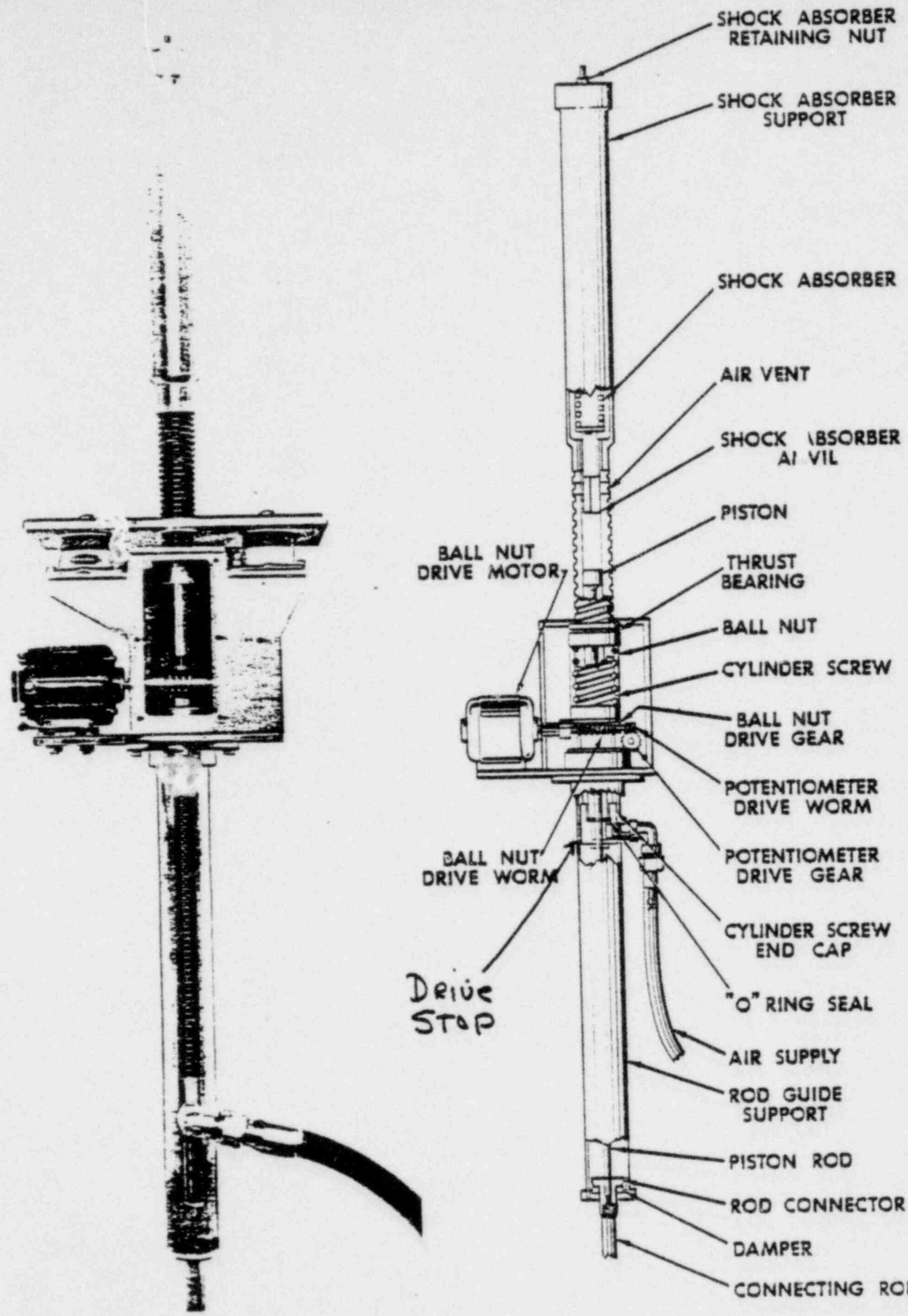
(2) Measured by comparison to other rods and fuel.

(3) Assumed starting temperature of 50°C.



Transient Rod

Figure 1



Transient Rod Drive Mechanism

Figure 2

K+E 10 X 10 TO THE CENTIMETER 18 X 25 CM KEUFFEL & ESSER CO. MADE IN U.S.A.

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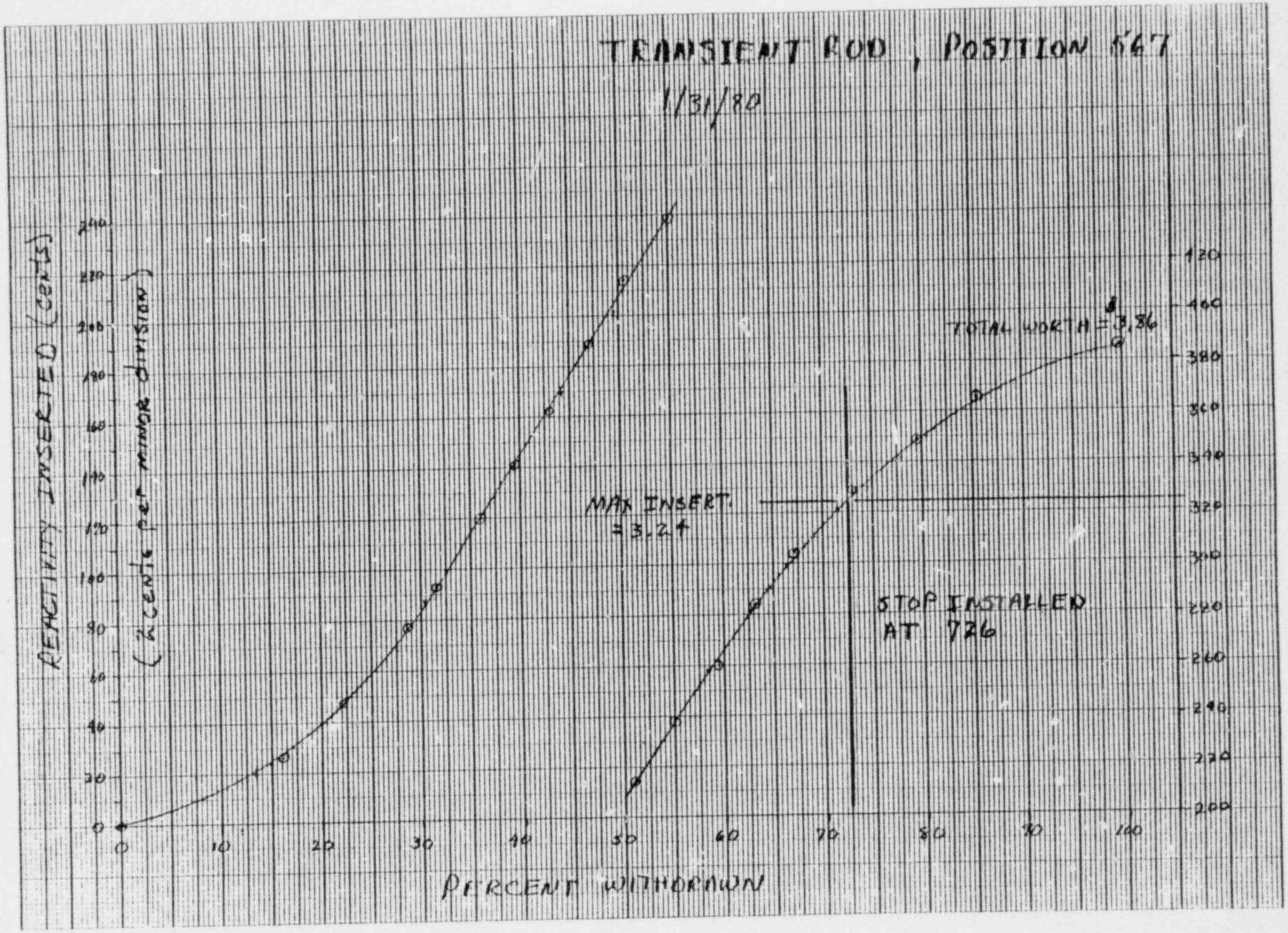
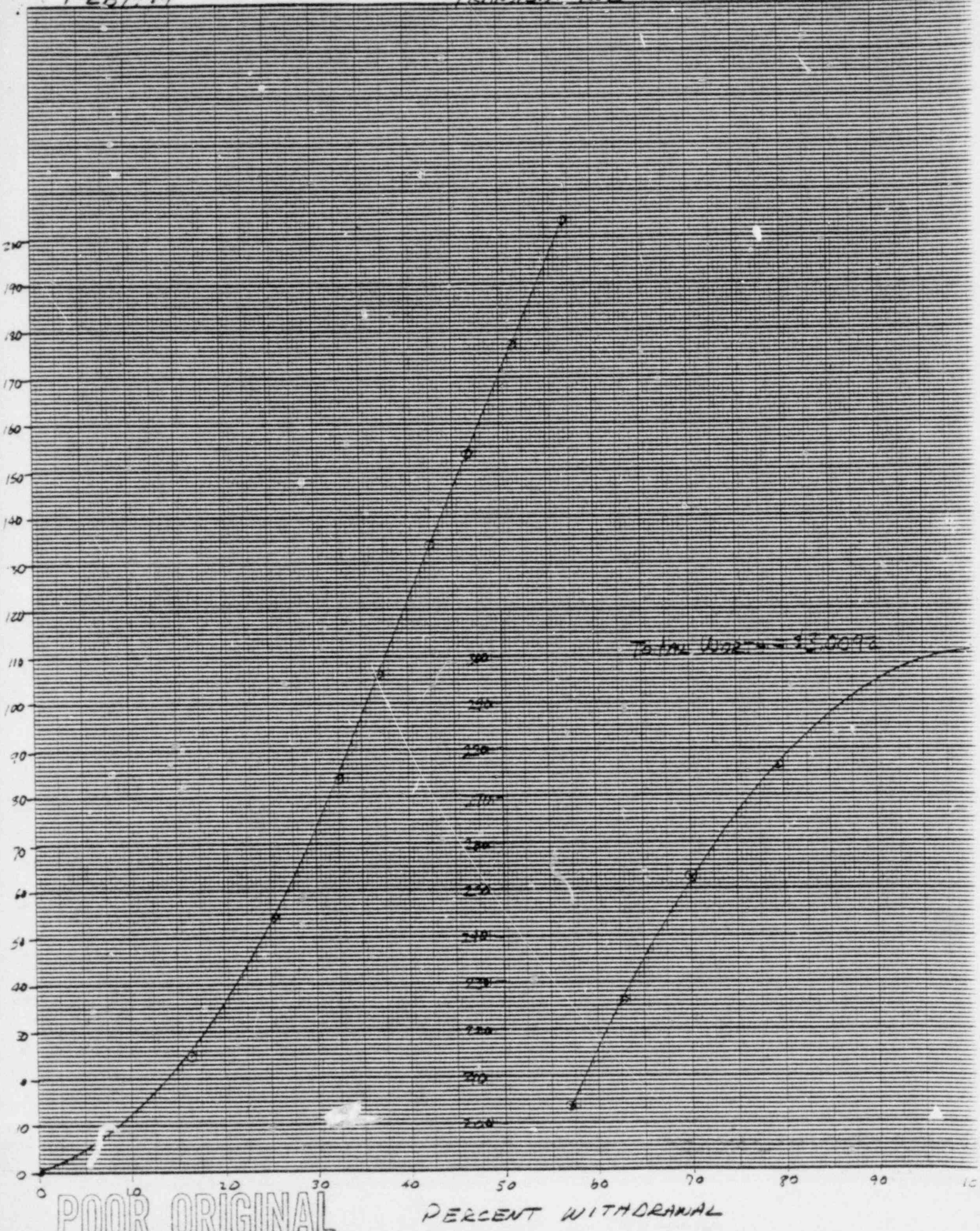


POOR ORIGINAL

FEB 1979

TRANSIENT ROD POSITION 903



FUCH'S MODEL

(average measured peak fuel temp.)

$$T = T_0 + 170.94 \times (\rho - 1)$$

where T_0 = 50 degrees C

$\rho - 1$ = size of pulse minus one dollar

For a pulse of $4.09 the maximum measured fuel temperature would be 578 degrees centigrade.

ELECTRONIC DAMAGE TO SAFETY CHANNEL ONE

I. History and Description of the Damage

On Saturday, 15 March 1980, the Physicist-in-Charge of the AFRRI Reactor Facility discovered that most scram indicators on the reactor control console were illuminated. Such a condition is consistent with a power failure or outage of the console. A query of the AFRRI Security Watch revealed that between approximately 1600 hours, 14 March, and 0400 hours, 15 March, a survey of the AFRRI electrical systems was in progress. The nature of the survey was such that power could have been inadvertently removed and reapplied to the console. All scram indicators were reset and all systems appeared to be functioning normally.

On Monday, 17 March 1980, a Weekly Nuclear Instrumentation Checklist was performed expressly to verify console operability in light of the events of 14 and 15 March, and prior to any reactor operation. This inspection revealed that Safety Channel One would not initiate a scram in accordance with the Technical Specifications of Reactor Licence R-84. Safety Channel Two and all other instrumentation were fully operational.

The operational amplifier on a Safety Channel One circuit board had been damaged when electrical power had been reapplied to the console after the power outage. Normally that amplifier would have had an output of 10 volts with an input of 1 milliamp and a scram set point for the circuitry at 11 volts. The damaged amplifier had a maximum saturated output of approximately 8.4 volts. Thus, at the maximum authorized steady state power of 1000 KW, Safety channel One would have indicated about 840 KW. Safety Channel Two and all other nuclear instrumentation would have performed properly.

II. Corrective Action

The circuit board containing the damaged operational amplifier was replaced. At that time, Safety Channel One was calibrated, fully operational, and capable of performing its intended function. Additionally, a relay requiring manual reset was installed in series with the console so that, in the event of a power outage, the console circuitry is more fully protected.

The electrical survey performed on 14 March, was repeated on 22 March 1980. This survey revealed that the electricians involved had inadvertently thrown a circuit breaker that provided power to the console. That circuit breaker has been more clearly marked to prevent the recurrence of that particular event.

III. Safety Review

The damage done to Safety Channel One does not represent an unreviewed safety question and did not endanger the health and safety of AFRRI personnel or of the general public. The AFRRI Reactor and Radiation Facility Safety Committee will review this report at its next scheduled meeting.

Enclosure (2)