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SM-1
RESEARCH AND DEVELOPMENT
QUARTERLY REPORT

October 1 to December 31, 1959

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ABSTRACT

This report covers Research and Development conducted at the SM-1 under Contract AT(30-3)-326 from October 1 to December 31, 1959, including evaluation and analysis effort at Schenectady, New York.

Progress made in plant water chemistry and health physics practices, and progress on individual tasks and tests is reported. Plant modifications and special projects which required engineering by the Research and Development staff are discussed.

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SUMMARY

WATER CHEMISTRY. Three incidents of high chloride content in the secondary system occurred. These were caused separately by a plugged evaporator blowdown line, improper evaporator operation, and defective commercial sodium sulfite.

No particular chemistry problems were observed in the primary system.

HEALTH PHYSICS. Health physics monitoring was necessary as a result of routine and certain non-routine shutdowns. No excessive demands were made by prolonged or involved shutdowns. One technical overexposure resulted from work on the rod drive mechanisms. No accidents or unusual circumstances developed. A proportional flow counter and additional laundry equipment were approved for purchase and use at SM-1. Modifications of the mobile air monitor were approved and new components will be installed upon receipt from the manufacturer.

TASK XI. A review of previous study efforts to establish criteria for evaluating all plant controls and instrumentation was completed. Conclusions and summaries were made and recommended solutions to problems given. Both stop-gap and redesign approaches to the solutions were evaluated. A final report of the study phase was issued in October 1959 as APAE No. 52, (Volumes I and II).[1]

TASK XII. The resistance-reactance load bank was delivered to Fort Belvoir on December 29, 1959. Instructions for operation and maintenance of this equipment will be delivered to Alco in January 1960. The location site for the skid was completely graded and foundation members and overhead power cable supporting poles were delivered. The points of electrical connection for this equipment were completed, but actual connection of the unit cannot be made until shutdown of the plant.

Delivery of high-pressure turbine parts for operation at higher throttle pressures is expected early in January.

Selection, procurement, and delivery of most instrumentation items for sensing and recording of selected system parameters was accomplished. A final list of these parameters was published.

Design of a steam by-pass system for 100% of plant rated steam capacity was completed and quotations received from vendors for control and desuperheating features of this by-pass line. Purchase approval was requested of the

AEC. Installation design of this steam by-pass system is under way, but the bulk of design work must await contact with the selected vendor.

Test procedures were written and are being reviewed.

TASK XIV. Task activity continued in the direction of the blocked channel method of instrumenting fuel plates. The time lag between meat temperature change and thermocouple response was determined. Nuclear perturbation from an instrumented plate was calculated. Design work was completed on the instrument "table" and on methods of supporting the instrument leads.

TASK XVI. Work on Task XVI was initiated in November 1959 with the objective of decontaminating the SM-1 steam generator as outlined in APAE No. 43, Volumes I, II and III.[2] Planning of the task was completed. A detailed program and schedule was prepared to accomplish the decontamination scheduled for the period March 15 to April 30, 1960.

A preliminary design and layout was made of a special plug assembly and of piping required to isolate the steam generator from the remainder of the primary system when decontaminating the steam generator. Corrosion specimens simulating metallurgical conditions in the steam generator were fabricated and made ready for testing in the decontaminating solutions.

TEST SERIES 100. Analytical chemistry data was obtained on Test 101. Conclusions and recommendations were developed.

Accumulation of data on samples of the inner shield tank water was continued in support of Test 102.

The Beckman O_2 Analyzer and pH instrumentation was installed in the secondary system. The pH instrumentation was placed in service and operation has been satisfactory.

Preparatory work continued on test evaluations and test procedures on other tests in this series which are scheduled in the plant during the next quarter.

TEST SERIES 200. Work continued in the accumulation and analysis of data for tests in phase I Activity Buildup, Core I.

Test 201 in phase 5 Fission Product Studies, Core I, was completed. From analysis of data it was concluded that a cladding defect exists in Core I.

Test 205 was completed and the final report issued.

TEST SERIES 300. Tests were run for core physics measurements. Data was recorded during a scheduled research and development plant shutdown. Preliminary data reduction was accomplished.

TEST SERIES 400. Test procedures for gamma and neutron flux measurements in the primary shield and gamma and neutron flux measurements in the instrument wells were revised. Particular attention was given to design of equipment and procedures for measurement of dose rates on spent element surfaces to determine the relative fuel burnup.

TEST SERIES 500. Test 500 was completed in the plant. The final report will be written and issued in the next report period.

TEST SERIES 600. Test 600, "Loss of Flow Accident," was evaluated and test procedures submitted. Studies and analyses show that SM-1 may be assumed safe from burnout in the event of pump failure without scram for periods of at least 5 seconds and probably longer.

PLANT MODIFICATIONS AND SPECIAL PROJECTS. Considerable engineering time was spent on plant modifications and special projects. Only those items on which engineering recommendations were made are included in the report. These are in summary form in Section 6.0.

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1.0 INTRODUCTION

The plant was scheduled for training in operating procedures during October and the first two weeks of November 1959. Research and Development tests proceeded whenever compatible with the training schedule. The plant was shut down for repairs to rod drive seals and the circulating water pump during the last two weeks of November.

Test 205, "Effectiveness of an SM-1 Demineralizer for Reducing the Activity of Radioactive Liquid Waste" and Test 500, "Field Test of Minneapolis Honeywell BF_3 Pulse Transformer Channel" were completed in the plant during the training period in October and November.

Throughout December, R & D testing was scheduled. A ten-day full power run for fission product buildup was completed on December 8 for Test 201. In-plant data for Test 201, "Iodine Levels in the Primary Coolant" and Test 101, "Secondary Water Chemistry at Elevated pH Levels" were completed in this period.

Test series 300, "Core Physics Measurements" were started on December 17 after approximately 13.5 MWYR of energy release. In-plant data was completed on December 23.

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PLANT WATER CHEMISTRY

2.0 PLANT WATER CHEMISTRY

2.1 SECONDARY SYSTEM

The evaporator continued to be the main source of unsatisfactory water conditions in the secondary system. On two documented occasions high chloride in the steam generator was traced to improper evaporator operation. On October 31 and November 4, 1959 routine analysis of the steam generator blowdown revealed the presence of chloride above specification limits. Correction of these conditions involved increasing steam generator blowdown to the capacity of the cooler (about 1 gpm) and insuring that a sulfite residual greater than 10 ppm was maintained in the steam generator. A check of the secondary system showed that high chlorides were due, in the first case, to a plugged evaporator blowdown line and in the second, to incorrect operation which resulted in an excessive steaming rate.

Another incident of high chloride and hardness occurred during performance of Test 101. As part of the test, bi-hourly measurements were made for chlorides in the steam generator blowdown. The analysis immediately prior to the discovery of high chlorides revealed that the chloride level was well within specification limits. As soon as the high chloride level was verified, an immediate investigation was made to determine the source of impurities. An analysis of the secondary system revealed that the chloride and hardness were present only in the steam generator. A check of the commercial sodium sulfite showed the presence of both chlorides and hardness. The use of this material was temporarily discontinued pending delivery of fresh stock, and C. P. grade sodium sulfite was substituted. During the period of high chloride content, the steam generator blowdown was maintained at 1 gpm. The time for chlorides to return to within the specification limit of 0.5 ppm was approximately 10 hours.

2.2 PRIMARY SYSTEM

During this report period no particular problem in chemistry was observed in the primary system. However, one demineralizer was replaced because of the acidic nature of the effluent water, which indicated that the anion resin was exhausted sooner than the cation resin. Since the SM-1 primary system should develop a higher cation loading than anion loading, the cause of the early exhaustion of anion resins is, as yet, undetermined. One possible explanation was that preferential deposition of suspended material on the anion resins rendered them ineffective. Future studies of demineralizer effectiveness will cover this condition.

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HEALTH PHYSICS

3.0 HEALTH PHYSICS

3.1 NEUTRON SOURCE CONTROL

During the seventh bi-monthly health physics meeting, action was initiated to establish procedures for transporting the Pu-Be neutron source to the training annex and provide for safe storage over an extended period of time or possibly on a permanent basis. Previously, the source was stored in a paraffin shield which was adequate for personnel shielding requirements; however, the shield presented a potential hazard in event of fire. It was recommended therefore, that a permanent storage well be constructed. Drawings of a suitable well were submitted to the Site Representative.

A request was submitted to Lexington Signal Depot to supply NTA neutron sensitive film for personnel monitoring purposes. The first shipment was to be available for use during the first week in January 1960; continued usage will be made as long as need for this type of monitoring exists. Assignment of badges is on a group basis, with badges positioned in highest dose rate areas occupied by personnel using the neutron source. Exposures are observed and recorded on the personnel dosimetry record DA 1141, as are results of the usual Lexington Photodosimetry Reports. The lower limit of sensitivity of these badges is 60 mrem.

3.2 PLANT MODIFICATIONS

A proposal recommending modifications in the Nuclear Measurement Corporation Mobile Air Monitor was submitted and approved by the Site Representative. In effect, the modification will update the equipment to make it comparable to currently available models. Advantages gained from this modification will be increased sensitivity and greater ease and reliability in calibration.

3.3 PERSONNEL MONITORING

During routine processing of personnel film badges, the film badge worn by M/Sgt. J. H. Nicoll was found to indicate an exposure of 880 mrem received during the period November 15-29, 1959. The exposure was within the limits that define a technical overexposure, i. e., the exposure was less than 900 mrem in any seven consecutive days. Since all information appears as after-the-fact, it is extremely difficult to make an exact analysis of the situation or circumstances leading to the incident. An extensive investigation of the circumstances indicated no unusual radiation conditions or descrepancies in monitoring techniques. A whole body count was made which indicated no increase in internal deposition of radionuclides. A formal technical overexposure report was submitted as well as a detailed investigation of the circumstances surrounding the incident.

3.4 INSTRUMENT CALIBRATION

A short paper outlining the present capability of the SM-1 to calibrate portable health physics survey instruments was made and submitted to the Site Representative. The calibration requirements of several instruments currently used were included along with a recommended solution.

3.5 WASTE RETENTION AND DISPOSAL

No shipments of radioactive waste were made during this report period.

A total of 1.01×10^{-2} curies of liquid radioactive waste was released to the Potomac River under controlled conditions. No adverse affect was observed on river biota or use of the water for human consumption. Chemistry analysis was responsible for much of the waste because large quantities of water were flushed through the system before samples were taken, and a large quantity of primary water was used for R & D testing of various demineralizer systems.

TASK XI

4.0 TASKS

4.1 TASK XI INSTRUMENT AND CONTROL STUDY

4.1.1 Introduction

In the preceding quarter (July-September 1959), a study to establish criteria capable of evaluating all the controls and instrumentation in the plant was largely completed. The criteria were employed for evaluation of SM-1 controls. Tentative conclusions were arrived at concerning the problems and their solutions. Details of the Task XI background, including the above progress, were included in the previous quarterly report, APAE Memo No. 237.[3]

4.1.2 Work During Period

The R & D group completed a review of its own study efforts and those of two consultants. Conclusions and summaries were drawn in final form stating the problems and recommended solutions. Both stop-gap and redesign approaches to the solutions were evolved, and these were evaluated. The instrument and control study report APAE No. 52, was written and issued in this period.

A list of suggested priorities was prepared for the stop-gap program and forwarded to the Site Representative in December. Material to be incorporated into the detailed specifications was accumulated on the basis of the priority list.

4.1.3 Principal Conclusions of the Final Report

1. SM-1 controls and instrumentation difficulties stem largely from obsolescence in both equipment and installation methods.

2. The three major categories of controls and instrumentation that have caused difficulties are listed in descending order of importance:

   a. nuclear instrumentation (operational)
   b. nuclear instrumentation (health physics)
   c. process instrumentation

3. Major difficulties with the controls and instrumentation are categorized as follows:

   a. equipment unreliability (inherent in design, which includes:

      (1) few or no examples of static and solid-state parts and circuitry

(2) lack of adherence to high-quality standards, such as MIL, JAN or other specifications

b. lack of human-engineering design factors which would make operations and/or maintenance reasonable simple and require minimum technical knowledge, including:

(1) centralization of controls, recording, and indication into compact units, such as the proposed Main Control Unit

(2) modular, plug-in and similar construction

(3) miniaturized techniques

(4) built-in and automatic test facilities andtest points

(5) adequate instrument shop and equipment facilities

(6) proper separation of manual controls for operators and maintenance technicians

(7) electrical safety requirements of a high order

(8) modern wiring, cabling and wireway techniques

(9) adherence to modern standards of marking, identification, etc.

(10) techniques minimizing noise, ground loops, etc.

4. The general redesign approach to controls and instrumentation will produce more than twice the advantages of a stop gap approach at approximately half again the the cost.

5. Some recommended equipment developments include:

a. solid-state annunciator
b. health-physics general-purpose instrumentation and controls
c. nickel-cadmium station batteries (installation)
d. new power supplies
e. scanning comparator for redundant circuitry

6. A study of emergency natural environmental effects on plant controls and instrumentation is recommended.

7. A vitally needed combined test facility, library, informational "clearing house," and environmental test facility is recommended and discussed.

4.1.4 Future Work

Detailed specifications will be prepared to describe accurately the requirements of equipment under the stop-gap concept. These will be employed to purchase remedial equipment to the monetary limits imposed by the 1959-60 fiscal budget. Purchases will be committed by June 30, 1960. It is expected that much of the equipment will be delivered for installation during the first half of FY 1960-61.

4.1.5 Problems

Except in the proposed area of developments, no problems are anticipated in carrying out the recommendations. Further evaluation in the development area will have to be made as these developments reach new stages.

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TASK XII

4.2 TASK XII PLANT RESPONSE AND SYSTEM PERFORMANCE

4.2.1 Introduction

Task XII is designed to investigate experimentally the various factors in SM-1 affecting transient response characteristics, in order that analytical techniques can be refined and design programs of future nuclear power plants for the military can proceed with a greater fund of proven techniques. A previous quarterly report, APAE Memo 237, included the detailed objectives and scope of this task.

4.2.2 Work During Period

1. Subtask 1 - SM-1 Plant Characteristics Under Transient Loading

   a. Instrumentation - Prior arrangements had been made to borrow a recording oscillograph but arrangements could not be confirmed for the period of the revised running schedule. Firm commitments for this instrument or a replacement will be made early in the next quarter. The Sanborn preamplifier was received and checked out. Work on calibrating and balancing the systems is in progress. Because of the cost and difficulties in obtaining an adequate electronic instrument, the frequency parameter (5.3) has been temporarily deleted to allow use of this instrument channel for the third stage turbine extraction. Instrument installation design is continuing.

   b. High Pressure Turbine Parts - We have received revised prints on the turbine modifications from the General Electric Company. The revised outline print of the turbine shows no dimensional changes, and the low initial steam pressure and motor operated load limit device have been deleted. These items are being checked for accuracy with General Electric.

   c. Steam Separator - Quotations were evaluated and purchase approval requested.

   d. Testing - Test procedures were reviewed and a final draft is being prepared.

2. Subtask 3 - Installation and Checkout of Electrical Load Bank

   The resistance-reactance load bank was delivered to the ERDL area at Fort Belvoir and is ready for installation at the site. Ground was graded and foundation supports placed on hand to permit location of this equipment. Electrical connection of the load bank must await anticipated shutdown of the plant at end of March. Laying of connecting cable and design of overhead supports and under-the-road conduit are in final planning stages, and some installation work has begun.

Installation drawings were completed and the 45-foot pole for the 4160-volt line was installed. Steps were taken to obtain an outside electrical contractor to make the 4160-volt tie-in and line run. The control wiring will be accomplished by plant personnel.

3. Subtask 4 - Installation and Testing of Steam Dump Line with De-Superheater

Systems design of the steam by-pass was completed and vendor quotations received for the control system. The control system selected contains a variable orifice de-superheater to permit close control of the steam inlet temperature to the condenser over the range of flow from 2% to 100% of plant-rated steam capacity. All other vendor quotations received included a steam assist de-superheater which will permit close control over only the range from 15% through 100% of a rated load. Purchase approval for this system was requested of the AEC. Installation designs for the steam by-pass system are under way, but a major portion of work cannot be completed until final purchase approval is granted and direct contact with the selected vendor established.

4.2.3 Future Work

Work on Task XII during the first quarter 1960 will cover an intensive program of installation design, equipment procurement and manufacture, as well as installation of many items and final preparation of test procedures, data sheets and instructions. Analytical phases of Task XII to be conducted during this period will include analog simulation of the SM-1 plant and some preliminary runs on the analog computer to give predicted plant transient response performance. Actual installation of many major items in Task XII must await plant shutdown at end of March.

TASK XIV

4.3 TASK XIV - TEMPERATURE AND FLOW MEASUREMENTS

4.3.1 Time Lag of Plate Temperature Instrumentation

4.3.1.1 Introduction

At the Task XIV conference on December 3, 1959 at ERDL, various methods of instrumenting a fuel plate were discussed. The "Blocked Channel" method was presented as the best choice, and approval for its subsequent development was requested.

It was recognized that by using this technique, the time lag between meat temperature change and thermocouple response would be greater than if the thermocouple were located directly in the meat. However, a detailed calculation of the lag increase and an evaluation of its effect on task objectives was not yet available at the time of the conference.

The determination and evaluation of this lag has now been completed, and is presented in this report.

4.3.1.2 General Theory

The determination of time constants in temperature measurement is most conveniently accomplished, both analytically and experimentally, by subjecting the temperature being measured to a step change and then noting the resulting rapidity in measurement response. Generally, this response is of the form shown in Fig. 4.1:

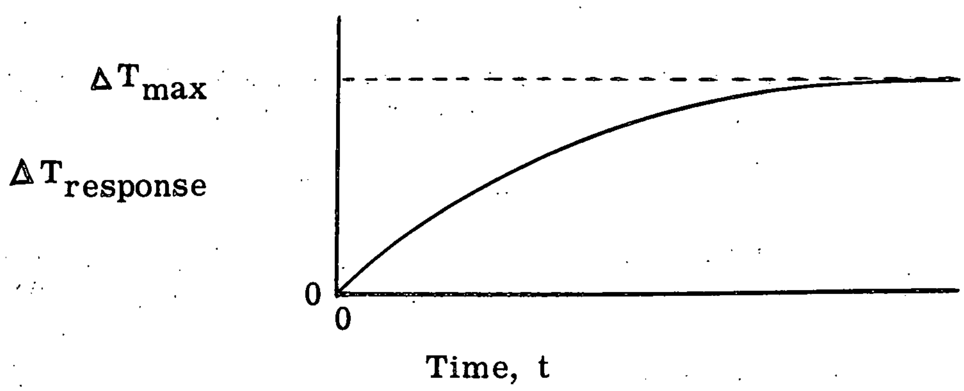

Fig. 4.1 - Linear Plot of Response to Step Input

where:

$$\Delta T_{response} = \Delta T_{max} (1 - e^{-t/\tau}) \qquad (1)$$

and e = natural logarithmic base = 2.71828 --

The value "τ" is called the time constant, and is conventionally evaluated as follows:

$$\text{Let } t = \tau$$

$$\text{Then } \frac{\Delta T_{response}}{\Delta T_{max}} = 1 - \frac{1}{e} = 0.632\text{--}$$

Therefore, the time constant can be evaluated as the elapsed time required to attain a 63.2% response to a given step change in the driving function. This definition is also used at times to define time constants for response curves which are not strictly of the form specified by equation (1).

Exponential functions are tedious to plot, and recourse is often made to log-log or semi-log plotting so that the function can be represented by a straight line whose slope indicates the exponential power. Consider the following algebraic treatment of equation (1):

$$1 - \frac{\Delta T_{response}}{\Delta T_{max}} = e^{-t/\tau} \qquad (1a)$$

This can be simplified by the definition:

$$E = 1 - \frac{\Delta T_{response}}{\Delta T_{max}}$$

E is therefore that fraction of the driving function step change <u>not</u> yet indicated by the response function. Equation (1a) can then be written:

$$E = e^{-t/\tau} \qquad (1b)$$

Taking the natural (Naperian) logarithm of each side of equation (1b):

$$\ln E = \ln e^{-t/\tau} = -\frac{t}{\tau} \ln e$$

$$\ln E = -\frac{t}{\tau} \qquad \text{where } o \leq E \leq 1 \qquad (1c)$$

A semi-log plot of equation (1c) can be made as shown by Fig. 4.2 to obtain the convenience of a straight line representation:

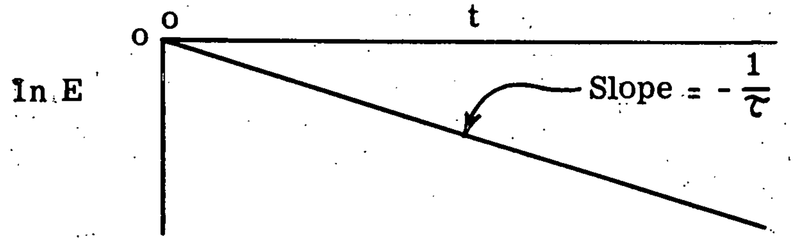

Fig. 4.2 - Semi Log Plot of Response to Step Input
(Natural or Naperian Log Ordinate)

Conventional semi-log graph paper is compatible with a log base of 10, and not e as required in Fig. 4.2. Therefore, taking the common (Briggsian) logarithm of each side of equation (1b):

$$\log_{10} E = \log_{10} e^{-t/\tau} = -\frac{t}{\tau}\log_{10} e = -\frac{\log_{10} e}{\tau}\, t$$

$$\log_{10} E = -\frac{0.4329---}{\tau}\, t \qquad \text{where } 0 \leq E \leq 1 \tag{1d}$$

A semi-log plot of equation (1d) can be made as shown by Fig. 4.3 to obtain the convenience of a straight line representation:

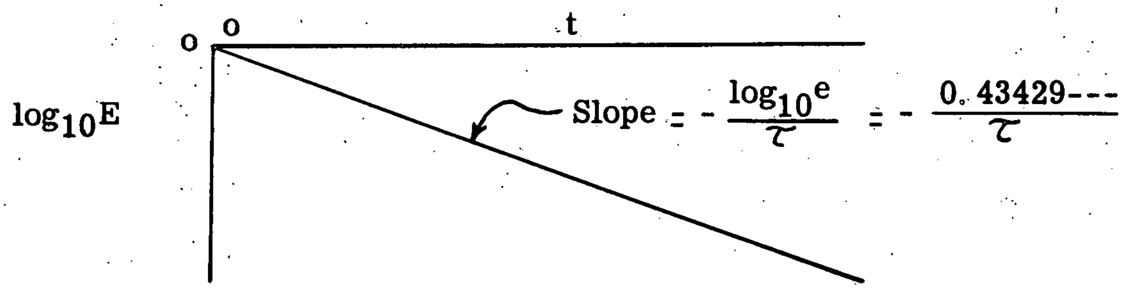

Fig. 4.3 - Semi Log Plot of Response to Step Input
(Common or Briggsian Log Ordinate)

The curve of Fig. 4.1 and its corresponding equation (1) appears frequently in engineering problems for many parameters besides temperature. Consider any process whose output or response function T_2 behaves according to equation (1) following a step increase in its input or driving function T_1. If T_1 is some arbitrary function of time (not necessarily a step), then the corresponding T_2 function is determined by solving the following linear ordinary differential equation:

$$\frac{d}{dt} T_2 = \frac{1}{\tau_{1,2}} (T_1 - T_2) \tag{2}$$

Consider two independent processes, each by itself behaving according to equation (2), coupled in series as shown by Fig. 4.4:

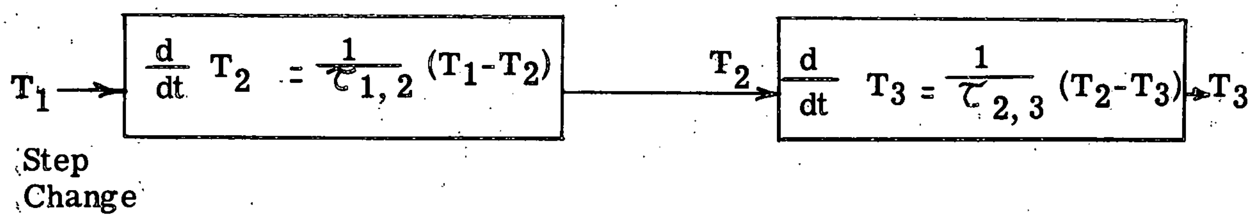

Fig. 4.4 - Two Simple Lag Processes Coupled in Series

It is further specified that T_1 undergoes a step change at time zero. Therefore, the specific solution to equation (2) is equation (1):

$$\Delta T_2 = \Delta T_1 (1 - e^{-t/\tau_{1,2}}) \qquad \text{where } \Delta T_1 \text{ is a constant} \tag{3}$$

The input function T_2 to the second process is not a step function, but it is defined so that the corresponding output function T_3 can be determined by the solution of the following differential equation:

$$\frac{d}{dt} \Delta T_3 = \frac{\Delta T_1}{\tau_{2,3}} (1 - e^{-\frac{t}{\tau_{1,2}}}) - \frac{1}{\tau_{2,3}} \Delta T_3 \tag{4}$$

If the two time constants $\tau_{1,2}$ and $\tau_{2,3}$ are not equal, then the solution to equation (4) is:

$$\Delta T_3 = \Delta T_1 \left[1 - \frac{\tau_{1,2}}{\tau_{1,2} - \tau_{2,3}} e^{-\frac{t}{\tau_{1,2}}} + \frac{\tau_{2,3}}{\tau_{1,2} - \tau_{2,3}} e^{-\frac{t}{\tau_{2,3}}} \right] \tag{5}$$

If the two time constants $_{1,2}$ and $_{2,3}$ are equal, then the solution to equation (4) is:

$$\Delta T_3 = \Delta T_1 \left[1 - (1 + \frac{t}{\tau}) e^{-t/\tau} \right] \qquad \text{where } \tau = \tau_{1,2} = \tau_{2,3} \tag{6}$$

Note that neither equation (5) or (6) is similar to the form of equation (1). If the time constant of the combined lag systems is still defined as the elapsed time t required to attain a 63.2% response in ΔT_3 to a given step change in the driving function ΔT_1, its evaluation is not simply the addition of the time constants of the two systems acting independently. The combined system time constant is therefore the time at which:

$$\Delta T_3 = 0.632\ \Delta T_1 = \Delta T_1 (1 - \frac{1}{e}) \quad ; \quad t = \tau_{1,3}$$

Comparing this expression with the form of equation (5):

$$\frac{1}{e} = \frac{\tau_{1,2}}{\tau_{1,2} - \tau_{2,3}} e^{-\frac{t}{\tau_{1,2}}} - \frac{\tau_{2,3}}{\tau_{1,2} - \tau_{2,3}} e^{-\frac{t}{\tau_{2,3}}} \quad \text{where } \tau_{1,2} \neq \tau_{2,3} \qquad (5a)$$

Comparison with equation (6) shows:

$$\frac{1}{e} = (1 + \frac{t}{\tau}) e^{-\frac{t}{\tau}}$$

$$t = 2.146\ \tau \qquad \text{where } \tau = \tau_{1,2} = \tau_{2,3} \qquad (6a)$$

Equation (6) shows that when the two individual time constants are equal, the combined constant is 7.3% higher than their sum. But as the relative difference between the individual constants is increased, the combined constant as given by equation (5a) converges to simply their sum.

4.3.1.3 Nomenclature

C Specific heat of filler, Btu/(lb)(°F)

K Thermal conductivity of filler, (Btu)(ft)/(hr.)(ft^2)(°F)

r Distance from meat inner surface to filler centerline, mils

t Time after step change in meat inner surface temperature, sec

T_c = Filler centerline temperature, °F

T_i = Initial uniform temperature of meat and filler, °F

T_m = Meat inner surface temperature after undergoing a step change, °F

T_t = Thermocouple temperature reading, °F

X = Ratio defined by equation (7), dimensionless

Y = Ratio defined by equation (8), dimensionless

ρ = Density of filler, lb/in^3

τ_f = Time constant of lag in thermocouple readings introduced by filler only, sec

τ_t = Time constant of lag in thermocouple readings introduced by thermocouple only, sec

$$X = \frac{K\,10^6}{3600(12)\rho C r^2}\,t \qquad \text{dimensionless} \tag{7}$$

$$E = \frac{T_m - T_c}{T_m - T_i} \qquad \text{dimensionless} \tag{8}$$

4.3.1.4 Instrumented Plate Constants

C = .067 @ 68°F (Higher temp. value not available)

K = 10.9 @ 500°F

r_1 = 81.5 (For 193 mil total plate thickness)

r_2 = 40.75 (For 111-1/2 mil total plate thickness)

X_1 = 2.43 t (Corresponding to r_1)

X_2 = 9.61 t (Corresponding to r_2)

ρ = .233

τ_t = .17 (Immersed suddenly in water)

4.3.1.5 Time Lag of Filler Alone

As outlined in the previous quarterly report,[3] a filler of 163 mils was originally contemplated in using the "Blocked Channel" method of instrumenting a fuel element. The corresponding total plate thickness is then 193 mils.

A modification of these dimensions is under study in which the filler is reduced to 81-1/2 mils, the corresponding total plate thickness being 111-1/2 mils. The water channel dimensions of an instrumented element are unchanged provided that an even number of these modified plates are contained by any one element. A schematic diagram of the modified plate is shown in Fig. 4.5.

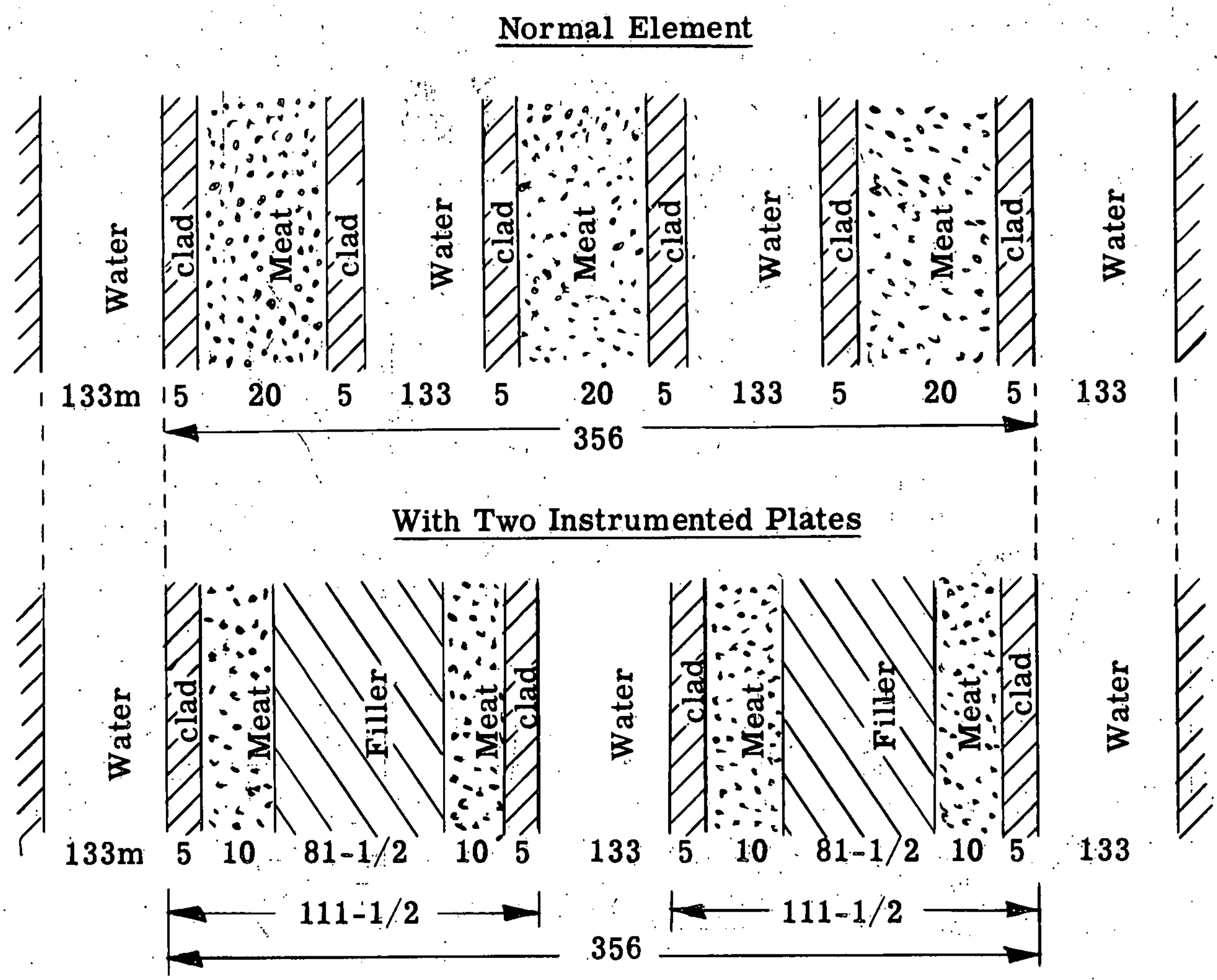

Fig. 4.5 - Schematic Diagram of Blocked Channel Method Using Even Numbers of Modified Instrumented Plates per Element.

The advantages of the thinner instrumented plate are (a) a smaller nuclear perturbation because of less added neutron absorbing material and less variance in the fuel to water ratio, and (b) a smaller filler contribution to the thermocouple time lag in measuring meat temperature changes.

The thermocouple is located in the center of the filler. The lag in filler centerline temperature T_c following a step increase in the meat inner surface temperature T_m can be readily determined without tedious calculation by the use of Hottel Charts.[4] These consist of non-dimensionalized transient temperature distribution curves for a slab following a step change in surface temperature, with or without a surface film coefficient acting. In this application of the charts, no film is considered.

By using the values of X_1 and X_2 as given in Section 4.3.1.4, the slab (or filler) center line temperature responses provided by the Hottel Charts are illustrated in Figs. 4.6 and 4.7 for the 193 mil and the 111-1/2 mil thick plates respectively.

A comparison of Figs. 4.6 and 4.7 with the form of Fig. 4.3 shows that the centerline temperature response can be represented by an initial small time delay followed by a simple time lag. The magnitude of the delay and the time constant of the lag are shown on the respective response curves.

4.3.1.6 Time Lags of Filler and Thermocouple Combined

If a sheathed 40 mil O. D. thermocouple is suddenly submerged in hot water, a time constant of about 0.15 sec is experimentally observed. This includes the effect of the water film heat transfer coefficient acting along the sheath surface. When such a thermocouple is placed within an oversize filler penetration, surface resistance between the ambient (filler material) and the sheath is much larger, and hence, the thermocouple time constant will increase appreciably.

In order to keep the surface resistance to a practical minimum, the following installation features are contemplated:

a. Clearance between thermocouple sheath and filler cavity is kept to a minimum.

b. End of cavity is necked down for snug fit around thermocouple tip.

c. Necked down portion of cavity is lubricated with powdered graphite for good thermal contact.

d. Cavity is kept dry during reactor operation by sealing at cavity opening in side plate.

e. Unfilled portion of cavity is flushed with helium prior to sealing for improved heat transmission.

The resulting surface resistance, and its associated thermocouple time constant, would be difficult to determine analytically. However, based upon various experimental tests by Bettis and by thermocouple manufacturers, it is conservatively estimated that the time constant for the thermocouple in the filler cavity is about 1.35 seconds.

To obtain the time constant of the filler and thermocouple combined, equation (5a) is solved and the time obtained is added to the filler time delay shown by Figs. 4.6 and 4.7. Results of these calculations are presented below in Table 4.1.

TABLE 4.1
TIME CONSTANTS OF INSTRUMENTED PLATES

Instrumented Plate Thickness	193 mils	111.5 mils
Filler Thickness	163 mils	81.5 mils
Time Delay, Filler	.035 sec	.009 sec
Time Const., Filler Alone	.170 sec	.042 sec
Time Const., Thermocouple	1.35 sec	1.35 sec
*Time Const., Filler and Thermocouple combined	1.57 sec	1.40 sec

* Defined as elapsed time for thermocouple reading to reach 63.2% of theoretical step increase in meat inner surface temperature.

One important benefit derived from an instrumented plate with a small time constant is the capability of detecting ϕDNB (heat flux at departure from nucleate boiling). Bettis personnel have used their BISA instrumented plates for this purposed and believe that a time constant as high as 3.0 to 3.5 would still be satisfactory for convenient ϕDNB detection. Reference to Table 4.1 shows that either of the plate thicknesses being considered would provide a time constant about one half of this maximum. The particular choice would therefore be based primarily on other considerations such as nuclear perturbation and ease of fabrication.

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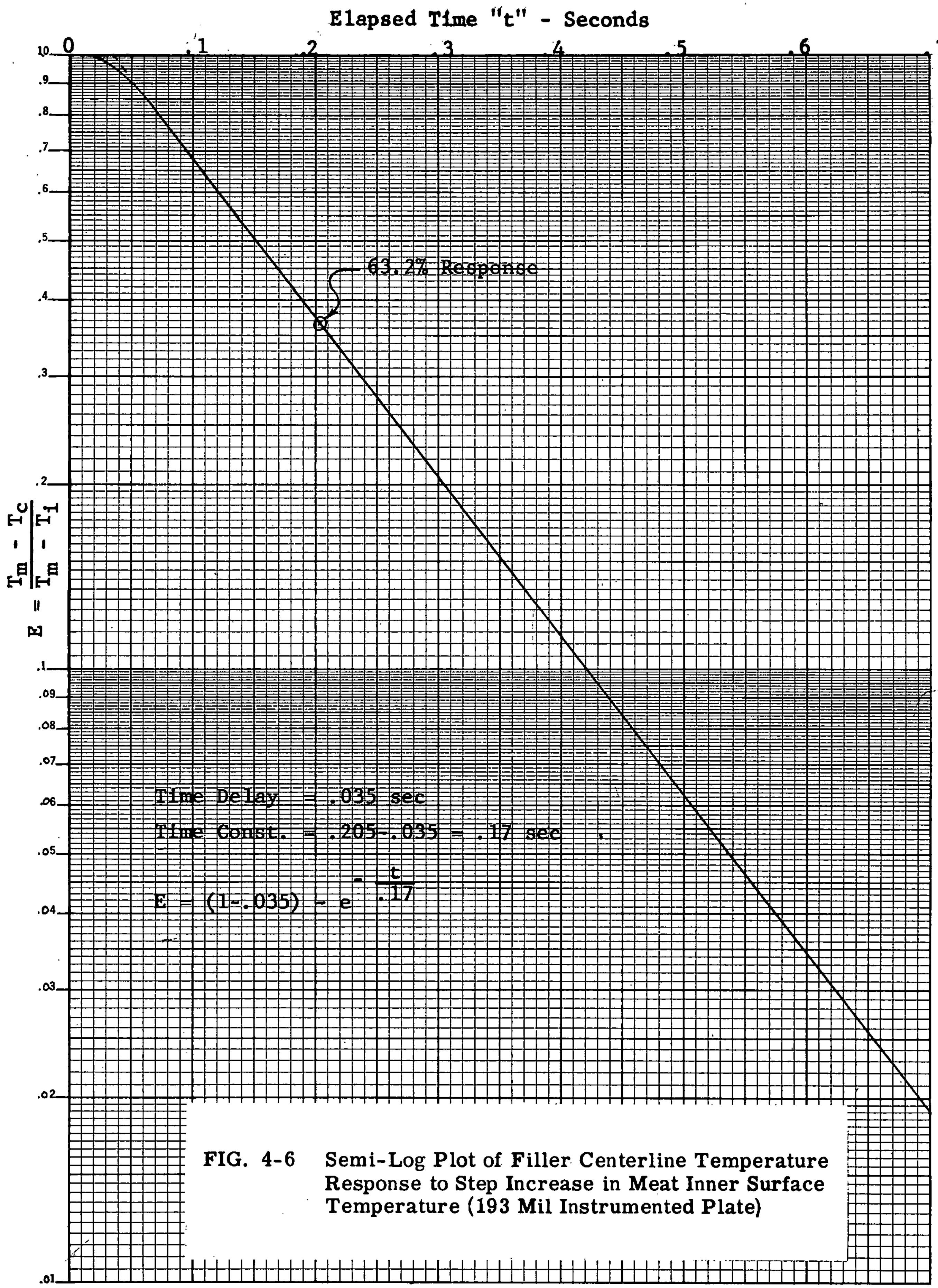


FIG. 4-6 Semi-Log Plot of Filler Centerline Temperature Response to Step Increase in Meat Inner Surface Temperature (193 Mil Instrumented Plate)

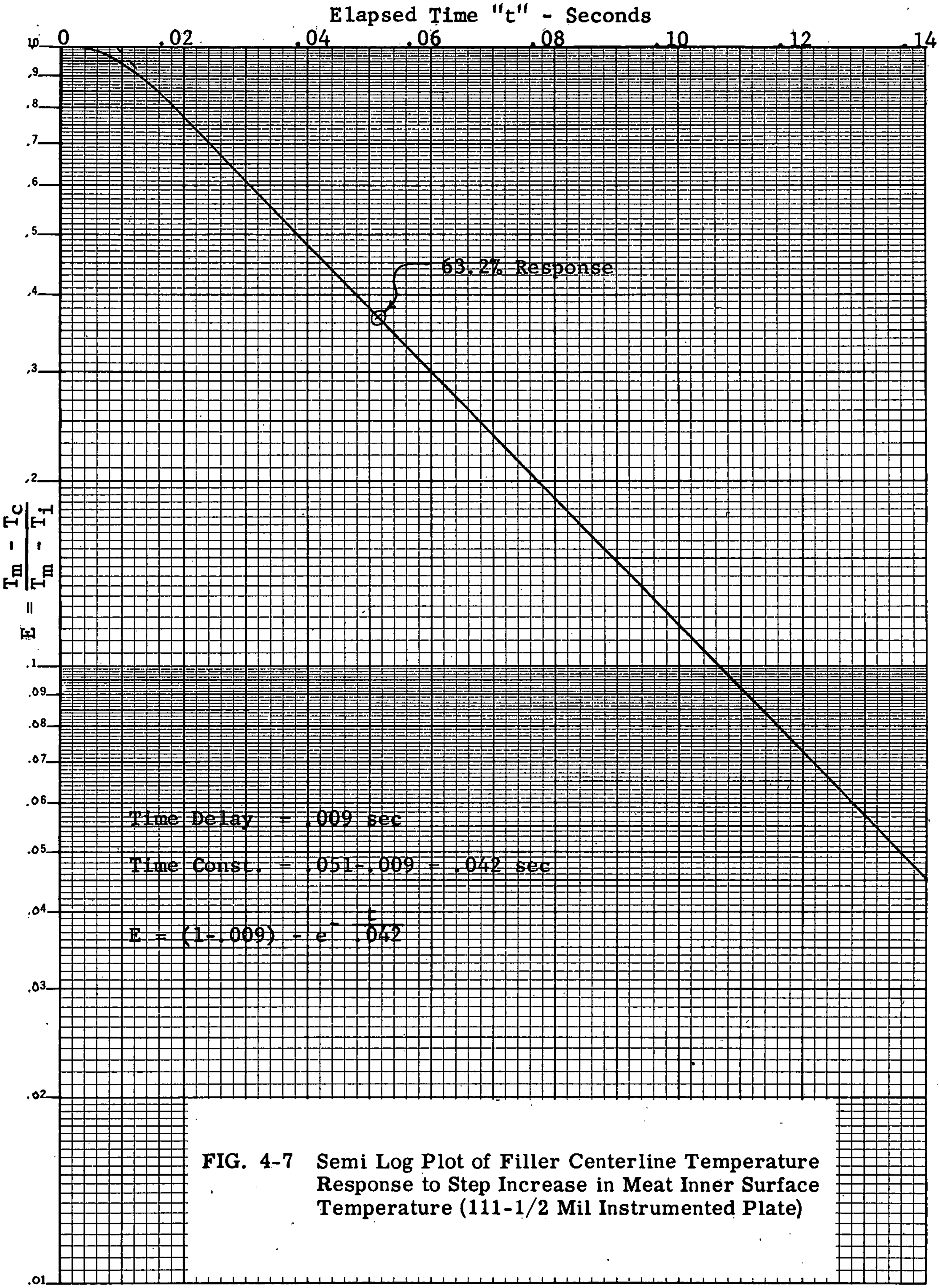


FIG. 4-7 Semi Log Plot of Filler Centerline Temperature Response to Step Increase in Meat Inner Surface Temperature (111-1/2 Mil Instrumented Plate)

4.3.2 Nuclear Perturbation from Instrumented Plate

The introduction of a filler material in the instrumented plate produces a local decrease in the neutron level. To determine the magnitude of this local perturbation, calculations were performed on an IBM 650 computer using a one-dimensional slab representation (5) of the core. To obtain the largest perturbation, all instrumented plates were inserted in the center element for calculation purposes, all elements being assumed as of the stationary type.

The results of these calculations, using various filler materials, is presented below in Table 4.2. Power distribution can be considered as being proportional to the thermal neutron flux distribution.

TABLE 4.2
POWER PERTURBATION AT INSTRUMENTED PLATE

No. of Plates Inserted	Filler Mat'ls. and Thicknesses	Perturbed Power / Unperturbed Power
1	163 SS	0.89
1	163 Nb	0.96
1	163 Yt	0.96
1	163 Zr	0.99
2*	163 SS	0.88
2*	10 Nb 153 Zr	0.99
2*	10 Nb 71.5 Zr	0.99

* Plates placed at opposite sides of center element.

A comparison of the 1st and 5th lines of Table 4.2 shows that a slight amount of plate interaction when two stainless steel filler plates are installed at opposite sides of an element. For the other filler compositions listed, interactions would be insignificant.

A comparison of the 6th and 7th lines shows that the difference in perturbations produced by the 163 mil filler and the 81.5 mil filler (see Section 4.3.1.5) appears only in the third significant figure. Such calculation precision is not indicated in Table 4.2 because of the filler homogeneity assumptions required by the calculation method.

In conclusion, either of the two filler thickness considered introduces a local flux reduction in the order of 1%, with little preference nuclearly between the two. The small magnitude of the perturbation is in excellent agreement with the overall instrumentation objectives of the task, as cited in the previous quarterly report.

4.3.3 Tolerance on Plate Surface Temperature Determination

4.3.3.1 Errors in Determination of Heat Generation Rates

Because of the complex geometry and heterogeneous nature of the core, relative neutron flux levels in the core are best determined by zero power experiments (ZPE). A variety of foil materials can be used in such a study, but bare uranium foils yield more accurate measurements and will therefore be used. Even so, there are various random and systematic errors involved in taking foil measurements.[6]

Once relative flux levels have been determined experimentally at various finite points, it is possible to normalize them with respect to the average core level. Because of the severity of local flux variations, and limitations imposed by the necessity of considering only a limited number of data points, significant errors are introduced in the normalizing process.[7]

Given then normalized flux distribution curves for a cold core, there remains the problem of adjustment for a hot operating core. Items involved are (a) higher temperature and pressure, with its affects on moderator density, various cross-sections, etc., (b) non-uniform burnup and buildup of xenon, and (c) slightly higher control rod withdrawal due to a negative temperature coefficient. The associated errors can be reduced but not eliminated by tedious calculation procedures.

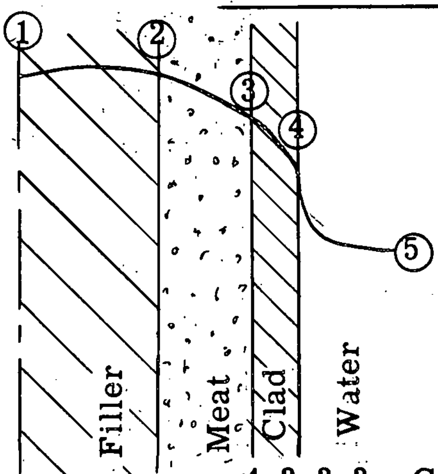
In the above discussion, it was assumed that all the elements used in the ZPE would be used in the actual operating core, and arranged in the same order. Although various fuel plates are made to the same specification, appreciable differences do occur (meat thickness, fuel density, etc.) as a result of manufacturing process limitations. These variations are sufficiently small to permit fuel element interchangeability on core loading, but a difference in ZPE and plant arrangements of the elements will necessarily produce changes in the relative flux distribution. Therefore, it is planned to use the SM-1 Core II in the Task XIV ZPE, indexing the elements so that the same core arrangement is maintained in the plant installation.

4.3.3.2 Accuracy of ΔT Calculations

The temperature in the center of the fuel plate is measured directly by thermocouples located in the filler. Temperatures at other penetrations of the plate are derived by calculation of the temperature change (ΔT) from the thermocouple to the point in question. The calculation of ΔT is dependent upon three types of information, (a) the heat generation ratio in the meat, (b) dimensions of the plate, and (c) physical properties of the plate materials. The first of these has been described above and is subject to the greatese percentage error.

It is estimated, considering the possible errors in the three types of information required, that the various ΔT calculations can be in error by as much as 10%. On this basis, the maximum temperature tolerances at various plate penetrations is shown in Table 4.3.

TABLE 4.3
MAXIMUM PLATE TEMPERATURE TOLERANCES



Temp.	How Obtained	Tolerance
T_1	By direct measurement	1-1/2°F
T_2	T_1 $T_{1,2}$	2°F
T_3	T_2 $T_{2,3}$	3°F
T_4	T_3 $T_{3,4}$	3-1/2°F
T_5	By direct measurement & curve extrapolations	1-1/2°F 2-1/2°F

4.3.3.3 Core Scheduling for Critical Facility

As explained in Section 4.3.3.1, it is highly desirable that the same core to be installed in the reactor is used in the zero power experiments. Fortunately, this situation is attainable without undue difficulty.

The SM-1 Core II is presently stored at the Alco Critical Facility, but must be shipped out soon to make room in the storage vaults for the SM-1a and SM-2a cores due in May, 1960. Both of these latter cores are scheduled for shipment out of the Facility at the end of June. The plates of the SM-1 instrumented elements are scheduled for completion by the end of June also, and can therefore be sent to the Facility prior to assembly into elements. Plates are more easily handled in a ZPE because of more convenient foil placement. The original SM-1 Core II elements can also be shipped back to the Critical Facility, and the ZPE performed in July-August, 1960.

4.3.4. Verification and Calibration Testing

4.3.4.1 Instrument Lead Disconnects

The gasket to be used in an instrument lead disconnect must be made of a material that (a) will not creep under compressive loading at 500°F. and therefore loose its sealing ability, (b) can satisfactorily withstand the exposure of a low-radiation field and (c) can withstand contact with 500°F water.

Two materials that appear particularly applicable are nickel and stainless steel 304, both annealed to a "dead" soft condition. Gaskets have been made from these materials and will be used in the planned disconnect autoclave testing program. These tests will also determine the feasibility of (a) various methods for purging the trapped water in the disconnect after connection and (b) the design of the ceramic insulators for the thermocouple lead terminals.

4.3.4.2 Reactor Vessel Penetrated Gasket

A detailed examination of the primary and secondary stresses in the reactor vessel head gasket is well underway. Results to date indicate that the proposed penetration technique[8] will be very satisfactory. The only experimental program that will be required in support of analytical determination is a series of compressive tests on segments of the penetrated gasket. A costly gasket test loop attempting to duplicate imposed stresses and temperature distributions is not necessary.

The compressive tests are scheduled for completion in the near future. A meeting between AEC, NPED and Alco personnel will be requested shortly thereafter to present the results of the gasket study.

4.3.5 Mechanical Design Aspects

4.3.5.1 Instrument Table

Under the initial mechanical design concept developed under Task V, thermocouple lead disconnects and pressure probes were to be mounted on an instrument "cradle" supported from the vessel flange[9]. However, the dimensional relationships between the flange and the core structure are subject to large tolerances. Since accurate placement is required, the mounting platform design was modified to use the existing pilot tubes as supports while the cradle cylinder was eliminated. The resulting structure is now referred to as the instrument "table", the table legs being mounted to the top of the core support structure.

4.3.5.2 Penetrated Gasket

The penetrated gasket is lowered into the reactor together with the instrument table. Long flexiable rods, acting as cantilever springs, are mounted vertically on the table top to support the gasket above in approximately the correct relative position. The appearance of these springs is similar to an inverted "L".

Lifting lugs on the gasket were revised and relocated to receive a standard "off the shelf" safety hook.

4.3.5.3 Instrumented Element Assembly

Thermocouple penetrations of the filler are from the side edges of the instrumented plates. The instrument leads are then mounted along the outer surfaces of the two side plates. Upon reaching the top end box of the element, the leads are bundled together for passage through the appropriate hinged core support door.

Two methods of supporting the instrument leads (both thermocouple and probe tubing) have been developed by the Welding Laboratory. Both involve the use of the capacitance welder previously acquired through task funds. Each method has its individual merits, so that both techniques will be used in assembly of the instrumented element.

In one methods, short lengths of 2 mil diameter stainless steel wire are spaced along the lead but inserted between the lead and the support plate, the wires being oriented perpendicular to the lead. The result is something similar to a railroad rail with cross-ties. One welder electrode is grounded to the support plate, the other electrode is place consecutively over the points where lead and wires cross. The result is a series of spot welds which (a) are simple to perform and control and (b) minimize weld penetration into the lead casing, (c) requires appreciable force on the lead to break, and (d) introduce negligible thickness to the assembly. This method can easily be extended to fasten a bundle of leads by welding individual leads to each other.

In the other method, small bands are cut from 2 mil stainless steel shim stock. These hands are bent into the shape of the Greek capital letter Omega (Ω), and inserted over a single or bundle of leads. The ends of the bands are then spot-welded to the support plate. The results is an attachment which (a) is simple to perform, (b) requires no weld penetration into a lead casing, (c) requires even greater forces than by the previous fastening methods to break, (d) introduces only 2 mils to the assembly thickness and (e) permits easy replacement of leads by simply cutting the band.

The existing clearance between adjacent stationary element side plates is sufficient for the passage of instrument leads. However, the clearance between the control rod fuel element side plates and the rod tube is too small. Therefore, the side plate separation will be reduced by 100 mils. This is conveniently accomplished by milling 50 mils off the "dead" edges of those conventional rod fuel plates which are scheduled for the instrumented rod element assembly.

4.3.5.4 Instrument Lead Disconnects

As determined under Task V, it is necessary to introduce instrument lead disconnects in the interval between fuel element and penetrated gasket. Preliminary designs for these disconnects were made and presented in APAE No. 37.(10)

Consultation with ceramic fabricators led to the conclusion that the design provisions made for ceramically insulating the pins and sockets from the metal disconnect housing were not satisfactory. A modified design is therefore being prepared in accordance with the suggestions received.

TASK XVI

4.4 TASK XVI - SM-1 DECONTAMINATION

4.4.1 Introduction

Radioactivity of the SM-1 steam generator and other primary system components and piping is increasing due to deposition of adherent radioactive corrosion products on walls of the primary system. Radiation levels within the steam generator water box may become so great as to seriously hinder or render impossible necessary maintenance in the event of tube failure.

The objectives of this task are to:

1. Decrease radiation levels in the steam generator and primary system in order to reduce the radiation hazard when performing reactor maintenance.

2. Determine the feasibility of isolating and decontaminating the SM-1 steam generator.

3. Provide the engineering experience necessary to design facilities for and perform decontamination of future reactor plants.

The schedule for accomplishment of this task is geared to availability of the SM-1 plant after burnup and removal of Core I. At the initiation of this task, the SM-1 master schedule indicated reactor availability starting March 15, 1960. Decontamination and system cleanup is estimated to require six weeks. All engineering and procurement, therefore, was scheduled to permit decontamination starting on March 15, 1960. The method of decontamination to be followed is that outlined in Volume III of APAE No. 43.[2]

The engineering requirements to be accomplished prior to actual decontamination were divided into the following areas:

1. Engineering, procurement, and fabrication of steam generator and primary pump isolation plugs.

2. Engineering, procurement, and fabrication of decontamination system piping, pumps, tanks and other equipment.

3. Corrosion testing and evaluation.

4. Evaluation of waste disposal of decontaminating solution.

5. Evaluation of samples to be collected and inspection techniques following decontaminating to determine the effectiveness of decontamination.

6. Preparation of detailed decontamination procedures to be followed.

4.4.2 Work During Period

Work was initiated on the first three engineering areas of the pre-decontamination task.

4.4.2.1 Task Planning

A detailed program, scope of work, and time schedule were completed. Because of long delivery times for certain components of equipment, difficulty in meeting the March 15 deadline was expected. However, it was anticipated that the SM-1 master schedule might be modified, with decontamination rescheduled for later in the year. Such rescheduling would eliminate the procurement problem.

4.4.2.2 Engineering - Isolation Plugs and System Piping

The reference design for decontamination of the steam generator was based on the engineering concept developed in Volume III of APAE No. 43. The steam generator is required to be isolated from the remainder of the primary system when circulating the decontamination solutions. A schematic drawing of the piping and special plugs required to accomplish such isolation is shown in Fig. 4 of APAE No. 43. A major engineering area was the design of the plug to be inserted in the outlet nozzle of the reactor pressure vessel. This plug prevents entry of the decontamination solutions into the pressure vessel, but allows passage of the solutions through the steam generator by means of external temporary decontaminating piping.

A preliminary design and layout of this reactor vessel isolation plug assembly was made. The design was based on the concept in Fig. 1 of APAE No. 43. Included in the design was necessary equipment and piping to lower the plug into the reactor pressure vessel and lock the plug in place within the vessel nozzle. A layout drawing was completed giving the location of the auxiliary 8-in. piping and circulating pump within the vapor container. Preliminary design and drawing of the plugs for isolation of the Westinghouse primary coolant pumps was completed.

4.4.2.3 Corrosion Testing and Evaluation

Before decontamination of the SM-1 steam generator would be undertaken, assurance had to be obtained that corrosion of the Type 304 SS steam generator during decontamination would not adversely affect its future operation or life expectancy. Previous studies indicated the need for such assurance. The objectives of this program were:

1. To obtain corroboration of the integrity of Type 304 SS when exposed to decontamination agents.

2. To demonstrate that decontamination does not increase the susceptibility of Type 304 SS to corrosion during subsequent exposure to primary coolant under operating conditions.

Specimens simulating the metallurgical conditions in the steam generator were to be subjected to a complete decontamination cycle in static autoclaves, followed by at least one month exposure to primary coolant conditions in static autoclaves. Specimens were to be prepared as:

(a) straight coupons simulating tube sheets and tubes, etc.,

(b) weld coupons simulating tube sheet welds,

(c) U-bend specimens simulating stressed areas in tubes, and

(d) crevice coupons simulating crevices present either by design or as result of partial cracking which may have occurred in three years of SM-1 life.

After each step in the test procedure, specimens were to be examined, weighed, and appropriate samples removed from test for metallurgical examination.

A detailed test procedure was prepared. Specimens of straight coupons consisting of Type 304 SS sheet cut to 3.8 x 3.8 x 0.3 cm size were prepared. These coupons were polished with 1/0 paper to provide a reproducible surface for testing. The weld coupons were similar, except that a bead of 308 SS was laid on each by the TIG welding technique. Crevice coupons consisted of two straight coupons bolted together, the mating surfaces providing the crevice. The U-bend specimens were prepared from 50-mil sheet and were bent around a 1/2" mandrel. Finally, the bends were bolted to provide a constant deflection. Fig. 4-8 shows a sample holder with the several types of coupons mounted and ready for testing.

4.4.3 Future Work

Since it is anticipated the SM-1 operating schedule will be revised and that decontamination will be scheduled for September or October, 1960, the scope of work of Task XVI will be reviewed and consideration given to full system decontamination (with core removed).

Work will include:

1. Engineering methods and design of equipment to allow drainage and cleanup of the entire primary system.

2. Corrosion studies to evaluate effects of decontaminating solutions on materials and components of the entire primary system. Testing of Type 304 SS samples already prepared will be carried out. It is expected that evaluation of results of this phase of corrosion testing will be completed during the next quarter.

3. Evaluation of problems pertinent to the storage and/or disposal of the radioactive spent decontaminating solutions.

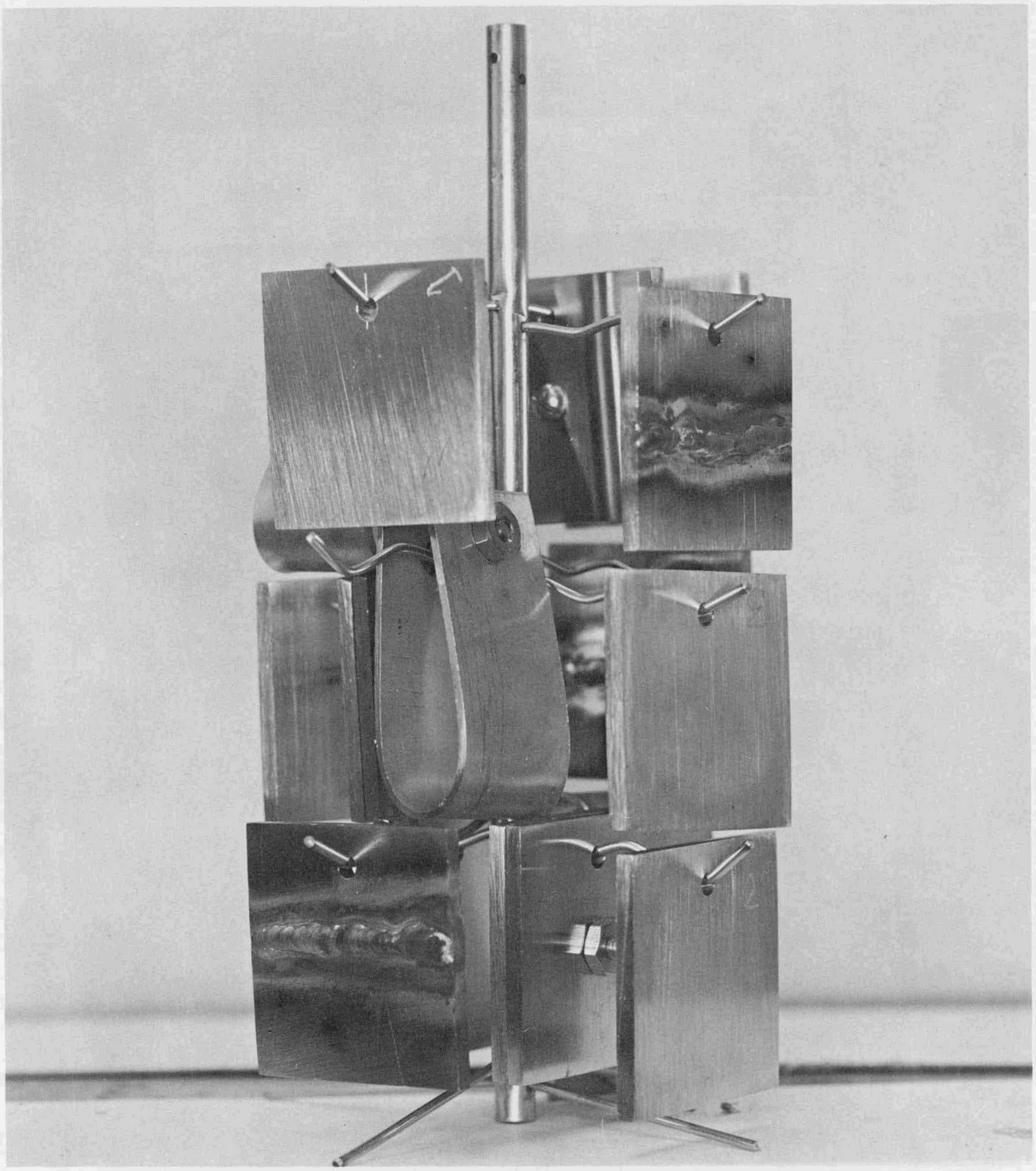


FIG. 4-8 Sample Holder With Coupons For Testing

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TEST SERIES 100

5.0 TESTS

5.1 TEST SERIES 100 - PLANT CHEMISTRY

During the report period, progress was made on accumulation of data in support of approved tests in the chemistry series. In addition, preparatory work was initiated for performance of several requested tests.

5.1.1 Test 101 - Secondary Water Studies at Elevated pH Levels

This test involved evaluation of water chemistry conditions including material corrosion at various alkaline hotwell pH levels. Analytical chemistry data was obtained during the test period December 1 - 8, 1959.

1. Conclusions

   a. The evaporator preheater is inefficient in removing dissolved gases to within specification levels. The oxygen level of the evaporator effluent is approximately 2 ppm and the carbon dioxide content is 15 to 20 ppm. However, very little of the dissolved gases reach the hotwell. This is attributed to the efficiency of the air ejectors.

   b. There is little, if any, measurable difference in the dissolved oxygen content of the hotwell, regardless of the source of makeup water, i.e. distilled water tank or evaporator.

   c. No hydrolytic decomposition of sodium sulfite could be detected in the SM-1 steam generator during the test.

   d. Morpholine undergoes some decomposition to ammonia. The ammonia content varies, in general, proportionally with morpholine concentration.

   e. Excessive corrosion of secondary materials takes place during plant shutdown. Once the plant reaches equilibrium after a start-up, the corrosion levels are low. However, it is felt that the analyses for corrosion products are not sufficiently sensitive to give a true indication of qualitative relationships under varied plant conditions.

   f. Copper corrosion is definitely related to ammonia concentration. It appears that at higher pH levels, especially during evaporator operation, the copper level increases. During shutdown, this same effect can be seen readily, i.e. copper content varies proportionately to ammonia buildup.

g. From a total secondary system standpoint, the hotwell pH should be maintained at about 9.0 to 9.2.

h. Manganese was found in the blowdown throughout the test. It is expected that manganese would be contributed chiefly by the stainless steel.

2. Recommendations

a. Initiate continuous feeding of both sodium sulfite and morpholine. As soon as adequate in-line instrumentation is available, instrument the feeders to function variable system demand.

b. Initiate an investigation to determine the reason(s) why the evaporator preheater is ineffective in reducing the dissolved gases content of the evaporator water to very low levels. If the problem with the preheater is inherent and cannot be corrected, consider other methods for dissolved gas removal, such as using liberated heat from secondary blowdown plus accessory plant power sources to de-aerate evaporator makeup water. Finally, if no methods are available for degassing the evaporator makeup water within a reasonable length of time, feed sufficient morpholine to the evaporator to scavenge the carbon dioxide.

c. Initiate an investigation to protect the secondary system during plant down time. The approach should involve the use of an auxiliary steam source to operate the air ejectors and maintain a steam head on the turbine gland seals. An oxygen scavenging resin may have to be used in conjunction with the auxiliary steam supply.

d. Consider the placing of corrosion coupons, in the secondary system to measure qualitatively the corrosion product formation under varied SM-1 secondary water conditions.

e. Investigate the parameters associated with air ejector operation to determine the effect of concentrating ammonia on system corrosion.

5.1.2 Test 102 - Analysis of Inner Shield Tank Water

The corrosion promoters, oxygen and chloride, were measured in the inner shield tank water on two occasions. The first analysis was made on October 23, 1959, three days after reactor shutdown. The following data was obtained:

	Chloride, ppm	Oxygen, ppm	pH	Resistivity, ohm-cm
Bottom of Shield Tank	0.204	7.65	7.75	120,000
Middle of Shield Tank	0.165	6.48	7.55	120,000
Top of Shield Tank	0.024	7.83	6.99	120,000

Prior to the second measurement, it was suggested that some loss of oxygen from outgassing might be occurring using Test 102 procedures. To check this possibility the suggested alternate method was adopted involving evacuating a McLean Flask and opening it at the desired sample depth. A comparative oxygen analysis from the bottom of the shield tank was made using both Test 102 procedures and the evacuated flask. The results obtained were as follows:

Method	Chloride, ppm	Oxygen ppm	pH	Resistivity, ohm-cm
Test 102 Procedures	6.89	4.80	6.89	120,000
Evacuated McLean Flask		4.30		

Since the results of the two methods are comparable it is planned to continue using Test 102 procedures.

5.1.3 Test 104 - Evaluation of Beckman Oxygen Analyzer and pH Instrumentation

A Beckman oxygen analyzer and pH instrumentation were received and installed in the SM-1 secondary system during the report period. The discharge of the condensate recirculating pumps was chosen as the sample stream because: (1) the hotwell is the pivotal process point of the secondary system; (2) this system operates at a fairly constant pressure; and (3) during plant shutdown, the condensate pumps continue to operate therefore giving a continuous measurement of hotwell pH.

The oxygen analyzer, its controls, and the pH amplifier were installed in a specially constructed panel unit. This unit was located between the condenser and the steam driven boiler feed pump. The recorders, mounted in a panel, were placed on the north wall just outside the electrical equipment room.

The pH instrumentation was placed in service during the report period. The oxygen analyzer, although completely installed, will not be in operation until early in January. This is due to scheduling problems with Beckman personnel. Operation of pH instrumentation has been continuous since startup. Since the pH measurement is dependent upon a small quantity of KCl entering the sample stream, measurements were made for the chloride pickup. Within the limits of the analysis, the chloride pickup was insignificant. For this reason, it was decided to return the sample stream to the system, thereby eliminating the wasting of process water. This arrangement appears satisfactory thus far and operation of the instrument has been acceptable.

5.1.4 Test 105 - Cleanup of Suspended and Settled Crud during Reactor Head Removal Operations.

This test was initiated by the Site Representative to detail the procedures and equipment requirements for removal of suspended and settled crud in the inner shield tank during a reactor head removal operation. During the head removal operation in March 1959, considerable difficulty was experienced due to the crud being disturbed and subsequently suspended in the inner shield tank water. This condition caused a high radiation level at the water surface and a general cloudy condition of the shield tank water. Correction of these conditions involved the installation of a temporary filter system. By planning for the occurrence of similar water conditions during the next head removal operation, it is expected that many of the former problems can be avoided.

5.1.5 Test 106 - Pressurizing the SM-1 Steam Generator During Plant Shutdowns

During SM-1 shutdown, atmospheric oxygen can enter the steam generator. In an effort to exclude oxygen, a test request was initiated involving pressurizing the steam generator with nitrogen during shutdown periods. An evaluation of this test request was made and submitted to the site representative.

5.1.6 Test 107 - Low Volatile Water Treatment (Morpholine and Hydrazine) In the SM-1 Secondary System

During the first several months of operation of the SM-1, hydrazine was used as the oxygen scavenger in the SM-1 secondary system. Secondary chemistry tests during this period indicated that hydrazine was ineffective in reducing the oxygen level. Therefore sodium sulfite was substituted and continued to be used as the water treatment chemical. A test request by Combustion Engineering, Inc. involving a re-evaluation of hydrazine was submitted. This request was evaluated and it was recommended that the test not be performed due to the uncertainty of the inhibiting properties of hydrazine in regard to stress corrosion.

5.1.7 Test 108 - Evaluation of Milton-Roy Chloride Analyzer

The SM-1 steam generator is constructed of AISI type 304 stainless steel. Since many corrosion studies have demonstrated the corrosive effect of chlorides on stainless steels under certain conditions, it is essential that chloride in the steam generator be maintained below the specification limit of 0.5 ppm. Present control of chlorides is dependent on periodic wet chemistry analysis. This gives only momentary coverage, at best. An effort to obtain a chloride analyzer on consignment from Milton-Roy Company for continuous coverage testing was successful. Milton-Roy was unwilling to consign the instrument to the SM-1 for testing. However, it was found that this instrument, along with other chloride analyzers were being evaluated by KAPL for the Navy. This information was forwarded to the Site Representative and a recommendation was made to defer evaluation of chloride analyzers pending completion of the KAPL study.

5.1.8 Test 109 - Evaluation of Industrial Instruments Dissolved Oxygen Analyzer

In addition to the evaluation of the Beckman oxygen analyzer (Test 104), an evaluation was requested for a dissolved oxygen analyzer manufactured by Industrial Instruments Company. Contacts with Industrial Instruments indicated their willingness to loan an oxygen analyzer to the SM-1 for testing. Accordingly, arrangements were made to obtain the instrument, which is scheduled for delivery in mid-February, 1960.

5.1.9 Test 110 - Evaluation of Lapp Microflo-Pulsafeeder Chemical Pump

Lapp Insulator Company was contacted regarding consignment of a Lapp Microflo Pulsafeeder Chemical Pump for test at the SM-1. A successful agreement was concluded and delivery of a Lapp pump to the SM-1 site was scheduled for the second week in January, 1960.

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TEST SERIES 200

5.2 TEST SERIES 200 - RADIOCHEMISTRY

Work during the period was restricted to Phases 1, 5, and 7 of the 200 series, and one independent test, Test 205. Work on Phases 2, 3, 4, and 6 were scheduled for Core II at SM-1.

5.2.1 Phase I - Activity Buildup, Core I

1. Introduction

Work continued on crud and water analyses, the measurement of long-lived dose rates and analysis of metal coupon samples.

One entire coupon holder upstream of the blowdown cooler, containing coupons, was removed on November 27, 1959. The holder was no longer serviceable and the coupons could not be removed at Fort Belvoir. The holder was shipped to Schenectady.

2. Work During Report Period

a. Laboratory work was completed on metal coupons collected October 1, 1959.

b. Laboratory work was completed on crud and water samples collected on September 23 and 25, 1959. The data is being reduced.

c. Coupon samples were removed on November 27, 1959 and shipped to Schenectady. Analyses are being performed.

d. Crud and water samples were drawn on November 27, 30 and December 5, 1959. These samples are being analyzed.

e. Dose rates were taken on the primary system on December 11, 1959. The data are given in Table 5-1.

3. Results

A radiochemical analysis for Co^{60}, Co^{58}, Fe^{59}, Mn^{54} and Cr^{51} was performed on the loose material and descaled material taken from metal coupons removed from the upstream sample location. The data are given in Table 5-2.

A chemical analysis for iron was performed on a solution of the loose and descaled material.

The activity associated with the firmly bound material was easily removed by chemical descaling from the carbon steel, Inconel and 304 SS specimens. The croloy 16-1 could not be completely descaled by this procedure. To obtain reliable nuclide ratios, the activity was removed from the croloy 16-1 by treatment with 9N HC1.

4. Future Work

Crud water and metal coupon samples obtained during November and December will be processed.

Sampling for crud and water will continue during the full power run to core burnup. Monel, nickel, 347 SS, croloy 16-1, 304 SS, Inconel and A212B carbon steel coupons were inserted for activity buildup studies during this period.

TABLE 5-1
LONG-LIVED DOSE RATES

Date: December 11, 1959
Time: 1100

New Point No.	Old Point No.	Location	mr/hr*
C	S. G. #3	S. G. above #2 outlet	170
E	S. G. #5	Upper collar S. G. above steam line	500
F	S. G. #6	Upper collar S. G. above inlet	150
5	5	Pressurizer elbow, bottom	750
I	-	S. G. #2 outlet, bottom	50
J	-	S. G. #2 outlet, top	100
21	21	S. G. #1 outlet, bottom	50
M	-	Reactor inlet pipe, bottom	150
N	-	Reactor inlet pipe, top	600
3	3	Side of S. G.	450
6	6	Reactor outlet	150
9	9	Reactor inlet on slant line	110
10	10	Flow tube, right	700
13	13	S. G. on upper ladder	80
19	19	S. G. 8'5" from #1 outlet	80
20	20	Inlet to S. G., top	100
23	23	Drain plug on bottom of S. G.	25
24	24	Bottom of B. D. cooler	50
25	25	4' from bottom of B. D. Cooler	50

* Measured with shielded Jordon meter.

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TABLE 5-2

MATERIAL REMOVED FROM METAL SPECIMENS

Sample Date: 10/1/59 °F days 8695.34 EFPH 10,608.3

Specimen	Blowdown Hours	Millions of dpm/mg Total Activity	PERCENT DISTRIBUTION Co^{60}	Co^{58}	Fe^{59}	Cr^{51}	Mn^{54}	NUCLIDE RATIOS Co^{58}/Co^{60}	Fe^{59}/Co^{60}	Cr^{51}/Co^{60}	Mn^{54}/Co^{60}
Upstream											
304 SS Pipe	4233	19.	54	23	2.7	13	7.2	0.42	0.05	0.25	0.13
A-212B CS	4233	2.5	53	22	4.9	11	8.6	0.42	0.07	0.21	0.16
A-212B CS	190	1.6	45	27	6.8	14	7.5	0.61	0.15	0.31	0.17
Inconel	190	18.	39	25	3.5	24	8.2	0.64	0.09	0.63	0.21
304 SS	9145	28.	46	24	1.7	19	7.9	0.53	0.04	0.40	0.17
304 SS*	2829	24.	48	22	2.3	20	7.9	0.45	0.05	0.42	0.17
Croloy 16-1	4233	15.	53	32	2.3	3.0	10	0.60	0.06	0.06	0.20

* New coupon put through two loop decontamination runs and put back in reactor.

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5.2.2 Phase 5 -- Fission Product Studies, Core I

1. Introduction

The major effort during the report period was the completion of Test 201, "Startup and Shutdown Test for I-131 and I-133".

One new test was requested. The test (No. 207) will evaluate three methods for determining fission product iodine in the primary system. Test 200 will be delayed pending the outcome of Test 207.

Vendors have been contacted for information on instream fission product monitors. More information will be available during the next report period.

2. Work During Period

a. Test 201, "Startup and Shutdown Run for I-131 and I-133" was completed.

b. A test request submitted by the Site Representative, "Evaluation of Procedures for the Determination of Fission Product Iodine," was evaluated. The evaluation has been submitted for approval.

3. Results

During the initial primary system cleanup operation for Test 201, the blowdown rate was maintained at 1.0 to 1.5 gpm for 72 hours. The iodine-131 level did not decrease steadily as expected, but increased as the blowdown rate was decreased. A similar observation was made during a previous attempt to run Test 201.[3] The findings indicate leakage from the core.

The iodine isotopes peaked during startup, as was observed in tests with defected elements (Figs. 5-1 and 5-2). The equilibrium ratio of the iodine activities was not that expected for a surface contamination source. The iodine-133 to iodine-131 ratio was 7.75 rather than 13 as predicted by a surface contamination source.

After shutdown, with no blowdown, the iodine-131 level increased by 50 percent. The iodine-131 level behaved as predicted if we assumed the iodine-131 increase after shutdown was due to leakage from the core. The data are shown in Figs. 5-3 and 5-4, and are listed in Tables 5-3 and 5-4.

The conclusion was reached that a cladding defect does exist in Core I.

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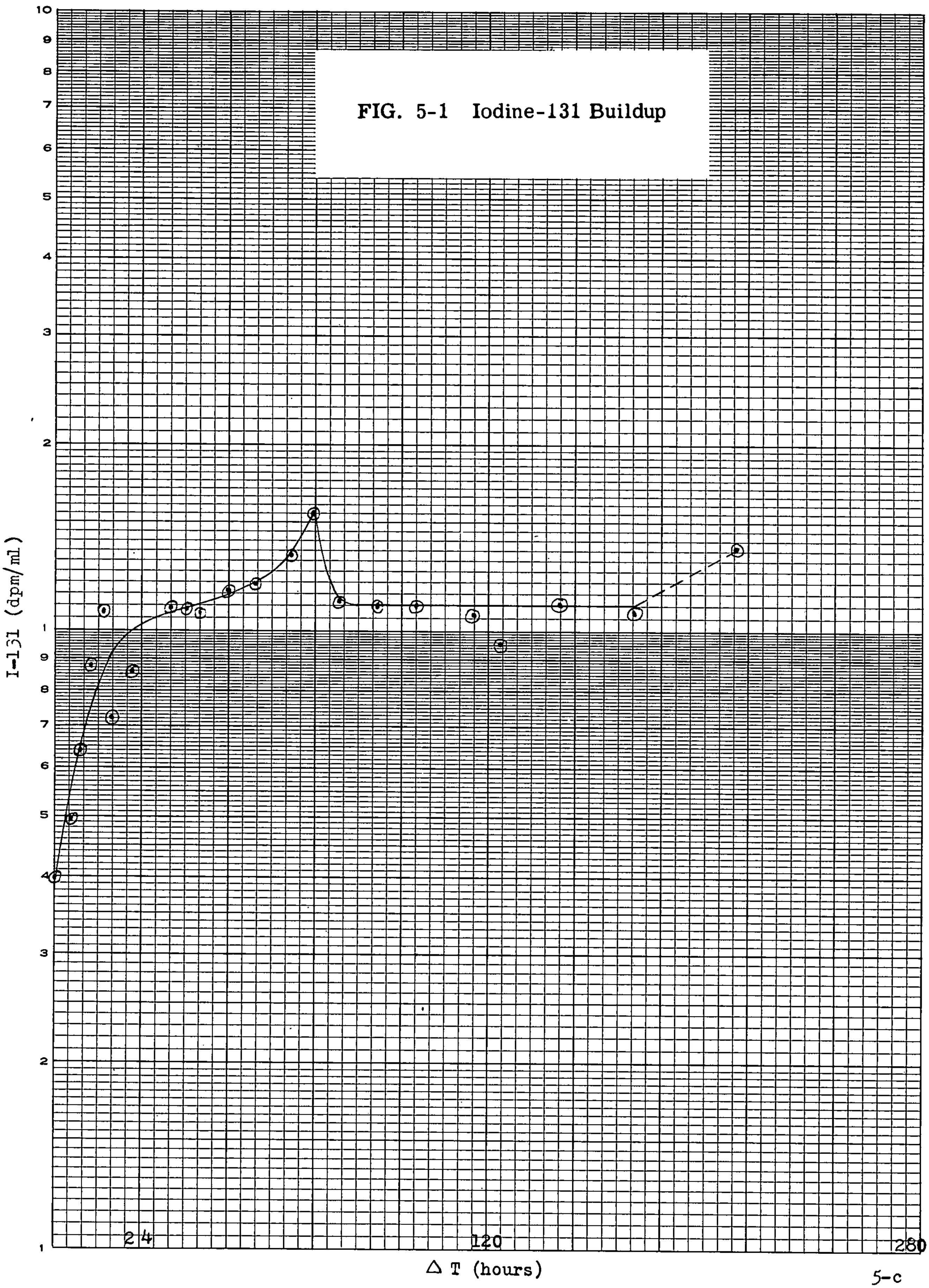


FIG. 5-1 Iodine-131 Buildup
I-131 (dpm/ml)
10
9
8
7
6
5
4
3
2
1
9
8
7
6
5
4
3
2
1
24
120
280
△ T (hours)

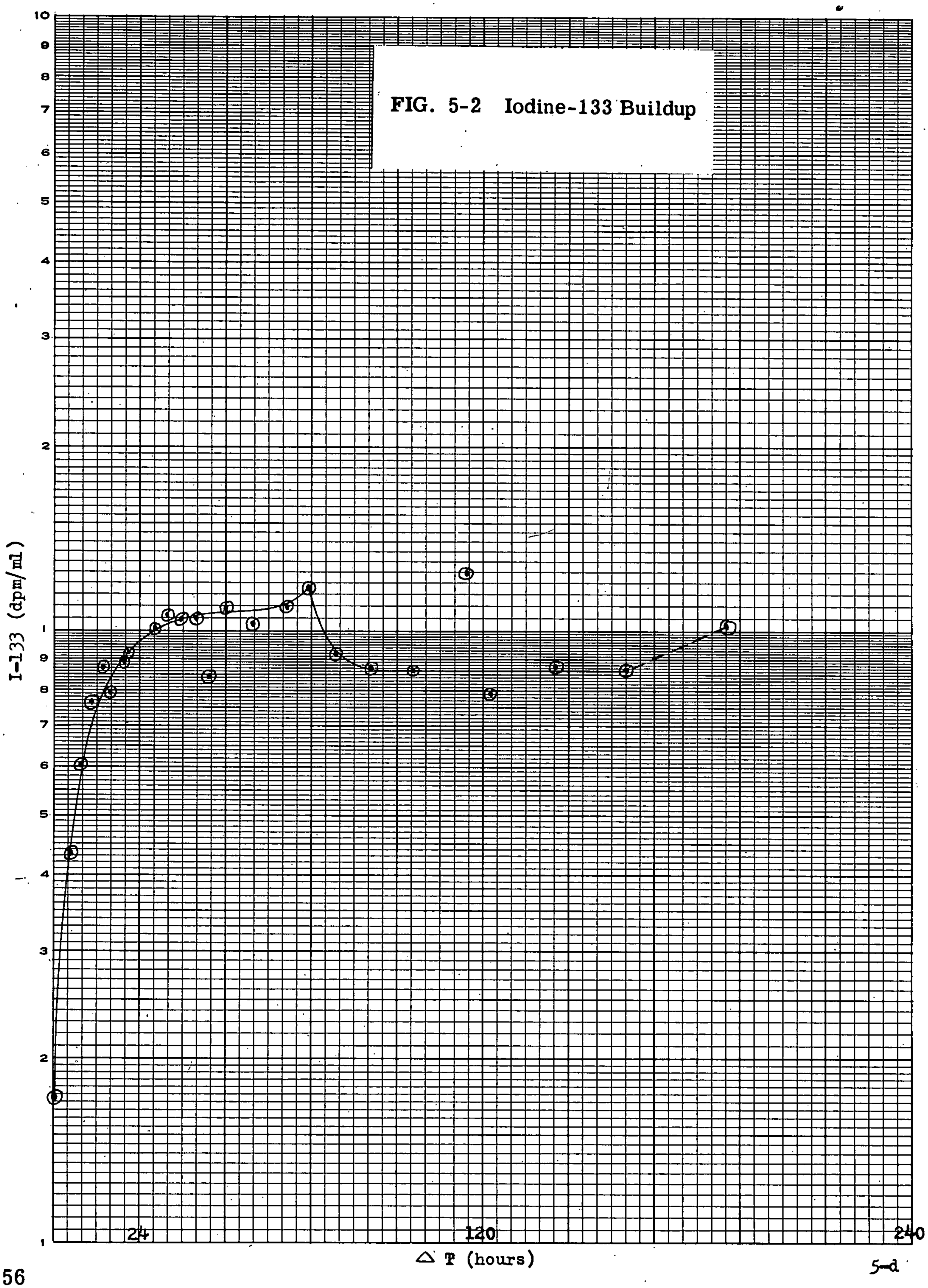


FIG. 5-2 Iodine-133 Buildup
I-133 (dpm/ml)
10
9
8
7
6
5
4
3
2
1
9
8
7
6
5
4
3
2
1
24
120
240
△ T (hours)
5-d

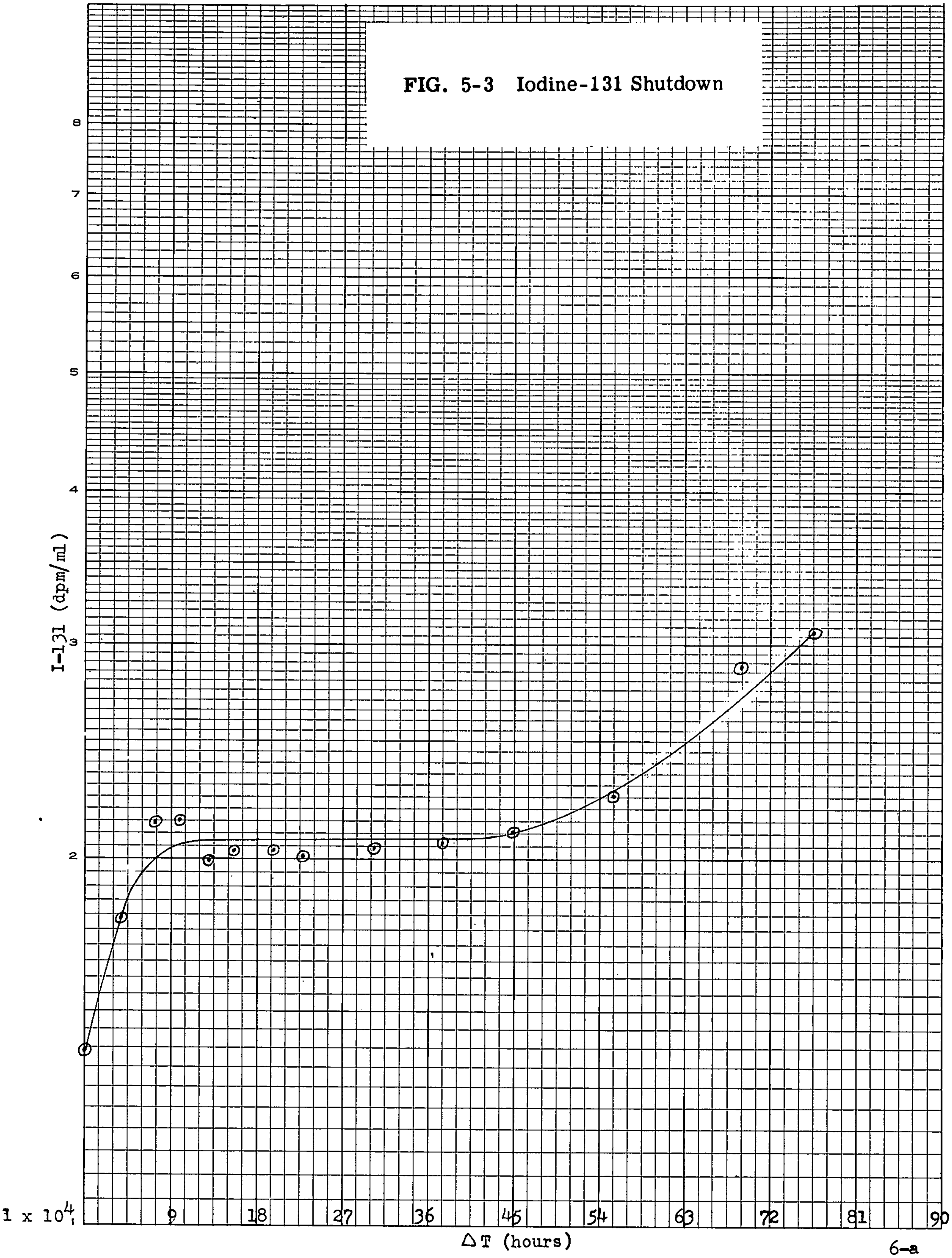
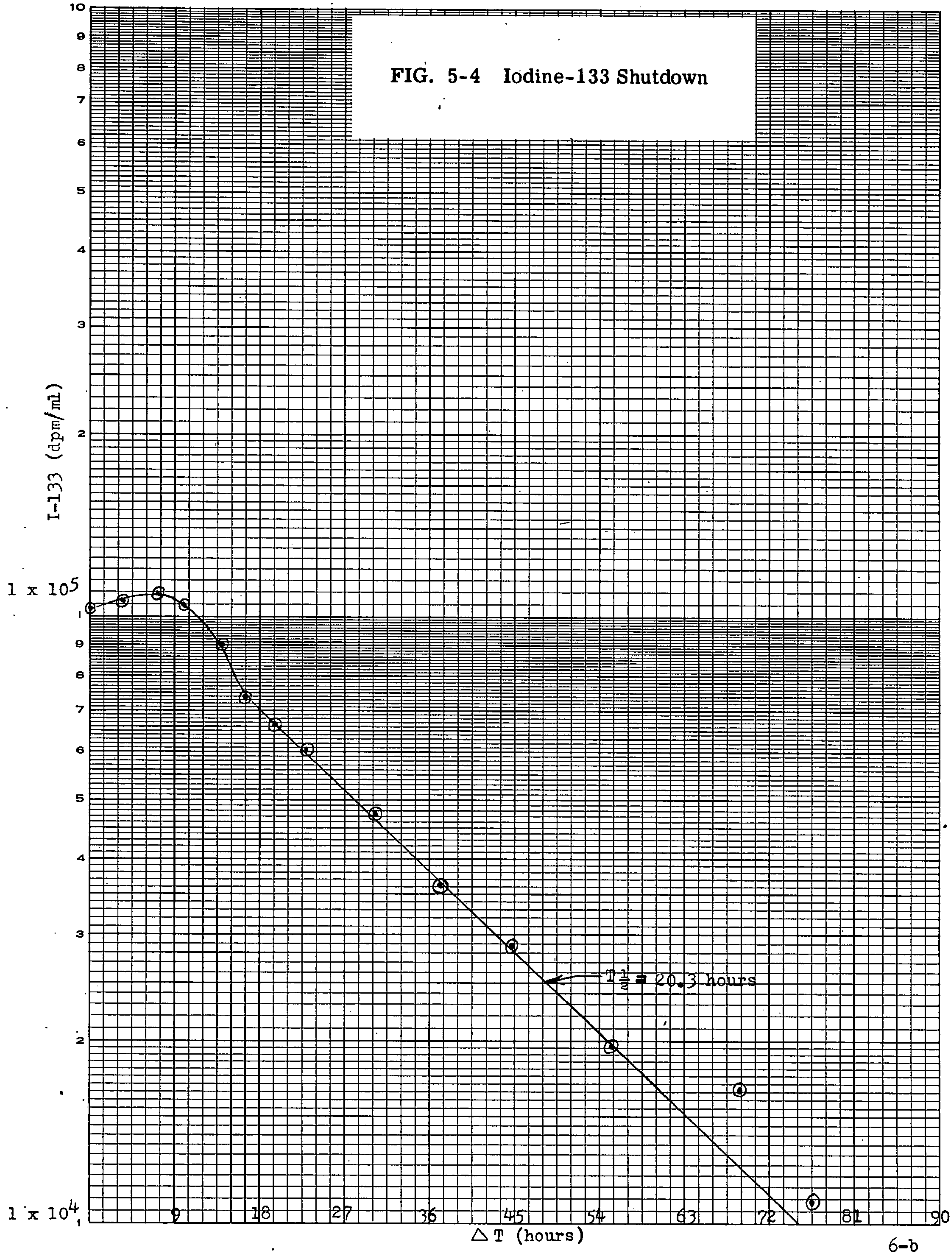


FIG. 5-3 Iodine-131 Shutdown
I-131 (dpm/ml)
8
7
6
5
4
3
2
1 x 10⁴
9
18
27
36
45
54
63
72
81
90
ΔT (hours)

6-a

FIG. 5-4 Iodine-133 Shutdown



4. **Future Work**

Work will commence on the evaluation of instream fission product monitors, the evaluation of three iodine separation methods, and the routine monitoring of the SM-1 primary coolant for α activity.

5.2.3 Phase 7 - SM-1 Decontamination

(See Task XVI, Section 4.4)

TABLE 5-3
IODINE BUILDUP

Sample	Δt*	dpm/ml I-131**	dpm/ml I-133**	Remarks
401		1.54×10^2		Cleanup
402		4.63×10^2		Cleanup
403		4.42×10^2		Cleanup
404		5.83×10^2	8.54×10^2	Started Heatup
405		8.72×10^2	9.93×10^2	
406		7.01×10^2	1.35×10^3	
407		9.60×10^3	2.91×10^4	
408		3.99×10^3	1.74×10^4	Reached full power
409	4.25	4.93×10^3	4.40×10^4	
410	7.25	6.44×10^3	6.10×10^4	
411	10.50	8.79×10^3	7.71×10^4	
412	13.42	1.07×10^4	8.73×10^4	
413	15.92			
414	20.00	8.56×10^3	8.90×10^4	
415	21.83	8.48×10^3	9.29×10^4	
416	28.00	9.67×10^3	1.01×10^5	
417	31.83	1.09×10^4	1.07×10^5	
418	35.91	1.08×10^4	1.05×10^5	
419	39.88	1.07×10^4	1.06×10^5	
420	43.75	9.45×10^3	8.45×10^5	
421	47.78	1.17×10^4	1.10×10^5	
422	55.08	1.19×10^4	1.03×10^5	
423	59.67			
424	65.75	1.31×10^4	1.10×10^5	
425	71.80	1.54×10^4	1.18×10^5	
426	77.92	1.12×10^4	9.20×10^5	
427	88.75	1.11×10^4	8.70×10^5	
428	99.17	1.10×10^4	8.66×10^5	
429	114.8	1.01×10^4	1.03×10^5	
430	122.4	9.53×10^3	7.89×10^4	
431	139.8	1.11×10^4	8.81×10^4	
432	160.4	1.07×10^4	8.64×10^4	
433	188.3	1.38×10^4	1.03×10^5	

* Time elapsed since reaching full power, hours.
** Average of duplicate analyses.

TABLE 5-4
IODINE LEVELS AFTER SHUTDOWN

Sample	Δ t*	I-131 dpm/ml**	I-33 dpm/ml**
434	3.67	1.78×10^4	1.06×10^5
435	7.25	2.14×10^4	1.09×10^5
436	9.85	2.15×10^4	1.05×10^5
437	13.8	1.99×10^4	9.05×10^4
438	16.4	2.03×10^4	7.36×10^4
439	19.7	2.03×10^4	6.67×10^4
440	22.8	2.01×10^4	6.04×10^4
441	30.3	2.04×10^4	4.76×10^4
442	37.1	2.07×10^4	3.62×10^4
443	44.7	2.11×10^4	2.89×10^4
444	55.3	2.27×10^4	1.95×10^4
445	68.8	2.88×10^4	1.66×10^4
446	76.6	3.07×10^4	1.09×10^4

* Time elapsed since shutdown, hours.
** Corrected for dilution due to makeup.

5.2.4 Independent Tests

1. Introduction

Test 205, "Effectiveness of an SM-1 Demineralizer for Reducing the Activity of Radioactive Liquid Wastes," the only active independent test during the period, was completed.

The purpose of the test was to determine the effectiveness of an SM-1 demineralizer for lowering the specific activity of radioactive liquid wastes. The demineralizers contain two cubic feet of resin (Rohm and Haas XE-150).

The data collected were the gross D. F. as a function of specific activity as the waste was recirculated through the demineralizer, and the specific D. F.'s for Co, Cr, Mn, Fe, Cs, Sr, and I_2. Data were also collected showing the dependency of the D. F. on the pH of the waste. These later data were obtained in the laboratory.

2. Work During Report Period

a. Summary

The gross D. F.'s across the SM-1 demineralizers were determined at various flow rates. Specific D. F.'s for Mn, Fe, Co, Cr, I_2, Sr-Ba, and Cs across the demineralizer were determined for a flow rate of 0.4 to 0.6 gpm. The gross D. F. as a function of pH was investigated in the laboratory.

b. Results

Test 205 consisted of two runs. The first run was made using the hot waste tank as a container for the liquid waste. The second run was made using two of the boron injection tanks as containers for the waste. In both cases the waste consisted of primary coolant which had aged for at least 48 hours and, therefore, contained primarily long-lived isotopes.

The test was run twice as the result of a high detergent level in the hot waste tank during the first run. It was suspected that this caused the low decontamination factors (D. F.) observed during the first run. The second run showed that this was not the case. The second run verified the data obtained from the first.

The initial specific activity of the waste water was 1.7×10^{-3} uc/cc and 1.5×10^{-3} uc/cc respectively for runs 1 and 2. The initial D. F.'s (upstream activity divided by downstream activity) in both cases were about 7.5 at a flow rate of 0.5 gpm. The D. F.'s decreased with increased flow rates through the demineralizer to a value of about 3.0 at 1.9 gpm. The D. F.'s at 0.5 gpm decreased during circulation from 7.5 to about 3.0 as the specific activity of the waste water dropped from 1.6×10^{-3} uc/cc to 0.30×10^{-3} uc/cc.

The D. F.'s for fission products were considerably higher than those for the induced activities, the latter ranging from 1.8 to 79 and the former from 69 to >400.

Laboratory investigations indicated that the D. F. was dependent on the pH of the waste. Gross decontamination factors as high as 290 were obtained at a pH of 1.5.

It is necessary to obtain additional information before a final conclusion can be reached concerning the use of demineralizers for waste disposal.

All data are given in Tables 5-5 through 5-9.

3. Future Work

Work will be initiated in the next quarter to determine the purification constant of the primary purification system. A survey will be made of the activity in the primary makeup tank.

TABLE 5-5
D. F. Vs. FLOW RATE

	Specific Activity			D. F.	
Flow Rate	Upstream of Filter	Upstream of Demineralizer	Downstream of Demineralizer	Across Filter	Across Demineralizer
Run 1	Pt. 1	Pt. 2	Pt. 3		
0.4 gpm	1.85×10^{-3} uc/cc	1.7×10^{-3}	2.3×10^{-4}	1.1	7.7
1.0	1.80×10^{-3}	1.6×10^{-3}	1.8×10^{-4}	1.1	9.1
1.3	1.61×10^{-3}	1.5×10^{-3}	2.9×10^{-4}	1.1	5.2
Run 2					
0.5	---	1.5×10^{-3}	2.1×10^{-4}	---	7.2
1.1	---	1.1×10^{-3}	4.1×10^{-4}	---	2.7
1.5	---	0.69×10^{-3}	2.0×10^{-4}	---	3.5
1.9	---	0.54×10^{-3}	1.8×10^{-4}	---	3.0
0.3	---	0.84×10^{-3}	1.7×10^{-4}	---	5.0

TABLE 5-6
D. F. Vs. Specific Activity

Hours of Recirculation	Specific Activity: Upstream of Demineralizer	Specific Activity: Downstream of Demineralizer	D. F.
Run 1 at 1.0 gpm	Pt. 2	Pt. 3	
1	0.99×10^{-3} uc/cc	1.73×10^{-4}	5.7
5	0.83×10^{-3}	1.66×10^{-4}	5.0
9	0.70×10^{-3}	1.45×10^{-4}	4.8
13	0.71×10^{-3}	1.91×10^{-4}	3.7
17	0.46×10^{-3}	1.15×10^{-4}	4.0
21	0.55×10^{-3}	1.48×10^{-4}	3.7
Run 2 at 6.0 gpm			
1	0.58×10^{-3}	1.20×10^{-4}	4.8
5	0.23×10^{-3}	0.70×10^{-4}	3.3
9	0.34×10^{-3}	1.10×10^{-4}	3.1
13	0.24×10^{-3}	1.10×10^{-4}	2.2

TABLE 5-7
SPECIFIC D.F.'s

Isotope	Specific Activity: Upstream of Demineralizer	Specific Activity: Downstream of Demineralizer	D. F.
Run 1 (0.4 gpm)			
Fe^{59}	27 dpm/ml	10 dpm/ml	2.7
Co^{60}	2.0×10^2	1.05×10^2	1.9
Cr^{51}	---	---	---
Mn^{54}	7.09×10^3	4.61×10^2	15.4
Co^{58}	84	18	4.67
Run 2 (0.6 gpm)			
Fe^{59}	57	31	1.8
Co^{60}	5.10×10^2	14	36.5
Cr^{51}	25	4	6.25
Mn^{54}	12.9×10^3	1.62×10^2	79.5
Co^{58}	3.6×10^2	8	45
I_2 (gross)	10551 cpm/100ml	153 cpm/100ml	69
Cs^{137}	1187 cpm/2 1	<3 cpm/2 1	>400
Sr-Ba (gross)	21649 cpm/2 1	1169 cpm/2 1	185

TABLE 5-8
CRUD LEVELS

Run No.	Pt. 1 ppm	Pt. 1 Sp. Act.	Pt. 2 ppm	Pt. 2 Sp. Act.	D. F.'s
1	1.31	8.27×10^6 dpm/mg	0.59	0.41×10^6	44.4 (2.22)*
2	0.83	7.06×10^6	0.51	0.25×10^6	45.4 (1.62)*

* Figure in parentheses is ppm Pt. 1/ppm Pt. 2.

TABLE 5-9
D. F. Vs. pH

pH	D. F.
7.0	10
2.2	72
1.9	80
1.6	120
1.5	290

TEST SERIES 300

5.3 TEST SERIES 300 - PHYSICS MEASUREMENTS

5.3.1 Tests 301 to 316 - Core Measurements

Core physics measurements have been made throughout the lifetime of SM-1 Core I. The integrated results of the measurements will be used to provide a basis for the evaluation of nuclear performance of the core. The results of experimental data will be used to normalize calculational models of other similar core designs. Measurements previously reported include: measurements through 9.1 MWYR - APAE Memo No. 178,(11) measurements from November 1958 through June 30, 1959 - APAE Memo No. 206, (12) and measurements taken after 12.1 MWYR - Test Report issued September 30, 1959.(13) APAE Memo No. 178 includes a description of the SM-1, the experimental techniques developed for power reactor core measurements, and data recording and reduction methods.

On December 17, 1959, after approximately 13.5 MWYR of energy release, the SM-1 was shut down for scheduled core measurements. Data for Tests 301 through 316 were recorded. Procedures for the tests were prepared during the previous quarter. The operating procedure for the test period was the integral of individual test procedures to produce the maximum useful data in the minimum testing time. Reduction of data recorded has been initiated. Preliminary results are reported here.

5.3.2 Test 301 - Transient Xenon

The negative reactivity introduced by transient xenon was determined by maintaining criticality with a calibrated rod and recording rod position as a function of time. During the decay from peak xenon concentration, the change in xenon reactivity provided a variable for the calibration of a control rod with other rods at a stationary position. The reactivity value of peak xenon will be determined from the change in critical rod position evaluated as reactivity from the rod calibration. Peak xenon concentration occurred approximately 7.2 hours after power reduction. The rod position was the same as the equilibrium position indicating equal reactivity worth of xenon approximately 19 hours after power reduction.

5.3.3 Test 302 - Equilibrium Xenon

The negative reactivity introduced by xenon at full power equilibrium concentration will be determined. The reactor was operated approximately 63 hours at full power to establish an equilibrium bank position. The equilibrium xenon reactivity worth may be calculated from the rate of change of xenon decay reactivity as a function of time and the worth of xenon reactivity as a function of time after shutdown relative to the equilibrium value.

5.3.4 Test 303 -- Five-Rod Bank Position; Peak Xenon, 440°F

The time of peak xenon concentration in the core after shutdown at 13.5 MWYR energy release was 7.2 hours determined from the transient xenon measurement. The critical five-rod bank position was 17.58 in. with the following conditions: rods A and B at 19 in., peak xenon concentration and mean core temperature at 440°F.

5.3.5 Test 304 -- Five-Rod Bank Position; Equilibrium Xenon, 440°F

The reactor was operated at full power for approximately 63 hours to approach an equilibrium concentration of xenon in the core. The critical position of the five-rod bank was 16.03 in. with the following conditions: rods A and B at 19 in., full power equilibrium xenon concentration and mean core temperature at 440°F.

5.3.6 Test 305 -- Five-Rod Bank Position; No Xenon, 440°F

The xenon in the core decayed for 60 hours following power operation, when the critical position of five-rod bank varied less than 0.05 in. in a two hour period it was assumed that xenon concentration in the core was at "no xenon concentration". The critical position of the five-rod bank was 13.50 in. with the following conditions: rods A and B at 19 in., no xenon and the mean core temperature at 440°F.

5.3.7 Test 306 -- Five-Rod Bank Position; No Xenon, Room Temperature

The xenon in the core decayed for 60 hours following power operation, the power was reduced to a very low level and the primary system temperature was reduced to room temperature. The critical position of the five-rod bank was 9.44 in. at the following conditions: rods A and B at 19 in., no xenon and the primary system at room temperature.

The critical position of the five-rod bank plotted as a function of core energy release is shown in Fig. 5-5. The figure shows four curves, one each for the conditions of tests 303, 304, 305 and 306.

5.3.8 Test 307 -- Calibration of Rod A; Peak Xenon, 440°F

Rod A was calibrated with peak xenon concentration in the core with the mean core temperature at 440°F. The rod was calibrated as a function of position versus the five-rod bank position; the period method was used. The calibration points obtained will be used to calibrate the five-rod bank.

5.3.9 Test 308 -- Calibration of Rod A; Low Xenon, 440°F

Xenon in the core decayed for 60 hours following power operation, with xenon at "low concentration", rod A was calibrated. The rod was calibrated as a function of position versus the five-rod bank position by the period method. The mean core temperature was maintained at 440°F. The rod worth in cents per inch will be evaluated at the calibration points and the rod worth will be plotted as a function of rod position.

5.3.10 Test 309 - Calibration of Rod A; Low Xenon, Room Temperature

Xenon in the core decayed for 60 hours following power operation and the power was reduced to a very low level so that the temperature of the primary system cooled to room temperature. With the core at these conditions, rod A was calibrated as a function of position versus the five-rod bank position. The period method of calibration was used. The rod was calibrated at intervals of approximately one inch, the rod worth in cents per inch will be evaluated at the calibration points and rod worth will be plotted as a function of rod position.

5.3.11 Test 310 - Calibration of Rod C; Low Xenon, 440°F

Xenon in the core decayed for 60 hours following power operation, the primary system operating temperature was maintained at 440°F. With the core at these conditions, the central control rod was calibrated as a function of position versus the four-rod bank position. The period method of calibration was used. The rod worth in cents per inch will be evaluated at the calibration points and rod worth will be plotted as a function of rod position.

5.3.12 Test 311 - Temperature Coefficient

The temperature coefficient from operating temperature to room temperature was measured by maintaining criticality with a calibrated rod and recording rod position and primary system temperature as a function of time. The change in critical rod position will be evaluated as reactivity from the rod calibration. The change in reactivity with change in temperature will be evaluated by means of the calibrated rod.

The critical position of the five-rod bank with rods A and B withdrawn to 19 in. was recorded as a function of temperature. The plot of the five-rod bank position versus temperature is shown in Fig. 5-6. The reactivity change from 70°F to 440°F will be evaluated from bank motion and the bank calibration.

5.3.13 Test 312 - Source Multiplication

The start up count rate and core reactivity was determined by insertion of a calibrated rod from the critical position. A plot of the neutron count rate versus core reactivity will be used to associate the sub-critical reactivity with a count rate at a given bank position. At 13.5 MWYR the count rate was 4.0-4.5 counts per sec. with all rods inserted, and 6.5 counts per sec. with rod A and B withdrawn to 19 in. and the five-rod bank fully inserted. The count rates were taken 111 hours after power reduction.

5.3.14 Test 313 - Gamma Heating in the Pressure Vessel

No data for this test was recorded during the quarter.

5.3.15 Test 314 - Five-Rod Bank Calibration; Peak to Equilibrium Xenon

Test 315 - Five-Rod Bank Calibration; Rod A Calibration, Low Xenon, 440°F

Test 316 - Five-Rod Bank Calibration; Rod A Calibration, Low Xenon, Room Temperature

Calibration points for the five-rod bank will be determined from the integral of reactivity measurements taken from Tests 314, 315, and 316. The reactivity worth of the five-rod bank is, in general, too large to be measured directly, by the period method; therefore, indirect methods of large reactivity changes as determined from the integral of several small changes are associated with five-rod bank motion and the bank is calibrated indirectly.

5.3.16 Test 317 - Spent Core Rearrangement

When SM-1 Core I is spent with elements in the original configuration, additional energy may be released if, due to non-uniform burnup, high burnup elements are interchanged with low burnup elements. The addition of a few new elements would add to the useful core life. On this basis, a program for SM-1 has been proposed that will rearrange the spent Core I and insert two new elements to gain an additional core energy release. However, the proposed program does not provide for the rearrangements under this test to use 18 new core II elements. Therefore, the test as presently scheduled will measure the change in core reactivity introduced by interchanging calculated high and low burnup elements.

During the corephysics measurements after 13.5 MWYR energy release, tests were run to show: (1) a calibration of rods A and B driven as a bank versus the five-rod bank position by the period method is feasible, (2) the SM-1 core after 13.5 MWYR energy release may be shut down with any three rods fully withdrawn with the following conditions - low xenon concentration, primary system at room temperature. These measurements verify the calculations presented in APAE Memo No. 225[14] and indicate that no criticality hazards are introduced by this test.

5.3.17 Future Work

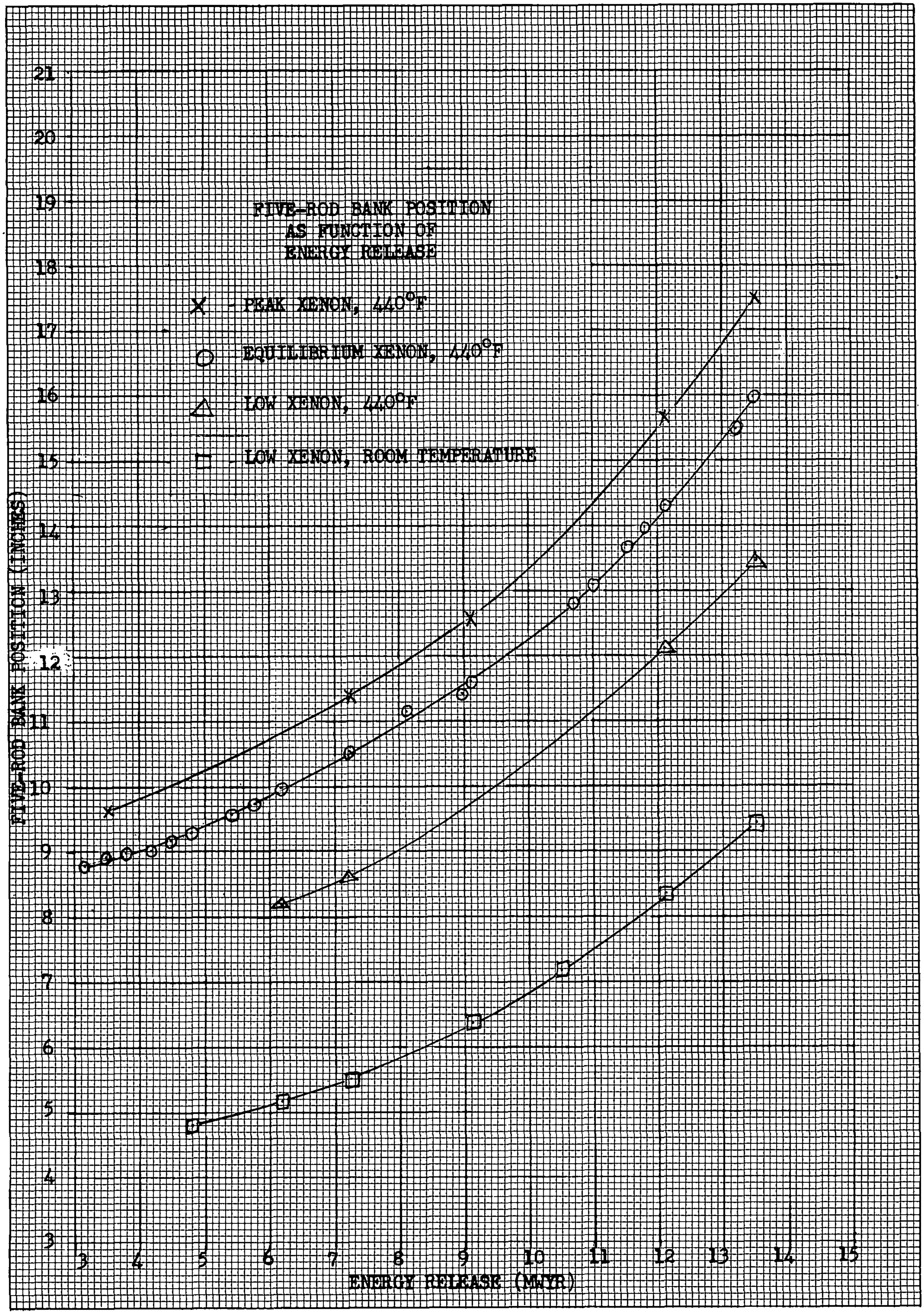
Reduction of data obtained during shutdown after 13.5 MWYR will be completed and a report will be issued giving results.

Additional core measurements will be taken at end of Core I life. These tests will include the spent core rearrangement.

The end of Core I life is expected early in April 1960. A final report on core measurements during the Core I lifetime will be issued when Core I is spent.

Core measurements will be made on Core II.

FIG. 5-5



FIVE-ROD BANK POSITION
AS FUNCTION OF
ENERGY RELEASE
X - PEAK XENON, 440°F
○ - EQUILIBRIUM XENON, 440°F
△ - LOW XENON, 440°F
□ - LOW XENON, ROOM TEMPERATURE
FIVE-ROD BANK POSITION (INCHES)
ENERGY RELEASE (MWYR)
21
20
19
18
17
16
15
14
13
12
11
10
9
8
7
6
5
4
3
3
4
5
6
7
8
9
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11
12
13
14
15

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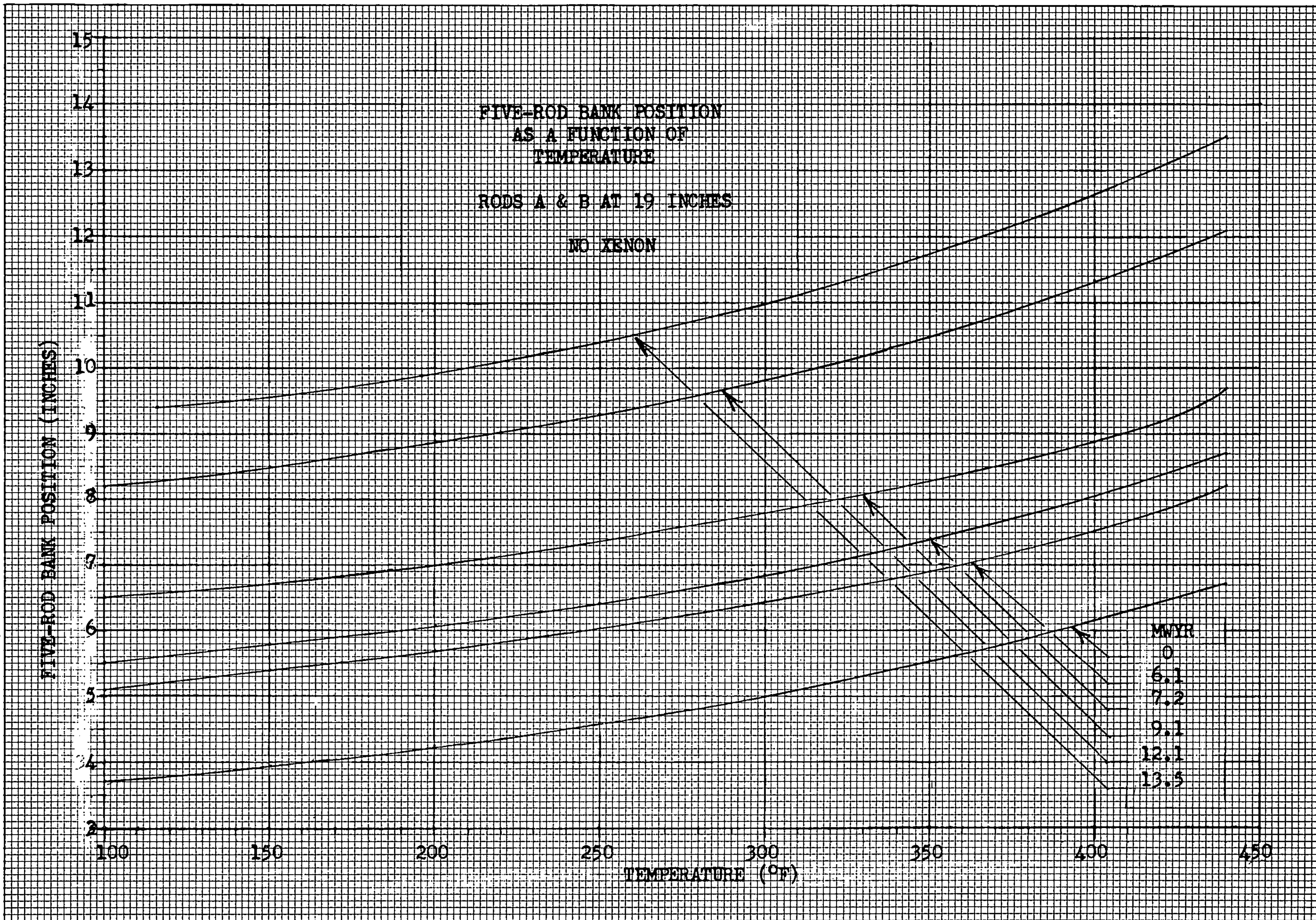

FIVE-ROD BANK POSITION
AS A FUNCTION OF
TEMPERATURE
RODS A & B AT 19 INCHES
NO XENON
MWYR
0
6.1
7.2
9.1
12.1
13.5
FIVE-ROD BANK POSITION (INCHES)
TEMPERATURE (°F)
100
150
200
250
300
350
400
450
3
4
5
6
7
8
9
10
11
12
13
14
15

FIG. 5-6

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5.4 TEST SERIES 400 -- SHIELDING MEASUREMENTS

Data obtained from investigation of SM-1 shielding will be used as a basis for optimization of future shield designs and for possible improvement of the existing shield. Particular attention was given during the quarter to tests to be done at the end of Core I life. Equipment and procedures for end-of-life tests are being designed.

5.4.1 Test 401 -- Primary Shielding Measurements - Neutron Flux and Test 402 - Primary Shielding Measurements - Gamma Flux

Procedures for these tests were prepared by the shielding unit at Schenectady during the previous quarter and reviewed by the Site Representative during this quarter. Comments by the Site Representative were received and revised procedures were subsequently submitted.

5.4.2 Test 403 - Neutron Flux in the Instrument Wells and Test 404 -- Gamma Flux in the Instrument Wells

Procedures for these tests were prepared by the shielding unit at Schenectady during the previous quarter and reviewed by the Site Representative during this quarter. After revisions to the procedures were suggested by the Site Representative, the procedures were revised and resubmitted.

5.4.3 Test 405 - Reactor Component Activation Studies

On the basis of the proposed SM-1 schedule, this test will be performed during the first quarter of FY-61. An investigation was initiated to determine required measurements and testing methods for this study; however, preparation of test procedures has been delayed by other priority work. The test will give particular attention to required measurements to determine feasibility of relocating the plant.

5.4.4 Test 406 - Neutron Measurements in the Rod Drive Pit

Test 407 -- Gamma Measurements in the Rod Drive Pit

Data taken on these tests was reported in APAE Memo No. 237. No data for these tests was taken during this period; additional data will be obtained at the end of Core I life.

5.4.5 Test 408 - Dose Rates from Spent Fuel Elements During the Fuel Transfer Operation

Procedures for this test were prepared by the shielding unit at Schenectady during the previous quarter. The procedures were submitted to the Site Representative during this quarter. Approval by the Site Representative was pending at the end of the quarter.

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TEST SERIES 500

5.5 TEST SERIES 500 - INSTRUMENTS AND CONTROLS

5.5.1 Test 500, Field Test of Minneapolis Honeywell BF_3 Pulse Transformer Channel

This test was completed in the plant and the test equipment removed during the week of November 16, 1959. Difficulties were experienced with the test equipment, mainly with amplifier drift. There was no evidence, however, that the pulse transformer caused any problems. The final report of this test has been delayed due to low assigned priority. We expect that the final report will be issued during the next quarter.

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TEST SERIES 600

5.6 TEST SERIES 600 - HEAT TRANSFER AND FLOW

5.6.1 Test 600 -- Evaluation of Loss of Flow Accident

The purpose of the test is to measure the variation in reactor and system parameters following a loss of the primary coolant, with the low flow scram in the system and with the low flow scram removed from the system.

Procedures were submitted which recommend conducting this test in a series of small steps. It is intended to conduct the test initially at a low power level, then at 1 MW step increments up to and including full power.

Several studies were made previously indicating that a loss of flow without a scram will have no detrimental effects on the reactor core or system.

The general approach to this problem was to develop an electronic model of the SM-1 on the analog computer. Using the results of this lumped simulation and the ratio of maximum to average power in the core, burnout ratios were calculated which indicated the safety of the SM-1 in the event of pump failure without scram.

The lumped kinetic model shows transients in average coolant temperature and average power drop that are very similar to those reported in APAE 39 [15] for the PM-2A. Utilizing the maximum to average power ratio and the average coolant temperature, burnout ratios were calculated. For the case of the reactor operating at 10 MW, the minimum burnout was found to occur at about 0.5 sec. after pump failure. This minimum equals 5.5.

The minimum burnout ratio of 5.5 is well above the minimum of 1.5 usually postulated for safe operation without scram. Hence the SM-1 may be assumed to be safe from burnout in the event of pump failure without scram for periods of at least 5 seconds and possibly longer.

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PLANT MODIFICATIONS

6.0 PLANT MODIFICATIONS

6.1 WINTERIZATION OF SCREENWELL INTAKES

Engineering was completed on winterization of screenwell intakes. In the past, the plant was shutdown due to low water in Gunston Cove and formation of ice on the screens at intakes for the condenser cooling water pumps.

The system designed will utilize the relatively warm water being discharged from the plant's general drainage lines. This drain water will be recirculated to the front of the screen to supplement the water supply and to spray over the screens, thereby preventing formation of ice.

Plans and specifications for this project have been prepared and invitations for bids have been issued.

6.2 PRIMARY MAKE-UP TANK PIPING

Low spots in the hydrogen pressure relief piping atop the primary make-up tank caused trouble in the past. Trouble was experienced during the winter when condensation in the piping froze, blocking the relief pipe passage.

This piping has been rerun to eliminate low spots and is now pitched to allow condensation to run off.

6.3 ROD DRIVE SEAL LEAKAGE

On two separate occasions during the report period, the temperature of the primary blowdown increased to an abnormally high level as a result of mixing the blowdown with the seal leakage water. Investigation of this condition revealed that the make-up lines to the seals were plugging with foreign material, thus permitting reactor water to flow through the seals. An analysis of the foreign matter is underway. Although analytical work is not complete, preliminary information indicates that the material is from the primary make-up pump packing. Based on this information, it has been recommended that filters be installed downstream of the pumps.

6.4 BLOWDOWN VALVE AND FLOWMETER

The existing blowdown valve does not perform satisfactorily because the plug and seat are not the type for required control in 0.1 gpm increments or for flows in range of 0-2.5 gpm. The design flow rate of 0.84 gpm is in the error range of the 0-5 gpm Brooks and Tel-O-Set instrumentation using the 3-5 psig air signal.

Specifications were completed for a motor-operated blowdown valve and blowdown flow instrumentation based upon the requirements of research and development, training and operation.

Specifications were transmitted to the Site Representative and to prospective vendors for quotations.

6.5 PRESSURE SWITCH - VAPOR CONTAINER

The existing United Electric Controls Company type J-11 differential pressure switches PS-15, PS-16 and PS-17 were tested and found the wrong type to operate at 20 psia. These switches are required to operate at 20 psia to actuate all trip valves in the event of major incident.

Type J-20 switches, recommended as replacements, were purchased. A pressure test housing for testing these switches was fabricated. The J-20 switches will be installed at first opportunity during a scheduled plant shutdown.

6.6 ORIFICE PLATES - MAIN STEAM AND FEEDWATER

The accuracy of the steam flow and feedwater flow measurements was questioned because of disagreement of the Tel-O-Set instrument indications. Investigation of the orifice plates was recommended due to probability of wear. The orifice plates were inspected, and although the sharp edge was questionable, there was no measurable wear in the orifice. Installation discrepancies in location of the pressure taps were noted. A recommendation to replace the orifice plates and install orifice flanges was submitted.

6.7 EVAPORATOR CONTROLS AND INSTRUMENTATION

The present instrumentation on the evaporator system was found inadequate to properly evaluate evaporator performance. An engineering investigation determined that service water flow and steam flow instrumentation is required.

Specifications for instrumentation of service water flow and integrator, steam flow, and steam flow control valve were written. Quotations were received from two vendors and a third quotation is being expedited.

6.8 WARNING HORNS

At request of the Site Representative, a recommendation was made for installation of warning horns in the SM-1 and Ponton Basin vicinity to include activation of the ERDL whistle in case of a nuclear incident. An emergency switch in the control room will be operated to sound the alarm to evacuate the areas.

The engineering was completed and recommendations made to comply with the request.

6.9 DESIGN OF NEW SOURCE FOR SM-1, CORE II

The limitations and specifications for a startup source for Core II were considered in selecting and designing the new source. The operations group at Ft. Belvoir specified that the new startup source should provide a neutron production 3 times higher than that of the initial 15 curie Po-Be source and the Be block now in Core I. (The strength of a 15 curie Po-Be source will have decreased by a factor of 240 after three years.) Spacial limitations of the startup channel, cost, and life of the source are other factors considered in designing and selecting the new startup source.

Considerations above eliminated most of the common neutron sources used to start up reactors. A Pu-Be source, through stable and of low gamma activity, was eliminated because of its prohibitive size; a Ra-Be startup source, also stable throughout the life of the reactor, was eliminated because of its size and cost. Photoneutron sources, such as Be blocks or Sb-Be capsules, usually depend on neutrons from the core to reach desired levels of radioactivity. Consequently, they cannot be used alone as startup sources. The relatively small sizes and costs of Po-Be capsules make this source the most practical of those considered for initial startup of Core II. In order to offset the loss in source strength of Po-Be as a function of time, it is desirable to add an auxiliary source which self-perpetuates itself through interactions with neutrons or photons from the core. The most desirable auxiliary source for Core II from feasibility and performance considerations is Sb-Be.

Since the initial Po-Be and auxiliary Sb-Be sources best meet the requirements of source strength, spacial limitations, longevity, and cost, it is recommended that they be used for SM-1 Core II.

6.10 SPENT FUEL PIT RECIRCULATING SYSTEM FILTER

The diatomaceous earth filter originally proposed for use in the spent fuel pit recirculating system presented the possibility of obtaining large amounts of radioactive waste when back-flushing, and the possibility of diatomaceous earth getting in the spent fuel pit and from there into the primary system. It was decided, therefore, to remove this filter from the system. The original filter was replaced with a disposable cartridge type unit using cellulose cartridges. A unit was ordered from Commercial Filters Corporation. The housing is constructed of 316 stainless steel.

6.11 GAMMA SCANNING OF SPENT FUEL ELEMENTS

It was determined that gamma scanning of SM-1 spent fuel, for measuring relative burnup throughout the core, was feasible and could be performed at the site in the spent fuel pit. A trip was made to Battelle Memorial Institute to discuss their method of gamma scanning and to obtain any information enabling Alco to adapt their methods for use in the SM-1 spent fuel pit. A program was initiated to design and build a gamma scanning apparatus. This apparatus is scheduled to be built and installed in the SM-1 spent fuel pit by the end of the first quarter 1960.

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