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SM-1

**RESEARCH AND DEVELOPMENT
QUARTERLY REPORT**

APRIL 1 TO JUNE 30, 1960

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ALCO PRODUCTS, INC.
NUCLEAR POWER ENGINEERING DEPARTMENT
FORT BELVOIR OPERATIONS
P. O. BOX 414, SCHENECTADY, N. Y.

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SM-1
RESEARCH AND DEVELOPMENT
QUARTERLY REPORT

April 1 to June 30, 1960

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ABSTRACT

This report covers Research and Development conducted at the SM-1 under Contract AT(30-3)-326 from April 1 to June 30, 1960, 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

PLANT WATER CHEMISTRY. With the exception of a brief period of slightly elevated chloride level in the secondary blowdown, water chemistry conditions during this report period were quite satisfactory. The increased chloride level was traced, in part, to the in-line pH instrumentation. Since the amount of KCl which enters the water from this source will vary from electrode to electrode, it was decided to install a cleanup demineralizer for use whenever the sample effluent chloride level exceeds 0.1 ppm.

HEALTH PHYSICS. During the report period, the reactor was shut down for end-of-core life testing and rearrangement. This, along with the major SM-1 building project, greatly increased the health physics workload. One technical over-exposure occurred during the period when readings were taken on a spent demineralizer being prepared for disposal to the Chemical Corporation. On several occasions, work was suspended because of laundering problems and overextension of personnel. Although the laboratory has been in a state of disrepair from construction, permitting maintenance of minimum health physics sampling schedules only, the enlarged facilities which will result, coupled with new monitoring equipment in operationally weak areas, will make a significant contribution to personnel safety.

TASK XI, INSTRUMENT AND CONTROL STUDY. A set of specifications covering all electronic and electro-mechanical mechanisms required to control the SM-1 reactor through the rod drive motors and clutches was prepared and issued. The specifications were designed to stress reliability and ease of operation and maintenance. Some innovations have been included and some development will be required of the designer-manufacturer. It is anticipated that acceptance tests of the installed equipment can be approved in eight to eleven months following the contract to proceed.

TASK XII, PLANT RESPONSE AND SYSTEM PERFORMANCE. Installation of instrumentation was virtually completed. Leak checking, calibration and channel checkout will continue into the next quarter. Turbine modifications and repairs were completed and the turbine was reassembled. The steam separator and trap have been installed. The electrical installation, including the load bank and 4160-volt tie-line, is complete. Hydrostatic testing and system cleanup will be accomplished in the next quarter prior to plant startup. Wherever possible, installation of the 100% steam dump line has been completed to points of shutoff to facilitate final installation on receipt of the material on order.

SUMMARY (CONT'D)

TASK XIV, TEMPERATURE AND FLOW MEASUREMENTS.

The engineering efforts of Task XIV described in this report are in four categories as follows:

- a) Evaluation of the plate temperature effects resulting from gamma heating in the instrumented plate zirconium filler.
- b) Evaluation of the experimental error associated with various plate temperature determinations.
- c) Quantitative examination of some of the benefits of the task instrumentation installations.

TASK XIV, SM-1 DECONTAMINATION. Preliminary Stellite corrosion data made impossible any recommendations of full system decontamination at this time. Equipment procurement for partial system decontamination has therefore begun. By the end of the next period all equipment will have been obtained and installed where possible.

TASK XVIII, ANALYSIS OF SM-1 CORE I ACTIVITY DATA.

Phase I - Induced Activity Buildup. Work on the interpretation of long-lived radiochemical data obtained at the SM-1 during Core I lifetime was continued during the second quarter of 1960. Dose rate buildup to the end of Core I life (April 28, 1960) on the exterior of the steam generator was plotted as a function of time since reactor startup. Equations to account for the activation of out-of-core corrosion products were derived and programmed for the analog computer. Work with personnel at KAPL involved in activity buildup in Naval reactors was started. The total activity in the coolant was found to increase as expected. At low crud levels, most of the activity is in the non-filterable fraction, while at high crud levels it is in the filterable fraction.

Phase II - Fission Product Activity Associated with Core I. Analysis of all fission product data collected during Core I life has started. An outline of the final presentation has been prepared. No results are offered or conclusions reached.

Phase III - Investigation of Fissionable Material Surface Contamination of SM-1 Core II Fuel Elements. Thirty-eight stationary and seven control sub-assemblies from SM-1 Core II were checked for alpha contamination by a gas flow proportional counting technique. The observed contamination, if

SUMMARY (CONT'D)

representative of all fuel plates, would sustain a fission product concentration of approximately 6×10^{-4} $\mu\text{c/cc}$ in the Core II primary coolant one hour after shutdown. This is negligible in comparison to corrosion product activity.

TASK XIX, ION EXCHANGE RADIOACTIVE WASTE DISPOSAL. The work on the final design of a waste disposal system for SM-1A was stopped and an investigation of an interim system containing a by-pass sampling system was undertaken. A report describing this interim system was prepared. Attention was then shifted to the design of a skid-mounted system to process waste at the PM-2A. A skid-mounted system is being designed which will provide a decontamination factor of at least 10 for all the high activity waste accumulated during a year. The unit will consist of a skid approximately 8 ft by 12 ft with the following major components: one 1000-gallon tank, two 270-gallon tanks, two canned rotor pumps, two shielded demineralizers, one resin fine filter, one rotometer. Flow diagrams have been prepared and the required instrumentation has been described.

TEST SERIES 100, PLANT CHEMISTRY. A final report detailing the performance of the Beckman in-line process instrumentation was issued ⁽¹⁾. Procedures established for Test 105 were used to clean the inner shield tank water. With the circulating system in continuous operation it appears that the water can be maintained in a high purity, relatively low activity condition. Installation of the Industrial Instruments dissolved oxygen analyzer was completed for testing during the next extended power run. A letter report on the performance of the Lapp Chemical Feed Pump was prepared.

TEST SERIES 200, RADIOCHEMISTRY. Work continued on Tests 202, 203, and 204 in the activity buildup phase of Test Series 200. Complete analysis of the data obtained under these tests will be performed under Task XVIII. Final reports were issued on independent Test 209⁽²⁾ and 210⁽³⁾.

TEST SERIES 300, PHYSICS MEASUREMENTS. Core physics measurements were taken at end of Core I life to complete the series of measurements made throughout the lifetime of the core. The integrated results will be used to provide a base for the evaluation of nuclear performance of the core. All other scheduled end-of-life tests were completed in the plant and data recorded. Preliminary results are reported pending completion of final reports.

TEST SERIES 400, SHIELDING MEASUREMENTS. Tests 408, "Fuel Transfer Dose Rate", and 409, "Spent Element Surface Dose", were run at end of core life. Data has been recorded and final reports are in preparation.

SUMMARY (CONT'D)

TEST SERIES 500, INSTRUMENTS AND CONTROLS. The final report of Test 500 was written and will be issued during July, 1960. The installation of equipment for Test 501, Bendix Nuclear Instrumentation and Control Channels, and Test 502, Westinghouse N¹⁶ Monitoring Equipment, continued throughout the period of the report and is virtually complete. These tests will be scheduled to run on the spiked core.

TEST SERIES 600, HEAT TRANSFER AND FLOW. The final report of Test 600, "Evaluation of Loss of Flow Accident", was issued (4).

TEST SERIES 1000, METALLURGICAL STUDIES. The rearranged core was spiked with one SM-2 and one PM-1M fuel element. The reactor analysis support work devoted to studying the nuclear and thermal consequences of rearranging and spiking SM-1 Core I is given in Section 6.0 of this report. The spiked elements will be left in the core through the life of Core II to obtain irradiation data.

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1.0 INTRODUCTION

The plant was operating at full power for core burnup at the beginning of this report period. It was shut down on April 19 for Test 320, Xenon Override Measurements. Following completion of this test, it was returned to full power and operated to the end of Core I life.

The end of core life was reached at 0801 hours on April 28, 1960. Operation at slightly reduced temperature was continued until 0838 hours on April 29, at which time end-of-life core physics testing in Test Series 300 was initiated.

The plant was shut down through the remaining period of this report for installation of Task XII equipment and the completion of core testing and core rearrangement. Instrumentation for Tests 501 and 502 was installed.

2.0 PLANT WATER CHEMISTRY

During this report period, prior to end of Core I life on 28 April 1960, the SM-1 was operated at full power. Except for a brief period when a slightly elevated chloride level was found in the secondary blowdown, the water chemistry conditions were quite satisfactory. An investigation of the cause of the increased chloride level was traced to the in-line pH instrumentation and faulty operation of the evaporator (corrected).

During the initial checkout of the pH equipment, the differential pressure between the process stream and the reference electrode was varied between 3 and 15 psi. The purpose of this work was to determine the quantity of chlorides entering the sample stream from the reference electrode at the various differential pressures. The results of this test were completely negative; there was no increase in the volume of mercuric nitrate in the chloride analysis over that required for the blank, within the differential pressure range. In fact, during the first two months of operation, only about 5 ml of saturated KCl solution was lost from the reference electrode. By establishing the extremely low rate at which the first reference electrode passed KCl into the process stream, it was possible to channel the discharge from the flow chamber directly back to the secondary system.

When the slightly elevated chloride level was found, the pH instrumentation had been in service approximately four months. A check of the influent and effluent sample stream at the pH flow chamber revealed a significant increase of chlorides in the effluent stream. This condition was not unexpected since over a period of time the electrode tip wears, permitting more KCl to enter the water in the flow chamber. However, upon replacing the reference electrode and again measuring the chloride level in the influent and effluent streams, a high chloride in the effluent stream was determined. It was assumed, and subsequently confirmed by Beckman, that the penetration in the electrode tip varies from electrode to electrode. As a result, it was decided to install a cleanup demineralizer in the sample effluent stream to be used whenever the chloride level exceeds 0.1 ppm.

At the end of Core I life when the plant was shut down, the water chemistry conditions were within specifications. It is expected that these chemistry conditions will not be maintained due to the extensive modifications to the secondary system piping. It is planned that prior to plant startup, both the primary and secondary water systems will be cleaned.

3.0 HEALTH PHYSICS

3.1 GENERAL

During this period, a plant shutdown to permit core rearrangement, major SM-1 building modifications, and new construction placed unusually heavy demands on the SM-1 health physics group. In addition to the coverage necessitated by replacement of the spiked core fuel, two uncommon situations arose which required control.

Several fuel elements were transferred to the spent fuel pit, placed in an underwater gamma scintillation counter, and scanned to determine fuel burnup patterns. During this operation, the scintillation counter needed repairs, which were done at the site by raising the scanner above the spent fuel pit's water level. Dose rates of approximately 100 mr/hr were observed on the fuel element carriage assembly, enabling repairs to be made without encountering health physics problems.

The primary make-up tank, which had been inadvertently collapsed, was hydro-tested at 75 psi to ensure its conformity to design specifications. A containment basin leading to a rubber tank reservoir was constructed to contain primary make-up water in event of tank failure under test. Approximately 2 liters of make-up water was collected during the test from around the inspection port seal. This leakage was monitored and disposed of in accordance with accepted procedures.

3.2 PERSONNEL EXPOSURE

Personnel radiation exposures for this period were much higher than normal. There was one technical over-exposure, and plant operating personnel have received exposures close to the weekly permissible dose. The number of project --- core rearrangement and replacement, gamma scanning of spent fuel elements, pressure testing of primary make-up tank, waste disposal, and SM-1 building construction --- have necessitated such exposures.

On June 13, 1960, a spent demineralizer was being prepared for shipment by M/Sgt. A. Levine and Sp/5 Wessley. The Chemical Corps has required direct monitoring of demineralizers to aid in design of concrete disposal casks. As M/Sgt. Levine attempted to obtain measurements on the demineralizer, the portable Jordan pegged on the 50 R/hr scale. He proceeded to lower the demineralizer into the lead cask. On completing this, he noticed that his as well as Sgt. Wessley's dosimeter was off-scale, indicating that

each had received an exposure in excess of 200 mr. The film badges were sent to Lexington Signal Depot for processing and a teletype report indicated that M/Sgt. Levine had received a dosage of 530 mrem and Sgt. Wessley had received 275 mrem.

Army Chemical Corps., Edgewood, Maryland, has been contacted regarding the advisability of continuing such monitoring. After inquiry into the hazards involved in this procedure, Edgewood has rescinded their request.

3.3 INSTRUMENTATION

NPFO approved Alco's recommendation for the purchase of additional health physics instrumentation which includes:

<u>Name</u>	<u>Description</u>	<u>Manufacturer</u>
Gas Proportional Alpha Meter	PAC-3G	Eberline Instrument Corp.
Scintillation Alpha Meter	PAC-1SA	Eberline Instrument Corp.
G-M Survery Meter (2 units)	E-500A	Eberline Instrument Corp.
Fast Neutron Survey Meter	FN-1A	Eberline Instrument Corp.
Fast Neutron Survey Meter	FNS-3	W.B. Johnson & Associates
Air Sampler	SE-16	Gelman Instrument Co.
Air Sampler	NA-7	Gelman Instrument Co.

Instruments purchased from Eberline incorporate recent design features, as:

1. Plug-in, transistorized, electronic circuit cards which can be readily removed for replacement and repair with a minimum time delay.
2. Strengthened meter construction.
3. Design permitting operation under a wide range of environmental conditions.
4. Units available built to military specifications from the manufacturer.

G-6 C14

It is felt that the procurement of these instruments will result in greater reliability during operation and reduce both the total amount of time required for repair and the frequency of maintenance. The instruments recommended for purchase are commercially supplied items and operation at temperature extremes of -40°F to $+130^{\circ}\text{F}$ is not possible; however, replacement of the D cell battery pack by the Silvercel pack will permit operation at these extremes.

3.4 RADIOACTIVE WASTE

The waste retention building had to be put in use before the building contractor had completed electrical wiring and plumbing installations, due to quantities of waste accumulated during shutdown and lack of personnel to prepare waste for shipping and disposal as it accumulated.

The quantity of liquid waste accumulated for the period was higher than usual, and on several occasions operations were suspended due to laundry equipment failures and lack of adequate liquid storage tanks. The quantities of water resulting from laundering contaminated clothing and decontamination procedures were greater than could be released to the river in effluent concentrations of $1 \times 10^{-4} \mu\text{c/cc}$ or less. The problem was intensified by not having 4000 gpm of cooling water for dilution and release of the waste. Plans are being made to obtain additional tanks, possibly from Edgewood Chemical Center, for retention of this waste. Despite this problem, health physics controls have insured that the quantities of waste released to the Potomac River have not adversely affected river biota or the use of the river water for human consumption.

3.5 PLANT MODIFICATIONS

An enclosed ventilation hood has been requested for installation over primary system sample points in the demineralizer room to insure that samples are taken under conditions that provide for maximum safety to operating personnel. Previously, personnel have been required to wear respiratory protection apparel, which, even though safe for radiological hazards, introduces an operational safety hazard because it limits normal sight and communications.

Equipment required to modify the AM-2 mobile air monitor was received and installed during this period. Operation of the instrument was satisfactory, and increased sensitivity and greater ease of calibration is now possible.

4.0 TASKS

4.1 TASK XI - INSTRUMENT AND CONTROL STUDY

4.1.1. INTRODUCTION

Most of the preceding quarter (January to March, 1960) was devoted to the accumulation, assortment and evaluation of material eventually to become a part of a set of specifications for a new control scheme for the SM-1 reactor, fully compatible with the modern state of the electronics art, to be issued under the following title:

"SM-1 Specifications for Complete Static & Solid-State Nuclear Controls & Instrumentation System"

4.1.2. PROGRESS

During the present report period, the work initiated in the preceding period was completed. The above specifications were released and distributed June 2 to interested parties in the Army and AEC for review and approvals. The scope of the specifications covers all the electronic and electro-mechanical mechanisms required to control the SM-1 reactor through its rod-drive motors and clutches, either automatically, through defined feedback mechanisms, or manually, through an operator intermediary. In order to make the system as reliable as possible and as easily operated and maintained as possible, for human-engineering reasons, several innovations are incorporated in the specifications:

- (a) Period sharing between the startup and intermediate-level channels.
- (b) Cutoff of startup channels by intermediate channels (auctioneered).
- (c) Cutoff of intermediate channels by power channels (coincidence).
- (d) A trickle-charged nickel-cadmium stored-energy supply for transistor circuitry and all power requirements.
- (e) All interlocks and control functions accomplished by a logic network of electronics.

- (f) All rods controlled by a new system involving magnetic amplification exclusively.
- (g) Recorder converters for all recorders.
- (h) All analogue functions compared.
- (i) Time delays on all scram functions, as well as "shaped-period" control functions.
- (j) Rigid requirements on analogue signal drift from all causes.
- (k) 400 cps inverters to supply both scram bus and rod-position synchros.

4.1.3. FUTURE WORK

After customer approval of the specifications is received, a contract will be let on bid (by the R&D group) to a nuclear instrumentation and controls designer-manufacturer which will design, develop, test, install and guarantee performance of the equipment as specified. The R&D group will closely follow the progress of all phases of this work to assure adherence to the specifications. It is expected that the acceptance tests of the installed equipment can be approved within eight to eleven months after a letter of intent to proceed is received by the designer-manufacturer.

4.1.4. PROBLEMS

Only the development areas are expected to present significant problems. However, it is believed that these have been studied by the R&D group sufficiently, in conjunction with conferences with the designer-manufacturers, for means of problem solution and avoidance of delays to the program.

6.6 17

4.2 TASK XII - PLANT RESPONSE AND SYSTEM PERFORMANCE

4.2.1. INTRODUCTION

The power quality specifications for anticipated nuclear power plant designs place extremely close tolerances on the allowable voltage and frequency fluctuations. Present plant design techniques lack the necessary background of experimental data to permit a reliable approach to plant designs which will meet these close specifications during sudden, severe electrical load changes.

Task XII is a test and analytical program to investigate experimentally the SM-1 plant response to step load changes and to evaluate the test results as a tool in the design of future plants.

4.2.2. PROGRAM

Task XII consists of two distinct test phases:

- A. Transient electrical loading.
- B. Transient steam loading.

In the transient electrical loading tests, a resistance, reactance load bank is connected to the generator output terminals. This electrical load absorbing apparatus is employed to impose step electrical load changes on the plant of $\pm 10\%$, $\pm 20\%$ and $\pm 30\%$ on base loads from 0 to a maximum of 100%.

In the transient steam loading tests, the turbine-generator is bypassed by the steam which is dumped directly to the condenser. The plant is subjected to step changes in steam flow by means of control valves in the steam bypass line. The schedule of base load settings and transient load changes is similar to that of the transient electrical loading tests.

During the test runs described above, the behavior of selected system parameters is traced by recording oscillographs. The data traces are then evaluated and compared to similar, analytically derived, traces for the SM-1 plant. This comparison will contribute to improved analytical representations for use in the design of future plants.

4.2.3. PROGRESS

The schedule of December 30, 1959 anticipated end of core life at the end of March with Task XII installation and testing taking place April through June. The schedule was changed due to actual plant shutdown on April 28.

The new schedule called for installation to take place during May and June with checkout and testing slated for September. With the new schedule and the long delivery of steam dump line system components, only work that could be completed or would allow the plant to start up would be undertaken. A plant shutdown in September will be necessary for the installation of components not received for installation at this plant shutdown (see 4.2.3.3.).

During this quarter, installation continued whenever and wherever possible while the plant was in operation. After the plant was shut down, work progressed as men were available, depending upon personnel radiation dosage limits.

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

4.2.3.1.1. Instrumentation

The SM-1 plant is being instrumented to sense, process and record the behavior of approximately 50 system parameters. These parameters include such items as steam flow, turbine throttle pressure, generator rpm, feed-water flow, primary coolant temperature, etc. These parameters are common to both the transient electrical and transient steam tests, with the exception of eight associated with the turbine generator and three used only with the steam bypass line.

Installation of the instrumentation outside the vapor container continued during this report period. After the plant shutdown, the instrument installation in the vapor container commenced. A leak detector was borrowed for checking the vapor container penetrations to the approved specifications used for Test 600. The checking took longer than anticipated due to problems resulting from a faulty probe. A new probe was procured and the leak checking continued to completion, as qualified personnel were available.

Procedure approval for the installation of the thermocouples in the primary system was obtained from the NPFO. This procedure was based upon successful bench testing conducted in the presence of an NPFO representative. The installation of the thermocouples is complete.

Leak checking, calibration and checkout of the channels of instrumentation will continue into the next quarter. When the power meter, the power factor meter and some diodes and resistors for the electrical parameters are received, all items for the instrument installation will be complete.

4.2.3.1.2. High Pressure Turbine Modifications

Prior to the plant modifications for Task XII, the steam pressure at the turbine throttle was maintained at approximately 175 psia at all levels of plant output. This was accomplished by means of pressure reducing valves in the main steam line. These P.R.V.'s are considered a detriment to fast plant response to sudden load changes and are to be blocked open during the Task XII tests. Steam generator pressures up to 425 psia will now be imposed directly on the turbine and compensating modifications are necessary.

Following cooldown of the turbine, the insulation was removed and disassembly of the unit began. The manufacturer's representative arrived on May 4 to supervise the modification and inspection. The turbine rotor and the new first stage wheel were shipped to the manufacturer's plant for rework. While there, the rotor was cleaned, inspected, repaired and balanced. The repair encompassed straightening some blading and metal spraying the shaft at the 2nd and 4th stage packing areas which were eroded.

Further inspection of the turbine showed extensive erosion/corrosion had occurred on various diaphragms and corresponding areas of the casings. The diaphragms were sent to the manufacturer's service shop for cleaning and repair. The casings were removed and repaired in the ERDL machine shop. Repair of the diaphragms and casings consisted of building up the areas of erosion/corrosion with weld rod and machining back to the original dimensions.

The valve seats for the new steam chest were received undersize and were sent to the manufacturer's service shop for plating. The chest was assembled and tested hydrostatically by the ERDL machine shop.

The turbine with the new steam chest was reassembled; after the manufacturer's representative left on June 3, various items of piping and valve linkages were still to be installed. These items have been completed and the unit will be lagged as soon as it has been leak-checked after plant start-up. A detailed inspection report from the manufacturer is forthcoming.

The safety valve (SV-2) downstream of the main steam line pressure control valves has been removed and the extraction blanked off. The function of this valve is superfluous with modification of the turbine for higher pressure. Higher pressures are anticipated after the pressure control valves are blocked open, a requirement for Task XII testing. The higher pressure rated safety valve (SV-5) on the outlet of the steam generator will protect the secondary system equipment.

The new element for the low initial steam pressure regulator has been received and will be installed in the next quarter.

4.2.3.1.3. Steam Separator

The sudden load changes to be investigated by Task XII will impose surge conditions on the steam generator tending to promote carryover of moisture into the main steam header. To prevent possible damage to the turbine, an entrainment separator has been installed in the main steam header just upstream of the turbine throttle valve.

An inspection trip to the steam separator manufacturer was made on April 11, 1960 by an Alco representative. This inspection was in accordance with an AEC AND NPFO request that the integrity of the welds and internal structure of the vessel be checked before buttoning-up. The manufacturer has since forwarded copies of their data report and certificate of compliance.

A steam trap was ordered and received in compliance with a requirement for the separator installation.

The separator and trap with their component parts have been installed. The installation was down simultaneously with the steam bypass installation and will be cleaned and hydrostatically tested during the next quarter.

4.2.3.1.4. Testing and Reports

The Hazards and Precautions Report has been reviewed and is in process of being drafted for final submittal. The calibration and checkout procedures are ready for review. The test procedure comments are being evaluated for incorporation into a final draft. These documents and the analytical report will be completed as time and manpower permit.

4.2.3.2. Subtask 3, Installation and Checkout of the Electrical Load Bank

The load absorbing medium for the transient electrical loading tests is an air-cooled assembly of resistance and inductive-resistance elements. This load bank was fabricated by the General Electric Company as part of the Task XII program.

The load bank was issued in qualifying tests on the PM-2A at Dunkirk, New York during the first quarter of 1960.

The load bank was received from Dunkirk and installed on site June 1. The 4160-volt tie-line installation by an outside contractor is complete. The interlock system for personnel and system protection has been designed and partially installed. Interlock units are on order that will be consistent with those supplied with the load bank.

When the load bank was received, a connection on the M-7 contactor required installation. A manufacturer's representative installed this connector on June 21 and at that time, at Alco's request, measured the various unit resistances. The values were low and the representative anticipates correcting these values the week of July 4. A program for checking the power factor in order to correct the problems encountered and reported on in the last quarter has been drafted as follows:

- A. Obtain from the manufacturer the tolerances to which the load bank was constructed and which are considered standard in the industry for this type of equipment.
- B. Examine tolerances and express acceptance or rejection.
- C. The manufacturer's personnel and Alco personnel to concurrently measure both resistance and reactance at Fort Belvoir.
- D. Further action to be determined by the results of the measurements.

4.2.3.3. Subtask 4, Installation and Testing of Steam Dump Line

The steam dump line is a piping system which bypasses the turbine and dumps steam directly to the condenser. It is being installed at SM-1 to investigate its feasibility as a system for conducting steam loading tests on newly installed nuclear plants. The system will provide a means for full-range, transient-load and steady-state testing of the thermodynamic characteristics of a plant without the necessity for complete installation of the electrical distribution system or for the availability of an electrical load bank.

Approval was received for the steam dump system control equipment at the beginning of this report period. The system layout and designs were accomplished and procurement was initiated. The installation began about a week after end of core life and proceeded as materials became available. It was the intent at that time to make the changes and penetrations to the plant piping and equipment to points of shutoff. The remaining installation could then take place after the plant was running. This has been accomplished except for teeing into the feedwater heater steam line (3rd stage turbine extraction) and the pressure reducing valves for the air-ejector and evaporator. There is doubt whether or not the materials and equipment for these installations will be available in time to complete work before the plant startup.

The penetrations to the condenser (steam and water spray), the bypass penetration to the main steam header, the spray water manifold at the condensate recirculating pumps, and various piping and valve rearrangements for acceptance of the dump system piping were accomplished in this report period.

The first Copes-Vulcan drawings were received for approval on May 5. According to the vendor's delivery date of ten weeks after Alco approval of their drawings, the control equipment is due for shipment the last week of July. At this time, the installation will continue. The drawings for the piping installation are being brought up to data and the control system schematics, including material lists, are being completed.

A review of the parameters for transient steam loading shows that a changeover of channels of instrumentation from the transient electrical loadings will result in the requirements of an additional linear transducer for step load control valve travel and a differential pressure transducer for bypass steam flow.

The hydrostatic testing and cleaning of the system already installed or modified will be accomplished in the next quarter prior to the plant startup.

The test procedures and reports are included with those reported under Section 4.2.3.1.4.

4.3 TASK XIV - TEMPERATURE AND FLOW MEASUREMENTS

4.3.1. EFFECTS OF GAMMA HEATING IN FILLER

4.3.1.1. Introduction

An objective of the fuel plate temperature measurement device is to minimize those perturbations introduced by the addition of the device. Measurements are then representative of conditions that would exist without for symmetrically located uninstrumented plates in the core.

Gamma heating levels within the cladding and meat are negligibly affected by the introduction of the filler in the blocked channel temperature measurement device and no significant contribution toward a temperature perturbation is involved under these considerations. However, the presence of new material as filler presents an additional gamma heating source, and the outward flow of this heat will significantly affect the temperature gradients of the filler, meat, clad and forced convection film.

The ideal filler material has negligible neutron and gamma cross-sections. Zirconium was selected as a filler material because of its close approach to the former category, but it has a gamma cross-section approximately 20% higher than that of stainless steel. Since an 81-mil filler thickness is required, its contribution toward a plate temperature perturbation from gamma heating is significant and must be evaluated.

Since the heat generated within the filler must pass out through the meat, clad and coolant film, increases in these temperature gradients will occur in addition to the gradient within the filler itself.

The radial and axial distribution of the gamma heating rate within the core at rated power is shown in Figs. 4.1 and 4.2 respectively, calculated on the basis of a cylindrical homogeneous core. The following evaluation of temperature perturbation is based upon a location at the core center and upon rated core power in order to yield the highest gamma heating effects. Each thermocouple measurement must be corrected on the basis of core location and plant load to realize the best attainable experimental accuracy.

4.3.1.2. Nomenclature

h	Center channel film coefficient, $\text{Btu/hr ft}^2 \text{ F}$
k	Thermal conductivity, $\text{Btu ft/hr ft}^2 \text{ F}$
q	Filler gamma heat generation, Btu/hr ft^3
Q	Max gamma heat flux = qt_m , Btu/hr ft^2
t	Thickness of material, ft
T	Temperature, F

Subscripts:

F	Film
M	Meat
S	Stainless cladding
Z	Zirconium filler

4.3.1.3. Data

h	= 3070 (based upon coolant velocity of 5.65 fps)
K_m	= 10.8
K_s	= 11.0
K_z	= 10.9
t_m	= 10 mils = 8.33×10^{-4}
t_s	= 5 mils = 4.17×10^{-4}
t_z	= 40.5 mils = 3.37×10^{-3} (half thickness only)
q	= 6.14 watts/gm = 3.83×10^6 (based on 0.233 lb/in ³)
Q	= qt_z = 13,000

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4.3.1.4. Gamma Heat Effects On Various Gradients

$$\text{Additional } \Delta T_F = \frac{Q}{h} = \frac{13000}{3070} = 4.2 \text{ F}$$

$$\text{Additional } \Delta T_S = \frac{Q_{ts}}{K_S} = \frac{13000 (4.17 \times 10^{-4})}{11.0} = 0.5 \text{ F}$$

$$\text{Additional } \Delta T_M = \frac{Q_{ts}}{K_M} = \frac{13000 (8.33 \times 10^{-4})}{10.8} = 1.0 \text{ F}$$

$$\text{Additional } \Delta T_Z = \frac{qt_z^2}{2K_Z} = \frac{Q_{tz}}{2K_Z} = \frac{13000 (3.37 \times 10^{-3})}{2(10.9)} = 2.0 \text{ F}$$

$$\begin{array}{l} \text{Total Increase in Max Plate Temperature} \\ \text{due to Filler Gamma Heating} \end{array} = 7.7 \text{ F}$$

The result of these derivations on the instrumented fuel plate temperature distribution is shown graphically in Fig. 4.3. Note that the zero datum level shown is the temperature distribution based upon an ideal filler material (no filler heat generation) and nominal plate dimensions.

4.3.1.5. Conclusions

From Fig. 4.3 it is seen that the temperature gradient in the filler is very nearly flat. Exact placement of the thermocouple in the center of the filler is therefore not critical as the error in temperature measurement associated with an off-center tolerance would be small (0.03° F temperature change for 5 mils off filler centerline.)

The various parameters used in the previous calculations may not be known very accurately, and an evaluation of the resultant maximum error based on gamma heating correction should be made. This analysis is shown in the succeeding sections of the report.

Fig. 4-1 Radial Distribution of Gamma Heating Rate

Fig. 4-2

Axial Distribution of Gamma Heating Rate

4.3.2. EXPERIMENTAL ERROR OF TEMPERATURE DETERMINATIONS

4.3.2.1. Introduction

A preliminary examination of experimental accuracies associated with the determination of plate temperature was reported in the SM-1 Research and Development Quarterly Report, October 1 to December 31, 1959⁽³⁾. A further refinement of this examination is presented in this report, with the final presentation scheduled for the next quarterly report to include conclusions of the current study outlined below.

An analysis of the maximum experimental error associated with the determination of volumetric heat generation rate within fuel plate meat is currently in progress. The various factors contributing toward the overall error are difficult to treat precisely, as can be seen from the descriptions of such factors in the above reference. However, on the basis of the study to date, it is believed that volumetric heat generation is conservatively definable within $\pm 10\%$. This figure is therefore used as a basis in this presentation. A revision to incorporate a modified value is easily obtainable from the calculation details provided.

Again as in section 4.3.1., a center of the core location with rated core power output is considered in the error analyses in order to obtain the maximum value of the error terms. Other core locations or core powers will have smaller associated errors. For definitive purposes, the following nominal and worst local condition are considered for developing ignorance factors:

Nominal Case
(Used for correction items)

Worst Local Case
(Used for error terms)

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Fig. 4-3 Temperature Gradient in Filler

4.3.2.2. Centerline Temperature, Instrumented Plate

Thermocouple errors

Pre-Installation Calibration	$\pm 0.8^{\circ} \text{ F}$
Reference junction box error	$\pm 0.2^{\circ}$
Extension wire & connector effects	$\pm 2.0^{\circ}$
Read out (Heiland Visicorder)	$\pm 0.1^{\circ}$
	<hr/>
Total error, max.	$\pm 3.1^{\circ} \text{ F}$

The thermocouple calibration itself, upon repeated temperature cycling and aging, may approach $\pm 2.0^{\circ} \text{ F}$. However, temperature measurements made in initial task runs are within $\pm 3.1^{\circ} \text{ F}$. This error could be reduced further by calibration with connector and extension wire in the circuit, but the task time schedule does not permit this. Some error allowance would still be necessary for effects of temperature gradients along the thermocouple leads. See Section 4.3.4. for further details on instrumentation.

4.3.2.3. Centerline Temperature, Uninstrumented Plate

A listing of the applicable error terms is first given, followed by an explanation of the various entries.

Instrumentation terms

Thermocouple errors	$\pm 3.1^{\circ} \text{ F}$
Filler removal correction terms	
Filler gamma heating effects	
Filler gradient error, $2^{\circ} (10\% \text{ flux})$	$\pm 0.2^{\circ}$
Clad and meat gradient error	
$0.5^{\circ} (10\% \text{ flux}) (\frac{16}{15} \text{ thick})$	$\pm 0.1^{\circ}$
Film gradient error, $4.2^{\circ} (10\% \text{ flux})(10\% \text{ coef.})$	$\pm 0.9^{\circ}$
	<hr/>
Total error, max.	$\pm 4.3^{\circ} \text{ F}$

The thermocouple errors are the same as detailed in Section 4.3.2.2. Additional errors are introduced through the correction calculation of Section 4.3.1.4 for gamma heating in the filler. In the above error calculations, the initial degree value refers to the calculated gradient while successive values within () refer to the applicable ignorance factors, as detailed in Section 4.3.2.1.

4.3.2.4. Surface Temperature, Instrumented Plate

Instrumentation terms

Thermocouple errors $\pm 3.1^{\circ}\text{F}$

Translational correction terms

Filler gamma heating effects

Filler gradient error $\pm 0.2^{\circ}$

Clad and meat gradient error $\pm 0.1^{\circ}$

Meat fission heating effects

Meat gradient error $10.0^{\circ} (10\% \text{ flux})(10\% \text{ thick})^2 = \pm 3.3^{\circ}$

Clad gradient error $9.9^{\circ} (10\% \text{ flux}) = \pm 1.0^{\circ}$

Total error, max.

$\pm 7.7^{\circ}$

Note that in tabulating the errors due to filler gamma heating, a film gradient term is not involved. Section 4.3.2.3 refers to an uninstrumented plate, and therefore all temperature effects introduced by filler gamma heating must be calculated and deducted. However, this section refers to an instrumented plate, retains filler heating considerations but considers a translation of the reference point from the plate centerline to the surface. As the film gradient is not traversed, no corresponding calculation error term is involved.

The maximum nominal meat and clad temperature gradients are 10.0°F and 9.9°F respectively and correspond to the same core axis location yielding the maximum gamma heating effects.

Note that the meat gradient error term contains the square of the meat thickness ignorance factor. This is due to the meat temperature gradient varying as the square of thickness, volumetric heat generation rate remaining constant.

4.3.2.5 Surface Temperature, Uninstrumented Plate

Instrumentation terms

Thermocouple errors

Removal correction terms

$$\pm 3.1^{\circ}\text{F}$$

Filler gamma heating effects

Filler gradient error

$$\pm 0.2^{\circ}$$

Clad and meat gradient error

$$\pm 0.1^{\circ}$$

Film gradient error

$$\pm 0.9^{\circ}$$

Translational correction terms

Meat fission heating effects

Meat gradient error

$$\pm 3.3^{\circ}$$

Clad gradient error

$$\pm 1.0^{\circ}$$

Total error, max.

$$\pm 8.6^{\circ}\text{F}$$

4.3.2.6 Changes of Temperature, Fixed Position

In Sections 4.3.2.1 through 4.3.2.5, the concern has been over the error possible in determining the absolute value of various plate temperatures. Consider next the error associated with the determination of a change in temperature, both measurements made at the same reference point and under steady state conditions.

Consider further, as an example, that the phenomena causing the temperature change is crud deposition on the plate surface, causing the instrumented plate centerline temperature to increase for a given core power level. Such a phenomena, should it occur, would become evident within a relatively short period of reactor operation, so that neutron flux redistribution due to fuel burn-up is insignificant. In making the before-and-after comparison, those error contributing terms of fixed algebraic sign and magnitude (yet unknown) cancel out by subtraction, thereby improving the precision of measurement of temperature change.

Instrumentation terms

Thermocouple error, per reading

Reference junction box uniformity

Readout repeatability

Total error per reading, max.

$$\pm 0.1^{\circ}\text{F}$$

$$\pm 0.1^{\circ}$$

$$\pm 0.2^{\circ}\text{F}$$

Core power control error terms

Rated temp. gradient across core	20.5°F
Repeatability of core gradient determination	± 0.1°
Max. gradient, local coolant to instr. plate center	133°
Plate center variation due to power control error	
$133^{\circ} \left(\pm \frac{0.1^{\circ}}{20.5^{\circ}} \right) =$	± 0.65°

Total error, change in instr. plate center temp.

$$2 (\pm 0.55^{\circ}) \neq 4 (\pm 0.2^{\circ}) = \pm 2.1^{\circ}\text{F}$$

The instrumented plate centerline temperature, as the parameter whose change is being investigated, is observed before and after, and thus involves two thermocouple readings. During each of these measurements, factors other than increased thermal resistance contributing toward an increase in plate center line temperature must be eliminated. Bulk coolant temperature, at the channel locations adjacent to the plate measurement points, and core power level must therefore be kept the same. This can be regulated by reading any reference coolant temperature and the total core coolant temperature rise. The former reading is held constant by control rod adjustment, while the latter by adjustment of plant load.

However, these latter two measurements have an associated instrumentation error which does not cancel out through before-and-after readings. Some variation in plate temperature may therefore be due to small actual variations in these two parameters although the readings are the same.

The before-and-after measurement of the reference coolant temperature brings the total number of thermocouple readings to four, with an associated error of 0.8°F. This error is directly additive to plate centerline temperature error. The measurement of core coolant temperature increase is made directly by the existing SM-1 platinum resistance bulb installation, and the repeatability error associated with a reading is ± 0.1°F. The corresponding error in plate centerline temperature, as shown by the calculation in the table, is ± 0.65°F. Two such measurements are necessary, so that the total error from all instrumentation effects is ± 2.1°F.

In the above table, the maximum temperature gradient between the local coolant and the centerline of the instrumented plate is given as about 133°F. This includes hot channel factors related to individual film, clad meat, and filler gradients, and is therefore a conservative value.

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Thus it is seen that the determination of changes in various core temperatures can generally be made with less associated experimental error than the determination of absolute temperature values. Also, no simple or standard definition of experimental accuracy magnitude can be made, since individual considerations related to the purpose of the measurements involved greatly affect this magnitude.

4.3.3. BENEFITS OF TASK INSTRUMENTATION

4.3.3.1. Introduction

The subject of objectives and benefits of the task instrumentation was first treated in the SM-1 Research and Development Quarterly Report, July 1 to September 30, 1959⁽²⁾. With the error information generated in Section 4.3.2 available, it is of interest to reexamine the subject in a more quantitative manner than was then possible.

The potential advantages of experimentally measuring core temperatures and flow rates can be classified broadly into two areas. In the first area lies an improved ability to predict fuel plate temperatures through better knowledge of the film coefficient, as applied specifically to the geometry of the instrumented core. In the second area lies a means of ascertaining the actual operating values of core temperatures and flows by direct observation, rather than by analytical prediction necessarily involving the possibility of large deviations.

As elaborated upon in reference⁽²⁾, the present variance of opinion on the proper coefficient for use with the Dohns-Boelter Correlation is 0.019 to 0.21, as applied to the SM-1 reactor. On the basis of a 0.019 coefficient, this represents a maximum uncertainty of 21%. If the uncertainty associated with an experimental determination of the coefficient were less than this 21%, the task could contribute toward benefits on the first area previously mentioned.

The maximum film gradient in the center core channel is nominally calculated at about 84°F. According to Section 4.3.2.4, a $\pm 7.7^\circ\text{F}$ error is possible in experimentally determining the instrumented plate surface temperature. From section 4.3.2.2, a $\pm 3.1^\circ\text{F}$ error is possible in determining the adjacent bulk coolant temperature. Also, from Section 4.3.2.1, a 10% error in the local heat flux is possible from local meat thickness variations, and another 10% error from neutron flux level information. By considering only these possible error terms, an experimental determination of the "D-B" coefficient would have an uncertainty of 32.9%. This percentage would increase significantly when considerations of error in flow measurement and in coolant physical properties associated with temperature uncertainty are included.

Therefore, justification for Task XIV is not based on improved analytical techniques, but on the direct experimental observation of operating temperatures and flows. Various aspects of this advantage are discussed in the following section. The investigations implied are not planned as part of Task XIV, but would be made possible by this task for initiation under a separate task number.

4.3.3.2 Monitoring Instrumentation

From Section 4.3.2.5, it is seen that the surface temperature of an uninstrumented plate can be evaluated within $\pm 8.6^{\circ}\text{F}$, based on direct measurements. The analytical determination of the corresponding temperature would have an uncertainty much greater since error factors involving film coefficient, local coolant velocity, local neutron flux level, core power output, etc. would be involved.

Thus the margin of safety between the actual operating plate temperature and the limit for proper core operation can be determined with a greater degree of precision. The core power output could then be adjusted to a higher value without the possibility of exceeding this limit. This becomes progressively more important with increasing core power outputs for the same sized core since safety factors are less liberal, and consequences of exceeding the "safe" temperature are more serious.

Since the plate centerline temperatures of instrumented and uninstrumented plates differ by about 7.7°F at the most, the thermocouple readings of the former can be used directly for monitoring purposes in general reactor plant operation.

4.3.3.3 Detection of Crud Deposition

At high heat flux levels (e.g. as in the SM-2 design) a phenomena has been observed wherein "crud" carried by the primary loop water is sometimes deposited at a rapid rate on fuel plate surfaces. The occasional nature of the phenomenon's occurrence indicates that other factors are involved besides the heat flux level, but not much information is presently available on the subject.

However, when the required factors are present, the deposition rate produces a significant rise in fuel plate temperature within a matter of a few days of reactor operation. For high heat flux cores, therefore, it is highly advantageous to have direct fuel plate temperature instrumentation installed, so that rapid detection of the occurring phenomenon is possible.

According to Section 4.3.2.6, crud deposition effects can be identified when the short range increase in plate centerline temperature exceeds about 2.1°F which would correspond to a very early stage of the phenomenon's development.

4.3.3.4 Flux Redistribution with Fuel Burnup

For a given fuel plate and an associated flow channel, the local temperature gradient between the plate centerline and the bulk coolant is proportional to the local heat flux. The heat flux in turn is a function of the local neutron flux level and the local fuel burn-up.

Various local gradients along the instrumented fuel plate can be determined from the plate and coolant thermocouple instrumentation. The local fuel burnup can be estimated from initial neutron flux distribution data together with the plant load history. It is therefore possible to obtain an estimate of the core neutron flux redistribution with reactor usage from the task instrumentation.

4.3.3.5 Other Uses of Core Instrumentation

Various chemical and metallurgical phenomena occurring within the core may be quite sensitive to fuel plate temperature within a limited range. An accurate determination of plate temperatures is therefore necessary in related studies. Examples of such phenomena are surface corrosion, fission product release from UO_2 , and helium release from Zr B_2 .

In a bulk-boiling type reactor, changes in core power cannot be measured by changes in the temperature rise of the core coolant. However, power output and safe operation from thermal burnup can be determined by fuel plate temperature instrumentation. The purpose of the task instrumentation program is not merely to prove its applicability to the SM-1 in particular, but to plate-type cores of similar geometry generally.

4.3.4 THERMOCOUPLE AND DIFFERENTIAL PRESSURE TRANSDUCERS CONSIDERATION

4.3.4.1 Introduction

The reasoning behind the specifications for the task temperature and flow instrumentation is described. The purpose of this section is to develop instrumentation necessary to correlate the relationship between fuel plate surface temperatures, coolant bulk temperatures, coolant velocity, and local power distribution of the present SM-1 reactor.

Temperature instrumentation is divided into the following categories:

1. Thermocouples
2. Thermocouple Reference Junction
3. Temperature Measuring Circuit

Flow instrumentation is categorized as follows:

1. Impact and Static Pressure Probes
2. Differential Pressure Transducer
3. Flow Measuring Circuit

4.3.4.2 Thermocouples

The thermocouples were chosen to give maximum reliability for the purposes of this task. Thermocouple reliability depends on:

1. Calibration
2. Insulation Resistance
3. Sheath Material
4. Size of Thermocouple Wire
5. Method of Swaging Process
6. Selection of Thermocouple Wire Material

4.3.4.2.1 Calibration

The thermocouples were calibrated at temperature points near 450°F and 650°F to an accuracy of $\pm 0.8^\circ\text{F}$.

Thermal electric material specified for all thermocouples is Iron Constantan, Type ISA "J" premium calibration quality, whose limits of error are within $\pm 2^\circ\text{F}$ without calibration within this temperature range.

4.3.4.2.2 Insulation Resistance

The insulation resistance of a swaged thermocouple plays a very important role in the thermo-electric measuring circuit. Insulation resistance depends on the type and density of insulation, distances between wires and wires and sheath, and the moisture content and degree of purity of the insulation material.

Three oxides commonly used as insulation material for swaged thermocouples are magnesium oxide (MgO), Zirconia (ZrO_2) and Alumina (Al_2O_3).

MgO has very good high resistance properties and is preferable for use above 1000°F . However, because of the hydrostatic properties of MgO , moisture absorption from the atmosphere at exposed thermocouple ends is sufficient to greatly reduce insulation resistance. An undetected fault in the sheath of the water-submerged thermocouple installation, exposing the insulation, can readily cause insulation expansion, resulting in sheath rupture. Also, one of the impurities of MgO is magnesium chloride (Mg Cl_2).

The combination of moisture and Mg Cl_2 will make the type 304 SS sheath material of the in-core thermocouples susceptible to stress corrosion.⁶

ZrO_2 is usable up to 1200°F only, because of low resistivity. Hafnium, an element which has a high neutron absorption cross-section, is present in Zr O_2 and would increase the perturbation of the neutron flux in a core installation.

Al_2O_3 , an insulating material of fairly high resistance properties, is usable up to 2300°F . Due to the disadvantages of MgO and ZrO_2 for the Task XIV application, Al_2O_3 was chosen for the insulation material of Task XIV swaged thermocouples. It is readily available at a purity of 99.4%.

4.3.4.2.3 Sheath Material

The sheath material chosen for in-channel thermocouples is type 304L SS because of the absence of tantalum as found in Columbium. Upon exposure to a neutron flux, the tantalum produces isotopes of long life.

The sheath material of the extension leads and end-box thermocouples is type 347 SS. As these installations are in a relatively low flux region compared to the in-channel thermocouples, type 304L is not required.

4.3.4.2.4 Size of Thermocouple Wire

Stability and life of thermocouples are directly proportional to wire size. The thermocouple wire size for Task XIV was specified as large as possible consistent with application requirements.

For the extension leads and end-box thermocouples, 0.065 in. sheath dia and 0.013 in. dia wire is used, as sheath dimensions are limited by the allowable size of drill holes in the reactor vessel ring gasket.

For the water channels and blocked water channels, thermocouples of 0.040 in. dia sheath and 0.005 in. dia wire are being used, as sheath dimensions are limited by the amount of clearance between stationary fuel element assemblies.

4.3.4.2.5 Method of Swaging or Reduction Processes

There is some controversy at present regarding the most desirable method of producing the metallic sheath of thermocouples for use in nuclear reactors. The difference of opinion is centered on the two manufacturing techniques used to reduce the sheath diameter: single-pass swaging and multipass swaging.

One manufacturer, E. C. Smith,⁷ contends that extreme internal sheath stresses, possibly resulting in interior sheath cracks, are built up by repeated reduction. These cracks may open up during thermal cycling of the thermocouple after installation and result in sheath failure. He also contends that repeated reduction damages wires excessively and requires annealing at the sheath anneal temperature, which is different than the thermocouple wire anneal temperature. This is said to affect thermocouple life and stability.

Mr. H. George Johannson,⁸ of the Bettis Plant in Pittsburgh, Pa., was contacted for his comments on the above statements. Mr. Johannson was instrumental in developing the Military Specification Thermocouples, corrosion-resistant metal sheathed, for nuclear reactors MIL-T-22300 (Ships), 11 January 1950. Mr. Johannson did not think this annealing problem was valid for chromel-alumel, but did not rule out its possibility for other thermocouple materials. He felt, however, that this question was still debatable.

The multiple swaging method causes thickening of the walls by radial material flowing along slip or stress lines. Hence the interior sheath wall cross section is made up of sharp peaks and valley which are potential rupture points. For single-pass swaging it is reasoned that the interior cross section is damaged to a lesser degree.

Other manufacturers contend that the single-pass swaging method results in a thermocouple cost. However, Alco's experience in reviewing various vendor quotations listing both options for Task XIV application was that the difference was frequently insignificant.

The arguments against single-pass swaging were based on the promise that the more costly technique was not necessary. No arguments were noted that advocated the multi-pass swaging on technical considerations along. Therefore, since opinion on the single-pass swaging thermocouples ranged from far superior to equivalent, and cost factors were insignificant, the single-pass method was specified.

4.3.4.2.6 Selection of the Thermocouple Wire Materials

Selection of suitable thermocouple materials was dependent on in-pile and out-of-pile stability, useful temperature range, and Seebeck Coefficient.

Five material combinations investigated were Iron-Constantan, Chromel-Alumel, Copper-Constantan, Platinum-Platinum 10% Rhodium, and Platinel.

Platinum-Platinum 10% Rhodium, and Platinel are not considered feasible for nuclear reactor employment, since these materials contain high neutron absorption cross section elements whose transmutation products cause calibration drift.

Thermal ("slow") neutrons cause the transmutation of the high thermal absorption cross section elements. Epithermal ("fast") neutrons are thought to displace atoms in the thermo-electric material, thus causing work hardening.

The above effects will cause thermocouple calibration drift and change⁹ only in radiation zones where the thermocouple wire is in a temperature gradient area. The greatest temperature gradient for Task XIV thermocouples will occur at the reactor vessel ring gasket. Here, the radiation level is considerably lower than in the core zone, and the effects of irradiation on thermocouples penetrating the ring gasket will be reduced.

In-core thermocouples, exposed to large thermal gradients and high radiation, will be subject to effects of irradiation. However, this combination is not encountered in the Task XIV installation.

Copper-Constantan is not feasible for nuclear reactor use because it oxidizes above 660°F. The brazing temperature needed to embed Task XIV thermocouples in fuel plates, etc., is well above the 660°F mark.

Leeds and Northrup⁹ have run drift tests on Iron-Constantan thermocouple wire. The wire was cycled for 200 times in 500°F to 1300°F temperature range for 3000 hr. MgO-insulated, stainless steel sheathed thermocouple wires of both 1/8- and 1/16-in., O.D. were used in the tests. The results of this test were that the Iron-Constantan wire drifted less than 0.1°F at 500°F. No hysteresis effects were observed.

Bettis tests⁹ on stainless steel-sheathed, ceramic insulated thermocouples of 0.010-in. Chromel-Alumel wire, conducted between room temperature and 1000°F, showed calibration drift up to 1-1/2°F.

Thermocouple manufacturers claim Iron-Constantan is more stable than Chromel-Alumel. Empirical evidence shown above supports this contention. Therefore, Iron-Constantan, when swaged into a stainless steel sheath, is preferable for Task XIV application on the basis of out-of-pile data.

Limited information is available concerning in-pile stability of thermoelectric materials. Reference (10) indicates that Copper-Constantan, Iron-Constantan, and Chromel-Alumel thermocouples receiving a total irradiation of 2×10^{19} n/cm² showed no change in calibration within ± 30 microvolts (approximately 1°F). This was within the accuracy of the experiment.

In Reference (11) Copper-Constantan, Chromel-Alumel, and Iron-Constantan thermocouples received a total irradiation of 2×10^{18} n/cm², and significant shifts in calibration were noted. Chromel-Alumel showed the greatest deviation of approximately 1.1°C. This seems to contradict Ref. (10) results, but it should be noted that the flux in Ref. (11) was slightly higher than in Ref. (10).

The General Electric Hanford Lab¹² conducted tests on Iron-Constantan and Chromel-Alumel thermocouples. A radiation dose above 4 or 5×10^{18} nvt (exact value classified) at approximately 300°C showed no deviations from calibration within 50 μ v, accuracy of the experiment. It was concluded that there was no significant change in calibration that would present any problems in engineering applications.

Since the EMF of Iron-Constantan in the temperature range of 400 to 600°F is approximately 30 $\mu\text{v}/^\circ\text{F}$ compared to 22 $\mu\text{v}/^\circ\text{F}$ for Chromel-alumel. A greater experimental accuracy is achievable from Iron-Constantan for a given value of μv drift.

Therefore, considering the above experimental data, Iron-Constantan was selected as the material for the Task XIV thermocouples.

One possible problem area exists with Iron-Constantan thermocouples. The iron is subject to rusting when exposed to an oxidizing medium. In the Task XIV installation, this problem is not encountered because of the construction of the instrument lead connector. The sheaths are brazed to the connector block, the wires to ceramic insulated pins mounted in the block, and the chamber in which the wires are exposed is weld sealed in the laboratory.

4.3.4.3 Thermocouple Reference Junction

A Pace thermocouple ambient reference junction box with a temperature accuracy including stability within $\pm 0.2^\circ\text{F}$ will be used in the thermo-electric measuring circuit, operating on a 115-volt, 60-cycle line.

The model to be used for Task XIV has a total of 36 channels and 72 input terminal pairs. With this arrangement, 36 Iron-Constantan or 36 Chromel-Alumel thermocouples (as well as any combination of both totaling 36) can be used for future application at the SM-1 site.

The Pace unit was chosen on the basis of high accuracy, good flexibility for future critical applications, and relatively low cost.

Line leads for the reference junction box will be shielded to minimize electrical pickup that may influence thermocouple output signals.

4.3.4.4 Temperature Measuring Circuit

The in-core thermocouples have a type 304 L SS sheath, 0.040-in. diam, and attach to one side of the connector assembly with their extension leads connected to the other side. Both the extension leads and the end box thermocouples will have a 0.065 in. dia, 347 SS sheath. The connector assembly pins will be made of stainless steel, but since ΔT across the connector pins is zero, the introduction of this third metal into the thermo-electric circuit will not affect the thermo-electric circuit EMF.

The end box thermocouples and extension thermocouple leads will then be connected to the reference junction box connections, made of thermocouple materials. From the reference junction box, shielded, untinned copper wire signal leads will be brought out through vapor container penetration plugs and then to Leeds and Northrup selector switches. These switches have a low contact resistance of 0.001 ± 0.0005 ohm and a thermal EMF of less than $1.0 \mu v$ at normal operating speed.

From the switches, the signals are fed to a Sanborn amplifier, Model 350-1500, and then to a Consolidated-Electro Dynamics recorder, Model 119-T4-36.

4.3.4.5 Thermocouple Specifications

The Military Specification Mil-T-22300 (Ships) Thermocouples, Corrosion Resistant Reactors (11 January 1960), was used when ordering all thermocouples.

4.3.4.6 Flow Instrumentation

4.3.4.6.1 Impact and Static Pressure Probes

Stainless steel tubing is used for the impact and static pressure probes which apply P_i and P_s across the differential pressure transducer diaphragms.

The in-core probes are 0.040 in. diam, while end box probes are 0.065 in. diam. The same reasons which fixed thermocouple and extension lead sizes again dictate pressure probe and extension lead sizes.

4.3.4.6.2 Differential Pressure Transducers

Fifteen Ultradyne differential pressure transducers will be used to relate ΔP to flow for core flow measurements. One each will be employed for the end boxes of fuel element assembly numbers 15, 36, 45, 53, 62, 67, and 74, and four each for the water channels of control rod element No. 33 and fixed fuel element No. 67. These transducers have a non-linearity of $\pm 0.5\%$ and are designed for use with a Sanborn Model 350-1100 system which is available at the SM-1 site.

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The differential pressure ranges for each fuel element assembly end box and fuel element water channel are listed as follows:

<u>Range</u>	<u>Fuel Element</u>	<u>Fuel Element Assembly End Box</u>
0 - 0.1 psi	67	
0 - 0.2 psi		67
0 - 0.3 psi	33	15
0 - 0.5 psi		62, 74
0 - 1.0 psi		36, 53
0 - 1.5 psi		45

After each pressure probe has been installed in an instrumented element, corresponding pressure transducers will be connected to the probe leads, and each flow velocity measuring point will be calibrated in a laboratory test loop.

The Ultradyne transducers were chosen on the basis of a high accuracy of 1/2% of full scale reading, relatively low cost, and compatibility with existing associated instrumentation at the reactor site.

To waterproof the transducers, Ultradyne uses Scotch Cast #253. This epoxy resin will also be used as an electrical insulator. This resin will break-down after a total dose of 10^8 rads. The transducers will be installed within the quadrant of the shield tank top that straddles the spent fuel chute centerline, and as close to the center of the shield tank top as possible. Gamma radiation ¹³ in the area will be 235 mr/hr or less at full power operation. At this dose rate, the epoxy will last the entire life of the SM-1 plant.

4.3.4.6.3. Flow Measuring Circuit

The flow measurements are derived from the differential pressure transducers, which provide an electrical signal proportional to ΔP . This signal is then fed to the Sanborn Model 350-1100 amplifier and then to a Heiland Visacorder, Model No. 1012.

Transducer excitation and signal leads will be shielded to minimize electrical pickup.

4.3.5 REFERENCES (TASK XIV)

1. Richards, W.M.S., "APPR-1 Research and Development Program; Design Analysis for Flow and Temperature Measurement Program, Task No. V," APAE No. 37, September 26, 1958.
2. "SM-1 Research and Development Quarterly Report, July 1 to September 30, 1959," APAE Memo No. 237, January 15, 1960.
3. "SM-1 Research and Development Quarterly Report, October 1 to December 31, 1959," APAE Memo No. 251, April 8, 1960.
4. "SM-1 Research and Development Quarterly Report, January 1 to March 31, 1960," APAE Memo No. 264, July 6, 1960.
5. Gebhardt, F.G., "SM-1 Reactor Vessel Penetrated Gasket, Design and Test Report," APAE Memo No. 254, June 10, 1960.
6. Posey, W.J., NP6787 Progress Report #46, April and May 1958.
7. E.C. Smith Manufacturing Co. Inc., Core-Type Thermocouple. Specification with Explanations.
8. Johansson, H.G., Westinghouse Bettis Plant, Personal Communication to L.A. Della Villa of Alco Products, Inc., June 15, 1960.
9. Fink E.W. and Hunt T.W., "High Accuracy Thermocouple Measurement Techniques," WAPD-AD-TH545 October 28, 1959.
10. Boorman, C., "Effects of Neutron Flux on the Thermoelectric Power of Thermocouples and on the Resistivity of Platinum and Nickel," AERE E/R 572, - September 1950.
11. Madsen, P.E., "The Calibration of Thermocouples Under Irradiation in BEPO," AERE M/R 649, January 1951.
12. Green, D.R. and Tobin, J.C., Hanford Atomic Products Operation, Richland, Washington, Personal Communication to L.A. Della Villa of Alco Products, Inc., July 18, 1960.
13. Rosen, S.S., Alco Products, Inc., Personal Communication to L.A. Della Villa of Alco, May 10, 1960.

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4.4 TASK XVI - SM-1 DECONTAMINATION

4.4.1 Introduction

A corrosion program was initiated in the previous (January - March 31) period to determine if full system decontamination would be feasible. Early in the current report period preliminary data indicated that stellites, located in the control rods and control rod drives, might be attacked by the solutions during a full system decontamination. In light of this data an extensive corrosion program would have been necessary to determine with confidence the behavior of the stellites during decontamination. The schedule did not permit such a program since the remaining time was necessary for procurement of equipment. The decision was made, therefore, to decontaminate only the steam generator. The engineering of full system decontamination and studies of the corrosion of other materials in the decontamination solutions were terminated. Engineering, procurement and fabrication of partial system decontamination materials and equipment were carried forward.

4.4.2. Program

The following Phase I pre-decontamination activities were performed during the period:

- A. Corrosion studies of stellites 3, 12 and 19, SS Types 410 and 17-4 PH, and butyl, buna-N, and buna-N, and buna-S rubbers.
- B. Review of status of work with ARMY - AEC representative.
- C. Estimation of volume and activity of decontamination wastes.
- D. Calculation of dose rate to be expected during removal of rod drive.
- E. Review and modification of seal-on steam generator isolation plug.
- F. Design and drawing of temporary shield tank cover.
- G. Completion of design and drawings of steam generator isolation plug and associated equipment.
- H. Resumption of circulating piping drafting.

- I. Specification of instrumentation for measurement of temperature, pressure and flow rate during decontamination.
- J. Calculation of pressure drop in the circulating loop and specification of pump.
- K. Review of drawings for material specification and engineering design.
- L. Specifications of equipment and materials from approved drawings.

4.4.3 Progress

Samples of SS Types 410 and 17-4 PH, and stellites 3, 12 and 19 were given a simulated full system decontamination. Both insulated samples and couples with SS type 304 were decontaminated. In addition, samples of stellite 3 and stellite 3 coupled to SS type 304 were decontaminated individually to more nearly simulate the surface to volume ratio that exists at SM-1. Westinghouse Electric Corp. (Bettis Plant) and Hanford Works have been contacted regarding their experiences with stellites. Several stressed and non-stressed specimens of butyl, buna-N and buna-S rubbers were given a simulated partial system decontamination; the seals on the steam generator isolation plug and special gaskets between pipe sections will be fabricated from one of these materials.

On May 9 a review of the task was presented to Army - AEC representatives. The corrosion programs for both full system and partial system decontamination were discussed. Full system decontamination could not be recommended in light of preliminary stellite corrosion data. Engineering designs for both types of decontamination were reviewed and future work on partial system decontamination was outlined.

A estimate of the volume and activity of decontamination wastes has been made on a decontamination factor of 100 (i.e., 99% of activity removed). The activity was assumed due to Co^{60} , Co^{58} , Mn^{54} , Fe^{59} , Cr^{51} , Sr^{90} - Y^{90} , Y^{91} and Ba^{140} - La^{140} , and was calculated from SM-1 coupon data. Approximately 3200 gal of liquid radioactive waste will be generated and temporarily stored in two 2000-gal tanks. Initially, the tank containing the caustic permanganate solution, citric acid solution and intermediate rinse will read 1100 mr/hr on contact and 450 mr/hr at one meter. Upon standing, the solutions will react to form approximately 200 gal of manganese dioxide slurry ($17 \mu\text{c/cc}$) which will settle to the bottom leaving a clear supernate ($6 \times 10^{-2} \mu\text{c/cc}$). The other tank, containing subsequent rinses, will read 86 mr/hr on contact and 35 mr/hr at one meter.

A rod drive will be removed so that the crud that has accumulated in the lower portion of the reactor can be removed. An estimate of the dose rate to be expected during removal of a rod drive was necessary. The activation of the rod drive components was calculated by determining the neutron flux in the regions of the core and lower reflector by use of the Valprod Code with the IBM-650. The materials considered in the activity calculation were SS types 302, 304 and 17-4 PH and the stellites. Then the gamma dose on the surface of the control rod drive shaft was calculated by considering the shaft as an infinite cylinder. To this was added the gamma dose from the pinion and shaft which were assumed to be a point source. The total dose at the back of a drive, where it will be handled during removal, will be approximately 6.7 R/hr. This dose will be reduced to a tolerable level by lead shielding.

A temporary shield tank cover to allow insertion of the steam generator isolation plug and associated apparatus has been designed and drawn. Several different seal designs were reviewed to determine the best method of sealing the isolation plug in the reactor outlet nozzle. A design was selected that comprises two rubber seals separated by a metal spacer. The assembly drawing of the plug and associated apparatus is shown in Dwg. D9-54-1015, Rev. A.

The main circulating piping layout is almost complete. Instrumentation, corrosion bypass loops and pump dimensions from a vendor to be selected are to be added. A bypass flow meter and Bourdon gage will be procured to measure flow rate and pressure; existing thermocouples will be used to measure temperatures. Coupon samples of the materials in the primary system will be inserted in bypass loops to determine their behavior during a decontamination. Two bypass loops will be required - one at 5 fps and the other at about 0.1 fps - to obtain the flows that would occur in various parts of the primary system during a full system decontamination. Pressure drop has been calculated and necessary information supplied to the procurement department to request bids on a pump.

4.4.4 Results

Weight losses of samples of SS types 410 and 17-4 PH given a simulated full system decontamination are shown in Table 4-1. Only insignificant corrosion occurred; no problems would be expected from these materials during a full system decontamination. Weight losses due to simulated full system decontamination and change in hardness of stellites 3, 12 and 19 are given in Table 4-2. The large weight losses (about 100 times that for SS type 304), drastic reductions in hardness and pits to a depth of 18 mils made it impossible to recommend full system decontamination at this time.

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TOOL FOR OPERATION PC-20 REF-20-96-1088

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08-10-60	1	ASSEMBLY	W. H. 27	
		FOR TYPING	W. H. 27	
		REVISION	W. H. 27	

25	1	ST. 304	CLAMP HALF
25	2	ST. 304	CLAMP HALF
27	1	ST. 304	SPACER
26	2	ST. 304	WEL NUTS 1/2-10
28	1	ST. 304	GUIDE STOP
29	2	ST. 304	GUIDE
23	2	ST. 304	CUTTER PIN 1/2-10
22	2	ST. 304	CUTTER PIN 1/2-10
21	2	ST. 304	CLUTTER NUTS 1/2-10
20	2	ST. 304	CLUTTER NUTS 1/2-10
19	2	ST. 304	CLUTTER NUTS 1/2-10
18	2	ST. 304	CLUTTER NUTS 1/2-10
17	2	ST. 304	CLUTTER NUTS 1/2-10
16	2	ST. 304	CLUTTER NUTS 1/2-10
15	2	ST. 304	CLUTTER NUTS 1/2-10
14	2	ST. 304	CLUTTER NUTS 1/2-10
13	2	ST. 304	CLUTTER NUTS 1/2-10
12	2	ST. 304	CLUTTER NUTS 1/2-10
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10	2	ST. 304	CLUTTER NUTS 1/2-10
9	2	ST. 304	CLUTTER NUTS 1/2-10
8	2	ST. 304	CLUTTER NUTS 1/2-10
7	2	ST. 304	CLUTTER NUTS 1/2-10
6	2	ST. 304	CLUTTER NUTS 1/2-10
5	2	ST. 304	CLUTTER NUTS 1/2-10
4	2	ST. 304	CLUTTER NUTS 1/2-10
3	2	ST. 304	CLUTTER NUTS 1/2-10
2	2	ST. 304	CLUTTER NUTS 1/2-10
1	2	ST. 304	CLUTTER NUTS 1/2-10

REV	DATE	DESCRIPTION
1	08-10-60	ASSEMBLY
2	08-10-60	FOR TYPING
3	08-10-60	REVISION
4	08-10-60	
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D9-54-1015

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However, since a large surface-to-volume ratio existed when the samples were decontaminated together in a static autoclave and the corrosion may be autocatalytic, several samples were individually decontaminated. Sample weight losses of 123 mg/dm² and 331 mg/dm² for insulated samples and 306 mg/dm² for a couple with SS type 304 were observed. No loss of hardness occurred. Further work would be required before the behavior of the stellites during a decontamination could be predicted.

TABLE 4-1
SAMPLE WEIGHT LOSS OF
SS TYPES 410 AND 17-4 PH, mg/dm²

	<u>SS Type 410</u>			<u>SS Type 17-4 PH</u>		
	<u>No. of Samples</u>	<u>Avg.</u>	<u>Range</u>	<u>No. of Samples</u>	<u>Avg.</u>	<u>Range</u>
Coupled with SS type 304	1	90		1	15.1	
Insulated	3	117	115-119	4	15.6	14.4-17.4
Weight loss of SS type 304 coupled with SS types 410 and 17-4 PH	1	6.3		1	7.2	

TABLE 4-2
SAMPLE WEIGHT LOSS, mg/dm^2 , AND
ROCKWELL HARDNESS, RC, OF STELLITES

	<u>Stellite 3</u>			<u>Stellite 12</u>			<u>Stellite 19</u>		
	<u>No. of Samples</u>	<u>Avg.</u>	<u>Range</u>	<u>No. of Samples</u>	<u>Avg.</u>	<u>Range</u>	<u>No. of Samples</u>	<u>Avg.</u>	<u>Range</u>
Weight loss coupled with SS type 304	2	1300	1300	2	700	700-800	2	900	700-1000
Weight Loss Insulated	3	900	700-1200	4	800	700-1000	4	900	800-1000
Weight loss of SS type 304 coupled with stellites	2	1.3		1	2.4		2	1.2	
Rockwell hardness before decontamination, RC			44-58			57-60			56-58
Rockwell hardness after decontamination, RC			~ 3			20-23			~18

The following quotations have been requested: the chemicals that will be used to prepare the decontamination solutions, the steam generator isolation plug and associated equipment necessary to lower and secure the plug, the piping and fittings for the circulating loop and the circulating pump.

4.4.5 Conclusions

Preliminary corrosion data indicated large weight losses, severe pitting and loss of hardness for the stellites so that full system decontamination could not be recommended. Engineering design of partial system decontamination is almost complete and equipment procurement is underway. No difficulty is expected in meeting the scheduled installation of equipment date of October 7.

4.4.6 Work in Next Period

By the end of the next period all the equipment will have been purchased and, where possible, installed.

4.5 TASK XVIII - ANALYSIS OF SM-1 CORE I ACTIVITY DATA

4.5.1 Phase I, Induced Activity Buildup

4.5.1.1 Statement of Problem

The purpose of Phase I of Task XVIII is to be interpret radiochemical data obtained at the SM-1 during Core I lifetime in order to account for the activity buildup. Phase I includes the study of those nuclides formed only by neutron bombardment.

4.5.1.2 Program

The program for this period included the analysis of crud and water samples obtained since July 1, 1959. The radiation levels at the end of Core I burnout were measured. Data obtained from the PWR were obtained and compared to SM-1 data. New equations were derived to account for the activation of corrosion products deposited in the SM-1 core. Equations used by KAPL were used to determine the activity buildup in the SM-1.

4.5.1.3 Progress

4.5.1.3.1 Analysis of Crud and Water Samples

Interpretation of crud and water samples obtained since July, 1959 was made. The results show that the ratio of the total activity in the filterable material (crud) to the total activity in the non-filterable material (less than 0.45 micron diam) varies with the crud level. When the crud level is around 0.5 ppm, about 75% of the total circulating activity is in the filterable fraction. Since July, 1959, the total crud level (filterable plus non-filterable) has averaged 0.44 ppm. The average ratio of the filterable to non-filterable fractions was about 0.8.

Previous to July, 1959 (October, 1957 to October, 1958) the total crud level averaged about 0.46 ppm. During that period the average ratio of filterable to non-filterable fractions was about 0.5. Thus, it appears that the total crud level has not changed significantly during Core I lifetime.

The data obtained from the crud samples taken between February, 1958 and February, 1960 indicates that the specific activity has increased by a factor of approximately 2.4 over the two-year period. Interpretation of crud and activity deposited on various types of metals exposed to the coolant was started in June.

4.5.1.3.2 Long-Lived Dose Rate Buildup

Dose rate measurements were taken at the end of core life and plotted on dose rate buildup curves. Figure 4-4 shows the buildup of radiation levels on opposite sides of the steam generator during Core I lifetime. The buildup curve for point 3 is approximated by the equation $y = x^2$ where x is the number of months since reactor startup and y is the radiation level in mr/hr. The buildup curve for point 19 can be approximated by the equation $y = 1/3 x^2$. The buildup at other points can also be approximated by equations of the same form. The greater radiation level at point 3 can be attributed to the geometric differences between point 3 and point 19. Theoretical equations derived and used previously (7) (APAE No. 51) do not give results which show a similar type of curve. Thus, equations better describing the observed buildup of activity are needed.

4.5.1.3.3 Mathematical Equations

Two different groups of equations have been programmed from the analog computer. One group consists of five differential equations. The equations are based on the assumption that all activity originates from the corrosion of the core. The activated corrosion products are subsequently released to the coolant, from which they deposit on the surfaces of the primary system. A portion of the deposits are later re-released to the coolant. The other group of equations consist of seven differential equations. These equations take into account activity arising from corrosion products deposited in flux as well as that activity due to corrosion release from the core. The re-release of deposits is also included. Constants required for both groups of equations were found from SM-1 experimental data.

Reduction of data for use in equations derived by personnel at KAPL was performed. These are a group of twelve differential equations which have been programmed for an IBM-704 computer. The equations are similar to the group of seven derived by Alco but distinguish between the filterable and non-filterable activity in the coolant. In addition, loss to crud traps are considered.

4.5.1.3.4 Other Reactor Data

Personnel at Shippingport PWR were visited to obtain activity data for comparison to SM-1 data. Induced activities found in the PWR crud and those observed in the SM-1 crud after 1440 EFPH operation were:

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Fig. 4-4 Buildup of Radiation Levels on Opposite Side of Steam Generator During Core I Lifetime

Nuclide	Activity (dpm/mg crud)		% of Total Activity	
	PWR	SM-1	PWR	SM-1
Co ⁶⁰	5.8×10^6	3.9×10^5	39.2	10.8
Co ⁵⁸	2.5×10^6	21.0×10^5	16.9	58.0
Fe ⁵⁹	1.8×10^6	5.0×10^5	12.1	13.8
Cr ⁵¹	2.2×10^6	4.9×10^5	14.8	13.5
Mn ⁵⁴	0.8×10^6	1.4×10^5	5.4	3.9
Zr ⁹⁵	0.97×10^6		6.5	
Hf ¹⁸¹	0.76×10^6		5.1	
	14.8×10^6	3.6×10^6		

The lower specific activity level at the SM-1 can be a result of several factors. The main source of the activity in the PWR apparently arises from corrosion and wear products deposited on in-flux surfaces. In the SM-1 the main source can be accounted for by corrosion of the activated in-flux components. Since the PWR is operated at high pH, the corrosion deposits could remain in-flux longer than they do during operation at low pH. The longer the deposits remain in-flux, the greater the specific activity would be. The greater contribution of Co⁶⁰ activity to the total activity in the PWR, when compared to the contribution found in the SM-1, suggests that another source of Co⁶⁰, such as wear products, is significant. Deposits removed from in-flux PWR surfaces show an enrichment in chemical cobalt and thus supports the theory. Many of the high cobalt areas in the PWR are in the control rod mechanisms located above the reactor core. Wear products presumably drop down and deposit on in-flux surfaces. Also, a higher Co⁵⁸ to Co⁶⁰ ratio would be expected in the SM-1 than in the PWR; since the SM-1 has a fast to thermal flux ratio of seven and in the PWR, the ratio is about one.

After approximately 4000 EFPH, a leak was repaired in one of the PWR boilers. Dose rates at manholes were 300 mr/hr measured with a cutie pie. Measurements as close to the tube sheet as practical were 2 r/hr. A comparison of these values and those found in the SM-1 steam generator after

7700 EFPH is given below.

	<u>DOSE RATE (mr/hr)</u>	
	<u>Tube Sheet</u>	<u>Hand-hole</u>
SM-1	6300	1800*
PWR	2000	300

*Dose rate at lowest part of water box, approximately equivalent to hand-hole position of PWR.

To make a comparison of the dose rates, the values measured at the SM-1 should be corrected back to 4000 EFPH. At the SM-1, radiation levels on the outside of the steam generator increased about a factor of 2.5 from 4000 EFPH to 7700 EFPH. Assuming radiation levels inside the steam generator increased in the same manner, the dose rates at 4000 EFPH are estimated to be 2500 mr/hr at the tube sheet and 720 mr/hr at the lower part of the water box. The radiation level at the SM-1 tube sheet is therefore comparable to that observed at the PWR. The difference in the dose rates away from the tube sheet is probably due to geometric factors.

4.5.1.4 Results and Conclusions

These conclusions were drawn from analysis of data to date:

1. It is difficult to accurately portray the buildup of radiation levels at various points on the primary system by the equations used previously.
2. The average total crud level (filterable plus non-filterable fractions) in the coolant has not significantly changed from October, 1957 to February, 1960. The average ratio of the filterable to non-filterable fraction increased from 0.5 to 0.8 during that period.
3. The specific activity of Co^{60} in the filterable crud has increased a factor of 2.5 during the two-year period from February, 1958 to January, 1960. The theoretical equations predict about the same increase during that period. However, the equations predict a specific activity nearly five times that observed. Thus, equations better describing the observed activity buildup are required.
4. Comparison of SM-1 activity ratios to those found in the PWR show the importance of minimizing the cobalt content of minor surfaces exposed to the coolant. In the PWR, many of the high cobalt areas are in the

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control rod mechanisms located above the reactor core. Cobalt wear products presumably deposit on in-flux surfaces. In the SM-1, the high cobalt surfaces, also in the control rod mechanisms, are located beneath the core in a low flow area. Thus, the effect of wear products is considerably diminished.

4.5.1.5 Program for Next Period

The new groups of equations will be run on the analog computer and the results analyzed. Results of the data run on the IBM-704 computer will also be interpreted. Interpretation of crud, water and metal coupon samples taken during Core I lifetime will be completed. The final draft of the report on activity build-up during Core I lifetime will be completed and will be issued in September, 1960.

4.5.2 Phase II, Fission Product Activity Associated with Core I

4.5.2.1 Statement of Problem

The objective of this phase is to analyze all fission product data collected at SM-1 during Core I life.

4.5.2.2 Program

The program includes the interpretation of the data in terms of the source of the fission products and the hazards associated with their presence in the SM-1 primary coolant.

4.5.2.3 Progress

An outline of the final presentation has been prepared. Analysis of the data has started.

4.5.2.4 Results

Any results reported at this time would be premature and are therefore not offered.

4.5.2.5 Conclusions

No conclusions have been reached at this point.

4.5.3 Phase III, Investigation of Fissionable Material Surface Contamination of SM-1 Core II Fuel Elements

4.5.3.1 Statement of Problem

The specification for the fabrication of SM-1 Core II fuel elements included a qualitative check for alpha contamination. Consequently, no quantitative data were obtained with which to calculate a reference fission product concentration in the SM-1 coolant at Core II startup. Experience with alpha contaminated fuel elements at other installations and the recent advances in alpha counting techniques made it desirable to obtain such data.

4.5.3.2 Program

The program for Phase III included the following:

1. Alpha counting of 38 stationary and 7 control elements in SM-1 Core II.
2. Determination of accuracy of the alpha counting technique by means of an irradiation test on a complete fuel element subassembly. The test will be performed at the Alco Products, Inc. Critical Facility in Schenectady. A suspected subassembly will be placed in a metal cylinder filled with water. The cylinder, in turn, will be placed in the core and irradiated at low power for approximately one hour. A radiochemical analysis for cesium-138 in the cylinder water gives an accurate indication of uranium-235 contamination.
3. Calculation of the fission product level in the SM-1 Core II primary coolant which can be sustained by the observed amount of alpha contamination.

4.5.3.3 Progress

All work on this phase of Task XVIII has been completed. A final report will be included as a section in the Phase II final report entitled "Review and Interpretation of SM-1 Core I Fission Product Activity."

4.5.3.4 Results

Alpha contamination on completed fuel element subassemblies was measured with a gas flow proportional counter supplied by Sylcor Corporation. The instrument consisted of two detectors connected to a common scaler. Each detector was placed against an outer surface of the outer plates of the subassembly. Consequently, only 5.6% of the 1592 fuel plate surfaces were actually counted.

626 060

The range of alpha contamination at the 95% confidence level for all SM-1 Core II subassemblies is given in Table 4-3. The wide ranges are due to poor counting statistics from low counting rates observed. The results indicate a range of cladding contamination of 0 - 0.09 micrograms (10^{-6} gms) uranium-235 per ft² of plate for 38 stationary elements and 0 - 0.20 micrograms uranium-235 per ft² of plate for 7 control elements. The maximum average level of contamination was calculated to be 0.11 micrograms U-235 per ft² of plate. This level can sustain a fission product concentration of approximately $6 \times 10^{-4} \mu\text{g/cc}$ in the SM-1 Core II primary coolant. The value is based on theoretical calculation and assumes that contamination is distributed uniformly in a surface volume which extends one alpha recoil distance (0.86×10^{-3} cm) below the stainless steel fuel plate surface. For any given amount of uranium, only 25% of the alpha particles reach a detecting device on the plate surface.

The lower power irradiation test was not performed because of the negligible alpha contamination. The maximum amount of alpha contamination found on any subassembly ($0.27 \mu\text{g U-235/ft}^2$ on 11v) would only give 54 dpm of cesium-138 at equilibrium. This was not considered sufficient for the experiment.

Radiochemical techniques for cesium-138 were reviewed in an effort to reduce the time required for analysis.

4.5.3.5 Conclusions

The fission product level in SM-1 Core II primary coolant, which can be sustained by the measured alpha contamination, is negligible in comparison to anticipated corrosion product activity. However, only 5.6% of all fuel plate surfaces were checked. A defective fuel plate could be present and not be found with this sampling plan. There may be considerable fission product activity (long-lived) from the SM-1 surfaces which are not decontaminated.

TABLE 4-3

**RANGE OF ALPHA CONTAMINATION OF SM-1
CORE II FUEL ELEMENT SUBASSEMBLIES**

<u>Element No. *</u>	<u>Range of Contamination $\mu\text{g U-235 per ft}^2$ of Cladding</u>	<u>Element No. *</u>	<u>Range of Contamination $\mu\text{g U-235 per ft}^2$ of Cladding</u>
2v (S)	0.0 - 0.20	28v (S)	0.0 - 0.12
4v (S)	0.0 - 0.20	29v (S)	0.0 - 0.13
5v (S)	0.0	30v (S)	0.0 - 0.08
6v (S)	0.0	31v (S)	0.0
7v (S)	0.0 - 0.18	32v (S)	0.0
9v (S)	0.0 - 0.17	33v (S)	0.0 - 0.14
10v (S)	0.0 - 0.20	34v (S)	0.0
11v (S)	0.04 - 0.27	35v (S)	0.0 - 0.11
12v (S)	0.0 - 0.17	36v (S)	0.0 - 0.14
13v (S)	0.0	37v (S)	0.0
14v (S)	0.0 - 0.18	38v (S)	0.0 - 0.16
15v (S)	0.0	39v (S)	0.0
16v (S)	0.0 - 0.16	40v (S)	0.0 - 0.15
17v (S)	0.0 - 0.13	41v (S)	0.0
18v (S)	0.0 - 0.16	43v (S)	0.0 - 0.12
19v (S)	0.0	CR-2-S (C)	0.0 - 0.23
20v (S)	0.0	CR-3-S (C)	0.0 - 0.20
21v (S)	0.0	CR-4-S (C)	0.0 - 0.21

* (S) - Stationary
(C) - Control

TABLE 4-3 (CONT'D)

RANGE OF ALPHA CONTAMINATION OF SM-1
CORE II FUEL ELEMENT SUBASSEMBLIES

<u>Element No. *</u>	<u>Range of Contamination g U-235 per ft² of Cladding</u>	<u>Element No. *</u>	<u>Range of Contamination g U-235 per ft² of Cladding</u>
22v (S)	0.0	CR-5-S (C)	0.0 - 0.13
23v (S)	0.0	CR-6-S (C)	0.0 - 0.18
24v (S)	0.0	CR-7-S (C)	0.0 - 0.22
25v (S)	0.0	CR-8-S (C)	0.0 - 0.24
27v (S)	0.0 - 0.19		

* (S) - Stationary
(C) - Control

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4.6 TASK XIX - ION EXCHANGE RADIOACTIVE WASTE DISPOSAL

4.6.1 Program

During this period the work on Phase II (final design of a waste disposal system for SM-1A) was stopped and an investigation of a by-pass sampling system was undertaken. The purpose of this system was to minimize the loss of primary water from the system. In addition to this, a back-up system was devised to handle large quantities of high activity waste, such as would result from a leak in the primary system. With the aid of dilution water, the suggested interim waste disposal system for the SM-1A provides a means to handle the radioactive liquid waste accumulated at this plant.

This system consists of a lab waste decay tank followed by a filter, three demineralizers in series, and a monitoring tank.

The laboratory and decontamination laundry waste is to be collected in the lab waste tanks which are already installed. The primary system leakage, primary blowdown diversion, and floor drains will be collected in the hot waste tank. Laboratory and laundry waste is expected to be directly dilutable and therefore should require no processing. A report describing this system has been completed and will be issued in July, 1960.

With the report on the interim waste disposal system for the SM-1A completed, attention has been shifted to the design of a skid-mounted system to process waste at the PM-2A. Permission to pump the waste to a disposal trench has been granted with the requirement that the waste is not higher than 10^{-3} μ c/cc and not more than 50 mc per year are released.

A flow diagram, Dwg. C9-47-1047, based on these conditions has been prepared and work is progressing on the design of a skidded unit to support the necessary components. The plan view of the skid, Dwg. C9-33-1032, will be the basis for structural calculation on the skid-mounted unit.

4.6.2 Progress

4.6.2.1 Waste Segregation

Not all waste accumulated at the PM-2A will be directly disposable into the disposal trench since the activity level is expected to be higher than the maximum allowable level of 10^{-3} c/cc resulting in a total accumulation of more than 50 mc per year. Therefore a system for segregating and treating that portion of the waste which exceeds these limits is being designed.

To segregate the waste, all high activity waste will be collected separately. The high activity waste includes the following sources:

1. O₂ analyzer effluent
2. Primary sample flushing
3. Primary floor drains
4. Primary blowdown diversion

This waste will be accumulated in the hot waste tank already provided.

The low activity waste is made up of the high solids waste which includes the radiochemistry lab waste and decontamination laundry waste. This waste will be stored in a 1000-gal tank.

Table 4-4 shows the estimated normal accumulation and activity from each source of waste.

TABLE 4-4

ESTIMATED NORMAL WASTE ACCUMULATION - PM-2A

<u>High Activity Waste (average)</u>		<u>Gal/Mo</u>	<u>μc/cc</u>
1.	O ₂ analyzer effluent	480	1.0
2.	Sampling flushing	35	1.0
3.	Leakage and diversion	65	1.0
4.	Floor drains from washing	50	1.0×10^{-2}
Total		630	.92
<u>Low Activity Waste (average)</u>			
1.	Lab Waste	170	2×10^{-3}
2.	Decontamination laundry & shower	100	5×10^{-5}
Total		270	1.3×10^{-3}

For design purposes, it is assumed that normal waste will accumulate for 11 months out of the year and that one month a year an abnormal volume of waste will be accumulated. The high activity waste for this one month would be 1130 gal, based on 600 gal of sampling waste and leakage instead of 100 gal.

The low activity waste will have 1170 gal for the one month of abnormal volume, based on a laundry waste volume of 1000 gal vs. 100 gal normally.

4.6.2.2 Method of Processing

The 5000-gal stainless steel hot waste tank will be used to accumulate all of the high activity waste. This is done to provide adequate decay for the waste and subsequent dilution of the incoming high activity waste. Two thousand gal will be accumulated before the first batch of this waste is transferred to the hot waste decay tank; a 270-gal stainless steel tank mounted on the waste treatment skid. The activity level in the hot waste tank at this time will be approximately 1.5×10^{-2} $\mu\text{c/cc}$ and may be determined by sampling after recirculating the hot waste with the hot waste tank pump. This allows a representative sample to be obtained.

The waste which has been transferred to the hot waste decay tank will be allowed to decay for at least 10 days. This permits the removal of all short-lived activity and results in an activity level of approximately 8×10^{-3} $\mu\text{c/cc}$ prior to processing.

Laboratory and laundry waste is expected to average 10^{-4} $\mu\text{c/cc}$ and is therefore directly disposable. However, it is necessary to accumulate and sample this waste prior to disposal, to be assured that the waste is below the maximum allowable level. It is also necessary to accumulate this waste so that the total activity discharged can be recorded. The total activity discharged from the plant must not exceed 50 mc per year.

On the basis of a yearly disposal of 4140 gal at 10^{-4} $\mu\text{c/cc}$, the laboratory and laundry waste activity will amount to only 1.6 mc per year, leaving a maximum of 48.4 mc to be discharged with the processed high activity waste.

The yearly accumulation of high activity waste, based on 630 gal/mo for 11 months and 1130 gal for one month, total 8.06×10^3 gal at an average

activity of $8 \times 10^{-3} \mu\text{c/cc}$ prior to disposal.

$$8.06 \times 10^3 \frac{\text{gal}}{\text{year}} \times 3.785 \times 10^3 \frac{\text{cc}}{\text{gal}} \times 8.0 \times 10^{-3} \mu\text{c/cc} =$$
$$2.44 \times 10^5 \mu\text{c/yr}$$
$$\text{or } 2.44 \times 10^2 \text{ mc/yr}$$

$244 \text{ mc} - 48.4 \text{ mc} = 195.6 \text{ mc/yr}$ must be removed by the waste processing system.

$4.84 \times 10^4 \mu\text{c}$ remaining after processing

$$6.08 \times 10^3 \text{ gal} \times 3.785 \times 10^3 \frac{\text{cc}}{\text{gal}} \text{ waste to be processed.}$$

$1.58 \times 10^{-3} \mu\text{c/cc}$ activity level required of waste after processing

Required decontamination factor is 5.06

$$\text{D F} = \frac{8.0 \times 10^{-3} \mu\text{c/cc prior to treatment}}{1.58 \times 10^{-3} \mu\text{c/cc after treatment}} = 5.06$$

but, since $1 \times 10^{-3} \mu\text{c/cc}$ is the maximum allowable discharge level, a DF of 8 is necessary. A system that can insure a DF of 10 or more appears adequate and can be accomplished by a series of two demineralizers and the necessary tanks required for collection and monitoring.

A skid-mounted waste disposal system for PM-2A is being designed which will provide a DF of at least 10 for all the high activity waste accumulated during a year. The unit consists of a skid approximately 8 ft by 12 ft, having the following major components:

1. - 1000 gal tank
2. - 270 gal tank
- 2 - canned rotor pumps
- 2 - shielded demineralizers
- 1 - resin fine filter
1. - rotometer

The flow diagram for this system is shown on Dwg. C9-47-1047 and a plan view of the skid is shown in Dwg. C9-33-1032.

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4.6.2.3 System Description - Lab and Laundry Waste

The Radiochemistry Laboratory waste and laundry waste is accumulated in a 1000-gal tank located on the waste disposal skid. During normal operation, this waste is expected to accumulate at less than 300 gal/mo and will have an activity level of about $10^{-4} \mu\text{c/cc}$. If the waste in the lab and laundry waste tank exceeds $5 \times 10^{-4} \mu\text{c/cc}$, it may be diluted to this activity level by water which is available for back-flushing the hot waste decay tank pump. It is not expected that this procedure will be necessary, since only coveralls and other garments having a 0.5 mr per hour* spot contamination will be laundered. Clothing having a higher dose rate will be disposed of as solid waste. By allowing the accumulation of about 500 gal of laundry and lab waste, decay should reduce the activity level to well below $5 \times 10^{-4} \mu\text{c/cc}$.

The lab and laundry waste may then be recirculated by the hot waste decay pump (Dwg. C9-47-1047). This permits a complete mixture of the waste in this tank prior to disposal. The hot waste decay pump may then be used to transfer the waste to the disposal trench.

* This value should be considered tentative until further investigation allows the selection of a final value.

NOTICE: This schematic diagram is intended to show the general arrangement of the system and is not to be used as a basis for construction. It is the responsibility of the user to verify the actual system configuration and to ensure that all components are properly installed and maintained. The user should consult the manufacturer's instructions for proper operation and maintenance.

ALCO		ALCO PRODUCTS, INC.	
NUCLEAR POWER ENGINEERING DEPT.		SCHENECTADY, N. Y., U. S. A.	
DATE	REV.	DR.	WILLIAMS 1-23-60
MATERIAL SPEC.		TR.	
		CHK.	9/1/60
		APP.	8-2-60
		APP.	
		REV.	
NAME WASTE PROCESSING SYSTEM			
PART NO. C9-47-1047

PLAN VIEW

UNLESS OTHERWISE SPECIFIED
DIMENSIONS ARE IN INCHES.
TOLERANCES ON FINISHED
POSITIONAL DIMENSIONS
TO BE:

- ✓ FINISH AS INDICATED
ON DIMENSIONS
- (1) MACHINE FINISH - 8000
- (10) FLAME CUT OR SAW

ALCO		ALCO PRODUCTS, INC.	
		NUCLEAR POWER ENGINEERING DEPT.	
		SCHENECTADY, N. Y., U. S. A.	
DATE	REV.	DR.	R.L. 8-3-60
MATERIAL SPEC.		TR.	
		CHK.	(20) 9/1/60
		APP.	J.H.L. 8-5-60
		APP.	
		NET	
NAME SKID ARR'GT - WASTE			
PROCESSING SYSTEM - PM-2A			
PART NO. C9-33-1032			

4. 6. 2. 4. Primary System Waste

All the high activity primary system waste is collected in the hot waste tank. By allowing 2000 gal of waste to accumulate prior to processing, a considerable reduction in activity level is possible because of the decay of old waste and subsequent dilution of the fresh waste. Although the activity level of the entering waste is expected to average nearly $1 \mu\text{c/cc}$, the waste in the hot waste tank should not exceed 1.5×10^{-2} anytime there is 2000 gal of normally accumulated waste in the tank.

The first step in the batch process is to recirculate the waste in the hot waste tank using the hot waste tank pump and then transfer about 250 gal of this waste to the hot waste decay tank.

In the hot waste decay tank, the hot waste is allowed to decay for about 250 hr, reducing the activity level of the waste to below $8 \times 10^{-3} \mu\text{c/cc}$. At the end of this period, the waste is processed through a rotometer, two demineralizers, a resin fine filter and into the monitoring tank. Before the waste is pumped to the disposal trench, a sample must be drawn from the monitoring tank and tested to be certain that the activity level is $1 \times 10^{-3} \mu\text{c/cc}$ or less. If a higher level than $1 \times 10^{-3} \mu\text{c/cc}$ is noted, the contents of the monitoring tank must be recirculated through one or both of the demineralizers until the maximum allowable discharge level is attained. It is not expected that recirculation will ever be necessary, since the decontamination factor demanded of the demineralizers is only 10.

4. 6. 2. 5 Instrumentation

Level indicators are required on all three of the skid-mounted tanks. These will indicate and alarm on the waste process room control panel.

The rotometer will allow the pumping rate to be adjusted to 2-5 gpm so as to prevent overloading of the demineralizers. The rotometer will have a range of 1-10 gpm.

A conductivity switch will actuate an alarm on the waste process room panel when the resistivity downstream of the first demineralizer falls below 1 megohm and a second alarm will be actuated when the resistivity upstream of the first demineralizer drops below 500,000 ohms. These settings may be changed to accommodate conditions as required. The downstream conductivity alarm will be provided to signal the exhaustion of the first demineralizer. The upstream conductivity alarm is to signal incorrect valve settings and the possible introduction of laundry waste to the demineralizers.

Differential pressure switches across each demineralizer and the resin fine filter will actuate alarms in the event that clogging occurs and results in excessive pressure drops.

Pressure indicators on each pump discharge will indicate remotely on the waste processing room panel.

Radiation monitoring for the waste processing skid will add two probes, indicating meters and accessories to the already installed PM-2A fixed radiation monitoring system. The two indicating meters servicing these probes will occupy channels VII and VIII in the radiation monitoring control unit located on the reactor console skid. The probe to indicating meter VII should be placed in the vicinity of the first demineralizer on the waste processing skid to be visible from the entrance of the room enclosing this skid. The probe to indicating meter VIII will be placed at the entrance to the hot waste building to be visible from the entrance to the area containing both the hot waste building and the waste processing building.

4.6.2.6 Recommended Location

It is recommended that the waste processing building be located at the end of the reactor and hot waste tank building tunnel. At this location it will be between the hot waste tank and the entrance of the proposed trench between this tunnel and the hot waste disposal trench (Dwg. A9-34-1014).

The waste processing building will be approximately 16 ft wide, 18 ft long, and 12 ft high and will have a 4 ft by 6 ft control panel and entry room.

A means of removing and replacing shielded demineralizers must be provided. Either removal through the roof by crane or an apparatus to lift the units onto a wheeled conveyance for transporting through the doors is possible.

The control panel and entry room allows operating personnel to monitor the waste disposal system in a low activity level area. The dose rate 1 meter away from the skid is expected to remain below 10 mr/hr and therefore allows the operator to adjust the necessary valves prior to and during the start of each batch process. The operator may remain at the monitoring panel to maintain close watch on the process.

ALCO

ALCO PRODUCTS, INC.
NUCLEAR POWER ENGINEERING DEPT.
SCHENECTADY, N. Y., U. S. A.

SCALE
 $\frac{1}{8"} = 1'-0"$

REV.

DR.

FB

7-29-60

MATERIAL SPEC.

TR.

CHK.

RLB

8-9-60

APPR.

RLB

8-2-60

APPR.

NET.

NAME RECOMMENDED LOCATION FOR
WASTE PROCESSING BLDG.
PM-2A TASK XIX

PART NO.

A-9-34-1014

5.0 TESTS

5.1 TEST SERIES 100 - PLANT CHEMISTRY

5.1.1 Test 104, Evaluation of Beckman In-Line Dissolved Oxygen Analyzer & pH Instrumentation

A report detailing the performance of the Beckman in-line instruments was issued on June 14, 1960 (1). The conclusions and recommendations made are summarized below.

Conclusions:

pH

1. The pH instrumentation performed in a satisfactory and reliable manner.
2. Maintenance requirements were low, averaging approximately 15 minutes per day.
3. Any change or fluctuation in plant operation affecting process stream pH was followed accurately by the pH instrumentation.
4. Continuous pH instrumentation can be presently used for process control at SM-1.

Dissolved Oxygen

1. The analyzer is a precision instrument that will measure the dissolved oxygen content of water as well or better than existing wet chemistry methods. However, because of the many difficulties experienced with the level control during the test period, the operation of the analyzer was unsatisfactory.
2. During analyzer operating periods, the plant variables affecting dissolved oxygen content of the hotwell were accurately followed, even when the change was as low as 1 ppb.
3. Normal maintenance requirements should average between 6 and 8 hr per week.

4. Sample water losses can become significant if a multiple sampling system is employed. At present, no means is readily available to return the sample water to the plant system, especially if one of the sample points is from the primary system.

Recommendations:

1. A cleanup purification system should be installed on the effluent stream from the pH flow chamber to remove chlorides that may be added from the reference electrode.
2. Additional instrumentation for continuously monitoring pH at SM-1 should be purchased and installed in the main stream, secondary blowdown, upstream and downstream of the primary demineralizer. The availability of these instruments will be beneficial in process control and research and development programs. For example, monitoring of pH around the steam generator will assist in performing material balances of the water treatment chemicals; also, any changes in the characteristics of chemicals affecting pH would be noted. Operational gains include more rigorous control of water conditions, chemical feed requirements, and indication of plant variables. A continuous pH measurement in the primary system will indicate such occurrences as air inleakage, system fluctuations, and demineralizer effectiveness.
3. Continued testing of the oxygen analyzer is required to verify the good performance obtained at the end of the test period and to determine the lowest possible sample flow rate consistent with accurate measurement.
4. A manifold distributor should be purchased for the oxygen analyzer to determine the performance characteristics.
5. For the present, it is not recommended that primary water be sampled by Beckman analyzer for dissolved oxygen content.
6. Associated with (5) above, an investigation should be undertaken to determine the best method for returning the analyzer effluent to the system. The purification methods investigated should include a return system for secondary water only and a cleanup and return system for both primary and secondary analyzer water.

5.1.2 Test 105, Cleanup of Suspended Crud in Inner Shield Tank

The purpose of this test is to establish procedures for maintaining clear, low activity water in the inner shield tank during work within the reactor vessel. Essentially, the method employed is to circulate the inner shield tank water through a filter and demineralizer. Preliminary evaluation of data obtained thus far indicates that the water can be maintained in a high purity, relatively low activity condition with the clean-up system in continuous operation.

5.1.3 Test 106, Pressurizing the Secondary System During Shutdown

The scope of this test was redirected by the Site Representative. A nitrogen blanket will be established in the steam generator and steam lines, prior to plant startup.

5.1.4 Test 109, Evaluation of Industrial Instruments Dissolved Oxygen Analyzer

Installation of the analyzer and component parts was completed. The analyzer will be tested during the first extended power run.

5.1.5 Test 110, Evaluation of Lapp Chemical Pump

A report on the performance of the Lapp pump was prepared. It was found that the pump was suitable for delivering water treatment chemical to the SM-1 secondary system for extended periods of time. However, the evaluation is considered incomplete since a controller to automate the operation of the pump was not available for testing.

5.2 TEST SERIES 200 - RADIOCHEMISTRY

Work continued on Tests 202, 203 and 204 in the Activity Buildup phase of Test Series 200. Tests 209 and 210 were completed. Work continued on Task XVI, SM-1 Decontamination, and Task XVIII, Analysis of Core I Activity Data.

5.2.1 Phase 1 - Activity Buildup, Core I

5.2.1.1 Summary of Work Accomplished

1. Two crud samples and water samples and an eight liter water sample from the primary system makeup tank were analyzed.

Experimental work has been completed on these samples and the data has been reported.

2. An investigation was made to determine the nature and origin of insoluble material in the filtered water samples taken from SM-1. The results indicate that the material is mainly iron.
3. Laboratory work has been completed on coupons sampled on November 27, 1959, and data has been reported on a number of these coupons.
4. Laboratory work is in progress on metal coupons and pipe samples removed on May 28, 1960.

5.2.1.2 Experimental Work Performed

Two crud samples, two water samples, and one water sample from the makeup tank were analyzed radiochemically for major long-lived gamma-emitting nuclides present in the SM-1 primary system coolant: Co^{60} , Co^{58} , Fe^{59} , Cr^{51} , and Mn^{54} . The data indicates that 61-62% of the total activity in the crud is due to Co^{60} and Co^{58} , while filtered water samples taken on the same date showed 72-75% of the total activity is due to Mn^{54} .

The crud samples were analyzed chemically for iron, cobalt, nickel, and manganese. The water samples for chemical analysis were analyzed for iron, cobalt, chromium, nickel, and manganese. The manganese comprised 0.27% and 0.29% of the crud, but the two filtered water samples contained 21.3% and 16.9% manganese. Iron, cobalt, and nickel were found in both crud and water. Chromium was found only in the filtered water sample.

Filtered water samples which contained no standard carriers were found to have a precipitate which was analyzed radiochemically for Co^{60} , Co^{58} , Fe^{59} , Cr^{51} and Mn^{54} . All of these nuclides were found. The material was analyzed chemically for iron, cobalt, manganese and nickel; only iron was detected. These results indicate that the precipitate probably is formed from particles, mainly iron and less than 0.45 microns in diameter, that pass through the Millipore filter. This precipitate will be considered as part of the soluble material when interpreting data under Test Procedure 203.

The precipitate present in the filtered water samples containing standard carriers, was radiochemically analyzed in a similar manner. This precipitate was found to consist mainly of iron, with small amounts of other standard carriers occluded. This precipitate can be redissolved by adjusting the pH of the solution.

Radiochemical and chemical analysis of the solutions obtained by dissolution of material removed from the following metal coupons has been completed:

<u>Upstream</u>		<u>Downstream</u>
S-58	C-130	I-1-11
S-60	C-165	I-1-22
S-74	C-167	I-1-6
I-1-23	C-113	A-1-21
F-1-31	C-116	A-1-36
F-1-20		F-1-34

These coupons were descaled by use of alkaline permanganate followed by a rinse with a citrate combination solution. The samples were analyzed radiochemically for Co^{60} , Co^{58} , Fe^{59} , Cr^{51} , and Mn^{54} . A chemical analysis for iron was done on all solutions.

The "S" series and "C" series coupons have a history of exposure to the primary system, followed by decontamination and reinsertion. The exceptions are C-165, C-166, and C-167, which were new, annealed specimens.

5.2.1.3 Future Work

Laboratory work will be completed June 30, 1960, on metal specimens obtained on May 23, 1960 from the upstream coupon holder:

S-81	I-2-8
F-1-30	N-1-12
C-146	A-2-12
I-1-14	A-2-10
M-1-10	UI
V0	

Results of the laboratory analysis will be tabulated and reported. Complete analysis of the data obtained under these tests will be performed under Task XVIII.

5.2.2 Phase 5 - Fission Product Studies, Core I

5.2.2.1 Introduction

The effort on Phase 5 during this period was restricted to the completion of Test 210.

5.2.2.2 Work During Period

1. Summary

Test 210, of Activity in the Primary System and Makeup Tank, was completed. Data are given in Table 5-1. The test report has been submitted⁽³⁾.

2. Results

The results of this test were discussed in the last quarterly report⁽⁸⁾. However, one erroneous statement was made. Strontium-90 was not detected in the makeup tank. Upon completion of the analysis, all of the strontium found in the makeup tank was Strontium-89.

TABLE 5-1

ACTIVITY LEVELS IN THE SM-1 PRIMARY COOLANT

Isotope	Date Sampled	dpm/ml Coolant Circulating	dpm/ml Tank Makeup	P/M*
I-131	2/17/60	1.7×10^4	< 10	$> 1.7 \times 10^3$
I-133	2/17/60	1.2×10^5	-----	-----
Cs-137	2/16/60	3.7×10^2	< 5	> 74
Cs-138	2/20/60	2.5×10^5	-----	-----
Bs-139	2/15/60	3.0×10^5	-----	-----
Ba-140	2/15/60	9.8×10^3	< 10	$> 9.8 \times 10^2$
Sr-91	2/23/60	7.5×10^4	-----	-----
Sr-90	2/16/60	35	< 1	> 35
Sr-89	2/16/60	2.1×10^3	27	78
Co-60	2/19/60	1.4×10^3	1.4×10^2	10
Co-58	2/19/60	1.5×10^3	40	38
Fe-59	2/19/60	5.3×10^2	20	27
Cr-51	2/19/60	90	5	18
Mn-54	2/19/60	9.4×10^3	50	190

* Activity in circulating coolant/activity in makeup tank.

5.2.2.3 Future Work

Test 207, Evaluation of Procedures for Determining Fission Product Iodine, will be run when the plant returns to power.

5.2.3 Phase 7 - SM-1 Decontamination

See Task XVI, Section 4.4.

5.2.4 Independent Tests

5.2.4.1 Introduction

Test 209, Determination of the Primary System Purification Constant, was completed with inconclusive results.

5.2.4.2 Work During Period

1. Summary

Test 209 was completed.

2. Results

The results of Test 209 were inconclusive, as discussed in the previous quarterly report. (8)

By using a value for the effective system volume for dissolved nuclides halfway between the limiting values, a purification constant within thirteen percent of the actual value (2) may be calculated.

5.2.4.3 Future Work

No work is planned at present.

5.3 TEST SERIES 300 - PHYSICS MEASUREMENTS

5.3.1 Tests 301 - 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. 178⁹; measurement from November, 1958, through June 30, 1959 - APAE Memo No. 206¹⁰; measurements taken after 12.1 MWYR - Test Report issued September 30, 1959¹¹; measurements taken after 13.5 MWYR - Test Report issued March 23, 1960¹². APAE Memo No. 178 includes a description of SM-1, experimental techniques developed for power reactor core measurements, and data recording and reduction methods.

On April 28, 1960, after approximately 16.43 MWYR of energy release, the SM-1 reached the defined end of core life. End of core life was defined as that time when during normal full power operation, equilibrium xenon, mean core temperature at approximately 440°F, all rods fully withdrawn, the outlet coolant temperature could not be maintained at approximately 450°F.

Data for Tests 301 through 316 were recorded. 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

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 variable for the calibration of a control rod with other rods at a stationary position. Peak xenon concentration occurred approximately 7.0 hr after power reduction. The rod position was the same as the equilibrium position indicating equal reactivity worth of xenon approximately 19.2 hr after power reduction.

5.3.3 Test 302 - Equilibrium Xenon

The negative reactivity introduced by xenon at full power equilibrium concentration was measured and the date is being reduced.

5.3.4 Test 303 - Five-Rod Bank Position; Peak Xenon, 370°F

The time of peak xenon concentration in the core after shutdown at 16.43 MWYR energy release was 7.0 hr determined from the transient xenon measurement at 370°F mean core temperature.

5.3.5 Test 304 - Five-Rod Bank Position; Equilibrium Xenon

The reactor was operated at full power for approximately 8 days to approach end of core life and an equilibrium concentration of xenon in the core. The critical position of the rods at end of core life (16.43 MWYR) were all 7 rods at 22.00 in. with the following conditions: full power equilibrium xenon concentration and mean core temperature at 440°F.

5.3.6 Test 305 Five-Rod Bank Position; No Xenon 370°F

The xenon in the core decayed for 60 hr following power operation; when the critical position of five-rod bank varied less than 0.05 in. in a 2-hr period it was assumed that xenon concentration in the core was at "no xenon concentration". The critical position of the five-rod bank was 15.67 in. with rods A and B at 19 in., no xenon, and a mean core temperature of 370°F.

5.3.7 Test 306 Five-rod Bank Position; No xenon, Room Temperature

The xenon in the core decayed for 60 hr 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 12.35 in. with 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. 5-1. The figure shows four curves, one each for the conditions of tests 303, 304, 305 and 306.

The curve for Test 303 was not extended to 16.43 MWYR because it was impossible to attain criticality with a mean coolant temperature of 440°F and peak xenon concentration.

5.3.8 Test 307 - Calibration of Rod A; Peak Xenon 370°F

Rod A was calibrated with peak xenon concentration in the core and a mean core temperature of 370°F. The rod was calibrated as a function of position versus the five-rod bank position.

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 is plotted as a function of rod position in Fig. 5-2. The calibration of Rod A was only accomplished for the top half of the rod due to limited available reactivity at that temperature.

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

Xenon in the core decayed for 60 hr following power operation, and power was reduced to a very low level and 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 and the rod worth in cents per inch plotted as a function of rod position in Fig. 5-3.

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

Xenon in the core decayed for 60 hr 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 is plotted as a function of rod position in Fig. 5-4.

5.3.12 Test 311 - Temperature Coefficient

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. 5-5. The reactivity change from 70°F to 415°F will be evaluated from bank motion and the bank calibration and the temperature coefficient calculated.

Fig. 5-1

Five Rod Bank Position as a Function of Energy Release

Fig. 5-2 Calibration of Rod A vs. Five Rod Bank Position. Burnup 16.43 MWYR, Temp. 440°F.

Fig. 5-3 Calibration of Rod A vs. Five Rod Bank Position. Burnup 16.43 MWYR, Temp. 120°F.

Fig. 5-4 Calibration of Rod C vs. Four Rod Bank Position.

Fig. 5-5 Five Rod Bank Position as a Function of Temperature.

5.3.13 Test 312 - Source Multiplication

The startup count rate and core reactivity was determined by insertion of a calibrated rod from the critical position. At 16.43 MWYR the count rate was 1.8 cps with all rods inserted, and 2.6 cps with rod A and B withdrawn to 19 in. and five-rod bank fully inserted. The count rates were taken 123 hr 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

The SM-1 Core I is defined as being spent when all rods are fully withdrawn at full load with equilibrium xenon concentration and the mean core temperature at 440°F. This occurred April 28, 1960, after approximately 16.43 MWYR of energy release. The change in reactivity at the end of life of Core I between the spent elements in the original configuration (Fig. 5-6) and after the elements were rearranged and the core spiked with one SM-2 and one PM-1M element was measured. This rearranged configuration is given in Fig. 5-7. Reduction of data recorded has been initiated. Preliminary results are reported here.

Rod A and Rods A and B as a bank were calibrated using the period method and the results shown in Figs. 5-8 and 5-9 respectively. The five rod bank was calibrated as a function of Rods A and B bank position. The reactivity of the rearranged, spiked core was taken directly from the integrated rod worth curves at their respective critical positions and corrected for temperature, 100°F to 68°F. The K_{eff} for the rearranged, spiked, cold, clean core is 1.096. This allows a cold, clean excess reactivity of approximately \$12.00.

Fig. 5-6 Original Core Arrangement

Fig. 5-7 Rearranged Core Configuration

616 091

Fig. 5-8 Rod A Worth as a Function of Rod Position -
(100°F Spiked Core)

Fig. 5-9 Rods A & B Bank Worth as a Function of Rod Position -
(100°F Spiked Core)

5.3.17 Test 318 - Gamma Scanning of Spent SM-1 Core I

The fuel burnout distribution in reactor fuel elements may be obtained by measuring the spatial distribution of gamma rays emitted by unsaturated fission products. During the report period, the design for a scanning device was completed, and the device fabricated, assembled and tested at the SM-1 site. Gamma ray energy distribution was obtained. After the energy distribution was established, a specific gamma ray energy was selected, and gamma ray intensity versus location obtained over the stationary elements of one quadrant of the SM-1 core. Difficulties in removing the control rod cap assemblies from the control rod basket caused termination of the scanning operations after fixed elements were scanned.

The scanner, basically, is a long shielded collimator tube equipped with removable inserts on each end. The inserts were fabricated with a range of aperture openings in order to regulate the gamma ray intensity at the detector. Fuel elements are driven past the lower end of the collimator with a hand-driven mechanical carriage. A mechanical readout system is attached to the carriage to indicate location of the fuel element. Details of the scanner design are shown in AEL-558 and AEL-563. Fig. 5-10 shows the gamma scanner in the spent fuel pit.

The detection equipment consists of a Baird Atomic model 815 BL probe, containing a 1-3/4 in. x 2 in. cylindrical N & L crystal optically sealed to a Dumont 6192 photo-multiplier tube followed by a cathode-follower type preamplifier. The signal from the preamplifier drives a strip-chart type spectrometer consisting of a linear amplifier, motor-driven differential pulse height analyzer, scaler, logarithmic count meter, and a strip chart recorder.

The gamma scanner was received at the SM-1 site on June 8, 1960 and assembled for testing by June 15, 1960. Several modifications to the assembly and the housing around the spent fuel pit were made and final assembly, testing, and scanner calibration completed on June 23.

Energy calibration of the spectrometer was by gamma ray source of known energy (Cs^{137} and Co^{60}). The calibration source was attached to the upper insert by means of a hollow tube and inserted into the collimator tube to obtain an energy calibration. Figure 5-11 illustrates the scintillation spectrum obtained from the 0.661 mev Cs^{137} source. It should be noted that the crystal was detecting gamma rays both from the Cs^{137} calibration source and from contaminated water in the spent fuel pit. The major peak shown in Fig. 5-11 is due to total absorption of the 0.661 mev isomeric transition gamma ray from Ba^{137} , the daughter product of the parent beta emitter Cs^{137} . The location of this peak was utilized to provide a conversion factor between the spectrometer base line voltage reading and the gamma ray energy, shown as the abscissa of Fig. 5-11 thru 5-15. Several such calibration spectra were run during the course of the experiment as verifying checks on the overall gain stability of the counting system.

GAMMA SCANNING APPARATUS IN THE SM-1 SPENT FUEL STORAGE PIT

Fig. 5-10 Gamma Scanning Apparatus, SM-1 Spent Fuel Storage Pit

Fig. 5-11 Scintillation Spectrum for Cs^{137} Source and Spent Fuel Pit Water

Fig. 5-12 Scintillation Spectrum for Spent Fuel Pit Water

Fig. 5-13 Scintillation Spectrum, SM-1 Fuel Element Structural Materials Plus Spent Fuel Pit Water

SM-1 ELEMENT #58

↓
□ X = OY = 10 6/27/60

BAIRD ATOMIC MODEL 815B1 SCINTILLATION
PROBE $1\frac{3}{4} \times 2$ " NA1 CRYSTAL 6192 DUMONT
PHOTO TUBE HV = 750 AMP GAIN = 2×74 SCAN
0.4 MEV WINDOW WIDTH 200 KEV

Fig. 5-14 Scintillation Spectrum for Fission Products Plus SM-1
Fuel Element Structural Materials and Spent Fuel Pit Water,
0-4 Mev

In order to identify the origin of each gamma ray present in intensities adequate to produce a unique scintillation, peak several gamma energy distribution measurements were made. These measurements include scintillation spectra of the spent fuel pit water, fuel element structural materials, and fission products.

The spent fuel pit water spectrum illustrated in Fig. 5-12 was obtained without a fuel element in the scanner carriage. Although the relative intensities do not justify quantitative analysis, significant peaks are apparent at energies of 0.2, 0.8, 1.1, and 1.3 mev. These energies are probably indicative of Fe^{59} , Co^{58} , and Co^{60} contaminants in the spent fuel pit water. These isotopes result from activation and subsequent corrosion of the fuel element structural materials and the Haines metal flux suppressors on the tip of the control rod fuel elements.

An energy distribution of the gamma rays emitted by fuel element structural materials was obtained from the steel just below the active core boundary of SM-1 element #79 (Fig. 5-13). The significant peak in the structural materials spectrums are identical to those found in the spent fuel pit water, and as background radiation in the energy calibration data presented in figure.

Having established the predominating peaks due to structural materials the scintillation spectrum for fission products plus structural materials was obtained. Fig. 5-14 illustrates this spectrum with predominant peaks at 1.55, 0.75 and 0.50 mev. A less significant peak was found at 0.15 mev. Only the 1.55 and the 0.75 new peaks were considered to have adequate intensity for purposes of scanning and subsequent analysis. A review of the literature indicated that the Zr^{95} - Nb^{95} and Ba^{140} - La^{140} pairs would account for the predominant peaks in the scintillation spectrum. The Zr^{95} - Nb^{95} pair is the source of the very strong 0.75 new peak. The parent product, Zr^{95} , with a 63.3 day half life, emits gamma rays of 0.754 and 0.722, followed by the Nb^{95} daughter with a 35 day half-life which emits a 0.764 mev gamma ray. Since the fission product yield of these materials is high, the net effect is an extremely potent source of 0.7 - 0.8 mev gamma rays for several months following reactor shutdown.

The Ba^{140} - La^{140} pair emits a multitude of gamma rays. The predominating energies emitted by the 12.8 day half-life Ba^{140} parent product are 0.53, 0.42, 0.16, and 0.03 mev, while the 40.2-hr half-life daughter product, La^{140} , emits 1.6, 0.82, and 0.486 mev gamma rays. In composite, the Ba^{140} and Zr^{95} spectra should show strong peaks in the regions of 1.6, 0.7 - 0.8, and 0.5 mev, and less significant peaks at 0.42, 0.16, and 0.03 mev. The identification established from Fig. 5-15 and other similar curves were considered adequate and the approximate energy interval, 0.7 to 0.8 mev, was

Fig. 5-15 Scintillation Spectrum for Fission Products Plus SM-1 Fuel Element Structural Materials and Spent Fuel Pit Water, 0-2 Mev

Fig. 5-16 Relative Gamma Ray Intensity vs. Axial Location

selected for scanning purposes for reasons of relative intensity and radioactive half-life. The fact that the La^{140} peak appeared at 1.55 mev rather than 1.60 simply represented a slight shift in the system gain in the time interval between energy calibration and scanning or a small error in the calibration factor. Considering the cumulative uncertainties in gain stability, low energy calibration factors applied to high energy gamma rays, and constancy of strip chart drive speeds, the error was not considered excessive and the procedures described below were established to prevent undetected gain shifts.

Figure 5-15 illustrates the scintillation spectrum over the 0-4 mev range. This scan was made with a window width of 0.2 mev in order to improve the counting statistics at the higher energy levels. The consequence of the increased window width, however, is some loss of resolution as evidenced by the virtual disappearance of the La^{140} peak. There were no significant new peaks found at the higher energies.

The stationary elements from one quadrant of the spent SM-1 Core I were scanned to obtain gamma ray intensity in the energy interval of 0.7 and 0.8 mev as a function of location on the fuel element. Since there was a very strong single peak in this interval, the spectrometer energy calibration was checked periodically by making small adjustments in the amplifier fine gain control to obtain a maximum count rate on the count rate meter, thus insuring that the discriminator window remained centered about the chosen scintillation peak.

Each fuel element was scanned along the centerline of both of the outside fuel plates and transversely at three axial elevations on the outside fuel plates. Fuel element orientation in the reactor core and gamma scanner carriage was established by periscope observations made as the elements were removed from the reactor core and again as the elements were mounted in the scanner carriage. The point of reference chosen for these observations was the dowel pin protruding from the lower fuel element end box.

The readout dial for the scanner carriage was adjusted to read zero when the upper nominal meat edge passed under the scanner collimator; hence the axial location shown in Fig. 5-16 indicates the distance below the top of the active reactor core.

Two scanning operations were performed on fuel element number 52 from the SM-1 lattice position number 15. These scans were made approximately one week apart in order to provide correction factors for the time behavior of the gamma ray activity.

Figure 5-16 presents the gross activity level in the approximate energy interval 0.7 - 0.8 mev versus location along element #52. It should be emphasized that these data are presented for illustrative purposes only. No quantitative significance should be placed on them until data reduction and analysis have been completed.

Analysis of this data is now in progress. Corrections to the data will include the Compton contribution from gamma rays of higher initial energy than 0.8 mev, correction for contributions due to structural materials, and photo contributions from isotopes other than Zr^{95} and Nb^{95} , such as Cs^{137} and La^{140} . The intent is to continue scanning the energy and intensity distribution in a single SM-1 Core I element in order to establish the time behavior of both the gamma ray energy and intensity distributions.

Since the Zr^{95} - Nb^{95} pair were essentially at saturation activity, they represent the power distribution in the reactor during the last few months of reactor operations rather than the integral effect of fuel burnout. It will be necessary to allow the intermediate half life products to decay in order to identify a longer half life product whose activity is in direct proportion to the total number of fissions in that fuel volume. Such a product is Cr^{137} , which emits a 0.661 mev gamma ray and has a 30 year half life. Cs^{137} counting was not permitted in these operations due to the very strong Zr^{95} - Nb^{95} activities which masked the Cr^{137} activity.

5.3.18 Test 319, Danger Coefficient Measurement (Coefficient of Reactivity)

A test to determine the coefficient of reactivity of a fresh SM-1 fuel element placed in several radial positions in the spent SM-1 core was performed May 13, 1960, during the end of core life experiments. Reduction of data recorded was initiated, and results are reported in Table 5-2.

TABLE 5-2
ASSOCIATED REACTIVITY CHANGE,
FRESH SM-1 FUEL ELEMENT IN SPENT CORE

	<u>Notes</u>	<u>Associated Reactivity</u> <u>Change in Cents</u>
1.	Fresh SM-1 element replaced spent element in grid position #62	+ 47.9
2.	Fresh SM-1 element replaced spent element in grid position #35	+ 94.7
3.	Fresh SM-1 element replaced spent element in grid position #16	+11.9

5.3.19 Test 320, Temperature Compensation for Xenon Override

Data for this test was recorded on April 19, 1960. At the start of the test, the reactor had been run at full power for 129.07 hr. Xenon was at an equilibrium concentration in the core and the mean core temperature was 440°F. Rods A and B were at 20.20 in. and the five-rod bank was at 20.70 inches. Energy release was 16.21 MWYR.

The data will be reduced and a final report will be written..

5.3.20 Test 321, Flux Mapping the SM-1 Spent, Rearranged and Spiked Cores

Flux mapping of the SM-1 cores has been completed. The data is now being reduced. This data will be correlated with the gamma scanning data, (Test 318), to determine the power distribution in the spent and spiked cores.

5.4 TEST SERIES 400 - SHIELDING MEASUREMENTS

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

The dose rate at the surface of the central shield tank above the core as a function of the position of a spent fuel element during the transfer operation was recorded and the results tabulated in Table 5-3. The distances were recorded as those from the surface of the central shield tank to the top of the meat section of the fuel element while suspended from the handling tool,

The data will be of value in optimizing the height of the water above the core and the height of the water in the spent fuel pit in future SM-1 type systems.

TABLE 5-3
DOSE RATE FROM SPENT
FUEL ELEMENT AT SURFACE OF SHIELD TANK

<u>Distance below</u> <u>water surface</u>	<u>Dose Rate (MR/HR)</u>
4 ft 6 in.	1200
5 ft	500
6 ft	150
7 ft	120
8 ft	80
9 ft	60
10 ft 2 in.	60
14 ft 6 in.	60

Note: Background at the surface was 60 mr/hr which gives rise to the 60 mr/hr readings from 14 ft 6 in. to 9 ft.

5.4.2 Test 409 - Dose Rates on the Spent Element Surfaces

Dose rate measurements on the spent fuel element surfaces were recorded and the preliminary results tabulated in Table 5-4. The purpose of this test was to obtain a measurement of the gamma dose rate at various positions on the surface of a spent fuel element. It is anticipated that these measurements will be used to normalize to the actual level the relative gamma activity as determined from the gamma scanning test (Test 318).

TABLE 5-4
DOSE RATES ON SPENT FUEL ELEMENT SURFACES

<u>Dose Rate</u> <u>r/hr</u>	<u>Detector Location</u>
1.4×10^4	Middle side plate - 1 ft away
4.9×10^4	Middle side plate - 5 in. away
5.4×10^4	Top of side plate - 2 in. away
1.4×10^4	Top of side plate - 1 ft away
1.2×10^4	Bottom end box - at contact
5.1×10^4	Bottom side plate - at contact
5.7×10^4	Bottom fuel plate - at contact
5.4×10^4	Middle fuel plate - 5 in. away

Note: Element #71 from position #66, 33 days after shutdown.

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5.5 TEST SERIES 500 - INSTRUMENTS AND CONTROLS

5.5.1 Test 500, Field Test of Minneapolis Honeywell BF₃ Pulse Transformer Channel

The final report of this test was written at the end of June, 1960, and will be released during the early part of the next quarter (July - September), to complete all work on the test.

5.5.2 Test 501, Field Test of Bendix Nuclear Instrumentation and Controls Channels

The technical difficulties reported in the preceding quarterly period have been resolved by the designer-manufacturer of the equipment. The equipment will be installed, calibrated and ready for testing in conjunction with the running of Task XII, as planned.

5.5.3 Test 503, Field Test of Westinghouse N¹⁶ Monitoring Equipment

During this report period, most of the remaining work required for checkout of the equipment outside the vapor container was completed. During the next quarter, the shields and other installations required inside the vapor container will be completed, with materials and equipment already on hand. The installation should be ready for initiation of the test in conjunction with Task XII and Test 501, with which its running has always been linked.

5.6 TEST SERIES 600 - HEAT TRANSFER AND FLOW

5.6.1 Test 600, Evaluation of Loss of Flow Accident

On the basis of several studies, it was predicted that the negative temperature coefficient would drive the reactor power down rapidly enough so that the SM-1 would not experience burnout for at least 5 sec after a primary coolant pump failure. Two of these studies implied that the reactor should be safe from burnout for a much longer time because of a rising burnout ratio.

The analysis of test data was completed and final report issued April 22, 1960. ⁽⁴⁾

The data shows that the negative temperature coefficient drives the reactor power down in a similar manner to that predicted for all power levels tested. The SM-1 is able to undergo a loss of the primary coolant pump without any detectable adverse affects for at least 15 sec after the pump fails.

Since the pump coastdown was found somewhat slower than that used in the analog model, it is believed that the minimum burnout ratio occurs somewhat later than predicted. Neither the magnitude of the minimum burnout ratio nor the exact time that it occurs could be determined from the experimental data because of a lack of bulk water temperature data.

It has been recommended that this test be rerun following the installation of Task XIV instrumentation. This would provide data on various temperatures in the core and steam generator as well as more accurate data on low coolant flows.

6.0 REACTOR ANALYSIS - SUPPORT WORK

6.1 INTRODUCTION

In the following paragraphs, a resume of the reactor analysis work which has been done in support of the SM-1 operation is discussed. Most of this work has been devoted to studying the nuclear and thermal consequences of rearranging and spiking SM-1 Core I.

The rearrangement and spiking of the SM-1 Core I involves placing the least burned up elements in the control portion of the core and the higher burnup elements in the outer positions ⁽¹³⁾. This results in an increase in reactivity and an extension of the life of Core I. Calculations of this increased reactivity and corresponding lengthening of core life have been made ⁽¹⁴⁾.

A steady state thermal analysis of the rearranged spiked core was conducted and reported in the last quarterly report. This study has been extended to include off-design and transient data. The effect of the step change load associated with Task XII transients experiments was investigated. In additions, the effect on the spiked elements of the loss of primary pump was studied and temperature distributions and burnout ratios at the power scram set point of 13 MW were calculated. As a check on the calculated power distributions for the rearranged and spiked core used in the above calculations, a comparison was made with data from experiments conducted at the Alco Critical Facility.

Other reactor analysis work on the SM-1 which was conducted during this quarter included an evaluation of the hazards associated with the rearranged spiked core.

6.2 PROGRAM

The transient behavior of the rearranged spiked core was determined through the use of the analog computer which utilized the lumped kinetic model described in APAE No. 38 ⁽¹⁵⁾.

Thermal analysis of the spiked elements at the off-design point of 13 MW operation was conducted with the aid of the ABWAC Code, which calculates temperature distribution through the element.

Burnout ratios during the pump failure problem were also calculated with the ABWAC Code for the IBM-650. Normalized average power and flow as determined from Test 600 ⁽⁴⁾ data was used as input. Experimental data was modified to account for maximum-to-average radial power variation and the effect of hot channel factors.

Increase in reactivity due to the spiked elements was determined by perturbation theory. The spiked, rearranged core lifetime was calculated by using the calculated reactivity change of the spiked, rearranged core and calculated SM-1 core lifetime data.

The power peak at the edge of an SM-2 element was found by comparing a radial power distribution calculation of the unspiked and spiked core. The radial power distribution calculation was performed with the spiked element in the central position of the core. The ratio of the spiked power distribution at the edge of the SM-2 element to the unspiked power distribution was used to scale up the unspiked power distribution at the point where the SM-2 element was placed. This spike must be further modified by the ratio of the SM-2 to SM-1 local peaking factors.

For comparison purposes with the calculated power peaking at the edge of the SM-2 element, experiments were conducted at the Alco Critical Facility. In these experiments, performed at zero power, an SM-2 element was placed in an SM-1 Core II mockup.

6.3 PROGRESS

(14) The analysis outlined in the previous section has been completed.
(16), (17) A complete summary of the conclusions of the studies is included in the following section.

6.4 CONCLUSIONS

6.4.1 13 MW Steady State Operation

At the power scram level of 13 MW, the minimum burnout ratio occurs in the hot channel of the SM-2 element in position 34. The minimum burnout ratio for an element in this position is predicted as 3.9. The minimum value for the burnout heat flux ratio at an off design condition for SM-2 is 1.5. Although it is realized that the burnout ratio predicted by the ABWAC Code is based on steady state conditions, transient effects should be small, since the Task XII transients are relatively slow. Since the final position selected for the SM-2 element is position 52, the burnout ratio will be higher than calculated for position 34 due to the lower power generation rate in position 25. Therefore, on the basis of calculations performed at 13 MW operation, burnout of the spiked elements will not occur during Task XII transient measurements.

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6.4.2 Analog Study of Response to Load Changes

The results of the analog simulation of the largest step transient imposed on the system (70% to 100% load change) are shown in Fig. 6-1. It is seen from the figure that the peak power during the transient is well below the scram set point of 13 MW. Since it has been shown above that the rearranged and spiked core can operate safely at 13 MW, the transients at lower peak powers called for in Task XII operation will present no problem as far as core safety is concerned.

6.4.3 Loss of Flow

The analysis of the pump failure problem in the rearranged and spiked core shows that the minimum burnout ratio occurs about 0.25 sec after loss of pump. The hot channel burnout ratio in the SM-2 element in position 34 decreases to 5.3 at this time, and the minimum burnout ratio in the hot channel of the PM-1M element in position 26 decreases to 6.9. This, once again, is well above the minimum of 1.5 postulated for safety during the transient phase of operation. Thus, loss of primary coolant pump should inflict no damages to the spiked rearranged core. As stated above, location of the SM-2 element in position 52 instead of position 34 makes the above analysis conservative.

6.4.4 Reactivity of Spiked Core

Although SM-2 elements have higher initial fuel loading than SM-1 elements, a higher boron content results in a new SM-2 element having less reactivity than a new SM-1 element. The K_{eff} resulting from replacing a 26% burnup element in position 26 by the PM-1-M element is $\Delta K_{eff} = +0.00365$; for an SM-2 element in position 52, $\Delta K_{eff} = -0.00096$. Hence, since rearranging the core results in a K_{eff} of $K_{eff} = 1.0175$, the final K_{eff} value of the spiked rearranged core is:

K_{eff} (rearranged core, eq. Xe and 440°F)	=	1.0175
(ΔK) SM-2 (position 52)	=	.00049
(ΔK) PM-1-M (position 26)	=	.00365
(ΔK) cc (Hot eq. Xe to cold no Xe)	=	.079
K_{eff} (spiked rearranged cold clean)	=	1.099

Fig. 6-1 Core Power After Step Load Change

At start of life, $K_{eff} = 1.182$. Since the rod worth remains essentially unchanged throughout core life, the hazards associated with spiking and rearranging the core are less than at start of life.

In the SM-2 element, the initial gain in reactivity due to boron burnup is greater than the reactivity loss due to the burnup of the fuel. Consequently, for the SM-2 element, reactivity initially increases with burnup.

Recent data on the SM-2 core shows an increase of K_{eff} over the first 2.5 MWYRS of burnup of $\Delta K_{eff} = 0.0040$ (uniform burnup of core). The average increase of one SM-2 element is $(0.0040) \times 1/45 = .00009$. This extremely small change in K_{eff} for the SM-2 element is negligible as far as hazards and lifetime calculations are concerned. Thus, the increased hazards due to spiking are negligible not only at startup of the spiked core, but also during core burnup.

6.4.5 Lifetime of Rearranged Spiked Core

Since the initial reactivity effect of spiking the core is negligible, the increase in core life comes from rearranging the core. This increase in core life may be predicted by reconstructing the experimental data of K_{eff} at equilibrium xenon and temperature as a function of core lifetime and assuming that an increase in K_{eff} at the end of life proportionately increases the lifetime. The values of K_{eff} (equilibrium xenon and temperature) are listed as a function of core life in Table 6-1 below.

TABLE 6-1

K_{eff} AS FUNCTION OF CORE LIFE

<u>Core Life (MWYRS)</u>	<u>K_{eff}</u>
12.1	1.034
13.5	1.023
16.43	1.000

Plotting the value of $K_{eff} = 1.02$ for the spiked rearranged core with the data of Table 6-1 and extrapolating (see Fig. 6-2), the increase in core life is seen to be 2.57 MWYR.

Fig. 6-2 K_{eff} Vs. Core Energy Release 440°F, Eq. Xe.

4.6 Comparison of Calculated Power Peaking in Spiked Elements with Experiment

The ratio of the spiked power distribution at the edge of the SM-2 element to the unspiked power distribution as determined by a multi-group diffusion calculation is 1.24. The ratio of the SM-2 to the SM-1 local peaking factors at position 25 is approximately $\frac{1.31}{1.2} = 1.09$. Hence, the final ratio of the spiked power distribution at the edge of an SM-2 element to the unspiked power distribution in position 52 is $1.09 \times 1.24 = 1.35$.

Experiments performed at the Alco Critical Facility for a spiked SM-2 element in SM-1 Core II mockup show the ratio of spiked to unspiked power generation to be in the range 1.34 to 1.40. This is in good agreement with the above calculated value of 1.35.

7.0 PLANT MODIFICATIONS

7.1 BLOWDOWN VALVE AND FLOW INSTRUMENTATION

The performance of the motor operated primary blowdown valve was found to be inconsistent with plant requirements. There were two main objections: first, the valve has a low lift plug which resulted in poor control when small increments of flow change were desired and second, it was not possible to decrease the flow rate below 0.5 gpm. It was decided that it would be more advantageous to purchase a new valve than to overhaul the present valve. Accordingly, specifications were prepared and sent to various valve vendors for quotation. On the basis of the replies, a Foxboro needle valve with an Askania operator was chosen.

Associated with the replacement of the present primary blowdown valve is the requirement to accurately measure the flow rate. The present flow instrumentation is installed downstream of the demineralizers. This point of installation results in an appreciable delay in indicating a flow change when the valve is actuated due to hydraulic resistance. In addition, the flow instrumentation is not accurate in the normal primary blowdown flow range. To correct these conditions, new instrumentation was ordered for installation immediately downstream of the primary blowdown valve. It is expected that the valve and flow instrumentation will be installed during the spiked core change.

7.2 EVAPORATOR CONTROLS AND INSTRUMENTATION

The plant has received the steam flow control valve, the steam flow indicator and the service water flow indicator and integrator. The valve is a 1-1/4 in. throttle type made by Edwards Valves, Inc. and will provide a means for controlling the steam flow when used in conjunction with the steam flow indicator. The service water flow indicator with integrator will be used to measure the quantity of makeup water into the evaporator and will eliminate the present method of estimating from the makeup rate to the distilled water tank.

7.3 WARNING HORNS

The installation of the warning horns at the SM-1 plant and Ponton Basin and the control circuit to ERDL whistle has been completed per plan.

The horns and whistle are actuated by manual operation of emergency type alarm switch located in the SM-1 control room. The alarm sound is on for a period of 3 min and is then de-energized with use of a timer.

The warning system is only for use in case of a nuclear incident.

7.4 VENTILATION IN ELECTRICAL EQUIPMENT ROOM

The fan, louvre, and thermostat have been received at the SM-1 plant for installation.

If the fan installation is adequate to remove the heat from the electrical equipment room, then the problem of the primary coolant pump breakers tripping due to excess temperature may be eliminated.

General Electric Company has been requested to furnish quotation for current trip-type breakers for the primary coolant pumps to replace the existing thermal trip-type breakers. No purchases will be made pending effect of exhaust fan installation.

7.5 ABSOLUTE PRESSURE RECORDER AND SENSING ELEMENT - VAPOR CONTAINER

A new model dynaformer with temperature compensation has been ordered. The existing dynaformer was affected by steam vapors when the primary coolant system thermocouple failed. At this time, the pressure recorder showed a decrease in pressure when the fact was known that the pressure increased.

7.6 CHEMISTRY LABORATORY

A new chemistry laboratory has been included in the building extension plan. The preparatory steps toward outfitting the new cold lab were initiated during this quarter. Bids from several vendors were received and were given a close examination. The vendor who was chosen (1) submitted the lowest bid, (2) supplies the same height furniture as presently used, (3) will supply the same exterior color finish, and (4) has furniture that matches the basic design of the existing furniture. The selected vendor was supplied with additional sketches and drawings from which final installation drawings are being prepared for our approval.

The addition of a new cold lab will ease the personnel congestion and radioactive contamination problem that has existed in the present facilities. All the drains in the hot lab have now been connected to the plant hot waste tank. The new hot lab island, which will replace the existing Kemrock topped island, will have a type 302, #4 finish, stainless steel top, curb, and ledge. The stainless steel top will be much easier to decontaminate than the present island top. Storage space, which has always been scarce in the single lab, will now be increased significantly with the addition of six wall cabinets.

REFERENCES *

1. Stein, R., "Final Report Test 104, Evaluation of Beckman Instruments pH and Dissolved Oxygen In-Line Instrumentation," June 14, 1960.
2. Hasse, R. A., "Final Report Test 209, Determination of Primary System Purification Constant," June 30, 1960.
3. Hasse, R. A., "Final Report Test 210, Activity Survey in the Primary System and Makeup Tank," June 10, 1960.
4. Morrison, J. H., Lee, D. H., "Final Report Test 600, Evaluation of Loss of Flow Accident in the SM-1," April, 1960.
5. APAE Memo No. 251, "SM-1 Research and Development Quarterly Report, October 1 to December 31, 1959," April 8, 1960.
6. APAE Memo No. 237, "SM-1 Research and Development Quarterly Report, July 1 to September 30, 1959," January 15, 1960.
7. Brown, W. S., et al., "SM-1 Research and Development Activity Buildup Program, Task I Final Report," APAE No. 51, August 10, 1959.
8. APAE Memo No. 264, "SM-1 Research and Development Quarterly Report, January 1 to March 31, 1960," July 6, 1960.
9. MacKay, S. D., et al., "SM-1 (APPR-1) Research and Development Program, Interim Report on Core Measurements - Task No. VII," APAE Memo No. 178, March 1, 1959.
10. MacKay, S. D., Tubbs, D. C., "SM-1 Research and Development Program, Interim Report No. 2 on Core Measurements - Task No. VII," APAE Memo No. 206, June 30, 1959.
11. MacKay, D. S., Tubbs, D. C., "SM-1 Research and Development Program, Test Report, Core Physics Measurements, Tests #301-316," September 30, 1959.
12. Tubbs, D. C., "SM-1 Research and Development Program, Test Report, Core Measurements at 13.5 MWYR, Tests 301-316," March 23, 1960.
13. Leibson, M. J., "Evaluation of Hazards Associated with SM-1 Core 1 Rearrangement," APAE Memo No. 225, September 30, 1959.

* Separate list of references given at end of Section 4, 3 for Task XIV.

REFERENCES * (CONT'D)

14. Birken, S.H., "Hazards Report, SM-1 Rearranged and Spiked Core," AP Note 255, Alco Products, Inc., May, 1960.
15. Brondel, J.O., "Plant Transient Analysis of the APPR-1 by Analog Computer Methods," APAE Memo No. 269, being issued (September, 1960).
16. Birken, S.R., "Kinetic Analysis of Rearranged and Spiked SM-1 Core I," AP Note 248, Alco Products, Inc., April, 1960.
17. Birken, S.H., Matthews, F.T., Lee, D.H., "Analysis of Rearranged and Spiked SM-1 Core I," AP Note 243, Alco Products, Inc., April, 1960.

* Separate list of references given at end of Section 4.3 for Task XIV.

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