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SM-1
RESEARCH AND DEVELOPMENT
QUARTERLY REPORT
JULY 1, 1959 TO SEPTEMBER 30, 1959

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ABSTRACT

This report covers the Research and Development conducted at the SM-1 under Contract AT(30-3)-326 during the period July 1 to September 30, 1959 including the evaluation and analysis effort developed at Schenectady, New York.

The report includes progress made in plant water chemistry and health physics practices, individual progress reports on tasks and tests, and a report on completed plant modifications or special projects on which engineering by the Research and Development staff has been required.

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SUMMARY

Water Chemistry. A survey of SM-1 water chemistry methods was initiated, covering (1) present water treatment chemicals, (2) chemical analysis, and (3) purification methods.

Certain questionable areas in the use of sodium sulfite were examined. A test run to measure pH levels with sulfite addition to the steam generator showed no decomposition of sulfite. After surveying sulfite feeding methods, it was concluded that considerable improvement in usage could be realized by reverting to a continuous feed system.

The present batch feeding of morpholine was examined. It was found that checking the pH once each shift was not adequate due to a six to ten hour decay cycle. Two or possibly three pH measurements per shift are required. It was further found that continuous feeding of morpholine is more desirable. This method will be adopted.

A review of analytical chemical procedures was made in chloride and oxygen analysis. Corrections were made to procedures to obtain more accurate results in chloride analysis. Further analytical work is believed desirable on the indigo-carmin technique for oxygen determination.

The primary system has been relatively free of water chemistry problems. An examination of the results of a test to determine the amounts of dissolved hydrogen as a function of the pressure and temperature of the gas blanket indicated that this method gives satisfactory results. A substantial reduction in the number of hydrogen analyses made in the plant is possible by using pressure-solubility relationships for hydrogen content control.

A future test will be made to evaluate the effect of a basic pH for reducing corrosion in the primary system. This will follow completion of work on identity and measurements of crud now in progress with the present neutral pH primary system.

Health Physics. Two possible technical overexposures were investigated during the period of this report: (1) Accumulative doses taken from direct reading pocket dosimeters indicated a possible radiation exposure of 340 mr to M/Sgt H. L. Allen on 16 July 1959. His film badge indicated an exposure of 290 mrem received in a period of five days; (2) Film badge processing reported that T/Sgt D. Sprey had received a dosage of 800 mrem for the two-week period ending 25 July 1959. A technical overexposure report was submitted.

The problem of radioactive waste retention has become more acute. An estimate of materials and cost of a proposed building for waste retention area has been submitted.

Task XI. A set of criteria for evaluating the controls and instrumentation was prepared. The SM-1 Operating and Maintenance logs were searched and organized to produce a record of instrumentation and control problem areas. An evaluation of performance was made against the criteria and tentative stop-gap remedial measures were established.

Task XII. The transient testing to be performed under Task XII will be done at a 0.8 pf. A resistance-reactance load bank has been ordered from the General Electric Company and will be delivered on about December 27, 1959. This equipment will permit testing at power factors of 0.8 and at unity. All major instrumentation items have been placed on order with the delivery of most items anticipated about December 1, 1959. A steam separator is to be installed in the main steam header ahead of the turbine; the order is scheduled to be placed during the month of November 1959. This equipment is considered necessary because of the danger of moisture entrainment in the throttle steam due the sudden severe step load transients which are to be imposed during these tests.

The originally proposed 30% capacity steam by-pass line has now been superseded by a 100% capacity steam dump line. Systems engineering of this by-pass line is essentially complete and orders are anticipated for placement during the month of November. This steam dump line system will require a desuperheater in the piping, plus additional desuperheater spray nozzles in the condenser. The dump line design will make possible the running of complete transient tests on the SM-1 with the turbine generator completely valved out of the system.

Installation engineering of the various equipment items to be used in Task XII is presently under way. The location of the resistance-reactance load bank skid unit has been placed in the open area beyond the fence opposite the spent fuel pit. Exact orientation of the skid has not yet been determined. The system for making electrical connection of this skid to the SM-1 electrical switchgear has been finalized.

A complete start up and test procedure has been written for use in the test work to be accomplished under Task XII, and is presently being reviewed.

The high pressure turbine components which will be necessary to withstand the higher throttle pressures which will be encountered during these tests have been placed on order with the General Electric Company and are

scheduled for delivery by December 21, 1959. A contract has been let to the General Electric Company to cover the supervision of the removal and installation of the new parts.

Task XIV. Various methods of instrumenting fuel plates are considered. Advantages, disadvantages, and problem areas of both a theoretical and practical nature are discussed with specific reference to an SM-1 core application.

After consideration of the methods presented, the conclusion is reached that an adaptation of the BISA technique originated by Bettis best meets the requirements and objectives of this task. Although the basic theoretical principles of the adaptation are similar to those of the Bettis plate, extensive metallurgical development must be undertaken to achieve a satisfactory design. The major problem is obtaining a strong bond between stainless steel and zirconium, with little additional contribution toward neutron absorption.

A high probability of obtaining this bond has been indicated by the introduction of a thin layer of a third material that is mutually bondable to both the stainless steel and the zirconium. Experimental efforts toward determining a satisfactory third material are being initiated. Related mechanical design and testing activities are proceeding on the assumption that a bonding technique will be achieved.

Some task activity is continuing on the instrumentation method of imbedding thermocouples in the fuel plate cladding. This latter method, as the original task concept, has been under considerable laboratory development, and is being continued on a small scale, (a) as a guard in case of major problems with the Bettis adaptation, and (b) in order to terminate past efforts at a more logical point as far as future applications are concerned.

Test Series 100. Preliminary tests for Test 101 indicate that morpholine should be fed directly to the hotwell rather than to the steam generator. Relatively large quantities of morpholine are required to control pH much above 9.0.

Samples of inner shield tank water were taken at each shutdown for chemical analysis to support any future metallographic examination which may be necessary on the pressure vessel studs.

Tests of this series during the period of this report were largely in evaluation and preliminary testing stage. Test 101 was started but had to be discontinued without conclusive results due to an unscheduled plant shutdown. The testing in plant will start with next full power run following the training cycle.

Test Series 200. Test procedures were prepared for sampling crud, water, metal coupons, and for the determination of long-lived dose rates. Samples were taken and analyzed.

Test procedures were prepared for fission product monitoring and for studying the behavior of I-131 and I-133 during startup and shutdown. An attempted startup test was terminated after 38 hours due to operating difficulties.

A review of fission product data collected to date indicates the fission product level at SM-1 increased by a factor of at least 100 between January and May 1958.

Test Series 300. Procedures for core physics experiments were prepared.

Physics experiments were performed on the SM-1 core at 12.1 MWYR of energy release. The experiments evaluated temperature coefficient, transient xenon, neutron source multiplication, and core reactivity.

The temperature coefficient was measured as $-3.6\text{¢}/^{\circ}\text{F}$ at 440°F . This value had been measured several times during core life, and no change with lifetime was indicated. The cold to hot reactivity change (70°F to 440°F) was 6.98 dollars.

The reactivity value of equilibrium xenon was determined as 3.02 dollars; peak xenon was determined as 4.47 dollars.

There is an estimated 4.6 dollars of core reactivity remaining to be used. The core life is expected to be 15 MWYR.

Test Series 400. During this quarter procedures were prepared for Tests 401, 402, 403, 404, and 408.

A study was performed to determine the consequences of a leak that developed in the water box that shields the control rod drive penetrations in the lower portion of the SM-1 pressure vessel. The results of this investigation are directly applicable to Tests 406 and 407. The significant conclusion that can be reached from this study is that most of the dose rate that is seen in the rod drive pit comes from the core and not from neutron activation within the pit. Although the neutron flux levels are high enough to cause significant activation, this activation is not a serious problem and will not interfere with proper maintenance of the drives even if the water box is run dry.

Test Series 500. Tests were initiated to determine the effectiveness of the pulse transformer in the Minneapolis Honeywell BF₃ start up channel and Bendix Aviation Corporation transistorized nuclear channels.

The test equipment for the pulse transformer test was installed and comparison start up charts were taken at each opportunity afforded by the plant operating schedule. The in-plant testing will run approximately three months. Progress on this test will be reported in a final test report, and is not included here.

The Bendix nuclear instrument test evaluation was completed. This test will be delayed pending delivery of recorders.

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1.0 INTRODUCTION

The R and D program at the SM-1 was broken down into specific and individual tests in the following categories:

- Test Series 100 - Plant Chemistry
- Test Series 200 - Radiochemistry
- Test Series 300 - Physics Measurements
- Test Series 400 - Shielding Measurements
- Test Series 500 - Instruments & Controls
- Test Series 600 - Heat Transfer & Flow
- Test Series 700 - System Tests
- Test Series 800 - Mechanical Components
- Test Series 900 - Health Physics

In addition to the above, work was continued on three active Tasks.

- Task XI - Instrument & Control Study
- Task XII - Plant Response & System Performance
- Task XIV - Temperature & Flow Measurements

The plant schedule during the period of the report precluded extensive scheduling of in-plant R & D programs. Core measurements were taken following the scheduled 60-day power run which ended on July 27, 1959. Following the core measurements, student training in operating procedures started on August 9, 1959 and ran through September 20, 1959. Test 500 for the evaluation of BF₃ Pulse Transformer was initiated in August and was run concurrently with the training program.

The period following the completion of "OP" training had been scheduled for Test 201 "Iodine behavior in the Primary System." This testing was interrupted on September 25, 1959 when the plant was shut down to collect data for checking a high flux level in the rod drive pit. Conclusive data for Test 201 was not obtained due to the plant shutdown. It was necessary to reschedule this test.

In the absence of an extensive in-plant program, the time was used productively in the preparation of test evaluations, specifications, and procedures. Considerable effort was expended in a survey and evaluation of present plant methods relating to water chemistry.

The final engineering requirements on the pressurizer cooldown line and the conversion to a helium gas blanket on the seal leakage tank were completed.

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2.0 PLANT WATER CHEMISTRY

The nature of the operations at the SM-1, with numerous shutdown and startup training cycles, have complicated the problems of water chemistry. It became obvious, particularly during the fiscal 1959 training, that methods required reevaluation. An SM-1 water chemistry survey was therefore initiated. Essentially, this survey covered three items: (1) Present water treatment chemicals, (2) Chemical analysis, and (3) purification methods. Since the particular requirements of the primary and secondary system are quite different, the aspects of water chemistry for each system are discussed below under individual headings.

2.1 SECONDARY SYSTEM

Since early 1958, chemical treatment of the SM-1 secondary system has relied upon the use of two chemicals, sodium sulfite and morpholine. Each chemical serves a distinct and separate function. The parameters associated with the usage of these chemicals were reviewed in an effort to gain information on methods for improving their chemical operational efficiency.

2.1.1 Sodium Sulfite

This chemical is generally accepted in the water treatment industry as being the most positive for eliminating dissolved oxygen. As used at the SM-1 it will scavenge oxygen quite rapidly at all water temperatures normally found. However there were certain questionable areas in its usage that needed to be resolved in order to take greater advantage of its potential. These were as follows:

1. Sodium sulfite is a strong electrolyte which markedly effects the conductivity of the steam generator blowdown. Advantage was taken of this fact by using conductivity limit as a set point for adding sulfite. However, this set point did not take into account the cumulative effect of other electrolytes that may be in the water during day to day operations, or a sudden increase of undesirable impurities may delay the decrease in conductivity until all sulfite protection is gone. Correction of these factors involved daily correlation by process control of sulfite residual vs. conductivity, and by forcing a sulfite feed every four hours.
2. Since sodium sulfite adds to the solids content of the secondary steam generator water, it is necessary to discard a comparatively greater portion of the water than desired. In order to

alleviate this situation a firm upper sulfite residual limit was set. This will permit lower blowdown rates if necessitated by short water supply.

3. Sodium sulfite is batch fed to the suction side of the boiler feed pumps. This method allows small quantities of air to enter the system. Although this problem cannot be eliminated as long as this method of feeding sulfite is used, it was reduced by requiring that the inlet to the suction hose always be submerged in the chemical solution. It was felt that considerable improvement in sulfite usage could be obtained by reverting to a continuous sulfite feed system which would permit oxygen scavenging to proceed in the feed water line.
4. It has been reported that sodium sulfite may undergo hydrolytic decomposition at the secondary pH levels maintained at the SM-1. A test to investigate this possibility by measuring the pH of the blowdown and steam before and after adding sulfite was initiated. No change in pH was observed. These results can be interpreted as an absence of sulfite decomposition.

2.1.2 Morpholine

The function of morpholine is to adjust the alkalinity of the secondary system, and to react with and tie up carbon dioxide. Since morpholine is a liquid, it is easily proportioned and readily fed to the system by continuous feeding methods. However, at the SM-1, morpholine was batch fed to the system through the suction side of the condensate recirculating pumps. This method was adopted due to the poor performance in the past of the chemical feed pumps.

Evaluation of the batch feeding methods revealed that 10 ml of morpholine was fed whenever the pH of the hotwell dropped to 8.5. When Process Control personnel were not on duty, it was the function of the equipment operator to measure pH once each shift. If the pH had dropped to the lower limit, morpholine was then added. To determine if this procedure was adequate a test was run to measure the hotwell pH from the time morpholine was added until the pH returned to the lower limit. It was found that the pH increased with time to about 8.9, then started to decay. The pH cycle varied between 6 and 10 hours regardless of whether the evaporator was on the line or not. This test showed that a pH measurement one per shift was not enough since the pH could fall well below 8.5 between measurements. Plant personnel were therefore directed to make at least two and possible three pH measurements each shift and add morpholine accordingly.

As part of Test 101, an evaluation was made of feeding morpholine continuously to the secondary system. It was found that this method of introducing morpholine to the process stream is entirely feasible, and this method will be adopted at the conclusion of the above test.

2. 1. 3 Evaporator

The evaporator is the principal source of make up water at the SM-1. Poor water produced at this point reflects in unsatisfactory water conditions throughout the secondary system. In many instances the production of poor water could be traced to extending the evaporator operation beyond its specification limit. This situation can be corrected by placing suitable instrumentation at the two critical operating points of the evaporator: the service water inlet and the process steam line. By bringing the service water flow and steam flow to the same set level each time, and correcting for scale buildup on the tubes, it should be possible to consistently produce good water. Accordingly, work to engineer the required instruments was initiated.

2. 1. 4 Chemical Analysis

A review of analytical chemical procedures reveals essentially two problem areas, namely, chloride and oxygen. The basic problem with the chloride analysis is the close titration required to obtain the endpoint. This analysis consists of titrating with mercuric nitrate until purple bubbles are observed. Since considerable effort is required of the chemist to titrate accurately, it is inexcusable to permit introduction of analytical errors by other controllable factors. The corrections applied to SM-1 procedures during this period were as follows:

1. Adjust all samples to a pH of 3.0 to 3.5. Samples for chloride analysis with pH ranging from 4.0 to 4.5 may be as much as 12% in error.
2. Plant Process Control personnel were instructed not to use de-mineralized water for a sample blank or chemical reagent unless the water was at least 10^6 ohm - cm. The purpose of the blank or reagent compensating sample is twofold: (1) the reagents may contribute to a delay in reaching the endpoint, and (2) the blank permits the chemist to titrate to an endpoint he can see. Obviously, the blank sample should not contain chloride to cause an abnormally high blank endpoint.

Present oxygen determination at the SM-1 involves the usage of a modified Winkler method. The primary disadvantage is that this method is time

consuming and requires analytical precision not normally developed by Plant Process Control personnel. Examination of other analytical techniques for oxygen determination revealed that the indigo-carmin technique offered the best possibility as a replacement technique. Accordingly, the information and materials necessary for testing the method were obtained. However, at this date no formal analytical work has been done.

2.2 PRIMARY SYSTEM

Excluding the crud and fission products formed in the primary system as a separate facet of water chemistry, it can be stated in general terms that the primary system has been relatively free of water chemistry problems. Purity control is maintained through a blowdown purification loop where a mixed bed demineralizer acts as both a filter and scavenger of electrolytic impurities, and by the addition of hydrogen which reacts with dissolved oxygen under the influence of a gamma flux field to produce water.

2.2.1 Demineralizers

The mixed bed demineralizers used in the SM-1 primary system blowdown purification loop produce essentially neutral pH effluent. A survey of literature available on experimentation with primary system at basic pH levels indicates that the corrosion can be materially reduced when compared with a neutral system. Examination of techniques for evaluating the effect of basic pH in the SM-1 system reveals that the best approach would be to use an ammonia cycle resin. Formal initiation of this work had been delayed until work on measurements of crud levels and their identity in the present neutral system have been completed.

2.2.2 Hydrogen

Approximately 15 to 30 cc of hydrogen is dissolved in each liter of primary water to promote the hydrogen - oxygen recombination reaction. The lower hydrogen limit provides sufficient hydrogen to insure the reaction will proceed. Concentrations of hydrogen as low as 10 cc per liter have been reported as being adequate. Control of the hydrogen content in these limits has never been any particular problem; however, in order to follow trends, it is desirable to know just how much hydrogen is dissolved in the water. The present analytical technique consists of a modified Orsat analysis where hydrogen is stripped from a sample of primary water and measured. This technique is not only time consuming and difficult to teach but is no more than 20% accurate. Captain S. R. Meeken of the Process Control Section proposed some time ago that the hydrogen content of the primary water be measured through the pressure and temperatures of the gas blanket in the primary make up tank. Accordingly, an experimental program was initiated at the SM-1 to measure hydrogen content in

the primary water by the use of these physical relations for comparison with analytical techniques. Hydrogen may be considered to dissolve in water according to Henry's law which states that the partial pressure of a component over a solution is proportional to its mole fraction in the liquid. Based on this relationship, the solubility of hydrogen is expressed graphically as a function of hydrogen pressure and water temperature in Fig. 2-1. To determine the effectiveness of the graph, the calculated solubility was compared with measured solubility. Comparisons are shown in Table 2-1. Orsat analysis for dissolved hydrogen in the primary make-up water was performed on the dates specified. Immediately prior to sampling, the gas blanket pressure and ambient temperatures were recorded. These pressures and temperatures were used in conjunction with Figure 2-1 to obtain calculated solubilities. A comparison of the calculated values with measured values can be readily noted in the Table. The average difference with all analyses considered is 2.67 cc per liter. Neglecting two of the largest deviations, the average difference between measured solubilities and calculated solubilities drops to 1.2 cc per liter. These results are considered to be excellent, especially when consideration is given to all the variables in the measurement. Some of these variables are listed below:

1. The physical relationships are based on hydrogen being the only gas in the gas blanket, and no consideration is given to the existence of nitrogen, oxygen, helium, fission product gases, etc.
2. Only a limited amount of solubility data is available for hydrogen at the pressures and temperatures of the SM-1 system.
3. The temperature measurements were obtained by reading a thermometer placed near the primary make-up tank. These were not true indications of the actual water conditions.

Even though the variables exist, it is felt that the physical relationships provide measurements which are sufficiently accurate that substantial reduction can be made in the number of hydrogen analyses performed weekly at the plant. Therefore, plant process control personnel were instructed to perform only one hydrogen analysis weekly, and to control hydrogen content of the primary make-up water by using the pressure-solubility relationships of Figure 2-1. However, it is pointed out that as long as hydrogen is used at the SM-1, the techniques of analyzing for hydrogen should be available, either by wet chemistry methods or by a hydrogen analyzer.

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Figure 2-1

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TABLE 2-1

HYDROGEN SOLUBILITY EVALUATION DATA

Date	<u>ccH₂ per liter H₂O</u>		<u>Temp. °F</u>	<u>Pressure, psig</u>	<u>Calculated Solubility,</u> <u>ccH₂ per liter H₂O</u>
	<u>Primary System</u>	<u>Primary Tank</u>	<u>Primary Tank</u>	<u>Primary Tank</u>	
1 June	29.4	43.3	66	22	44
3 June	26.6	29.8	76	12.5	30
5 June	27.4	33.6	91	18.2	33
8 June	20.5	20.7	94	17.2	31
10 June	25.5	24.2	95	17.0	31
12 June	29.9	31.1	90	15.2	30
15 June		33.04	78	15.0	32
16 June	24.8				
17 June		28.92	68	14.0	33
19 June	23.97	33.9	78	16.5	34
24 June		31.75	89	15.2	30

2.3 CHEMICAL LABORATORY

It has been recognized that the present SM-1 Laboratory is inadequate for the extensive work performed daily. The laboratory is used not only for routine analysis associated with plant chemistry studies, but for health physics (routine and experimental) and Process Control Training. It is difficult to produce work of the high quality desired, with many of these operations proceeding simultaneously. In addition, performing secondary chemistry analysis and radiochemistry investigation in the close quarters of the laboratory may lead to an unsafe condition.

To correct this condition, Alco proposed that 341 sq. ft. of laboratory space be added to the present SM-1 Laboratory facility. This would permit the laboratory to be divided into three sections: radiochemistry laboratory, chemistry laboratory, and counting room. Each section would be completely functional and provide for efficient and safe practices in the chemistry and health physics phases of plant research, development, and operation. Examples are given below of the work to be performed in each laboratory.

2.3.1 Radiochemistry Laboratory

All radiochemistry and routine analysis of primary and secondary water would be performed here. Also, all routine health physics measurements greater than 10^{-7} uc/cc would be made in this laboratory.

2.3.2 Chemistry Laboratory

This laboratory will serve several functions:

- (1) All secondary water chemistry studies;
- (2) routine plant and R & D health physics studies, provided activity is less than 10^{-7} uc/cc;
- (3) training of Process control students in actual plant operations.

2.3.3 Counting Instrumentation Room

For precise work in radiochemistry and health physics, it is necessary that a separate room be provided for activity counting. This room should be divided by a wall into two areas for further separation of the health physics and radiochemistry instrumentation.

3.0 HEALTH PHYSICS

Routine monitoring of the work areas is accomplished with portable survey instruments which provide dose rates for establishing working time limits. In addition, direct reading personnel dosimeters and film pockets provide cumulative exposure data. Radiation monitoring and record maintenance is the responsibility of SM-1 health physics personnel. In the past quarter the following incidents required investigation and reporting:

On 16 July 1959, while repairs were being made to correct leakage from a reactor outlet line thermocouple well, the film badge of M/Sgt H. L. Allen was sent to Lexington Signal Depot for evaluation of possible radiation exposure of 340 mr. The exposure was unauthorized and was "discovered" when cumulative doses reported on Radiation Work Permits were being totalized. The exposures reported on the Radiation Work Permit were obtained from direct reading pocket dosimeters. The film badge indicated an exposure of 290 mrem which was received in a period of five days. No further reporting was necessary.

On 31 July 1959 a technical overexposure was reported by Lexington through routine film badge processing. The film badge was worn by T/Sgt D. Sprey and indicated a dosage of 800 mrem for the two-week period ending 25 July 1959. The exposure, as recorded on Radiation Work Permit and determined through use of direct reading pocket dosimeters at the SM-1, was 240 mr. Through several explanations exist on the cause of overexposure as well as the discrepancy between the doses, there is no concrete evidence supporting these. Whole body counts and urinalysis showed no increases in body burden. A formal technical overexposure report was submitted as well as a more detailed and comprehensive investigation of the circumstances surrounding the incident.

3.1 WASTE RETENTION AND DISPOSAL

During this period the problem of radioactive waste retention has become more acute. The quantity of radioactive material has increased due to a more intensive research and development program. The increased amounts of radioactive materials and higher radiation intensities have made it necessary to expand the area so that the dose rate at the perimeter fence would not exceed 5 mrem/hour. An estimate of the material and construction costs for the proposed waste retention area was submitted to NPFO. Alco is awaiting approval and funding on the building which will be erected in the area now used for temporary storage.

Shipments of radioactive spent demineralizers were made to the U. S. Army Chemical Center at Edgewood, Maryland on August 11 and October 6. In both cases the activity content exceeded 2.7 millicuries and shipments had to be approved through the Bureau of Explosives.

A total of 8.22×10^{-3} curies of liquid radioactive waste was released to the Potomac River without adversely affecting biota and use of the water for human consumption. Routine health physics and chemistry analysis was responsible for only a small quantity of the waste. Decontamination of the vapor container after the primary system thermocouple well leak contributed to the majority of the waste.

3.2 HEALTH PHYSICS MANUAL

Approximately one-fourth of the time during the quarter was devoted to revising and implementing the SM-1 Health Physics Manual.

4.0 TASKS

4.1 TASK XI - INSTRUMENT AND CONTROL STUDY

4.1.1 Statement of Problem and Background

Task XI was initiated in late winter of 1958-59 following the original proposal by Alco Products, Inc., December 12, 1958, and authorization was received by letter from New York Operations Office, February 19, 1959. The basic problem originated from both the unreliability and the inconvenience of nuclear power plant controls and instrumentation in general, and those of the SM-1 in particular. A preliminary investigation, prior to the proposal, indicated that the controls and instrumentation of such plants limit the progress in the nuclear power field more than any other category of plant equipment. The general objective was to determine an optimum approach to the problem.

4.1.2 Description of Work During Period

During the period the R & D work consisted of:

1. Numerous interviews and communications with maintenance crews in the SM-1, operating personnel in the SM-1, personnel in the SM-1 training facility, manufacturers of instruments and controls, and the two consulting subcontractors.
2. Checking, organizing and evaluating SM-1 maintenance and operating logs.
3. Producing a set of criteria for evaluating the SM-1 controls and instrumentation and those of other nuclear (or fossil-fuel fired) power plants, chemical plants, refineries, etc. (This set of criteria is recommended for use in determining the nature of future programs in nuclear power plant controls and instrumentation.) It was necessary to produce the criteria, since no evaluating "yardstick" had been previously established for such controls, to make the study basis possible and practical.
4. The SM-1 controls and instrumentation were evaluated against the criteria and the log data analysis, etc. The shortcomings of the controls were established, as well as a tentative program for stop-gap remedial measures. Preliminary requirements were also established for the general change-over of SM-1 instrumentation.

During the same interval, the two subcontracting consultants completed their parallel studies. Preliminary reports were received from the consultants by the Alco R & D group.

In general, the reports and evaluations at the end of this period presented tentative rather than conclusive information to deal with the problem.

4.1.3 Future Work

During the next quarter the study efforts mentioned above will be completed. The final report of the study phases of the task will be released and distributed as authorized by the customer. The priority list of stop-gap remedial measures for the SM-1 controls and instrumentation will be produced. The study phases will be concluded, and the "hardware" phases will be initiated. During the remainder of the 1959-60 fiscal year the hardware will be placed on order. During 1960 a substantial portion of the stop-gap instrumentation can be installed.

4.1.4 Decisions

On the basis of the investigations to date, some tentative decisions regarding generalized causes of difficulties with the SM-1 controls and instrumentation are:

- a. In terms of the criteria now established, SM-1 controls and instrumentation are generally obsolete. There is shown to be a need for modifications to improve the controls to secure both inherent and human-engineering advantages. With modernization, maintenance and operating difficulties should greatly diminish.
- b. Facilities for testing future controls and instrumentation under various conditions, including environmental conditions, are needed at Fort Belvoir. An associated library is needed.
- c. The SM-1 (and other plant) controls and instrumentation should rapidly be brought to closely agree with the criteria and should follow the evolution as described in the criteria.

4.1.5 Evaluation

The Task XI study should produce results greatly needed in the field. Tentative conclusions and recommendations reached indicate that operations

and maintenance will be greatly enhanced in the SM-1 and similar plants. It is believed imperative that the recommended approaches be executed as soon as possible, to make the plants adequately reliable and reasonably convenient to operate and maintain.

4.6.1 Problems

The major problem areas, will likely prove to be:

- a. Nuclear controls and instrumentation (operational).
- b. Nuclear controls and instrumentation (health physics).
- c. Process controls and instrumentation

These problem areas will be discussed in the final report, to be released in the second quarter of fiscal year 1960. Recommendations will be on a stop-gap and redesign basis, with advantages of each given. The needed developments, facilities, and associated studies will also be included.

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4.2 TASK XII - PLANT RESPONSE AND SYSTEM PERFORMANCE

4.2.1 Objective

The objective is to collect transient response data from the SM-1 for use in the SM-2 and subsequent design programs. At present no adequate set of transient response data on a complete nuclear power plant is available. The data will be used to verify or revise the analytically derived analog computer models currently employed in the SM-2 transient analysis program. Methods of improving plant response to load transients will be indicated by these tests and applied to the SM-2 design. The exacting specifications for SM-2 power quality make this objective of primary importance.

4.2.2 Original Scope

The original scope required three complete, identical sets of transient runs, each set encompassing the full range of base load settings and step load changes. Each set was to be run in a separate subtask as follows:

Subtask 1 - Determination of SM-1 plant characteristics under transient load conditions for improved reliability and isolated operation. This required complete instrumentation of the plant, but no other plant changes. A full set of transient response data would be obtained from the following step transient (electrical) schedule:

Base Load	Step Transient		
0 %	+10%	+20%	+30%
10	+10	+20	+30
20	+10	+20	+30
30-70	+10	+20	+30
80	+10	+20	-30
90	+10	-20	-30
100	-10	-20	-30

Having a total of fifty-four (54) steps.

Subtask 2 - Operation of SM-1 without the main steam line pressure reducing valves. The pressure reducing valves were to be removed completely and the turbine-generator was to be altered to withstand maximum steam generator pressure at no load conditions. This subtask was to determine the effects of the pressure reducing valves on the overall plant response to transient loading. It was felt that these reducing valves were a major factor in the response characteristics of

the SM-1 pressurized water plant design, and they should be investigated separately with a series of tests identical to those outlined above in subtask 1.

Subtask 4 - Installation and testing of a steam dumpline with desuperheater as a control system for improving turbine generator response. This system would consist of a properly instrumented steam by-pass line around the turbine-generator, from the main steam header to the condenser. It was to be tested as a means of meeting the increasingly rigid power output quality specifications of present and future nuclear power plant designs for the Military. The test schedule was to be the same as that of Subtask 1.

4. 2. 3 Revised Scope

A re-examination of the program indicated that the greatest value could be derived in the limited time available by combining subtasks 1 and 2. The decision was based on the fact that sufficient data is presently available on the effects of the values and that the full set of transient runs scheduled under subtask 1 would be sufficient.

Subtask 4 was revised, at the request of the AEC and NPFO, to cover the design and testing of a 100% capacity steam bypass line as a means of field-checking newly installed plants, without dependence on the load absorbing capacity of the distribution network. The original design was a 30% by-pass system which was intended as a means of turbine generator control during transients. An outline of the more realistic revised scope for Task XII is:

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

- a. Purchase and install all instrumentation
- b. Install high pressure turbine and related parts
- c. Block open pressure reducing valves
- d. Run a complete set of transient tests

Subtask 3 - Installation and check-out of dummy electrical load at SM-1 for absorption of power during transient tests.

- a. Purchase and install dummy electrical load
- b. Shakedown and check runs

Subtask 4 - Installation and testing of a steam dump line with de-superheater.

- a. Install equipment including additional instrumentation
- b. Shake down and check runs
- c. Run a complete set of transient tests

4.2.4 Original Schedule

The original master schedule for Task XII, which started in February 1959, was programmed so that specifications, ordering, procurement lead time, and installation engineering would be complete by the end of September, 1959. The installation of equipment was scheduled for October, to be followed by two weeks of check out. Test runs were to be completed by January 30, 1960.

4.2.5 Revised Schedule

A new schedule that reflects a revised SM-1 operating schedule and core work now delays the completion of the test runs until the middle of 1960.

4.2.6 Progress

4.2.6.1 Subtask 1

- a. Instrumentation - This task requires the simultaneous collection of data throughout the system to detect plant response to transients in the order of 0.01 to 15 sec. duration. Since the present plant instrumentation is not designed for this type of experimental work, it is necessary to use high speed recording instruments. The design of the connections by which the parameter sensing equipment will be installed in the SM-1 plant is underway. The recording oscillographs that will record the history of each of these parameters during the proposed transient load changes will be located in the SM-1 control room. The use of the recording oscillographs on hand has been arranged. Delivery of most of the instrumentation items is anticipated about December 1, 1959. The original program listed 32 parameters to be detected and recorded to determine the response characteristics of components during load changes. The current list is parameters for read-out instrumentation is included in Table 4-1. This list reflects the changes brought about by the engineering effort as the instrumentation work progressed. It also includes some final adjustments relating to instrument accuracy and overall ranges.

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TABLE 4-1

LIST OF PARAMETERS TO BE DETECTED & RECORDED - TASK XII SCHEDULES

<u>Steam</u>	<u>Description</u>	<u>Location</u>	<u>Accuracy</u>	<u>Time Constant</u>	<u>Range</u>	<u>Signal-to-noise Ratio</u>
*1.1	Steam Flow	Main Steam Line	$\pm 5\%$	0.1 sec or less	Full Range 0-45,000 #/hr	
1.2	Steam Pressure	Inside Steam Generator shell above liquid level	$\pm 5\%$ of range	< 0.020 sec	150 psia-500 psia	
*1.3	Steam Temperature	Steam Generator outlet	Bare Thermocouple of fast resistance type	0.1 max.	300-500°F	
*1.4	Steam Pressure	Steam Generator at Outlet	$\pm 5\%$ of range	< 0.020 sec.	150-500 psia	
1.5	Steam Pressure	Pressure reducing valve inlet	$\pm 5\%$ of range	< 0.020 sec.	150-500 psia	
1.6	Steam Pressure	8-10 Pipe diameters downstream of pressure reducing valves	$\pm 5\%$ of range	< 0.020 sec.	150-500 psia	
1.7	Steam Pressure	8-10 Pipe diameters downstream of steam flow measuring orifice	$\pm 5\%$ of range	< 0.020 sec.	150-500 psia	
*1.8	Steam Pressure	Turbine inlet	$\pm 5\%$ of range	< 0.020 sec.	150-500 psia	
*1.9	Steam Temperature	Turbine inlet	As near as possible to position of 1.8	Bare thermocouple or fast resistance type	300-500°F	
*1.10	Position	Turbine speed control valve	$\pm 5\%$ over full range of travel	0.1 sec. or less	Full Travel	
*1.11	Steam Pressure	Just upstream of turbine blades. This will be within the turbine casing between the control valve plate and the first stage turbine nozzles	$\pm 5\%$ of range	< 0.020 sec.	150 psia -500 psia	
*1.12	Steam Temperature	Just upstream of turbine blades as close as possible to 1.11	Bare thermocouple or fast resistance type	0.1 max.	300-500°F	

TABLE 4-1 (CONT'D)

<u>Steam</u>	<u>Description</u>	<u>Location</u>	<u>Accuracy</u>	<u>Time Constant</u>	<u>Range</u>	<u>Signal-to-noise Ratio</u>
*1. 13	Steam Quality	In steam header as close as practical to turbine control valves. This will be a manual reading and recording and will not appear on the readout charts.				
*2. 1	Pressure	Turbine exhaust inlet to condenser	$\pm 5\%$ of range	< 0.020 sec	5-5" Hg absolute	
2. 2	Temperature	Condenser hotwell	Bare Thermocouple	1.0 sec.	90 - 160	
*2. 3	Hot Well Level		$\pm 5\%$	1 sec. or less	Full range (several inches)	
*3. 1	Position	Minimum flow valve.	$\pm 5\%$ over full range of travel	0.1 sec. or less	Full Travel	
3. 2	Temperature	Air ejector outlet	Bare Thermocouple or fast resistance type	0.1 sec.	90 - 160°F	
3. 3	Steam flow to air ejector		$\pm 5\%$	0.1 sec. or less	Full range 0-300 #/Hr	
*3. 4	Air Ejector Bypass Feedwater Flow		$\pm 5\%$	.1 sec. or less	0 - 10,000 #/Hr	
*4. 1	Feedwater Flow		$\pm 5\%$	0.1 sec. or less	Full Range 0 - 45,000 #/Hr	
4. 2	Pressure Drop	Across feedwater throttle valve	$\pm 5\%$ of range	< 0.020 sec.	0-200 psi	
4. 3	Pressure	8-10' Pipe diameters downstream of feedwater throttle valve	$\pm 5\%$ of range	< 0.020 sec.	150 psia - 500 psia	
*4. 4	Temperature	Feedwater, at feedwater heater exit nozzle	Bare thermocouple or fast resistance type	0.1 sec.	150 - 400°F	
*4. 5	Pressure	Feedwater pump discharge	$\pm 5\%$ of range	< 0.020 sec.	150 psia - 500 psia	

TABLE 4-1 (CONT'D)

<u>Steam</u>	<u>Description</u>	<u>Location</u>	<u>Accuracy</u>	<u>Time Constant</u>	<u>Range</u>	<u>Signal-to-noise Ratio</u>
*4. 6	Pressure	Feedwater heater shell side	+5% of range	< 0.020 sec.	0 psia - 50 psia	
*4. 7	Position	Feedwater throttle valve	+5% over full range of travel	0.1 sec. or less	Full Travel	
*4. 8	Level	Steam Generator	+5% of range	0.1 sec. or less	~17"	
4. 9	Steam Flow Signal	To three element controller	+5% of range	0.05 sec.	0 - 125% full load signal	
4. 10	Feedwater Flow Signal	To three element controller	+5% range	0.05 sec.	0 - 125% of full load signal	
4. 11	Steam Generator Level Control Signal	To three element controller	+5% Range	0.05 sec.	17 Inches equivalent	
4. 12	Feedwater Control Valve Control Signal	Output of these element controller	+5% Range	0.05 sec.	0-125% of Full load signal	
*5. 1	Generator Power Output		+2%	0.01 sec.	125% of full load current at 4160 volts	≥ 10
*5. 2	Generator Voltage	a. rectified b. unrectified	+1% over 800 volt range (4160 + 400)	0.01 sec or less		≥ 10
*5. 3	Generator Frequency		±.1%	0.01 sec. or less	90%-110% of full power RPM	≥ 10
5. 4	Exciter Output Voltage	input to comparator	±5%	0.01 sec.	0 - to exciter ceiling voltage 10%	≥ 10
5. 5	Exciter output current	Input to comparator	±5%	0.01 sec.	0-200% Full load current	≥ 10
5. 6	Exciter Voltage Error	Output of comparator	±1%	0.01 sec.	+400 to -400 Volts	10

TABLE 4-1(CONT'D)

<u>Steam</u>	<u>Description</u>	<u>Location</u>	<u>Accuracy</u>	<u>Time Constant</u>	<u>Range</u>	<u>Signal-to-noise Ratio</u>
5.7	Turbine Shaft RPM		$\pm 1\%$	0.01 sec. or less	0-125% of full RPM	≥ 10
5.8	Power Factor		$\pm 5-10\%$	0.01 sec.	.4 to 1.0 (estimate)	
5.9	Current	a. rectified b. unrectified	$\pm 1\%$	0.01 sec or less	0-450 AMP	
*6.1	Reactor Flow		$\pm 5\%$ of range	1.0 sec.	95-105%	
*6.2	Pressure	Primary System pressurizer	$\pm 5\%$	1.0 sec.	1000-1400 psia	
*6.3	Temperature	Reactor Inlet	$\pm 5\%$	1.0 sec.	400 - 500°F	
6.4	Temperature	Reactor Outlet	$\pm 5\%$	1 sec.	400 - 500°F	
*6.5	Temperature	Pressurizer liquid above heaters	$\pm 5\%$	1 sec.	500 - 600°F	
6.6	Temperature	Pressurizer Vapor	$\pm 5\%$	1 sec.	500 - 600°F	
*6.7	Level	Pressurizer Liquid	$\pm 5\%$	1 sec.	$\pm 10''$	
*6.8	Linear Power	Reactor output	$\pm 5\%$	1 sec.	0 - 125% Full Power	
*6.9	Log N		$\pm 5\%$	1 sec.	0 - 125% Full Power	

<ul style="list-style-type: none"> * Steam Generator Temperature - inlet * Condenser Cooling Water - inlet * Condenser Cooling Water - outlet * Position - main steam valve 	}	Deleted items
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* These items were included in the original scope prior to Task XII engineering effort.

Ordering of instrumentation was delayed where problem areas were not resolved. The main concern was that the number of parameters exceeded the number of points that could successfully be recorded on an existing recording oscillograph. Rather than duplicate transient tests, it was arranged to borrow a supplemental recording oscillograph from December 1, 1959 to March 1, 1960. The major portions of the instrumentation then were ordered.

An investigation of instrumentation required for the fast response of the electrical parameters (power, power factor and frequency) of the generator was conducted. "Off-the-shelf" items were found not suitable for transient response. As a result, specifications were written and invitations to bid sent to vendors. Quotations are pending receipt. The instruments sought will provide a more expedient data reduction effort, accurate instrument read-out of the parameters on the recorder, and a versatile package of instrumentation. The original schedule did not allow lead time for ordering these special meters. It would have been necessary to calculate the electrical parameters from the chart recordings of the three phases of voltage and current.

The instrument installation design has not progressed as anticipated since other phases of Task XII and R & D effort on SM-1 were expedited.

- b. High Pressure Turbine Parts - The transient response characteristics at SM-1 are believed largely limited by the presence of pressure reducing valves in the main steam line. Operation with the pressure reducing valves blocked open makes alteration of the SM-1 steam turbine necessary to permit operation at higher throttle steam pressures. The decision to block the valves open was based on pressure drop versus flow information received from Mason-Neilan Regulator Company. With the valves fully opened and at rated steam flow, information showed that only two psi pressure drop can be expected. The throttle steam pressure, however, will be higher because of loss of function of these pressure reducing valves. The present SM-1 steam turbine operates at a throttle pressure of 175 psi absolute. Under the new conditions with the valves blocked open, the throttle pressure at the no-load condition will be 425 psia.

The high pressure turbine parts were ordered from the General Electric Company June 11, 1959. The delivery schedule for these parts is:

	Week of
Nozzle plate, spacers and steam chest	11/16
Valves, stems and associated valve gear parts	11/23
Bucketed forged wheel	12/7
Trip throttle valve	12/21

On-the-site alterations will be done by labor supplied from SM-1. The General Electric Company will supply supervision. Arrangements for transportation of the rotor and other equipment requiring work by the General Electric Company at Baltimore, Maryland will be handled by Alco, Fort Belvoir personnel.

- c. Steam Separator - The possibility of moisture carryover during transients, particularly sudden load increases from low level base loads, required a steam separator in the line to protect the turbine. The turbine manufacturer has informed us that no carryover of moisture will be tolerated under warranties of equipment.

Quotations from various manufacturers revealed that no off-the-shelf items would fit in the present piping scheme. SM-1 requirements were reviewed and a second invitation to bid was sent out to vendors along with an installation sketch, for which quotations have not been received.

- d. Testing - A complete start-up and testing procedure to be followed when running Task XII tests has been compiled, and is being reviewed by the plant operations staff at Schenectady.

4.2.6.2 Subtask 3

The absence of special apparatus for controlled absorption of turbine generator electrical output constitutes a great inconvenience in tests at SM-1 requiring prescribed power levels.

Inquiries were sent to vendors to obtain a supplier for a load bank apparatus. Specifications called for a two-system quotation, one with a power factor provision and one with power factor equal to unity. The cost

information was to decide the final requirements of the load dump.

The General Electric Company (proposal #304-1-28402-A, alternate 2) was chosen as vendor for the load bank apparatus. Final approval to purchase the load bank apparatus was received from NYOO on June 23, 1959.

The apparatus is designed as a single structural unit having approximate overall dimensions of 29.6 ft. long by 9.7 ft. wide by 9.8 ft. high, and weighing about 38,000 lbs. The base loads and step load changes will be set and actuated respectively, by manipulation of two control switches on a remote control panel which will be located in the SM-1 control room.

Incorporation of power factor (resistance and reactance) in the load bank is necessary for true representation of the electrical load imposed on plants in the field. The degree to which a given power plant will respond to sudden load transients will vary, depending on the size of the reactive component in the electrical distribution which is absorbing the plant output.

The load bank can be set for operation at 0.97 power factor or 0.81 power factor or 0.81 power factor, depending on the type of electrical load simulation desired. The power rating of this equipment is 2500 kva at 0.8 power factor and 2000 kva at unity (0.97) power factor, and the voltage rating is 4160 volts. The loading units of resistances and reactances can be connected in various combinations to permit the desired base load settings and step load transients. The conversion from 0.8 power factor to 0.97 power factor can be made by manually disconnecting the reactances from each of the resistance-reactance unit combinations. The heat produced by the resistance units in the load bank will be dissipated by a large motor-driven electric fan which blows air across the finned resistance units and discharges it to the atmosphere.

The load bank unit is completely enclosed and of all-weather construction. It will be installed outdoors behind the SM-1 power plant building, opposite the spent fuel pit. The exact orientation of the unit from the standpoint of cooling air inlet and heated air exit has not been definitely established. We have requested information from the vendor concerning the temperature pattern around the exit of the cooling air duct. All wire leads from the SM-1 power plant to the electrical load absorbing bank have been arranged to connect at the air inlet end. This conforms with the general decision that the exit air must be discharged in a direction away from the plant.

The points of electrical connection of the load bank to the SM-1 switchgear have been determined. Connection of the high voltage 4160 lines will

be made at the power line pole on the hill leading up to the substation. This point of connection will permit use of breakers already installed in the ring bus electrical switchgear. It will also permit running of these transient tests with the SM-1 power plant completely isolated from the external distribution system with breaker "A" open. See Figure 4-1.

The approximately forty control and instrumentation leads and the 480 volt cooling fan power leads will be run from the SM-1 plant through conduit under the surface of the road. With these wires passing under the road, and with the high voltage power line connected to the existing power line pole on the hill outside the fenced enclosure of SM-1, it will not be necessary to install any overhead wires across the road. Interference with cranes and other equipment involved in refueling and spent fuel removal operations will be avoided.

The physical design of the electrical hookup systems for the load bank skid is underway. The army has been contacted for various stages of intallation (grading, pole installation and conduit installation under the road). A disconnect switch, not included in the original G. E. quotation, was requested and approved. The inclusion of the switch will provide a more convenient connection between the skid and the electric generator terminals since no special provision for intermediate switchgear will be necessary.

A meeting was held September 3 with G. E. representatives to discuss preliminary drawings and identify areas requiring clarification. Two important items discussed prevented approval of the drawings at the meeting. These were the resistance versus temperature coefficient of the load bank resistors, and the lag between the 10% and 20% step load contactors when actuating a step load change of 30%. A transfer switch was proposed to eliminate the 6% variation in resistance due to temperature coefficient during transient. This was not approved pending experimental proof that the problem is not serious. Approval was received for incorporating a 30% step load contactor.

Further investigation of the installation requirements indicated the need for auxiliary switches in the main disconnect switch. These switches are required for interlocking the load bank with the SM-1 system breakers. They have been approved. The incorporated switching (mechanical and electrical) will provide means for the following: (Refer to Figure 4-1.)

1. Permit closing of breakers "B" and "C" when breaker "A" is open and "Z" is closed,

Figure 4-1 SM-1 Electrical System Schematic
Showing Load Bank Connection

2. Prevent closing breaker "A" with breakers "B" and "C" or "Z" closed, and
3. Prevent opening and closing "Z" with breakers "A", "B" and "C" closed.

During transients the plant services will be fed through a construction tie in breaker "P". Breakers "F" and "G" will be blocked open in order that all the generator output will be fed to the load bank.

4. 2. 6. 3 Subtask 4

In the original proposal the steam dump line was described as a method of aiding the main steam throttle valve and turbine governing system in controlling turbine-generator RPM during severe load transients. The combination of main steam line volume, steam generator volume, turbine throttle and governing system, pressure reducing valves, etc., in a given secondary system design affects the load response characteristics of the system. The transient response characteristics of the plant may be independent of the components upstream of the throttle valve if a properly designed system-bypass is used. It should also permit reactor and steam system test work to be done.

The capacity of the steam dump line was not specified in the proposal because the requirement was not then known. In February 1959, the capacity of the steam dump line was set at 3% of rated plant steam flow, based on information from the Bell Laboratories, which is active in the design of the SM-2 distribution system.

Based on the original proposal and on the 30% step load transient requirement, the technical design of the steam dump line for Task XII proceeded throughout the summer. At a meeting with the Army and the AEC at Ft. Belvoir September 3, the decision was made to cease all work on a 30% steam dump line, and place the entire effort on the design and estimating of a 100% capacity steam dump line system. This 100% system will be tested under Task XII as a means of checking and initial testing of a newly installed nuclear power plant in the field. The advantage of the 100% steam dump line would be that the complete reactor system and steam generation system could be completely tested under all desired conditions of steady state or transient loading without the necessity of first installing the complete electrical distribution system. Tests could also be run before the turbine-generator is installed.

The system was analyzed and a general specification with sketch was prepared for submittal to vendors. We are awaiting receipt of quotations before expending further effort on this phase of the project.

The steam by-pass system requires that steam at turbine throttle conditions be expanded down to the low vacuum of the condenser. The equipment elements in the steam dump line system are shown in Figure 4-2. The steam dump line system must be capable of imposing step transient loadings on the reactor and steam generating systems identical to those to be imposed under the test schedule of sub-tasks #1. The setting of both control valves is accomplished by a remote operator's panel which is located in the control room. The setting of both valves can be adjusted manually to the desired base flow rate. For a step load increase the initial steady state base load is set on the large control valve while the small control valve is closed. The final desired setting of the small control valve is then preset from the operator's control panel. The step load transient is accomplished by tripping the small control valve open to the preset position. For step load decreases, the initial base flow setting is adjusted on both the large control valve and the small control valve. The step load decrease is then initiated by tripping the small control valve fully closed. The bypass system is intended to accomplish 30% step load increases and decreases and any desired step load changes below 30#. More severe step loadings can be imposed, however.

The first step in the design of a system of this nature is to determine the critical $\Delta P/P$ ratio. For this calculation the throttle conditions were assumed to correspond to rated plant output. The corresponding steam header pressure is 200 psia, and the condenser pressure was assumed to be 1.5 psia. The corresponding $\Delta P/P$ is 0.992. For the overall resistance coefficient of the steam dump line of 160, the critical $\Delta P/P$ is found to be 0.94. Under these conditions, only 188 psi pressure drop can be accomplished through the actual piping of the steam dump line system. Since the pressure drop from the header to the condenser must be 198.5 psi it is evident that critical flow conditions exist in the system. The major portion of the overall pressure drop is accomplished through the two parallel control valves. The pressure drop through the valve system is approximately 162 psi, and the exit steam velocity, after expanding into the down stream piping, is estimated at 57,000 ft. per minute, which is about one half the speed of sound, i. e., one half the critical velocity. This velocity is considerably reduced after passing through the desuperheater where water spray is added to reduce its temperature before the steam is introduced to the condenser. The velocity upon leaving the desuperheater is approximately 21000 ft. per minute.

The velocities and pressure drops described in the above paragraph are those associated with pipe sizes in general arrangement shown in Figure 4-2. Various other pipe sizes were examined carefully and

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100% Capacity Steam Dumpline
Fig. 4-2

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velocities computed during the course of the steam dump line design. It was found that the pipe sizes shown here gave the most favorable combination of pipe size and steam flow velocity. One of the big factors considered here is the noise level associated with the extremely high velocity of the steam at the low pressures and densities which accompany the expansion.

A major problem in determining the feasibility of the 100% capacity steam dump line was the method of introducing the steam to the condenser. Several conferences were held with the Lummus Heat Exchanger Division in Honesdale, Pa. to discuss the best way of accomplishing this in the installation at SM-1. When the 30% capacity steam dump line was being investigated Lummus felt that the baffle at the inlet of the existing smaller by-pass line at the SM-1 condenser would be adequate to handle the 10,500 lbs. per hour of steam involved. This meant that the 30% capacity steam dump line could be connected to the existing line, external to the condenser. With the change in emphasis to a 100% capacity system, further conference with the Lummus Division revealed that the 35,000 lbs. per hour steam flow associated with the 100% capacity line could not be handled by the existing condenser inlet arrangement and a special design would have to be made. Several alternatives were considered. One would bring the steam in the hot-well just above the water level and eject it upward toward the tubes of the condenser where it would be condensed and then drip back to the hot-well through the incoming steam, thus accomplishing, to a large degree, the desuperheating action required. This was ruled out because of internal space limitations as well as the high volume flow. Another system would bring the incoming by-pass line to a header which would distribute the flow around the condenser and inject it through the condenser wall above the tubes at 6 points. This was ruled out again because of external space limitations. The multiplicity of header pipes required with such a system could not be accommodated because of existing piping and support members. The final decision on the method of introducing the steam to the condenser calls for the incoming flow to be split into two branches before entering the condenser. Each of these branches would then penetrate the condenser wall and connect to a perforated header extending across the condenser from wall to wall above the tube level. The steam would then enter the condenser through the perforated holes in these headers. A spray system is required within the condenser to accomplish the final desuperheating action and to protect the last stages of the turbine wheels from the relatively high steam temperatures which might cause dangerous thermal stresses. Therefore, two spray nozzles are required just below the connection from the turbine exhaust hood to the condenser inlet. The water flow rate for these spray nozzles will be less than 5 gallons per minute.

The above discussion shows that two stages of desuperheating will be involved in the 100% dump line system. The basis for this can be seen by

tracing the flow of the steam from the header to the condenser and observing the changes in pressure and temperature which take place during this transition. At the steam header, the temperature of the steam is approximately 400°F. After expanding from the header pressure of 200 psia through the control valves to the down stream pressure of approximately 28 psia, the temperature of steam would drop from 400°F to about 340°F if no desuperheating were provided. After dropping the remaining approximately 28 psi from the exit of the valves to the condenser, the steam would arrive in the condenser at a temperature of approximately 330°F. This corresponds to a superheat of greater than 200°F. To avoid this high superheat entering the condenser it was considered necessary to install a desuperheater near the exit of the control valves. This desuperheater would then reduce the temperature of the steam from the 340° temperature at the exit of the valves down to about 260°F at the exit of the desuperheater. Direct expansion of this steam to the condenser would result in a superheat of about 150°F, and therefore it was decided to install the two water sprays in the condenser to accomplish this final desuperheating and also to serve as a protection to the exit stages of the turbine so that hot steam could not find its way up the exhaust and cause detrimental thermal stress conditions.

Vendors were asked to quote on the flow control valves, the desuperheater, and the control panel as a package. At present, one vendor has responded. This bid quotes a base figure for the complete package, which includes a variable orifice superheater. Accompanying this base quote is an alternate quotation, lower by \$1500, which includes a steam assist desuperheater. The base quotation, having the variable orifice desuperheater, is guaranteed by the manufacturer to control the exit temperature over the full range of 100% of rated flow down to 2% of rated flow. The steam assist type will control the temperature between 100% and 15% only. The step load transient schedule calls for testing at 10% load, and the necessity for the extra expense to obtain the variable orifice desuperheater is being evaluated. Of the three bids invited, one has been received as described above, one is not yet received but is expected in November, and the third has declined to participate. Delivery is quoted at ten weeks from the date of placing of the order.

The installation design of the steam dump line equipment is in its preliminary stages. The piping arrangement has not been finalized but studies have been made of the most advantageous way of arranging the piping at the approaches to the condenser. The complete steam dump line system installation cannot be finalized until the choice of vendor of the control equipment package has been made.

4.3 TASK XIV - TEMPERATURE AND FLOW MEASUREMENTS

4.3.1 Introduction

As part of the continuing SM-1 Research and Development Program, Task V was established to investigate the feasibility of in-core flow and temperature instrumentation with the objective of experimentally determining fuel plate temperatures and heat transfer film coefficient. This study was reported in APAE No. 37 dated September 1958. ⁽¹⁾ A development, design, fabrication and testing program to further these objectives was initiated in April 1959 under Task XIV.

As the design of pressurized water reactors has progressed, successively higher power densities have been demanded of the cores. Accurate fuel plate temperature predictions have become more and more critical as higher temperature levels are reached, and ignorance factors must be reduced.

One of the major obstacles toward better evaluation of plate temperatures is the determination of the heat transfer film coefficient. The most common method of calculation is the "Dittus-Boelter Correlation" ⁽²⁾ which is based upon flow rate and fluid properties at bulk temperature. However, experimental determinations of the coefficient have shown variations from this correlation by as much as -20%. ^{(3), (4)} Recent experimentation related specifically to pressurizer water reactor design by various investigators has partially reduced this figure, but not without some controversy. Some authorities in the field believe the maximum negative deviation for plate-type geometries will not be greater than 9% (use of 0.021 coefficient in the D-B Correlation) while others believe a more conservative 17% (0.019 coefficient) is advisable.

The advantages of experimentally measuring core temperatures and flow rates in the SM-1 reactor can be broadly classified into two areas. Although they are equally important, the former is much more difficult to attain, and the latter area must necessarily be a major influence in the justification of this task development program.

In the first area lie the advantages of more accurate prediction of fuel plate temperatures through better knowledge of the film coefficient, as applied specifically to SM-1 type cores. To experimentally determine this coefficient, measurements of relative flux levels, channel flows, total core flow, local bulk temperatures, fuel plate temperatures, total temperature base across the core, various core dimensions, and various fuel plate properties must be made with as high a degree of accuracy as practical. Because all these measurements each have an error band, the total error

band associated with the film will be difficult to keep within $\pm 10\%$. A detailed evaluation of this combined error band is strongly a function of the particular method used in instrumenting fuel plates, and will therefore be given in later reports.

In the second area lie the advantages of accurately knowing fuel plate temperatures after a reactor is operating. This information is based only upon one measurement; namely, plate temperature at a specified plate penetration. Adjustment of this temperature to other penetrations of the plate can be made with accurate analytical methods. Direct experimental observation of plate temperature to within a couple of degrees is practical.

4.3.2 Benefits of Task Instrumentation

Various phases of reactor design which would benefit by successful completion of the task instrumentation objectives are partially listed below:

1. Improvement or verification of analytical methods and model testing techniques involving the thermal and hydraulic design of SM-1 type reactors. Information on film coefficient can reduce temperature safety margins so that higher core powers are permissible without nucleate boiling. Element flow measurements can be correlated with data from model flow rig testing based on air. Deductions of actual neutron flux levels (see next paragraph) can be compared with predicted levels, and discrepancies evaluated in terms of "Hot spots."
2. If various individual film coefficient determinations are averaged, and this average is assumed to be the true coefficient, then the calculation process can be reversed through deductive reasoning to give information about relative flux levels. This technique would be especially useful in observing flux level profile variations with core age.
3. 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 (not formation) from UO_2 , and helium release from ZrB_2 .

4. 3. 3 Considerations in Instrumentation Design

In designing and developing Task XIV Instrumentation, the following considerations are kept in mind. Since all desirable qualities may not be met simultaneously by any one design, compromises must be reached by weighing of relative merits.

1. Accuracy of parameter involved (must be kept to the very minimum practical).
2. Probability of success, time required, and cost involved of any necessary specialized development programs.
3. Probability of instrument failure due to handling, fragile construction, etc.
4. Adaptability to permanent core instrumentation. Includes deterioration with age and replaceability of damaged or malfunctioning parts.
5. Ability to measure transients, reversing the versatility of the instrumentation.
6. Temperature measurements of instrumented plates representative of corresponding temperatures on an uninstrumented plate.
7. Minimum disturbance to core neutron flux levels.
8. Minimum introduction of new fabrication techniques and materials.

4. 3. 4 Methods of Instrumenting Fuel Plates

4. 3. 4. 1 Fuel Insertion Method

By this method, the thermocouple bead is inserted through the clad and into the center of the fuel meat. Entry is either through the side edge or the heat transfer surface. In the latter case, the leads must be at an angle to the surface so that the external lead does not create a water film disturbance in the proximity of the thermocouple location, thus increasing the local heat transfer film coefficient.

A cross-sectional view looking toward the side edge of an element is shown to scale in Figure 4-3, using 1 mil diameter wire coated with 1 mil thick insulation.

FIG. 4-3 - Fuel Insertion Method, Surface Entry

Advantages of this method are:

1. The thermocouple bead is located in a region of relatively small temperature gradients. Error from precision of placement and temperature gradient across bead are thereby minimized.
2. The basic fuel plate may be used in the fabrication of the instrumented plate assembly.
3. The instrumentation is suitable for the measurement of plate temperature transients.

Disadvantages of the method are:

1. Drastic miniaturization is necessary in order that temperature

measurement has any significance. Large perturbations are introduced in the normal flux level, heat generation rate, and heat flow patterns in the region of measurement by the thermocouple insertion. Even the 1 mil wire illustrated is too large for the accuracy objectives of this task.

2. Miniaturization of thermocouples to the extent required greatly increases the probability of early instrumentation failure.
3. Related problem areas are the achieving of adequate seals between meat and coolant to prevent fission product escape and between insulation and coolant to prevent water damage.

In view of the above disadvantages, this method is not considered to hold much promise, and is relegated to the "consider as a last resort" category.

4.3.4.2 Fuel Rolling Method

By this method, relatively large diameter sheathed thermocouples are inserted through the picture frame and into the meat by means of a drilled hole prior to the rolling manufacturing process. The inserts are parallel with the direction of rolling. Under continuous annealing conditions, a 14/1 diameter reduction would be achieved by the present Sylcor manufacturing practice as used in the SM-1/Core II. The proposed Batelle Memorial Institute fuel plate procedure would result in a 7.5/1 diameter reduction.

Oak Ridge is reported to have initiated development of such a method at one time. More information is being sought on their experience.

Advantages of this method are:

1. The thermocouple is located in a region of relatively small temperature gradients.
2. Except for the initial thermocouple insertion, the basic fuel plate fabrication procedures are followed for the instrumented plate.
3. Although the reduced size of the thermocouple renders it very fragile, no subsequent installation is required, and good support and protection are afforded.
4. The instrumentation is suitable for the measurement of plate temperature transients.

Disadvantages of the method are:

1. The diameter reduction ratio associated with the present Sylcor manufacturing procedure is more than twice as large as that presently attempted commercially by thermocouple suppliers doing pioneering work in miniaturization (one supplier contacted now offers 10 mil O. D. sheathed two-wire thermocouples).
2. Drastic miniaturization is necessary for temperature measurement to be significant. A reduced sheath diameter of 3 mils would be too large by accuracy considerations.
3. A rather costly development program would be required to achieve both structural integrity and miniaturization requirements with small prospect of success.

In view of the above disadvantages, this method is also deferred to the "consider as a last resort" category.

4. 3. 4. 3 Clad Imbedment Method

By this method, a thermocouple is imbedded with a groove cut in the clad surface. The groove is then filled in and the original surface contour re-established so that the heat transfer coefficient is not affected in the local area of measurement. The thermocouple could be either sheathed or have bare wire with sufficient insulation around the leads.

A side cross-sectional view of a 5 mil bare wire installation is shown to scale in Figure 4-4.

FIG. 4-4 - Clad Imbedment Method Using Bare Wire

Advantages of this method are:

1. The basic fuel plate may be used in the fabrication of the instrumented plate assembly.
2. The instrumentation is suitable for the measurement of plate temperature transients.

Disadvantages of the method are:

1. Additional clad thickness is required. Because stainless steel has a relatively high neutron cross-section, even the additional 7 mils of clad associated with Figure 4-4 introduces a large perturbation in the neutron flux level, as shown in Figure 4-5.
2. Thus the increased clad produces flux levels and temperatures in the region of the instrumented plate that are not representative of conditions in similarly located plates, or in the same location were a non-instrumented plate used. Although the heat transfer coefficient can still be determined, other benefits of the instrumentation are lost or reduced.
3. The instrumentation is located in a steep temperature gradient region (about $1^{\circ}\text{F}/\text{mil}$ maximum for the SM-1 and $3^{\circ}\text{F}/\text{mil}$ maximum for the SM-2). Thickness of the thermocouple bead in the direction of the gradient therefore produces significant error.
4. The groove and its imbedded thermocouple also introduce a local distortion of the heat flux path through the meat and clad in the area of measurement. To determine the relationship between heat generation rate and heat flux at a specific point on the clad surface, a one-dimensional analysis is normally sufficient. However, with the introduction of the imbedded instrumentation, a complex geometry three-dimensional analysis is required since the surface involved is partially blocked by the instrumentation itself.
5. In order to reduce these errors to a level compatible with task requirements, a miniaturized bare wire thermocouple installation is required. The failure probability of such instrumentation is relatively high.

In the initial stage of the task, no instrumentation method appeared more practical when considering task objectives, cost, development time and probability of achieving a successful fabrication procedure. Although there are severe disadvantages, other methods appeared worse. Laboratory

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FIG. 4-5
RELATIVE POWER DISTRIBUTION THROUGH
ONE SIDE OF THE SM-1 CORE

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development was therefore initiated on miniaturized instrumentation techniques using the clad imbedment method.

Satisfactory thermocouple beads and attachment of bead to stainless steel plate were attained using a capacitance welder. It was anticipated that a water insoluble cement would be found for use as a filler in the groove and as sealer at the entry to the externally mounted thermocouple sheath.

Various suppliers of high temperature insulation and cements were contacted for their recommendations. In general, there was little knowledge of the behavior of the materials in water at elevated temperatures, but all were known to be reliable in air at temperatures above those anticipated in the SM-1. Suggestions were received to test eleven materials for possible application. These are listed below in Table 4-2.

TABLE 4-2

HIGH TEMPERATURE CEMENTS TESTED

<u>Name</u>	<u>Type</u>
1) G. E. H-147 (glass frit)	Silicate Bonded Cermic
2) G. E. Phosphate Cement	Phosphate Bonded Cermic
3) G. E. RTV 20	Silicone Rubber, Filled
4) G. E. RTV 60	Silicone Rubber, Filled
5) Scotch Cast CRB #251	Epoxy Resin, Unfilled
6) Quigley AAA #1925	Ceramic (Strain Gage Cement)
7) Biggs #385	Epoxy Resin, Asbestos Filler
8) Pyroceram (glass frit)	Silicate Bonded Ceramic
*9) Saueriesen Cement #31	Ceramic, Acid Resistant
*10) Union Carbide formula, Alco prep.	Epoxy Resin, Unfilled
*11) Union Carbide formula, Alco prep.	Epoxy Resin, Unfilled

* Test coupons in preparation

With the exception of the last three materials listed, test coupons with these cements were prepared and exposed in an autoclave for two weeks by submersion in degassed high purity water at 600°F. All materials tested were found to have been dissolved or damaged upon subsequent inspection. The remaining three cements to be tested hold little promise since they are members of classes which did not pass the autoclave test.

Laboratory development is now being directed toward using a brazing alloy as the groove filler material. The thermocouple wire insulation would then be protected by metal from attack by the high temperature water. Another

possible technique is to fill the groove with a cement, coat the cement surface with graphite for conducting purposes, and then electroplate a protective layer over the area subject to attack.

Another method of fuel plate instrumentation has recently been devised, and is described in the following section. After extensive consultations and analysis in attempting to evaluate the probability of success in the problem areas that must be treated, it is felt that this recently devised method is preferred. Further task efforts are being concentrated in this direction, therefore, while some activity is continuing on the original task method of thermocouple imbedment (a) as a guard in case of major obstacles with the new method, and (b) in order to terminate past development effort at a more logical conclusion as far as future applications are concerned.

4.3.4.4 Blocked Channel Method

This method is an adaptation of the BISA (Base Instrumented Sub-Assembly) used by the Bettis Plant. The basic geometry and functional principles are the same. However, major metallurgical development must be undertaken to adapt the method to the SM-1 stainless steel type fuel plates. A general schematic diagram of the blocked channel method of instrumenting a plate is illustrated in Figure 4-6.

FIG. 4-6 - Schematic Diagram of Blocked Channel Method, Top View.

Advantages of this method are:

1. The thermocouple is located in a region of no temperature gradient.
2. No instrument miniaturization is necessary. Probability of instrumentation failure is therefore relatively low since 40 or 60 mil O. D. sheathed thermocouples can be used.
3. No complex geometry considerations are involved, such as holes in the meat or grooves in the cladding.
4. The plate temperatures of instrumented and un-instrumented plates in the same core location are the same.
5. Relative flux distribution in the core is only slightly affected by the introduction of an instrumented plate.

Disadvantages of the method are:

1. A major departure from the un-instrumented plate fabrication procedure is required.
2. A low neutron cross-section material is required for the filler. Metallurgical considerations on its compatibility with the stainless steel of the clad and fuel matrix may prove troublesome.
3. Response of the thermocouples to transients in meat temperature are not as rapid as can be achieved by the other instrumentation methods.
4. Slightly higher neutron levels in the core would be required when instrumented plates were used to compensate for the meat reduction.

In order to minimize instrumentation error and the effect of the instrumented plate on the general core flux levels, the filler material used must have a very low neutron cross-section. Zirconium is an excellent material for this application, and is used by Bettis. Since the cladding and meat matrix of the BISA plates are also of zirconium, no significant bonding problem between meat and filler arises. However, the SM-1 cladding and meat matrix are stainless steel (Type 304) and the bond between these dissimilar materials is not strong. Alternatives are (a) use another low cross-section material for the filler which will bond to stainless steel, or (b) introduce a thin layer of material (not necessarily of low cross-section)

which will bond to both the stainless steel and the zirconium. Strongly recommended for trial in a development program are yttrium and niobium (columbium) for part (a) with the additional possibility of nickel for part (b).

Macroscopic cross-sections of the various materials involved, along and in combination, are given below in Table 4-3. The combined cross-sections are computed using the homogeneous approximation method, and the relative proportions determined on the basis of 163 mil combined thickness.

TABLE 4-3
FILLER MATERIAL CROSS-SECTIONS

Material	Macroscopic Cross-Section, Cm ⁻¹
Ni	0.420
SS-304	0.222
Nb	0.060
Yt	0.048
Zr	0.008
$\frac{4}{163}$ Ni + $\frac{159}{163}$ Zr	0.018
$\frac{10}{163}$ Nb + $\frac{153}{163}$ Zr	0.011
$\frac{10}{163}$ Yt + $\frac{153}{163}$ Zr	0.010

As Table 4-3 shows, the use of two 5 mil layers of either niobium or yttrium between the stainless steel meat and the 153 mil zirconium filler introduces extremely little additional neutron absorption compared to the use of 163 mil zirconium filler alone. Thus a method of solution to the bonding problem presents itself without sacrificing experimental accuracy.

After weighing (a) the probability of success and the anticipated time required for the necessary development programs, (b) the experimental accuracy attainable, (c) the reliability of the instrumentation against failure, and (d) other consequences of minimizing disturbances to normal core neutron flux levels, it is strongly felt that the blocked channel method of instrumenting fuel plates offers the best solution to Task XIV objectives. Accordingly, continuing task activity is concentrating in this direction.

5.0. TESTS

5.1 TEST SERIES 100-PLANT CHEMISTRY

The purpose of test series 100 is to develop and evaluate techniques and water treatments for improving the water systems in the Army Pressurized Water Reactors. Methods for accomplishing this goal range from evaluating in-line chemical instrumentation to placing a filter in a recirculating system. The 100 series tests initiated during this report period are outlined below, and where applicable, a summary of the results obtained is included.

5.1.1 Test 100

The purpose of this test is to evaluate the effect of pH levels above 9.0 on corrosion of secondary system materials. The secondary system is pH controlled by morpholine. The effect of increased morpholine concentration associated with pH above 9.0 is not rigidly known at the SM-1. It is therefore necessary that corrosion parameters be evaluated before pH levels above 9.0 be specified for SM-1 secondary water. Since the primary objective of Test 101 is to increase and operate the SM-1 at a pH level of 9.5, it was decided that Test 100 could be evaluated in combination with Test 101 below.

5.1.2 Test 101

The objective of this test is to investigate chemistry conditions at various pH levels and eventually to evaluate water conditions at a pH level in the hotwell of 9.5. The normal pH in the hotwell varies between 8.5 to 9.0. Since it has been reported that possibility exists of sodium sulfite decomposing at these pH levels, it appears from the test requester's viewpoint that the pH should be increased to 9.5 thereby minimizing possible decomposition of sulfite.

Prior to initiating this test, a preliminary test was run to obtain the following information:

1. Chemical feed pump performance against a secondary head.
2. Characteristics of sodium sulfite as a feed solution.
3. Feed rates and concentrations of morpholine necessary to reach and maintain the required hotwell pH level.

The more important conclusions reached in the preliminary test are as follows:

1. The feed system will reliably reproduce feeding rates and should be suitable for test purposes.
2. Morpholine is not the chemical to use for pH control much above 9.0 because of the relatively large quantities required.
3. Morpholine is better fed directly to the hotwell rather than the steam generator.

The actual test was initiated, but due to an unscheduled plant shutdown was prematurely terminated without obtaining conclusive results. The test has been rescheduled for the next full power operation of seven days or longer duration.

5.1.3 Test 102

The purpose of this test is to obtain chemistry data of the inner shield tank water for possible correlation with corrosion of the reactor vessel studs. During the reactor head removal in March 1959, it was found that two holding studs had failed. Examination of the studs showed the presence of corrosion products. Since the impurity content of the inner shield tank water was not known, it was difficult to specify the exact nature of the corrosion attack. As a prerequisite for replacing the broken studs and the other holding studs, a metallographic examination had to verify the absence of all flaws. Periodic chemical analysis of the water will supply data for possible correlation with any irregularities that may be observed in future examinations of the studs.

5.1.4 Test 103

The SM-1 primary water pH is maintained essentially neutral by using a mixed bed demineralizer for purification. In work done at Westinghouse Corporation (Bettis), primary corrosion rate was decreased by operating at a high pH (9.0). It is desirable to make the same evaluation at the SM-1. This may be done by substituting the present SM-1 demineralizer for one in which the resins are in the ammonium cycle. This test will be deferred until crud and activity measurements are completed in the present neutral system.

5.1.5 Test 104

A Beckman dissolved oxygen analyzer and pH meter was purchased for evaluation at the SM-1. The planned use of these instruments will serve three functions: evaluation of performance for future installations, process control aid at SM-1, and training. The instruments will be set up initially to analyze the condenser hotwell water.

5.2 TEST SERIES 200 - RADIOCHEMISTRY

Test series 200 includes all work concerned with the buildup of radioactive nuclides. The work has been divided as follows:

- Phase 1 - Activity buildup, Core I
- Phase 2 - Activity buildup, Core II
- Phase 3 - Effect of temperature on crud deposition
- Phase 4 - Short-lived activity buildup
- Phase 5 - Fission product studies, Core I
- Phase 6 - Fission product studies, Core II
- Phase 7 - Decontamination of SM-1 steam generator

In addition, tests not directly connected with the phases above are initiated as the need arises.

5.2.1 Phase 1 - Activity Buildup, Core I

1. Introduction

The activity buildup program at SM-1 was initiated in February 1958. The purpose of this program is to characterize the nature of activity buildup, establish the mechanism by which it occurs, and predict future radiation levels. A final report covering all work through June 1959 has been issued as APAE No. 51. (5)

The isotopes responsible for essentially all of the long-lived dose rate on the primary system are Co^{60} , Co^{58} , Fe^{59} , Cr^{51} and Mn^{54} . For this reason, the activity buildup program has been limited primarily to the study of these isotopes in the primary system.

This program consists of three integral parts: the measurement of waterborne activity (soluble and insoluble), the measurement of corrosion and deposition rates and the activity of the deposits, and the measurement of dose rates on the primary system.

Three tests have been initiated to carry out the present program:

Test 202 -- Measurement of Long-Lived Dose Rates on the Primary System.

Test 203 -- Sampling of Primary System Water and Crud for Chemical and Radiochemical Analyses.

Test 204 -- Sampling of Metal Coupons and Pipe Samples for Chemical and Radiochemical Analyses.

A set of equations describing activity buildup has been developed. The various constants in these equations, i. e. corrosion rates, and purification rates, must be obtained from experimental data. This phase of the test program is designed to obtain these data and evaluate the equations in terms of the observed activity buildup for Core I.

Phase I will terminate with the removal of Core I. Another phase of the activity buildup program will commence with the installation of Core II.

2. Work During Report Period

- a. Test procedures were prepared for sampling crud, water and metal coupons, and for determination of long-lived dose rates.
- b. Four crud samples obtained May 15, 16 and 17 and two crud samples obtained July 3 and July 11 were analyzed. The data shows that the $\text{Co}^{58}/\text{Co}^{60}$ ratio was 0.50 in May 1959, and Co^{58} plus Co^{60} accounted for 74-79% of the total specific activity of the crud. See Table 5-1.
- c. One water sample, obtained July 11, was analyzed. The data shows that 83% of the total specific activity in the water is due to Mn^{54} , and that Co^{58} plus Co^{60} account for only 13%. The $\text{Co}^{58}/\text{Co}^{60}$ ratio was 0.31 in July 1959. This water sample was filtered through a Millipor Type WH filter and therefore represents what is considered to be soluble material.
- d. A chemical analysis on the crud samples indicated that iron represents 52-60% by weight of the ignited crud.
- e. Three coupons and one pipe sample were removed from the upstream and downstream sample points on August 3, 1959. Radiochemical and chemical analyses of the loose and descaled deposits were performed. The loose deposits were removed by shaking the specimens in deionized water. The firmly adherent deposits were removed by chemical descaling. The data indicates that there is no significant difference in the $\text{Co}^{58}/\text{Co}^{60}$ ratio in the loose deposits and the firmly adherent deposits. The time of sampling determines the $\text{Co}^{58}/\text{Co}^{60}$ distribution. A carbon steel coupon (A-212B) with 9498 exposure hours exhibited the same $\text{Co}^{58}/\text{Co}^{60}$ ratio as metal coupons with only 1365 exposure hours.

- f. The accurate determination of microgram quantities (less than ten micrograms) of cobalt and iron is now being studied. Procedures in use at KAPL and Westinghouse are being evaluated along with a modification of the procedures in use at the Alco Radiochemistry Laboratory. The possible chemical effects of the solutions used to dissolve the crud ($\text{HF} + \text{HClO}_4 + \text{HNO}_3$) on the chemical analysis are also being studied.
- g. Long-lived dose rates were taken during September. The data are reported in Table 5-2.

3. Results

The nuclide ratios listed in Table 5-1 reflect short term reactor operations rather than the long term trend in nuclide distribution. The samples taken during May 1959 were taken after an extended shutdown of about two months. Since the shorter-lived nuclides would undergo a relatively greater decay during this period than Co^{60} , the ratios are lower than would be expected had the reactor been operating at normal power levels. The $\text{Mn}^{54}/\text{Co}^{60}$ ratio is less affected than the others since the Mn^{54} half-life is much greater (310 days) than those of Co^{58} (72 days), Fe^{59} (45 days) and Cr^{51} (28 days). The decay of these shorter-lived activities is also demonstrated by the relatively lower specific activity of the crud.

The crud sample drawn July 11, 1959 was taken about one month after returning the reactor to normal operation. As expected, the specific activity and nuclide ratios increased. The nuclide ratios for this sample are similar to those observed in March 1959. The water sample drawn at the same time shows a $\text{Co}^{58}/\text{Co}^{60}$ ratio of 0.31. This is abnormal since the ratio usually approximates that in the crud. The other ratios compare with the crud ratios in the same manner as observed previously. (5)

Table 5-3 lists the radiochemical data obtained from the coupon samples. The data exhibit the characteristics as reported earlier. (5)

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TABLE 5-1
WATERBORNE CRUD

<u>Sample Date</u>	<u>EFPH</u>	<u>Total Activity dpm/mg</u>	(% Distribution of Activity)						(Nuclide Ratios)			
			<u>Co⁶⁰</u>	<u>Co⁵⁸</u>	<u>Fe⁵⁹</u>	<u>Cr⁵¹</u>	<u>Mn⁵⁴</u>	<u>%Fe</u>	<u>Co⁵⁸/Co⁶⁰</u>	<u>Fe⁵⁸/Co⁶⁰</u>	<u>Cr⁵¹/Co⁶⁰</u>	<u>Mn⁵⁴/Co⁶⁰</u>
5/15/59	8,964	16.1 x 10 ⁶	52	27	9.5	3.4	8.2	60	0.51	0.18	0.06	0.16
5/16/59	8,964	17.0	52	26	8.5	5.2	7.7	58	0.50	0.16	0.10	0.15
5/16/59	8,964	17.8	52	26	7.9	6.0	8.0	52	0.50	0.15	0.11	0.15
5/17/59	8,964	20.8	47	27	7.9	7.8	10	59	0.58	0.17	0.17	0.22
7/11/59	10,088	36.8	29	33	16	16	6.8	58	1.1	0.54	0.53	0.23
Water Sample		dpm/ml										
7/11/59	10,088	16.67 x 10 ³	10	3.1	1.5	1.7	83		0.31	0.15	0.17	8.2

TABLE 5-2

VAPOR CONTAINER SURVEY WITH
SHIELDED JORDON METER

Reactor Down: September 25, 1959

Date: September 28, 1959

Time:

<u>New Point No.</u>	<u>Old Point No.</u>	<u>Location</u>	<u>Dose Rate</u>
C	S. G. #3	S. G. above #2 outlet	95 mr
E	S. G. #5	Upper collar S. G. above steam line	460
F	S. G. #6	Upper collar S. G. above inlet	60
5	5	Pressurizer elbow, bottom	125
I	-	S. G. #2 outlet, bottom	50
J	-	S. G. #2 outlet, top	80
21	21	S. G. #1 outlet, bottom	50
M	-	Reactor inlet pipe, bottom	120
N	-	Reactor inlet pipe, top	150
3	3	Side of S. G.	360
6	6	Reactor outlet	125
9	9	Reactor inlet on slant line	80
10	10	Flow tube, right	1R
13	13	S. G. on upper ladder	50 mr
19	19	S. G. , 8' 5" from #1 outlet	50
20	20	Inlet to S. G. , top	800
23	23	Drain plug on bottom of S. G.	13
24	24	Bottom of B. D. Cooler	50
25	25	4' from bottom of B. D. Cooler	50

TABLE 5-3
COUPON SAMPLES (8-3-59)

			(Percent Distribution)					(Nuclide Ratios)			
Material	Sample No.	%Fe in Descale	Co ⁶⁰	Co ⁵⁸	Fe ⁵⁹	Cr ⁵¹	Mn ⁵⁴	Co ⁵⁸ /Co ⁶⁰	Fe ⁵⁹ /Co ⁶⁰	Cr ⁵¹ /Co ⁶⁰	Mn ⁵⁴ /Co ⁶⁰
Upstream											
304 SS	A-1-49 L*		32	24	6.1	25	13	.75	.19	.80	.41
	A-1-49 D*	55.7	26	19	3.6	45	6.1	.74	.14	1.8	.24
A-212 B CS	C-1-27 L		27	23	3.5	25	21	.87	.13	.93	.79
	C-1-27 D	76.7	28	22	6.6	3.6	6.9	.78	.23	1.3	.24
Inconel	I -1-15 L		33	11	6.6	40	8.9	.32	.20	1.2	.27
	I -1-15 D	Erratic	17	12	2.0	60	8.1	.70	.12	3.4	.46
304 SS Pipe	UQ L		36	23	5.8	26	8.2	.64	.16	.72	.23
	UQ D	68.4	37	22	4.6	29	7.7	.60	.13	.78	.21
Downstream											
304 SS	A-1-33 L	No Analysis	34	26	9.1	16	14	.77	.27	.46	.42
	A-1-33 D		37	20	13	19	12	.54	.35	.51	.32
Inconel	I -1-16 L		32	26	72	22	7.7	.79	.38	.69	.45
	I -1-16 D	44.3	36	18	13	23	10	.51	.35	.64	.29
A-212B CS	C-1-11 L		23	14	4.4	13	45	.63	.19	.56	2.0
	C-1-11 D	43.6	32	13	4.1	20	30	.39	.13	.62	.93
304 SS Pipe	DM L		41	24	9.8	15	10	.59	.24	.36	.25
	DM D	58.4	42	21	10	15	12	.49	.25	.36	.28

* L - Loose deposits

D - Descaled deposits

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4. Future Work

Work under Phase 1 will continue to the end of Core I life. At this point Phases 2 and 3 will be initiated. The remainder of the work under Phase 1 will consist of additional analyses of crud, water, coupon and pipe samples, and the determination of dose rates. Maximum use will be made of the extended full power run scheduled from the termination of the training cycle to Core I burnout.

5.2.2 Phase 2 - Activity Buildup, Core II

Phase 3 - Effect of Temperature on Crud Deposition

Phase 4 - Short-Lived Activity Buildup

Work on these phases will start with the installation of Core II. No effort has been made during this quarter.

5.2.3 Phase 5 - Fission Product Studies, Core I

1. Introduction

Fission product studies on Core I started in February 1958. The purpose of this program is to characterize the fission product behavior, monitor the fission product levels, determine the source, and evaluate any health hazards that exist. Two APAE reports have been issued covering work completed through June 1959. (6, 7)

It has been concluded on the basis of work performed to date that cladding defects do exist in Core I. (6)

The present program consists of two tests:

Test 200 -- Fission Product Monitoring at SM-1

Test 201 -- Startup and Shutdown Run for I-131 and I-133 in the SM-1 Primary Coolant

These tests are designed to monitor the fission product level at the SM-1 and to characterize their behavior. One additional test, "Monitoring for $\propto$ Activity in the Primary Coolant", will be initiated as soon as the proper counting equipment is purchased.

2. Work During Report Period

- a. Test procedures were prepared for fission product monitoring and for studying the behavior of I-131 and I-133 during startup and shutdown.
- b. A review was made of all data collected to date. The conclusion was reached that the fission product level at SM-1 increased by a factor of at least 1000 between January 12, 1958 and May 14, 1958.
- c. An attempt to run Test 201 was made on September 20, 1959. However, it was necessary to terminate the test on September 25 to determine if the rod drive water box was suffering excessive neutron activation.

3. Results

An evaluation of the short-lived data obtained under Task III was made and reported in the Task I final report. ⁽⁵⁾ This evaluation showed that approximately 80% of the short-lived dose rate in the primary system in January 1959 was due to fission products. In addition, the short-lived dose rate at one point in the primary system increased from 50 mr/hr on January 12, 1958 to 350 mr/hr on May 14, 1958. The short-lived dose rate at the same point on August 24, 1957 was 54 mr/hr. It is thus apparent that the increase between January and May was a result of some condition that did not exist before January 1958.

An analysis of the resin taken from a plant demineralizer during January 1958 showed the total rare earth (fission product) activity on the resin was less than 10^6 dpm per gram of resin. ⁽⁸⁾ Had the fission product level in the primary system previous to sampling this resin been equal to that observed during October 1958, the rare earth activity on the resin would have been at least 10^8 dpm per gram of resin.

It has been concluded from the information given above that the fission product level at SM-1 increased by a factor of at least 100 between January and May, 1958. Such an increase can be explained only by the occurrence of a cladding defect.

Table 5-4 shows the data collected during the attempted iodine startup and shutdown run. Samples 301 through 306 were drawn during the initial system cleanup, and 307 and 308 during steady power operation. The data obtained during power operating were inconclusive due to the abbreviated nature of the test.

TABLE 5-4

I-131 LEVELS

Sample No.	Date & Time Sampled	Blowdown Rate	Ln N	Degree F Days	Consecutive Hrs. at Power	Atoms/ml*
301A	9/20/59 1445	1.9	0	8667.24	cleanup started at 0001, 9/20/59	3.26×10^7
301B						4.40×10^7
302A	9/21/59 1750	1.7	0	"	---	3.65×10^7
302B						2.21×10^7
303A	9/23/59 0325	0.15	0	"	---	4.20×10^7
303B						4.55×10^7
304A	9/23/59 0630	0.50	0	"	---	4.55×10^7
304B						4.77×10^7
305A	9/23/59 1448	0.20	0	"	---	7.40×10^7
305B						7.10×10^7
306A	9/23/59 1803	1.5	1.0	"	---	6.20×10^7
306B						6.06×10^7
307A	9/23/59 2023	0.55	1.0	"	---	6.47×10^7
307B						7.03×10^7
308A	9/24/59 0020	0.52	65	8667.53	23m	6.60×10^7
308B						6.90×10^7
309A	9/24/59 0300	0.50	75	8669.42	3.05h	8.25×10^7
309B						7.99×10^7
310A	9/24/59 0525	0.45	75	8671.25	5.47	9.05×10^7
310B						9.74×10^7
311A	9/24/59 0813	0.50	75	8673.25	8.27	1.09×10^8
311B						1.26×10^8
312A	9/24/59 1113	0.55	75	8675.40	11.27	1.25×10^8
312B						1.32×10^8
313A	9/24/59 1353	0.60	76	8677.43	13.94	1.28×10^8
313B						1.38×10^8
314A	9/24/59 1703	0.50	75	8679.73	17.11	1.42×10^8
314B						1.52×10^8
315A	9/24/59 2015	0.55	75	8682.12	20.31	1.52×10^8
315B						1.68×10^8
316A	9/24/59 2330	0.55	75	8684.58	23.56	1.68×10^8
316B						1.65×10^8
317A	9/25/59 0225	0.52	80	8686.74	26.48	1.88×10^8
317B						1.75×10^8
318A	9/25/59 0450	0.50	78	8688.53	28.90	1.81×10^8
318B						1.79×10^8
319A	9/25/59 0814	0.46	80	8691.01	32.30	2.15×10^8
319B						1.87×10^8
320A	9/25/59 1108	0.48	80	8693.17		**
320B						**
321A	9/25/59 1403	0.41	82	8695.34	38.08	2.07×10^8
321B						2.07×10^8

* May be converted to dpm/ml by multiplying by 5.97×10^{-5}

** Seal leakage pumps were on when these samples were drawn, invalidating any information to be obtained.

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4. Future Work

Work under Phase 5 will continue to the end of Core I life. Test 201 has been scheduled for November 26 through December 12, 1959. Test 200 will start at the end of the training cycle.

5.2.4 Phase 6 - Fission Product Studies, Core II

Work on this phase will start with the installation of Core II. The work will include the evaluation of instream fission product monitors and more rapid methods of iodine analyses, presently impractical because of the high fission product level with Core I.

5.2.5 Phase 7 - Decontamination of SM-1 Steam Generator

Decontamination is scheduled at the end of Core I life. No work was done during the report period.

5.2.6 Independent Tests

No tests were initiated under this phase during the period.

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5.3 TEST SERIES 300 - Physics Measurements

5.3.1 Tests 301-316 - Core Measurements

Core measurements have been made on the SM-1 several times during Core I lifetime. The results of these measurements will be used to provide a basis for the nuclear performance evaluation of the core. Some of the experimental data may be extrapolated and applied directly to other cores and some will be used to normalize calculational models. The results of measurements through 9.1 MWYR are reported in APAE Memo 178⁽⁹⁾ which includes a description of the SM-1 experimental techniques developed for core measurements, and the methods of data recording and reduction. The second interim report, APAE Memo 206, ⁽¹⁰⁾ gives the results of measurements during the period from November 1958 to June 30, 1959.

On July 27, 1959, after approximately 12.1 MWYR of energy release, the SM-1 was shutdown for core measurements. Data for tests 301 through 316 were obtained. Procedures for individual tests were prepared and integrated into an operating procedure to yield the maximum amount of useful data in the shortest testing time. ⁽¹¹⁾

The results of the tests at 12.1 MWYR are given below.

5.3.2 Test 301 - Transient Xenon

The negative reactivity introduced by xenon plotted as a function of time after power reduction is shown in Figure 5-1. The reactivity value of peak xenon concentration at 12.1 MWYR is 1.45 dollars more negative than the equilibrium value. Peak concentration occurred 7.5 hours after power reduction and the reactivity introduced by xenon was the same as the equilibrium value 19.5 hours after power reduction.

5.3.3 Test 302 - Equilibrium Xenon

The reactivity worth of full power equilibrium xenon was determined as 3.02 dollars after 55 hours of continuous operation at full power.

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

At 12.1 MWYR energy release, 7.5 hours after power reduction peak xenon concentration in the core occurred. The critical position of the five-rod bank with rods A and B at 19 inches, peak xenon concentration in the core, and the mean core temperature at 440°F was 15.73 inches.

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FIG. 5-1

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

The reactor was operated continuously at full power for 55 hours to approach equilibrium xenon concentration in the core. The critical position of the five-rod bank with rods A and B at 19 inches, equilibrium xenon concentration in the core, and the temperature at 440°F was 14.38 inches.

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

At 12.1 MWYR, xenon was allowed to decay for 60 hours following power operation. The operating temperature was maintained at 440°F. With the reactor at the conditions described above, the critical position of the five-rod bank with rods A and B at 19 inches was 12.13 inches.

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

At 12.1 MWYR, xenon was allowed to decay from equilibrium concentration and the temperature was reduced to 70 F. With the reactor at the conditions described above, the critical position of the five-rod bank with rods A and B at 19 inches was 8.37 inches.

Figure 5-2 shows the critical position of the five-rod bank for the core conditions defined in Tests 303, 304, 305, and 306 plotted as a function of energy release.

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

Rod A was calibrated with peak xenon concentration in the core at a mean core temperature of 440°F. The calibration was done as a function of position versus the five-rod bank position using the period method. Three calibrated points were determined; the calibration was used to obtain a calibration point for the five-rod bank.

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

Rod A was calibrated with xenon concentration in the core at a low level. Low xenon concentration is achieved by allowing the xenon to decay for at least 60 hours following power operation. The mean core temperature was maintained at 440°F. The calibration was done as a function of position versus the five-rod bank position using the period method. Figure 5-3 shows the worth of rod A in cents per inch as a function of rod position.

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

Rod A was calibrated with low xenon concentration in the core with the mean temperature at 130°F. The calibration was done as a function of

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FIG. 5-2

Five Rod Bank Position as a Function
of Energy Release

ROD WORTH (CENTS/INCH)

FIG. 5-3

position versus the five-rod bank position. The period method of calibration was used. Figure 5-3 shows the worth of rod A in cents per inch as a function of rod position.

5. 3. 11 Test 310 - Calibration of Rod C; Low Xenon, 440°F

The central control rod was calibrated with low xenon concentration in the core and the mean core temperature maintained at 440°F. The calibration was done as a function of position versus the four-rod bank position with rods A and B at 19 inches. The period method of calibration was used. Figure 5-4 shows the worth of rod A in cents per inch as a function of rod position.

5. 3. 12 Test 311 - Temperature Coefficient

The temperature coefficient from operating to room temperature was measured. The plot of temperature coefficient as a function of temperature is shown in Figure 5-5. The curve is drawn from data taken several times during core life. The data points shown as 12.1 MWYR data are the average of the coefficient evaluated in increments of approximately 25°F. The temperature coefficient at 440°F is -3.5 cents/°F. The integral reactivity change from 70°F to 440°F evaluated from the temperature coefficient is 6.63 dollars.

The critical five-rod bank position with rods A and B at 19 inches was determined as a function of temperature. The plot of bank position versus temperature is shown in Figure 5-6. The reactivity change from 70°F to 440°F was evaluated from bank motion and bank calibration as 6.98 dollars.

5. 3. 13 Test 312 - Source Multiplication

The neutron count rate 97 hours after shutdown plotted as a function of $1 - K_{\text{eff}}$ of the core is shown in Figure 5-7. The subcritical reactivity associated with a given bank position may be determined from the neutron count rate using Figure 5-7. At 12.1 MWYR, with all control rods fully inserted, the neutron count rate was 4.8 counts per second, indicating a negative reactivity of 0.11. The neutron count rate with rods A and B withdrawn to 19 inches and the five-rod bank fully inserted was 8.4 counts per second indicating a negative reactivity of 0.065.

5. 3. 14 Test 313 - Gamma Heating in the Pressure Vessel

Data was recorded for this test during the full power run preceding the research and development shutdown for core measurements at 12.1

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FIG. 5-4

FIG. 5-5

Five Rod Bank Position as a Function of Temperature

Rods A and B at 19"

No Xenon

MWR
0
6.1
7.2
9.1
12.1

Temperature (°F)

FIG. 5-6

FIG. 5-7

MWYR. Temperatures on the outside surface of the pressure vessel are measured by thermocouples located around the vessel at the core centerline. No direct method for measuring the temperatures on the inside surface of the pressure vessel is available. The inside surface temperature is determined from the reactor inlet and outlet temperatures. Due to uncertainty of the temperature on the inside surface of the pressure vessel, no conclusions as to gamma heating have been reached. Methods for obtaining an accurate measurement of the temperatures are being investigated.

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

The calibration of the five-rod bank drawn from data obtained several times during core life is shown in Figure 5-8. The dotted curve is drawn through points determined from 12.1 MWYR measurements. The measurements during this period clearly indicate an increase in bank worth over that expected from a calibration in a uniform core. (12)

5.3.15 Test 317 - Spent Core Rearrangement

The SM-1 Core I is defined as being spent when at full load with equilibrium xenon concentration and the mean core temperature at 440°F, all rods are fully withdrawn. The change in reactivity at the end of life of Core I between the spent elements in the original configuration and after the elements are rearranged or replaced in three configurations is to be measured. The configurations to be investigated are:

1. Rearrangement of spent Core I elements by interchanging calculated high and low burnup elements.
2. Replacement of 18 Core I elements with calculated high burnup with 18 new Core II elements and rearranging to meet the following objectives:
 - a. To achieve the best neutron flux distribution.
 - b. To achieve the highest reactivity gain.

Procedures for this test have been prepared. An evaluation of the hazards associated with the rearrangement has been made and is reported in APAE Memo No. 225. (13)

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FIG. 5-8

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Future Work

A shutdown for core measurements after 14.5 MWYR of energy release has been proposed. Possible conflicts with other proposed reactor use at that time must be resolved. A final report on core measurements during the entire Core I lifetime will be issued in early 1960.

The spent core rearrangement test will be performed at the end of core life, which is expected in early March 1960.

Core measurements will be made throughout lifetime of Core II.

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5.4 TEST SERIES 400 - SHIELDING MEASUREMENTS

Shielding experiments will be performed on the SM-1 to obtain data to be used for the optimization of future shield designs and for the possible improvement of existing shielding. Procedures describing methods for obtaining data for some of the tests have been prepared by the shielding unit at Schenectady; other tests and possible test procedures are being investigated. The results of a study made in connection with a leak that developed in the water box shielding the control rod drive penetrations in the lower portion of the SM-1 pressure vessel are directly applicable to proposed shielding tests in this area. Progress during this quarter on this test series is described in the following paragraphs.

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

Procedures for these tests were prepared. These tests will obtain a neutron and gamma flux map of the SM-1 primary shield. The neutron flux will be measured by the irradiation of both bare and cadmium covered gold foils. The gamma flux will be measured by exposing gamma sensitive monitor films. Vertical traverses will be made in the water spaces between each of the shield rings and above the core in the central shield tank. The gamma traverses will be made both during reactor operation and as a function of time after shutdown.

These neutron and gamma flux maps will be used to optimise future shield designs of the same type and supply data for the evaluation of shield design methods.

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

Procedures for these tests were prepared. These tests will obtain a neutron and gamma flux distribution in the SM-1 instrument wells. The neutron flux will be measured by the irradiation of both bare and cadmium covered gold foils. The gamma fluxes will be measured by exposing gamma sensitive monitor films. Vertical neutron and gamma traverses will be made in each of the instrument wells. The gamma traverses will be made during reactor operation and as a function of time after shutdown.

These neutron and gamma flux measurements will be used to accurately determine the radiation environment of the instruments as a function of position within the tube, and most importantly give a value of the neutron to gamma flux ratio as a function of position. This data is necessary to properly choose the instrumentation, associated connectors and wiring.

5.4.3 Test 405 - Reactor Component Activation Studies

An investigation was started to determine measurements required and testing methods for a study of reactor component activation with regard to feasibility of relocation of the plant.

5.4.4 Test 406 - Neutron Measurements in the Rod Drive Pit Test 407 - Gamma Measurements in the Rod Drive Pit

A study was performed to determine the consequences of a leak that developed in the water box that shields the control rod drive penetrations in the lower portion of the SM-1 pressure vessel. The leak introduced a possibility that reactor operation with the water box empty might occur. The possibility that the loss of moderating material might increase the neutron flux level in the rod drive pit to the point where activation of pit components would be a serious problem was investigated. The results of this investigation are directly applicable to Tests 406 and 407. Additional data for these tests will be obtained from gamma measurements taken in the rod drive pit prior to and following removal of the spent core.

Calculations and measurements of the neutron and gamma levels in the rod drive pit were performed for the study. The information was required to determine if the high shutdown gamma dose rate in the pit originated from activation of pit components or from the reactor core itself. If activation were the main source, corrective measures would be necessary. The corrective measure would be either: (1) seal the leak to hold water in the box, or (2) replace the water in the box by some other moderating material that would resist radiation damage and would not leak from the box. If activation were not a problem, the shutdown dose rate in the pit could be reduced by placing a lead shield against the water box front surface.

5.4.4.1 Neutron Fluxes

1. Calculation of Neutron Fluxes

The neutron flux at the surface of the steel plate forming the front face of the rod drive water box was calculated utilizing two methods.

a. Method 1 - Removal theory

This method assumes that the core is a point source of fission neutrons and utilizes the point kernel to calculate the flux at the dose point. The flux which is calculated is an uncollided flux having essentially a fission neutron spectrum. This flux is only a portion of the total fast flux but it is indicative of the shield efficiency for the attenuation of fast neutrons. The flux calculated by this method is expressed by the following equation: (14)

$$\phi_u = \frac{5.4 S_0 T_w^{0.29}}{4 \pi R^2} e^{-[0.928 T_w^{0.58} + \sum_o T_w + \sum_m T_s]} \quad (1)$$

where:

- ϕ_u = Uncollided flux - n/cm² sec
- S_0 = Core neutron source - n/sec
- T_w = Thickness of water between sources and dose point - cm
- T_s = Thickness of steel between sources and dose point - cm
- $\sum_o$ = Oxygen removal cross-section for fission neutrons - cm⁻¹
- $\sum_m$ = Steel removal cross-section for fission neutrons - cm⁻¹
- R = Distance between dose point and source - cm

This method was applied to a dose point at the upper edge of the water box where a ray from the core to this point does not pass through any of the water in the box. The method was also applied at a position at the lower portion of the water box where a ray from the core will pass through the box and any water it may contain. The calculation was performed for the case of the box with and without water. The results are presented below:

Top edge of water box front plate:

$$\phi_u = 6.7 \times 10^8 \text{ n/cm}^2 \text{ sec}$$

Lower portion of water box front plate:

- 1) with water in box $\phi_u = 5 \times 10^3 \text{ n/cm}^2 \text{ sec}$
- 2) without water in box $\phi_u = 9.25 \times 10^5 \text{ n/cm}^2 \text{ sec}$

b. Method 2 - Two group diffusion theory

This method utilizes the Valprod IBM-650 code which is a one dimensional two group diffusion calculation. In this particular application, the calculation was performed in spherical geometry. Two group diffusion theory calculations were shown to agree with experimental results very closely. (15)

This method was applied to three positions on the surface of the water box front plate, the top edge, the central portion and the lower portion. The results of this calculation are presented below:

Top edge of water box front plate:

- Fast flux $2.6 \times 10^9 \text{ n/cm}^2 \text{ sec}$
- Thermal flux $5 \times 10^2 \text{ n/cm}^2 \text{ sec}$

Central portion of water box front plate:

With water in box - Fast flux 4×10^7 n/cm² sec
Thermal flux 5×10^1 n/cm² sec

Lower portion of water box front plate:

1) With water in box - Fast flux 2.1×10^5 n/cm² sec
Thermal flux < 1 n/cm² sec

2) Without water in box - Fast flux 5×10^8 n/cm² sec
Thermal flux < 1 n/cm² sec

2. Measurement of Neutron Fluxes (*)

The neutron fluxes were measured on the surface of the rod drive water box front plate at the lower left hand portion. The measurements were made by exposing both bare and cadmium covered gold foils for a timed interval while the reactor is operating at full power.

These foils were then counted with a gamma ray spectrometer which had previously been calibrated with a standard gold foil. The flux is then calculated with the following equation.

$$\phi = \frac{AD}{N_0 \sigma W} (1 - e^{-\lambda T}) e^{-\lambda t} \quad (2)$$

where:

ϕ = neutron flux - n/cm² sec

A = atomic weight of gold = 197 gm/mole

N_0 = Avogadro's number = 6.02×10^{23}

σ = thermal neutron cross-section for gold = 98 barns

W = weight of gold foil - gms

λ = decay constant for Au - hrs⁻¹

T = irradiation time - hrs.

t = time from end of irradiation to start of counting - hrs.

D = disintegrations per second corrected for counter efficiency.

The neutron flux calculated from the activity of the cadmium covered foil is subtracted from the flux calculated from the activity of the bare foil resulting in the total flux below the cadmium cut-off of 0.4 ev. This flux can be considered the thermal flux at the point of the measurement. The flux calculated from the activity of the cadmium covered

* These measurements were performed by the personnel at Fort Belvoir under the direction of R. A. Hasse.

foil is not the fast flux as the thermal cross-section of gold-197 is used in the calculation. The fast flux as such, cannot be calculated, as the proper cross-section to use cannot be determined without a detailed knowledge of the fast flux spectrum. Most of the fast activation of the gold probably occurs in the resonance region and is, therefore, due mainly to neutrons in the epi-thermal range. However, the "fast flux" calculated from the activity of the cadmium covered foil utilizing the thermal neutron cross-section for gold, when compared with the calculated thermal flux, gives a good indication of the relative fast and thermal activation to be expected at the dose point.

Three foil irradiation runs were made, one with the rod drive box without water and the other two runs with the rod drive box filled with water. Five bare and five cadmium covered foils were irradiated during each run. The average values of the measured fluxes are reported below:

Rod drive box full of water:

"Fast flux" 2.06×10^7 n/cm² sec
Thermal flux 2.24×10^7 n/cm² sec

Rod drive box without water:

"Fast flux" 6.86×10^7 n/cm² sec
Thermal flux 5.84×10^7 n/cm² sec

3. Comparison of Measured and Calculated Fluxes

The removal theory calculation indicates that the water in the rod drive box attenuates the fast flux by a factor of 185 over that if there were no water present. The two group diffusion theory calculation results in an attenuation factor for the water in the box of 2,380. However, a direct comparison of these two methods cannot be made as the removal theory flux has a much higher average energy. The ratio of the "fast fluxes" as measured with gold foils results in an attenuation factor of 3.3 for the water in the box. This fast flux however has a very low average energy (essentially epithermal) as compared to the calculated fluxes. Therefore, a direct comparison of the calculated and measured flux values cannot be made.

The calculated thermal flux values are very low as compared to the calculated fast flux. This can be explained by the fact that the 4

inch thick steel front plate of the water box attenuates any thermal neutrons that are present at its rear surface without appreciably thermalizing any of the fast flux.

4. Conclusions

The total flux as determined from the activity of the bare gold foils utilizing the thermal neutron cross section for gold in the calculation is 4.3×10^7 n/cm² sec with the box full of water and 1.27×10^8 n/cm² sec with the box empty. These fluxes can be considered as the total activation flux and can be utilized to calculate the activation of other materials in the area although some small error will be introduced because these other materials might not have the same ratio of thermal to resonance neutron cross sections.

The two group diffusion theory calculations show that the fast flux is attenuated by a factor of 4 by the water box front plate. The activation flux determined by the gold foils has a much longer average energy than the two group fast flux and is, therefore, attenuated more than a factor of 4 by the water box front plate. The average activation flux within the plate will, therefore, be considered to be 4 times that measured by the foils on the plate surface.

5. 4. 4. 2. Gamma Dose Rate

1. Calculation of Gamma Dose Rates

The gamma dose rate in the rod drive pit arises from the two major sources. One source is from the primary core gammas along with the secondary capture gammas from structural material near the core. The second source is from the activation of material within the drive pit by the neutron flux.

a. Dose rate due to core and secondary gammas

The core is assumed to be a point source of gammas and the capture gammas are taken into account by doubling the core source. This is a good approximation since both primary and secondary gammas contribute about equally to the dose rate. The point source equation is used to calculate the dose rate but an additional factor is included in the equation to account for the self absorption of the core. This self absorption factor is calculated as follows:

of sub elements $\frac{1}{4\pi R^2}$ law of
 radiation using the equation for

(3)

where:

- (2) A = self absorption factor
 μ_c = attenuation coefficient of core - cm^{-1}
 Z = core self attenuation distance - cm

The core self attenuation distance (Z) is calculated by the method given in Rockwell (14) for special sources. The dose rate in the rod drive pit can then be expressed by:

$$D = \frac{BS_0 A e^{-b}}{4 \pi R^2 C_E} \quad (4)$$

where:

- D = dose rate - mr/hr
 B = buildup factor
 A = self absorption factor
 b = number of relaxation lengths of absorbing material
 between source and dose point
 S₀ = source strength - mev/sec
 R = distance between source and dose point - cm
 C_E = conversion factor - mev/cm² sec per mr/hr

Using the above equations the dose rates were calculated at the surface of the rod drive water box front plate during reactor operation and at various times after shutdown. For reactor operation the calculation was performed utilizing the 5 gamma energy groups of 7, 6, 4, 2 and 0.75 mev. The shutdown dose rates were calculated utilizing 2 energy groups of 2 and 0.75 mev. The results of these calculations are presented below.

- Reactor operation - 880 r/hr
 5 hours after shutdown - 8.34 r/hr
 1 day after shutdown - 7.34 r/hr
 8 days after shutdown - 4.58 r/hr

b. Dose rates due to induced activity in the rod drive pit

The largest source of induced activity in the rod drive pit is the 4 inch thick steel plate which makes up the front face of the water box. This plate is exposed to a much higher neutron flux and is also geometrically much larger than any other item in the rod drive pit. Therefore, the induced activity of the other items in the pit will be negligible compared to the activity of the water box front plate.

The dose rate on the surface of the water box front plate due to activation within the plate can be calculated using the equation for an infinite slab source.

$$D = \frac{BS_V}{2\mu_S C_E} \left[1 - E_2(M_{St}) \right] \quad (5)$$

where:

D = dose rate - mr/hr

B = buildup factor

μ_S = absorption coefficient of steel slab - cm^{-1}

C_E = conversion factor - mev/cm² sec per mr/hr

t = thickness of steel slab - cm

S_V = activity of steel slab - mev/sec cm³

The activity of the steel slab (S_V) is a function of the neutron flux in this slab. There are two approaches to the calculation of the induced activity in the plate. The first approach is to use the calculated fast and thermal flux in the plate with fast and thermal activation cross sections and gamma yields to produce an activation source. This method can be expressed by the following equation:

$$S_V = \left[F \phi_{th} + G \phi_f \right] E \quad (6)$$

where:

S_V = activity in slab - mev/sec cm³

ϕ_{th} = average thermal flux in slab - n/cm² sec

ϕ_f = average fast flux in slab - n/cm² sec

F = photons/cm³sec for a unit thermal flux

G = photons/cm³sec for a unit fast flux

E = energy of gamma photons - mev

The F and G factor have been tabulated for reactor operation and various times after shutdown for the various materials expected in the shielding and structural parts of a reactor. The method calculating these factors is described in APAE 35. (15)

The steel slab activity is calculated with equation (6) and the results used in equation (5) to calculate the dose rate at the slab surface. The calculation is performed for the case of water in the rod drive box and the case of the box without water. The fluxes used in the calculation are average values within the slab as obtained from the two group diffusion theory method and are listed below along with the calculated dose rates.

	<u>Fast flux</u>	<u>Thermal flux</u>
water in box	5×10^6	5×10^5
box without water	10^9	negligible

Dose Rates on Water Box Front Plate Surface - r/hr

	<u>Water in box</u>	<u>No Water in Box</u>
Operating Reactor	2.24	176
5 hours after shutdown	0.008	0.760
1 day after shutdown	0.004	0.406
8 days after shutdown	0.004	0.392

The second approach is to use the experimentally determined activation flux with an experimentally determined activation cross section. The cross sections for various reactor structural materials have been measured by exposing the materials to a known neutron flux and measuring their activity. These cross sections are reported by Bopp and Sisman.⁽¹⁶⁾ The activity of the carbon steel plate at the front of the water box can be expressed by the following equation:

$$S_v = \phi E \Sigma \rho e^{-\lambda t} \quad (7)$$

where:

- ϕ = activating flux - m/cm² sec
- E = gamma energy - mev
- Σ = activation cross section - cm²/gm
- ρ = density of slab = 7.85 gm/cm³
- λ = decay constant of activation product - days⁻¹
- t = time after shutdown - days
- S_v = activity in slab - mev/cm³ sec

The active nuclides formed during the irradiation of carbon steel are Mn⁵⁶, Fe⁵⁹ and Co⁶⁰. The parameters required for input to equation (7) are listed in the following table.

<u>Nuclide</u>	<u>E (mev)*</u>	<u>(cm²/gm)*</u>	<u>(days⁻¹)*</u>
Mn ⁵⁶	1.15	1.6×10^{-3}	6.65
Fe ⁵⁹	1.2	3.6×10^{-5}	0.0154
Co ⁵⁶	1.25	5.0×10^{-4}	3.58×10^{-4}

* From Reference 16, listed in references at end of this report.

As in the previous method, the steel activity is calculated by equation (7) and the results used in equation (5) to calculate the dose rate at the slab surface. This calculation is also performed both with and without water in the box. The fluxes used are those measured with the bare gold foils multiplied by a factor of 4 to account for the attenuation through the water box front plate and are listed below along with the calculated dose rates.

Activation Flux n/cm^2 sec

Water in box	1.72×10^8
Box without water	5.08×10^8

Dose Rates on Water Box Front Plate Surface - r/hr

	<u>Water in Box</u>	<u>No Water in Box</u>
5 hours after shutdown	0.549	1.62
1 day after shutdown	0.173	0.510
8 days after shutdown	0.146	0.430

Both methods of determining the dose rate at the surface of the water box due to activation are in very good agreement for 1 and 8 days after shutdown for the case of no water in the box. The dose rates for 5 hours after shutdown can be considered to be in fair agreement for the same case when the overall accuracy of most shielding calculations is considered. The agreement for the case with water in the box is very poor. Since the latter of the two above methods is based on experimental data those results are more believable than the purely calculational method and will be used in the calculation of the total dose rate in the next section.

c. Total dose rate at the surface of the water box

The total dose rate at the water box surface is the sum of the dose rates from the core and secondary gammas and the gammas from the activation of the water box front plate and are presented in the table below.

Total Dose Rate at the Water Box Front Surface - r/hr

	<u>Water in Box</u>	<u>No Water in Box</u>
During operation	882	1056
5 hours after shutdown	8.89	9.96
1 day after shutdown	7.51	7.85
8 days after shutdown	4.73	5.01

The above table and the previous two sections indicate that the major portion of the dose rate is from the core and not from the activation in the rod drive pit. Although the presence of water in the rod drive box does reduce the activation about a factor 3 it does not change the total dose rate very much whether the box is run full or empty of water.

2. Measurement of Gamma Dose Rates

During the initial startup program at the SM-1, the dose rates at the front surface of the water box were measured during low power operation. (17) These dose rates were then extrapolated to full power operation and resulted in a dose rate of 700 r/hr.

During period from July through September 1959 periodic dose rate measurements were made in the rod drive pit after shutdown to try and ascertain the effects of the loss of water in the rod drive penetration water box. The results of these measurements are presented in the table below.

Gamma Dose Rates on the Surface of the Water Box Front Plate

<u>Position</u>	<u>Date</u>	<u>Time after shutdown-days</u>	<u>Dose rate-r/hr</u>
Upper right	7/20	8	1.0
Upper right	8/4	8	2.0
Upper right	9/22	3	3.0
Upper left	7/20	8	0.95
Upper left	8/4	8	1.0
Upper left	9/22	3	1.5
Bottom	7/20	8	0.21
Bottom	8/4	8	0.20
Rod Penetration Sleeve	9/22	3	30.0
General pit Area	9/22	3	0.300

As can be seen from the table there is considerable variation depending on position. Also the rod drive penetrations seem to provide a streaming path for radiation which probably originated from the control rod fuel elements which are in the lower portion of the pressure vessel during shutdown.

3. Comparison of Measured and Calculated Values

The measured value of 700 r/hr for the operating dose rate was obtained with the water box full and shows fairly good agreement

with the calculated value of 882r/hr for the same situation.

The measured shutdown values are anywhere from 0.2 to 30 r/hr depending on position and therefore are very difficult to compare with the calculated values except to mention that the calculated values fall within the range of the measured ones.

4. Conclusions and Recommendations

The significant conclusion that can be reached from this study is that most of the dose rate that is seen in the rod drive pit comes from the core and not from neutron activation within the pit. Although the neutron flux levels are high enough to cause significant activation, this activation is not a serious problem and will not interfere with proper maintenance of the drives even if the water box is run dry.

The object that will contribute most of the dose rate due to neutron activation in the pit is the 4 inch thick steel slab making up the front face of the water box. Therefore the dose rate in the rod drive pit can be greatly reduced if a lead slab of about 2 inches in thickness is placed in front of the water box. This will shield both gammas originating from within the pressure vessel and any activation gammas from the water box front plate.

These conclusions can be substantiated if careful gamma measurements are made in the rod drive pit prior to, and after, the removal of the spent core. If these conclusions are correct, when the spent core is removed the gamma dose rate in the rod drive pit should drop sharply. This test should be made when the present core is removed and the lead slab installed at this time if the result of the test is positive.

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

Procedures for this test were prepared. The objective of this test is to measure the dose rate at the surface of the water in the central shield tank above the core as a function of the position of a spent fuel element during the transfer operation. An ionization chamber will be mounted near the water surface and the spent element will be raised toward the water surface. The dose rate will be recorded as a function of distance of the spent fuel element below the water surface.

This data is necessary to optimize the height of the water above the core of an SM-1 type system and to optimize the water height in the spent fuel pit.

5.4.6 Test 409 - Dose Rates on the Spent Element Surfaces

An investigation was started to determine methods to establish relative fuel burnup from gamma dose rates on the spent element surfaces.

5.4.7 Future Work

During the next quarter work will continue on the investigations for Tests 405 and 409. Test procedures 401 and 402 will be revised on the basis of comments received on these procedures. Testing on this series will be done at the end of life of SM-1 Core I, and during operation with Core II.

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6.0 PLANT MODIFICATIONS COMPLETED

6.1 HELIUM IN THE SEAL LEAKAGE TANK

A hydrogen blanket was maintained in the seal leakage tank to prevent in-leakage of air with the resulting increase of oxygen in the primary water. A combination of equipment failures could possibly result in large quantities of hydrogen discharging into the vapor container. Moving the seal leakage tank outside the vapor container or running a line from the seal leakage tank relief valve outside the vapor container would involve extensive equipment modification and a special dispensation from the code committee. A simpler and quicker way was proposed to eliminate the remote possibility of accumulating large quantities of hydrogen, by using helium as the pressurizing gas.

The substitution of helium provides a gas which is completely compatible with nuclear systems. It is inert chemically and has no corrosive effects on engineering materials. The neutron capture cross-section for helium is negligible. In addition, helium is heat stable and resistant to the effect of radiation.

Information in connection with replacing hydrogen with helium was presented to the Army. The Army approved the modification. Helium was installed on the seal leakage tank during the week of September 13, 1959.

6.2 PRESSURIZER COOLDOWN LINE

There are occasions during plant operation when it is desirable to rapidly cool the primary system within the time allotted by metallurgical considerations. Since the pressurizer is on a dead leg, depending on radiation losses for cooling, it cannot be cooled at the same rate as the remainder of the system. This extends the system cooldown period 4 to 5 times the period actually necessary. A pressurizer cooling line was installed from the primary blowdown line to the top of the pressurizer. It was necessary to verify the integrity of this installation by stress analyses prior to actual use of the system. Professor Sayre, an Alco consultant, performed the necessary calculations to prove the safety of its use.

The system was tested by pressurizing and heating the primary system to various plateaus and then cooling the pressurizer until a stable cooldown rate was achieved. Finally, the cooldown installation was tested at full system pressure and temperature. At primary design conditions, a cooldown rate in the pressurizer stabilized at 46°F per hour.

On the basis of our engineering calculations and tests, we recommended that use of the cooldown system as an operational function be approved. Army approval of the recommendation was given.

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