APAE-108

ARMY PWR SUPPORT AND DEVELOPMENT PROGRAM

Second Half-Year Summary Report, April 1, 1961–September 30, 1961

December 15, 1961

Nuclear Power Engineering Department
Alco Products, Inc.
Schenectady, New York

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APAE-108

REACTOR TECHNOLOGY

SECOND HALF-YEAR SUMMARY REPORT ARMY PWR SUPPORT AND DEVELOPMENT PROGRAM

April 1, 1961 - September 30, 1961

M. H. Dixon
Project Engineer

Issued: December 15, 1961

Contract No. AT(30-1) - 2639
with U. S. Atomic Energy Commission
New York Operations Office

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ABSTRACT

Progress is reported on research and development tasks under the Army PWR Support and Development Program, Contract AT(30-1)-2639, during the 6-months' period April 1, 1961 to September 30, 1961. Anticipated work to be performed during the first half of FY -62 is also presented.

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SUMMARY OF RESULTS, CONCLUSIONS AND RECOMMENDATIONS

TASK 1 - ENGINEERING SERVICES

Item 1.1 - Cyclic Stress Analysis of SM-1 Primary System

No progress is reported beyond that presented in the mid-year summary report. The final report is scheduled for editing and publication in the next quarter.

Item 1.2 - Closure Stud Investigation

Corrosion tests were performed in two stages of 1000 hours duration for the purpose of evaluating hardness, stress level, and the effect of thread fit. Magnetic particle inspection and metallographic examination were conducted on test specimens, and a report is being prepared.

Item 1.3 - Procurement of Replacement Cores for SM-1A and PM-2A

The brazing problem has been resolved by nozzling the atmospheric gas inlet pipe so as to increase the gas flow directly into the ends of each fuel assembly.

Dimensional relaxation of fuel plate sag and water gap dimensions have been studied and allowable deviations from the specifications have been established.

Thirty stationary elements and five control rod elements for PM-2A Core II have been accepted and received at Alco. The SM-1A Core II is still in fabrication and is approximately 35% complete. It is expected that the SM-1A Core II will be completed and delivered to Alco early in November.

Item 1.4 - Report Establishing Restrictions on SM-1 Experiments

The criteria for evaluating hazards involved in proposed tests on and/or modifications to the SM-1 have been established and limiting system parameters set up. A report covering these criteria has been prepared and will be issued in the next quarter.

Item 1.5 - Preliminary Study of Radiation Damage to SM-1 Reactor Vessel

A feasibility study of annealing or replacing the SM-1 reactor vessel has been completed. Results indicate that replacement of the reactor vessel is possible, but highly impractical because of health physics problems which compound the major mechanical problems involved. Four methods of annealing the reactor vessel were considered to be thermodynamically and mechanically feasible. However, it is recommended that a design and development program be initiated prior to deciding which method of annealing is the most feasible.

Item 1. 6 - SM-1 Core II Hazards Report

Addenda to the SM-1 Core II hazards report have been prepared. These addenda cover SM-1 Core II without special components, insertion of the PM-1-M-2 element into SM-1 Core II, SM-1 Core II without the SM-1 Core I high burnup elements and with the PM-1-M-2 element, and replacing SM-1 Core I high burnup elements with SM-1 Core I spare elements. The variations considered in these addenda do not represent any hazards greater than those defined in the hazards summary report for the SM-1.

Item 1. 8 - Technical Assistance, SM-1 Core I Reprocessing

Plans have been completed for removing the flux suppressor combs from the spent SM-1 control rod elements.

Alco will supply procedures, manpower, and equipment for sampling the activity release rate of the spent control rods before and after removal of the combs. Procedures have been prepared and submitted for approval. Technical supervision and test procedures will also be provided for determining the cladding integrity before and after removal of the combs.

Alco will assist in preparing the SM-1 Core I for shipment to the reprocessing facility.

Item 1. 9 - Evaluation of Precipitation Hardened Stainless Steel

An investigation of all PH steel components in Alco-built reactors was conducted and the information compiled in interim reports issued during this period. Integrity of the control rod drives has been demonstrated by several years of satisfactory operation without incident due to 17-4 PH material failure, by proof calculations of low stress conditions, and by extensive destructive and non-destructive examination of SM-1 and SM-1A racks and pinions.

A specification for procurement procedures for 17-4 PH steel has been developed which can be applied to any 17-4 PH component with reservations as required.

Additional non-destructive examination of three SM-1 irradiated water seal shafts and the spare PM-2A rack and pinion will be performed in the first half of FY-62 and a final report issued to complete this task.

Item 1. 10 - Evaluation of SM-1A Core I Elements

Inspection and evaluation of SM-1A Core I fuel elements was initiated. All eight control rods and 26 stationary elements were found acceptable for unrestricted placement in the core. The remaining 13 stationary elements were acceptable provided they are custom-placed within the core. Seven SM-1A Core I stationary elements were rejected pending additional metallurgical examination. These elements have been replaced by eight SM-1 Core II stationary elements available at Fort Belvoir.

An SM-1A Core I loading procedure was established so that acceptable minimum coolant passage clearances are maintained between all fuel elements, based on considerations, of initial fuel plate ripple magnitude and potential ripple growth upon exposure to thermal stresses during rated power operation.

TASK 2 - PERFORMANCE OF EXISTING CORES

Five cores are being analyzed under this task. These cores are SM-1 Core I, SM-1 Core I Rearranged and Spiked, SM-1 Core II, PM-2A Core I, SM-1A Core I.

SM-1 Core I (Including Rearranged and Spiked)

A summary report covering the operation of SM-1 Core I from startup through the final operation of the rearranged and spiked core has been completed and will be issued in the next quarter. The data obtained indicated noticeable changes in various rod worths; however, in the areas of temperature and xenon measurement, very slight changes occurred. The total core energy output was 18.0 MWYR. An estimated 1.1 MWYR additional energy release could have been achieved had the core been operated to complete burnout. Thus, the rearranged and spiked core could have extended the Core I life to approximately 19.1 MWYR.

SM-1 Core II

The SM-1 Core II loading was completed according to detailed procedures set up at the Alco Critical Facility. Core performance was demonstrated to be satisfactory. An addendum to the original Core II loading procedures has been prepared to cover the replacement of the two Core I high burnup elements with new Core II fuel elements.

Following the SM-1 Core II loading, a special test series (300) was performed to evaluate core performance. A detailed report covering core physics testing and Core II operation through September 30, 1961, has been prepared and will be issued early in the next quarter. Routine and special tests will continue during the next six months in order to obtain additional information on core performance and to evaluate the effects of the special components.

In connection with the planned insertion of Task XIV instrumented fuel assemblies in SM-1 Core II in April 1961, the penetrated gasket braze problem was evaluated and a supplement to APAE-79 issued. Conclusions stated that the gasket should be used as is. AP Note-349 (Task XIV installation and test procedures) has been revised per comments of TEB and NPFO and will be issued by November 15, 1961. The purge system was reviewed for automatic control. Conclusions drawn are that it would be advisable to install the automatic control equipment. A preliminary hazards evaluation of the 600 series tests has been completed.

SM-1A Core I

Revised loading procedures have been issued which incorporate positioning of various elements in locations such that thermal effect of plate rippling is minimized. A revised test program has been prepared; however, initiation of this program has been postponed until after the plant has been accepted. Routine measurements are planned to provide data for monitoring SM-1A Core I performance.

PM-2A Core I

Physics measurements performed on the PM-2A have been reviewed and prepared for inclusion in an initial startup and test report. Results of the measurements have indicated satisfactory performance of the core. These initial startup measurements, plus routine and special measurements which should be performed as a function of core life, will be employed to evaluate the PM-2A Core I and provide a basis for improving future core designs.

TASK 3 - REPLACEMENT CORE DEVELOPMENT

Item 3.1 - Critical Experiments for Use of Type 3 Elements in SM-1, SM-1A, and PM-2A

Critical experiments were conducted utilizing Type 3 fuel elements (SM-2) in SM-1, SM-1A and PM-2A configurations. Measurements included power distribution, temperature coefficients, control rod critical positions and calibrations, critical rod configurations and material worths. A final task report is to be issued in the next quarter.

Item 3.2 - Analysis of Power Distribution, Core Burnout, and Rod Operation Using SM-2 Fuel Elements in the SM-1, SM-1A and PM-2A

Comparison between the measured and analytical power distribution for the three cores has been completed. Core burnout calculations have been initiated for each core with SM-2 Core I uranium and poison loadings. Control rod operation considerations have been completed and, based on stuck rod data, it is recommended that SM-1 and SM-1A be operated with a 7-rod bank when Type 3 cores are installed.

Item 3.3 - Steady State and Transient Thermal Behavior of SM-1, SM-1A and PM-2A Using SM-2 Core I Fuel Elements

The final steady state and transient analysis has been completed. This analysis indicates that SM-1A and PM-2A with SM-2 elements can operate at design power and at scram power with adequate thermal safety. No extensive local boiling is evident in either of these cores and DNBR's are well above 2.0 at design power.

The SM-1 does show some local boiling in stationary elements adjacent to the reflector. Single element flow tests indicate channel-to-channel flow maldistribution as high as 55% for elements in this region; however, a conical type diffuser fix was developed which proved to reduce the maldistribution to 23%, thus reducing local boiling in the hot channels. Since DNBR's are well above 2.0 for design power and scram conditions, it is considered safe from a thermal standpoint to operate the SM-1 with SM-2 elements.

Items 3.4 - Single Element Flow Test of SM-2 Fuel Elements for SM-1, SM-1A and PM-2A

All test work has been completed and the results indicate that SM-2 stationary fuel elements modified to fit SM-1A and PM-2A core support structures are within the $\pm$ 12% maldistribution factor specified in the preliminary thermal analysis; control rod assemblies did not fall within the $\pm$ 12% maldistribution factor; and some modifications are desirable in order to reduce the unexpectedly high maldistribution factor found for SM-1 stationary elements with small orifices.

Item 3.5 - Metallurgical Development

Irradiation of six high burnup ETR instrumented capsules continued during this period. Two capsules were removed on July 10, one will be removed on October 2, and the remaining three will be removed November 13, 1961.

The two SM-2 elements were examined after 1.6 MWYR operation in the SM-1 rearranged and spiked core, found satisfactory, and reinserted into Core II for continued irradiation.

Completion of the metallurgical examination of SM-1 Core I element S-79 revealed extensive intergranular and transgranular clad cracking. As a result, it has been decided to remove Core I elements 80 and 81 from Core II in October for examination.

Item 3.6 - Metallurgical Studies

A cost survey was conducted for development of replacement cores of the latest proven type for SM or PM type reactors. Major cost reduction is possible by using Type 3 cores (40-mil fuel plates, 25 w/o UO_2, welded assembly) in reactors currently using Types 1 or 2. Other significant savings are possible by multiple core reprocessing, relaxation of cobalt and tantalum limits in Type 347 stainless steel, and multiple core procurement. Minor savings may be realized by quantity procurement of UO_2 and special Type 347 stainless steel.

TASK 4 - PRIMARY SYSTEM AND COMPONENT PERFORMANCE

Item 4.3 - PM-2A Steam Generator Moisture Carryover Test and Analysis

A special multi-port steam calorimeter in conjunction with fast response, highly accurate instrumentation will be used to measure the moisture in steam from the PM-2A steam generator during maximum increasing load transients (idle to full load).

Design of the test equipment and instrumentation, along with detailed fabrication and installation drawings, has been completed and a set of test specifications submitted for approval.

Item 4.4 - Shielding Measurements

Shielding measurements obtained during the SM-1 Core II and PM-2A Core I startups are being analyzed. These measurements provide data for evaluating the SM-1 and PM-2A shields.

PM-2A data indicates no major problems in the shielding as modified. This data will be presented in detail in the PM-2A initial startup and testing report (to be issued in the first half of FY-62).

SM-1 data indicates higher gamma dose rates than those previously reported. More detailed analysis of this data will be performed and a report issued.

TASK 5 - PRIMARY SYSTEM ACTIVITY CONTROL

Radiochemical analyses for induced and fission product nuclides of SM-1 and PM-2A coolant and crud samples were continued. The results suggested that a major portion of the coolant activity is either in the colloidal, ionic, or very small particulate (less than 0.01 micron), state. Evaluation of limited PM-2A data indicates that a considerable amount of coolant induced activity is due to release of out-of-flux corrosion products deposited and activated in-flux. If this is the case, the use of extremely low cobalt impurity stainless steel in-core may not be justified, when a considerable amount of Co^{60} activity arises from "normal" cobalt impurity stainless in out-of-core areas.

Photographs of the inlet and outlet edges of the fuel plates of SM-1 fuel element No. 79, showed a considerable amount of crud buildup on the inlet edges of the fuel plates. It was estimated that 30% of the total in-core crud was located in these inlet deposits. The release of these activated deposits can account for the observed increases in coolant crud specific activity shortly after a reactor shutdown and subsequent startup.

SUMMARY (CONT'D)

Further evaluation of SM-1 data showed that 33% of the Co^{60}, 16% of the Co^{58}, and 85% of the Mn^{54} activities were removed by the purification system. Thus, it appears that the effective purification constant is not the same for all nuclides.

The study of methods for improved activity control techniques was continued. A literature review of presently available or contemplated methods was issued. No equipment is available commerically which is satisfactory for activity control. At present, studies of methods for activity reduction are being pursued along two avenues:

(a) bench studies to develop additives for further investigation, and

(b) in-plant (SM-1) testing of promising techniques, such as high pH operation.

A radiochemical method for the analysis for europium in reactor coolant, was developed. The method will be checked at the SM-1 for sensitivity and practicability of application at the other reactor sites.

TASK 6 - ANALYSIS OF REACTOR PRESSURE VESSELS DURING OPERATING LIFE

One irradiated flux monitor capsule was removed from the rearranged SM-1 Core I, and sent to NRTS for analysis. The data from this monitor will be reduced by NRL. A second capsule installed in the SM-1 in April, 1961 will be removed in October 1961 for analysis. A flux monitoring system has been installed in the SM-1A for irradiation.

Data on transition temperature shifts with respect to both integrated flux and irradiation temperature for various vessel materials has been obtained and analyzed.

Experimental data has been obtained for determining the vessel wall dosages for the SM-1 and PM-2A mock-up cores. Data reduction and analysis has been completed for the SM-1 mock-up and initiated for the PM-2A mock-up. Along with this, analytical calculations have been performed for use in predicting spatial and energy dependence of the neutron flux in the core and at the vessel wall. A report correlating these experimental and analytical results will be issued in the next quarter.

Design criteria to serve as a guide for the design of future reactor vessels and as a means for determining safe operating limits for existing vessels have been published. Radiation damage studies based on these criteria have been completed for each vessel, establishing safe operating limits for each plant.

SUMMARY (CONT'D)

TASK 7 - SECONDARY SYSTEM AND COMPONENT PERFORMANCE

Work under this task was originally performed under Contract AT(30-3)-326. An SM-1 transient analysis report was issued and a report has been prepared in draft form describing the Task XII status as of June 30, 1961.

Continuation of this work will be performed under Contract AT(30-1)-2639 and will include issuance of a final Task XII report, evaluation of the steam dump line, rechecking and troubleshooting of instrumentation already calibrated and the performance of test runs during normal plant training cycles.

TASK 8 - WASTE DISPOSAL

The design of the waste disposal skid has been completed, the skid has been assembled, and all major processing units have been installed.

Laboratory work accomplished to date indicates that chemical treatment and preliminary solids removal followed by distillation and ion exchange are tenable steps in waste processing; however, operation of the skid with actual plant waste may prove that the process can be shortened, perhaps to only distillation and ion exchange.

It appears from the data that the unit operations incorporated in the skid design can handle any type of liquid waste and decontaminate it safely to MPC levels.

TASK 9 - INSTRUMENTATION AND CONTROL

The proportional counter lifting mechanism has been received, inspected and calibrated at Alco, Schenectady. Installation and test procedures have been prepared and submitted for approval. Installation and testing of the mechanism is expected to start during the October, 1961 shutdown.

BIBLIOGRAPHY OF REPORTS ISSUED ON CONTRACT AT(30-1)-2639, APRIL 1, 1961 - SEPTEMBER 30, 1961*

Progress Reports		Issued
APAE No. 86	Mid-Year Summary Report, Army PWR Support and Development Program Oct. 1, 1960-March 31, 1961	6/2/61
AP Note 354	Progress Report, April 1961 - Engr. Support and Development of Army Pressurized Water Plants	6/20/61
AP Note 360	Progress Report May 1961 - Engr. Support and Development for Army Pressurized Water Power Plants	7/19/61
AP Note 379	Progress Report June 1961 - Engr. Support and Development for Army Pressurized Water Power Plants	8/31/61
AP Note 382	Progress Report July - August 1961 - Engr. Support and Development for Army Pressurized Water Power Plants	9/28/61

Program Plans - Contract AT(30-1)-2639		
AP Note 286	Program Plan for Engineering Support and Development of Army Pressurized Water Reactor Power Plants	10/10/61
AP Note 286, Add. 1	Program Plan for Engineering Support and Development of Army Pressurized Water Reactor Power Plants	3/31/61
AP Note 286 Add. 1, Rev. 1	Program Plan for Engineering Support and Development of Army Pressurized Water Reactor Power Plants	5/1/61
AP Note 378	Fiscal Year 1962 Program Plan for Engineering Support and Development of Army Pressurized Water Reactor Power Plants	9/6/61

* Bibliographies of topical reports appear at the end of each task.

TASK 1 - ENGINEERING SERVICES

ITEM 1.1 - STRESS ANALYSIS OF SM-1 PRIMARY SYSTEM
Task Engineer - R. A. Chittum

Task Definition and Objective

The purpose of this task is to determine, through a preliminary stress analysis, whether there are any areas in the SM-1 primary system that may be subject to fatigue failure due to the extensive cyclic operation of the plant during training and research and development testing. Those components found to be highly stressed will be recommended for a more detailed stress analysis. Based on the results of the detailed analysis, changes in operating procedures or replacement of components may be necessary to assure safety of operation and maximum utilization of the existing plant.

As a part of this study, an estimate of the total cyclic operating history of the SM-1 during its design life will be made. It will be based upon a compilation of the past history of plant operation extrapolated to cover the training programs and research and development tasks planned for the future.

Summary of Second-Half Results

A report(1) on work performed in the first half has been reviewed and is ready for publication.

Conclusions

None.

Recommendations

None

Future Work

1. The report on SM-1 primary loop stress analysis will be issued.

2. Work will continue under this task to cover a preliminary stress analysis of the PM-2A steam generator and pressurizer.

ITEM 1.2 - CLOSURE STUD INVESTIGATION
Task Engineer - H. R. Clayton

Task Definition and Objective

A detailed failure analysis of SM-1 original studs will be made to determine measures necessary for trouble-free operation of the SM-1 and SM-1A. In addition, corrosion problems associated with the female threads in the reactor vessel body will be investigated.

Summary of Second-Half Results

The corrosion test was completed in two stages, each of 1000-hr duration. The initial test extended from March 24 to May 15 and the second test extended from June 12 to September 4. These tests were set up to evaluate the following variables: hardness, stress level, and effect of thread fit.

Magnetic particle inspection (Magnaglo) of the Type 410 stainless steel stud specimens after 1000-hr of testing showed no evidence of cracking. No difference in corrosion attack of the female threads in the carbon steel plate was observed due to varying the thread fit. Stresses of 25,000 and 30,000 psi were used during this initial test.

For the second test, the stress load was increased to 58,000 psi. Magnetic particle inspection and metallographic examinations were repeated on the test specimens after the second 1000-hr testing period. A reproduction of an actual SM-1 stress failure was observed. A detailed report is presently being prepared on this work.

Conclusions

1. Non-destructive testing (dye penetrant inspection and magnetic particle inspection) revealed no evidence of cracking after 1000 or 2000-hr of testing.

2. Metallographic examination showed micro-cracks resulting from both test conditions.

Recommendations

None.

Future Work

A detailed report will be prepared and issued during the next quarter to complete this subtask work.

ITEM 1.3 - PROCUREMENT OF REPLACEMENT CORES FOR SM-1A AND PM-2A
Task Engineer - H. R. Clayton

Task Definition and Objectives

To provide technical liaison and inspection on the fabrication of PM-2A Core II and SM-1A Core II, and to evaluate the usability of SM-1A Core I stationary fuel elements.

Summary of Second-Half Results

The brazing process problems mentioned in the Mid-Year Summary Report (APAE-86) were resolved early in June 1961. The trouble encountered during the brazing operation was determined to be twofold. First, the use of an insufficiently dry inert gas during the initial purging operation was resulting in oxidation of the braze metal powder, thus preventing the powder from being properly reduced during the brazing heat cycle. Secondly, the hydrogen atmosphere was not circulating adequately over the inside surfaces of the fuel plates.

In order to improve the braze quality of the production elements, the atmosphere gas inlet pipe was nozzled so as to increase the gas flow directly into the ends of each fuel assembly. No fuel elements have been rejected because of poor braze quality as a result of this modification.

Waiver requests by Olin Mathieson for PM-2A and SM-1A fuel elements for dimensional relaxation of both top and bottom fuel plate sag and water gap dimensions have been studied by the Thermal and Hydraulics Group at Alco Products. Maximum allowable deviations for these dimensions have been established, as well as conditions under which deviations beyond this specification will be permissible. Included in the deviations were considerations of potential fuel plate ripple growth upon exposure to reactor operating temperatures. The work is reported fully in APAE Memo-300(2). Further work was done on several elements that exceeded the permissible allowed. The relaxed deviations and analysis of fuel assemblies beyond the permissible were analyzed utilizing "STDY-3, A One-Dimensional Steady State Thermal Code for Hot Channel Analysis," developed at WAPD.

PM-2A Core II

The delivery of 30 stationary and 5 control rod fuel elements was completed on August 29, 1961. These fuel elements were delivered by UNC to the Alco Products Critical Facility at Schenectady.

The following is the production record of fabrication of the PM-2A Core II:

Operation	Stationary	Control Rod
Cores started	827	225
Acceptable fuel plates	563	90
Plate rejection rate	31.9%	64.4%
Fuel assemblies brazed	31	5
Fuel elements accepted	30	5
Rejection rate	3%	0%

SM-1A Core II

Production of SM-1A fuel elements and absorber sections continued through the end of this report period. About 35% of the SM-1A fuel elements have been completed. The remaining elements are in process. About 50% of the absorber plates have been "hot" rolled. Development work on the welding of the absorber boxes has been completed.

SM-1A Core Stationary Fuel Elements

The top and bottom fuel plate sag measurements made at Fort Greely, Alaska by Mr. R. J. Beaver of ORNL were received during this period and were analyzed by the Materials Technology and the Thermal and Hydraulics Groups. It was reccommended that two stationary fuel elements be rejected because of braze metal splatter over the active core area. It was also recommended that four other elements be returned to ORNL for measurements of scratches with a Zeiss Light Section Microscope to determine if these scratches exceed the allowable depth of 0.001 inch. These fuel elements have left Fort Greely but have not been received at ORNL.

Conclusions

The SM-1A Core II fuel loading should be completed and delivered to the Critical Facility at Schenectady by early in November. This will complete this portion of the contract covering assistance in procurement.

Future Work

1. Technical liaison and inspection on the fabrication of SM-1A Core II will continue until completion.

2. A final report[3] on the fabrication of SM-1A Core II and PM-2A Core II will be issued.

3. Future work of this kind will be carried on Task 10 in FY-62.

ITEM 1.4 - REPORT ESTABLISHING RESTRICTIONS ON SM-1 EXPERIMENTS
Task Engineer - J. R. Coombe

Task Definition and Objectives

To delineate critical areas of hazard control and develop a set of limiting parameters and guidelines to facilitate evaluation of hazards for SM-1 tests and modifications. Hazards reviews of proposed tests will be made.

Summary of Second-Half Results

APAE-106[4] has been written to fulfill the requirements of the task definition and objectives. This report is a modification and amplification of AP Note 301, and supersedes the original document. General guidelines for pressurized water reactors, limiting technical specifications for the SM-1, and a summary check list of items to be considered in hazards reviews were presented in the subject report.

A hazards review of test series A300, A400, and A600 has been completed.

Conclusions

Restrictions and criteria have been established which provide a reasonable basis for evaluating and reducing hazards associated with PWR tests at SM-1.

Recommendations

These criteria should be critically examined by all interested parties and communications should be established where a difference in approach is believed warranted.

Future Work

1. The hazards and precautions sections of the A300, A400 and A600 test series will be modified in accordance with the findings of the previous hazards reviews.

2. Supplement 1 to the criteria report will be prepared, incorporating review comments.

3. Supplement 2, providing limiting system parameters and additional guidelines for the PM-2A and SM-1A will be developed.

4. This work will continue in FY-62 as Subtask 1.4.

ITEM 1.5 - PRELIMINARY STUDY OF RADIATION DAMAGE TO SM-1 REACTOR VESSEL

Task Engineer - D. W. McLaughlin

Task Definition and Objectives

Recent studies have shown that irradiation damage to A-212-B steel, the SM-1 reactor vessel material, is more severe than indicated by data available at the time the vessel was designed. In view of this and the severe cyclic operation of the SM-1 plant during training programs, the possibility exists of overstressing the vessel while it is at temperatures where its ductility and notch-toughness are impaired.

This task is to provide a basis for safe operation of the SM-1 vessel until September 1961. It is to define operating limits in terms of temperature, pressure and temperature transients. Limiting pressures and temperatures will be determined from a stress analysis using existing metallurgical data and conservative limiting stress criteria.

This task is also to determine the feasibility of annealing the vessel in place to reduce the cumulative effects of radiation damage, or of replacing the vessel in the event that the non-recoverable damage reaches a level that makes further operation unsafe.

Summary of Second-Half Results

The first objective, to provide safe operating limits for the SM-1 vessel, was accomplished during the first half. The results of this study were published in AP Note-290[5].

A feasibility study of annealing or replacing the SM-1 reactor vessel, was performed and the results published in AP Note 373[5]. The four principal methods of annealing considered thermodynamically and mechanically feasible were steam heating, hot gas heating, hot water heating and radiant heating. Due to design problems, each will require more detailed study and a design and development program must be initiated before final decision on the best method of annealing. The study of replacing the reactor vessel indicated that replacement was possible but highly impractical because of health physics problems which compound the major mechanical problems involved.

As a result of new physics data on integrated flux values, the SM-1 reactor vessel will not require annealing during the design life of 20 years, providing the overall load factor does not exceed 46%. This is approximately the overall load factor experienced to date, but is less than the 60% load factor used for planning purposes.

Conclusions

Objectives of this task have been met. Latest physics information on flux values at the vessel wall indicate that the radiation damage to the SM-1 vessel is not a severe problem and annealing of the vessel is not required, provided that the current 46% load factor is not exceeded.

The annealing concepts discussed in AP Note-373 would be applicable to the PM-2A and SM-1A reactors but consideration must be given to available plant facilities and transportation problems.

Recommendations

More detailed annealing studies should be initiated for the PM-2A and SM-1A since the radiation damage problem is more severe for each of these reactors, particularly the PM-2A.

Future Work

Related work for the PM-2A will continue under Subtask 6.12 in FY-62.

ITEM 1.6 - SM-1 CORE II HAZARDS REPORT
Task Engineer - J. R. Coombe

Task Definition and Objectives

Prepare a hazards evaluation of SM-1 Core II with Special Components. Using the nuclear and thermal analysis and the critical experiment measurements performed on SM-1 Core II, determine the effect, if any, of special components on the hazard potential involved with the use of this core in SM-1.

Summary of Second-Half Results

Addenda to APAE-84, "Hazards Report for the SM-1 Core II with Special Components,"[6] have been written.[7][8][9][10] These addenda contain nuclear and thermal characteristics and analyses for Core II with the separate variations. Conclusions related to the hazard potential involved with these changes are made.

Conclusions

The variations to Core II as defined do not represent any hazard over and above that defined in the hazards summary report for the SM-1.

Recommendation

Each variation to the SM-1 core where any novel substitution is made in the core should be subjected to a hazards analysis. This hazards analysis should then be compared to the SM-1 Hazards Summary Report for comparison of the hazards potential.

Future Work

1. It is planned to evaluate each change in the SM-1 Core with an applicable hazards report. One such change involves the Cd-In-Ag control rod absorber to be inserted in SM-1 Core II.

2. Preparation of such hazards reports will be performed in FY-62 as part of the various subtasks to which they will apply. Hazards reports for replacement cores will be performed under Task 10.

ITEM 1.8 - TECHNICAL ASSISTANCE, SM-1 CORE I REPROCESSING
Task Engineer - H. R. Clayton

Task Definition and Objectives

This subtask will provide the technical assistance required in connection with reprocessing SM-1 Core I. Technical data on the spent core will be provided for reprocessing facility. Test procedures will be provided for determining the integrity of fuel plate cladding of the spent control rod elements before and after removal of the flux suppressor combs. Technical supervision of the cladding integrity test will be provided. Assistance in preparing SM-1 Core I for shipment to the reprocessing facility will be provided.

Summary of First-Half Results

AEC Form 434, "Fuel Element Data" was reviewed and revised to reflect the latest data available concerning SM-1 Core I.

A trip was made to Neutron Products, Inc. and the University of Virginia to discuss the method of flux suppressor comb removal and to determine responsibilities. A sample of SM-1 spent fuel pit water was taken in order to obtain an estimate of the expected activity level in the spent fuel shipping cask before removal of the combs.

A draft of test procedures was written to determine whether the flux suppressor comb removal process increases the rate of fission product release from the fuel plates. A preliminary equipment list and estimated costs were prepared.

Conclusions

Alco will be responsible for measuring the activity release rate of the control rod fuel elements before and after removal of the flux suppressors.

Recommendations

None.

Future Work

1. Test procedures for sampling the activity release of the control rod fuel elements before and after removal of the flux suppressors will be forwarded to NYOO for comment and approval.

2. The possibility of any heating problems while the elements are being transferred and while in the sampling container water is being checked. The test procedure will be amended, if required, to allow for proper cooling.

3. The testing will be performed at the University of Virginia's hot cell facility.

4. Procedures will be written for sampling the water in the fuel element shipping cask, before shipment to the reprocessing facility. Assistance in shipping Core I to the reprocessing facility will be provided.

5. This work will be a continuing effort under Subtask 1. 8 in FY-62.

ITEM 1. 9 - EVALUATION OF PRECIPITATION HARDENED STAINLESS STEEL

Task Engineer - E. S. Haraway

Task Definition and Objectives

To insure safe operation of SM-1, SM-1A and PM-2A through investigation and resolution of the problem presented by failure of 17-4 PH steel components observed in other reactors.

To prepare reports to assist in AEC Hazards Review of SM-1 operation with Core II, of the SM-1A startup and of PM-2A operation; and to provide the AEC with data on 17-4 PH steel obtained from the SM-1, SM-1A and PM-2A plants.

Summary of Second-Half Results

1. We performed an investigation of all 17-4 PH steel components utilized in the SM-1 prototype, SM-1, SM-1A, and PM-2A which provided a general description of the components, their application and their calculated mechanical and thermal loads. In addition, the manufacturing specifications, procedures and testing were reported.(11)(12) Although no failure of control rod drive components is considered probable and an additional safety feature is available in the form of shutdown margin, a boron injection system is provided which is capable of shutting down the reactor in its most reactive condition with all control rods fully withdrawn.

2. We performed non-destructive and destructive tests on the SM-1 prototype racks and irradiated seal shafts removed from the SM-1. None of the five racks showed any evidence of cracks and all Magnaglo indications proved by sectioning to be harmless subsurface non-metallic inclusions or surface scratches. No deleterious effects had resulted due to straightening by locally applied heat.

   One of the two seal shafts tested revealed stress corrosion cracking had occurred prior to chromium plating but none of the cracks had propagated after the chromium had filled and sealed the cracks. Efforts to reproduce the cause of cracking were successful and the purchase description for future components has been written to eliminate that possibility of stress corrosion cracking during manufacture.

3. Monitored the shipment from the site, testing, re-heat treatment of racks and pinions, re-straightening of racks and return shipment to the site of SM-1A racks and pinions. During the testing several Magnaglo indications were noted but none were confirmed by dye penetrant and ultrasonic examination and most of the indications were removed by a surface polish. It was concluded that the balance of the indications were harmless, sub-surface, non-metallic inclusions which had been demonstrated by the sectioning of an SM-1 prototype rack with a similar indication.

4. Based upon the results of this investigation, a specification for procurement procedure for 17-4 PH steel components was developed and issued as ALCO-P-84. Although the specification was developed for racks, pinions and seal shafts, it can be applied to any 17-4 PH component with reservations as required.

5. Monitored the non-destructive and destructive testing by ORNL of the irradiated rack taken from the SM-1. At ORNL, visual examination at 5x magnification, magnetic particle inspection, liquid penetrant inspection, and metallographic examination were performed on the rack with no cracks being noted.

6. An analysis was made of the problems involved in the identification of failed components. It was noted that although a component failure may not be identified immediately, the delay would not impair the ability to scram. It was noted also that some work has been initiated on the removal of an irradiated pinion and, indeed, the actual removal of an irradiated rack had been accomplished.

Conclusion

In view of the extensive test and operational experience of 17-4 PH components in the SM-1, SM-1A and PM-2A reactor systems with no record of material failure, and considering the relatively low stresses and fatigue cycles imposed on these components, it is concluded that the components do not jeopardize the integrity of these reactors.

Recommendations

1. The integrity of the 17-4 PH steel components in the SM-1 should not be a deterring factor in the continued operation of SM-1 with Core II.

2. The integrity of the 17-4 PH steel components in SM-1A should not be a deterring factor in the SM-1A startup.

3. The integrity of the 17-4 PH steel components in PM-2A should not be a deterring factor in its continued operation.

4. All new 17-4 PH steel components for these reactors be procured in accordance with ALCO-P-84, Purchase Description for 17-4 PH Reactor Components.

Future Work

Three SM-1 irradiated water seal shafts and the spare rack and pinion from PM-2A will be non-destructively tested and the parts returned to their respective plants. Following this testing, a final report,(13) will be issued to complete the task.

ITEM 1.10 - EVALUATION OF SM-1A CORE I ELEMENTS
Task Engineer - J. O. Brondel

Task Definition and Objectives

The objective of this task is to evaluate manufactured SM-1A Core I elements for SM-1A application on the basis of site inspection data, incorporating into the evaluation the current knowledge on fuel plate ripple growth upon continued exposure to reactor operating temperatures.

Summary of Second-Half Results

Inspection of the SM-1A Core I elements was made under ORNL supervision at Fort Greely. Following an evaluation of the inspection data*, Alco recommended

* "Report of Examination of 39 Stationary Fuel Elements at Fort Greely, Alaska," R. J. Beaver, Oak Ridge National Laboratory, August 23, 1961.

to NYOO that 7 stationary elements (2S, 8S, 29S, 34S, 36S, 40S, 47S) be rejected on metallurgical grounds pending more critical examination*. These elements have subsequently been sent to ORNL for examination by ORNL and Alco metallurgists.

A thermal evaluation of the elements was separately conducted by Alco, as reported in AP Note 341,(14) including considerations of initial fuel plate ripple magnitude and potential ripple growth upon exposure to thermal stresses during rated power operation. All 8 control rod fuel elements and 26 of the stationary elements were acceptable for unrestricted placement within the core, but the 7 rejected stationary elements are all from this unrestricted category. The remaining 13 stationary elements were usable provided custom placement within the SM-1 Core 1A loading was used. These findings are summarized in Table 1.1.

TABLE 1.1
CLASSIFICATION AND DISPOSITION OF ELEMENTS EVALUATED

Thermally Unrestricted			Custom Core Placement	
CR2S		30S	3S	24S
CR3S		31S	15S	33S
CR4S		32S	16S	38S
CR5S		34S (R)	17S	43S
CR6S		35S	18S	50S
CR7S		36S (R)	21S	52S
CR8S		39S	23S	
8C	(S)	40S (R)		
		46S		
1S		47S (R)		
2S	(R)	51S		
6S		137S		
8S	(R)			
9S		17S		
10S		18S		
12S		22S		
13S		24S		
14S		39S		
25S		40S		
26S				
27S		S63 (U)		
28S		S83 (U)		
29S	(R)			

(R) - Temporarily rejected for metallurgical considerations pending further examination.

(S) - Spare control rod fuel element.

(U) - Unassigned, stored at Ft. Belvoir.

* Letter, W. S. Brown of Alco Products to New York Operations Office of AEC, attention I. Adler, dated July 11, 1961, file: AE-90-Task 1.10, contract AT(30-1)-2639, subject: Evaluation of SM-1A Core I Fuel Elements Inspection Data.

To obtain replacements for the rejected elements, the Army measured 8 stationary elements available at Fort Belvoir and submitted the data to Alco* for evaluation. All were acceptable for unrestricted core placement in the SM-1A Core I although 2 elements (S63, S83) were not required to complete a full loading complement and remain unassigned at Fort Belvoir.

The recommended SM-1A Core I loading chart[14] is shown in Fig. 1.1 achieving acceptable minimum coolant passage clearances between all fuel elements. On the basis of this chart, a revised SM-1A Core I loading procedure was prepared**.

Recommendations

None.

Future Work

Further examination and evaluation of the 7 temporarily rejected SM-1A Core I stationary fuel elements will be accomplished in FY-62 by an extension of Subtask 1.10. The necessary additional pages to AP Note 378 are being prepared.

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* Letter, I. M. Adler of New York Operations Office, AEC, to W. S. Brown of Alco Products, dated August 9, 1961, file: Ra:IMA:2639/3.1.10, subject: Inspection of SM-1 Spare Elements.

** "SM-1A Core I Loading Procedure TP-46 Revised," K. C. Sontheimer, Alco Products, August 11, 1961.

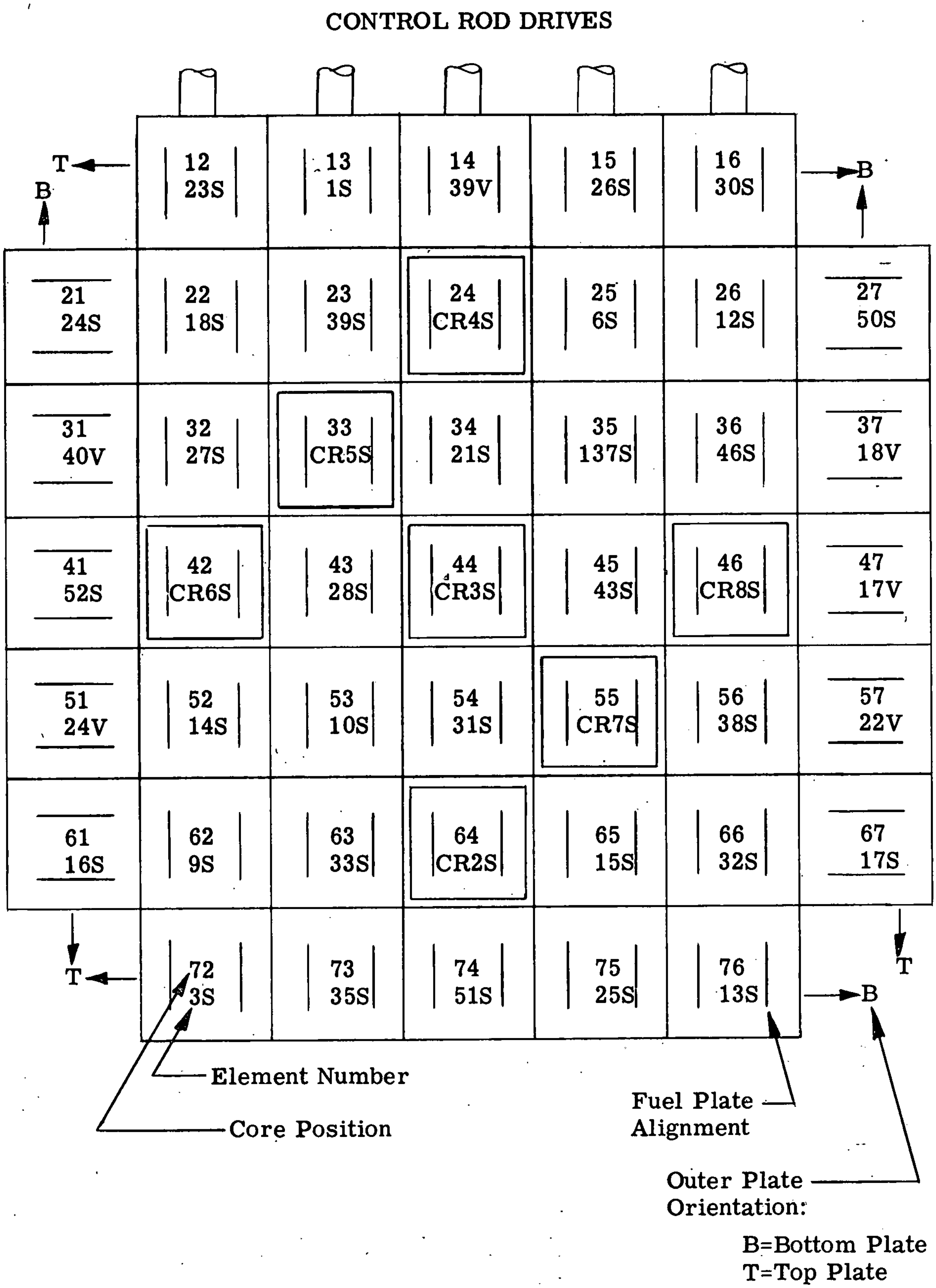


Figure 1. 1 - Recommended SM-1A Core I Loading Chart

BIBLIOGRAPHY - TASK 1

1. APAE Memo-283,"Preliminary Stress and Fatigue Analysis of the Primary System of the SM-1," to be issued December 1961.

2. APAE Memo-300, "Supporting Analysis and Derivation of Dimensional Tolerance Specifications for Core II of SM-1A and PM-2A," November 3, 1961.

3. APAE-94, "Final Report on Fabrication of SM-1A Core II and PM-2A Core II," to be issued.

4. APAE-106, "Criteria for Evaluating Hazards Involved in Proposed Tests on and/or Modifications to the SM-1," October 18, 1961.

5. AP Note-373, "Feasibility Study for In-Place Annealing or Replacing the SM-1 Reactor Vessel," August 9, 1961.

6. APAE-84, "Hazards Report of SM-1 Core II with Special Components," April 30, 1961.

7. APAE-84, Addendum I - "Hazards Report for the SM-1 Core II Without Special Components," April 19, 1961.

8. APAE-84, Addendum II - "Hazards Report for the PM-1-M-2 Element in SM-1 Core II," September 1, 1961.

9. APAE-84, Addendum III - "Hazards Report for SM-1 Core II Without the SM-1 Core I High Burnup Elements and with the PM-1-M-2 Element," October 7, 1961.

10. APAE-84, Addendum IV- "Hazards Report for SM-1 Core II with the SM-1 Core I High Burnup Elements Replaced with SM-1 Core I Spare Elements," October 9, 1961.

11. AP Note-352, "Interim Report Task 1.9 - Investigation of 17-4 PH Steel," June 1, 1961.

12. AP Note-374, "Interim Report Task 1.9 - Investigation of 17-4 PH Steel Components for SM-1A Control Rods, Control Rod Drives and Core Structure," August 4, 1961.

13. APAE-97, "Final Report Task 1.9 - Investigation of 17-4 PH Steel," to be issued.

14. AP Note-341, "Supporting Analysis for Thermal Suitability of Fuel Elements for SM-1A Core I Loading," to be issued November 1961.

15. AP Note-290, "PWR Task 1 Report - A Radiation Damage Study for the SM-1 Reactor Vessel," October 14, 1960.

TASK 2 - PERFORMANCE OF EXISTING CORES

Task Engineer - J. G. Gallagher

Overall Task Definition and Objectives

Evaluate by measurement and analysis the nuclear and thermal performance of existing cores. The cores presently covered under this task are: SM-1 Core I, SM-1 Core I Rearranged and Spiked, SM-1 Core II with Special Components, SM-1A Core I, and PM-2A Core I. This task will provide the basic technology, carefully checked by experiment, for use in the design and development of improved replacement cores.

Introduction

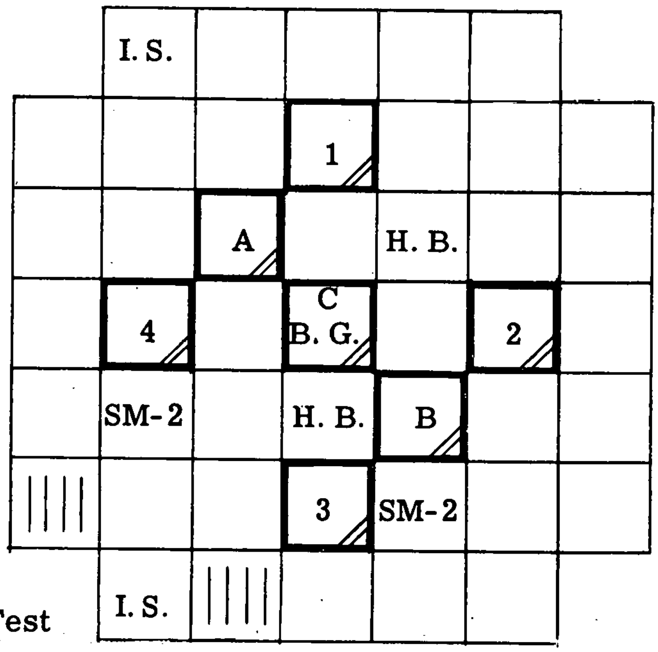
To facilitate reporting of progress on this task the report is broken down by core rather than by subtask numbers as listed in the FY-62 PWR Support and Development Program Plan. The wire scanner evaluation is presented after the discussion of the various cores.

The loading of the SM-1 Core II with special components was completed on June 2, 1961. Low power testing and heatup followed and full power operations started on June 25, 1961.

Figure 2.1 presents some of the important characteristics for the SM-1 Core II with special components which is presently in operation at Ft. Belvoir, Va. It should be noted that the core layout and data illustrated in Fig. 2.1 differes from that previously published in the Mid-Year Summary Report (APAE-86) due to some late revisions in the authorized loading for the SM-1 Core II with special components. As finally loaded the SM-1 Core II with special components did not include the PM-1-M-2 element and the Task XIV instrumented fuel assemblies and the boron gradient absorber. Present planning (October 1, 1961) is for the PM-1-M-2 element to be inserted into, and the two Core I high burnup fuel elements removed from, the SM-1 Core II during the late October 1961 plant shutdown. Insertion of the Task XIV instrumented fuel assemblies into SM-1 Core II has been delayed until the spring 1962 shutdown.

The SM-1A plant is presently estimated to be fueled in November or December 1961. The SM-1A Core I and PM-2A Core I compositions are unchanged from that previously reported[7]; however several of the original SM-1A Core I fuel elements have been rejected and replaced by SM-1 Core II fuel assemblies.

A report[1] covering nuclear technology data of all three existing cores was issued.



Legend

B. G. - Boron Gradient Absorber

SM-2 - SM-2 (Type 3) Test Fuel Elements

H. B. - High Burnup SM-1 Core I Fuel Element

I. S. - Impact Samples

Figure 2.1 - Configuration for SM-1 Core II With Special Components

ITEMS 2.1 AND 2.2 - SM-1 CORE I AND SM-1 CORE I REARRANGED AND SPIKED - PHYSICS MEASUREMENTS

Task Definition and Objectives

To conduct end-of-life core physics measurements on the SM-1 Rearranged and Spiked Core I and to prepare a summary report of all physics measurements performed on SM-1 Core I and the SM-1 Rearranged and Spiked Core I. The characteristics covered will be control rod bank positions, control rod calibrations, stuck rod critical positions, temperature and pressure coefficients, etc. This report will provide a comprehensive source of all physics measurements on SM-1 Core I and related critical experiments. Analysis of these measurements will serve as a basis for establishing the accuracy of core design and analytical methods and for evaluating the benefits of fuel element rearrangement on SM-1, SM-1A and PM-2A.

Summary of Second-Half Results

The SM-1 Rearranged and Spiked Core I was shutdown in April 10, 1961, after 1.6 MWYR operation, for changeover to Core II, Core physics measurements were performed at shutdown.

A comprehensive review was made of all data on SM-1 Core I from TP A-300 series core physics measurements. The summary report of all core physics measurements has been prepared(2) and will be issued in the next quarter. A test report on gamma scanning of SM-1 Core I spent fuel elements has been issued(3).

Figure 2.2 presents a summary of five rod bank critical positions at various core conditions as a function of energy release for SM-1 Core I and SM-1 Rearranged and Spiked Core I.

A. SM-1 Core I Experimental Results

1. SM-1 Core I life was 16.4 MWYR energy release.

2. Temperature coefficient measurements did not indicate any significant change in temperature coefficient during burnup. At 440^{o}F, low xenon concentration, the temperature coefficient is 3.6 $\pm$ 0.1 cents per ^{o}F.

3. The hot-to-cold reactivity change, with low xenon concentration in the core, was \$6.7 $\pm$ 0.5, and, within experimental uncertainty, remained constant with burnup.

4. The average pressure coefficient at low temperature is 1.06 $\pm$ 0.33 cents per 100 psi and at operating temperature is 3.35 $\pm$ 0.68 cents per 100 psi.

5. The integral rod A worth showed little variation with burnup. The integral worth of rod at 440^{o}C is approximately 20% greater than the same integral at 68^{o}F.

6. The total rod C integral worth at 440°F showed a 4.4% decrease compared to the same integral at 68°F and a decreasing integral worth with core burnup.

7. Rod C and rod 3 integral worth curves, obtained during zero power experiments, show a decrease in rod worth as poison is added to the core and the bank withdrawn.

8. The worth of the five rod bank increases with burnup in the upper regions of the core. In the interval from 13 to 22 in., the bank worth increased from \$5.2 at 0 MWYR to \$9.0 at 16.4 MWYR.

9. The five rod bank integral worth at 0 MWYR and 68°F is \$27.4 ± 1.0 from the fully inserted to the fully withdrawn.

10. The seven rod bank integral worth at 0 MWYR and 68°F is \$34.0 ± 2.0.

11. The integral worth of rods A and B as a bank at 0 MWYR and 68°F is \$6.6 ± 2.2.

12. The shutdown margin for the five rod bank plus rods A and B at 0 MWYR and 68°F is \$10.2.

13. The "80% stuck rod shutdown" requirement was met throughout core life.

14. The most reactive core condition met the "80% stuck rod shutdown" requirement by - 30 cents.

15. Peak xenon concentration was reached 7 to 9 hr after power reduction and had decayed to equilibrium concentration after a total of 19 to 21 hr.

16. Least squares fits of transient xenon reactivity data indicate:

    a. The reactivity worth of equilibrium xenon relative to low xenon changed from -325 at 0 MWYR cents to -309 cents at 16.4 MWYR.

    b. The reactivity worth of peak xenon relative to equilibrium xenon changed from -133 cents at 0 MWYR to -157 cents at 16.4 MWYR.

    c. The reactivity worth of peak xenon relative to low xenon changed from -458 cents at 0 MWYR to -466 cents at 16.4 MWYR.

17. The relative axial neutron flux distribution curves indicate that as the bank is withdrawn from the core, the location of the axial flux peak moves upward.

18. At a constant energy release, the peak-to-average flux ratio decreased as the core temperature increased.

19. The peak-to-average flux distribution ratio decreased with core burnup.

20. Source multiplication data shows an increase in shutdown margin with lifetime and that the worth of rods A and B appeared to remain constant with core life at $6.5 $\pm$ 0.2.

21. The excess reactivity of the core based on critical water height and worth was $17.2. This is considerably lower than the value of $23.8 obtained by integrating the five bank rod calibration curve from the cold clean critical bank position to 22 in.

B. SM-1 Rearranged and Spiked Core I Experimental Results

1. The SM-1 Rearranged and Spiked Core I operated for 1.6 MWYR for a Core I total energy release of 18.0 MWYR, at which time it was shutdown for core changeover.

2. The estimated total potential energy release of the SM-1 Core I and the Rearranged and Spiked Core I was 19.1 MWYR assuming operation to complete burnout.

3. The reactivity increase due to Core I rearrangement and spiking was $2.7.

4. The removal of the defective PM-1-M element and substitution of the SM-2A element had a negligible effect on core reactivity.

5. The integral rod A worth under cold, low Xe conditions shows an increase of approximately $1.0 over similar measurements made on Core I prior to rearrangement. The integral rod A worth at hot, low Xe conditions shows an increase of $0.9 over similar previous Core I measurements.

6. Rod C shows a greater integral worth than that measured in Core I prior to rearrangement.

7. Within the rather large experimental uncertainties there was no change in temperature coefficient in comparison to the measurements made prior to Core I rearrangement and spiking.

8. Five rod bank calibrations at startup of the Rearranged and Spiked Core I (16.5 MWYR) are similar to the 0 MWYR measurements on the original Core I. Calibrations at the end of life of the Rearranged and Spiked Core I (18.0 MWYR) show an increase in bank worth of $1.2 from 16.5 MWYR in the interval from 13 to 22 in. This increase in bank worth with core burnup is in agreement with the measurements made on Core I as a function of burnup, prior to rearrangement and spiking.

9. The location of peaks in the relative axial neutron flux distribution curves obtained in the rearranged and spiked core were in agreement with the data obtained from Core I prior to rearrangement and spiking. However the peak-to-average flux ratios were lower than the ratios obtained previously, indicating a flattening of the neutron flux distribution in the Rearranged and Spiked Core I.

10. Transient xenon reactivity data was in agreement with the data obtained on Core I prior to rearrangement.

    a. Peak xenon concentration was reached 7.5 hr after power reduction and had decayed to equilibrium concentration after a total of 20 hr.

    b. Peak xenon relative to equilibrium xenon had a negative reactivity of $1.39.

    c. Equilibrium xenon relative to low xenon had a negative reactivity of $2.97.

Conclusions

The temperature coefficient measurement techniques were not accurate enough to detect the slight changes in temperature coefficient predicted by calculations. (Reference Task 2.3).

Recommendations

The following suggestions for further work should be considered for the purpose of further clarifying and evaluating the data presented in the Summary Physics Report. Recommendations on improving present core physics measurements and techniques which would add greatly to the usefulness of the data are also presented.

1. Investigate possible methods for obtaining more accurate temperature coefficient measurements. Temperature coefficient could be measured during heatup, maintaining criticality with the calibrated five rod bank or with a single calibrated control rod. The bank or rod would be calibrated during the heatup as a function of temperature. In this manner the calibrations would be obtained for the actual core conditions.

2. Investigate obtaining temperature coefficients at power, with equilibrium xenon concentration in the core, using a method similar to that used at PWR*.

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* Gray, J. E., et al, "Shippingport Operations, From Startup to First Refueling, December 1957 to October 1959," DLCS-364.

3. Investigate the difference in excess reactivity of the core obtained by critical water height measurements and by the integral bank worth. This would involve an investigation of the techniques used at other facilities as well as experimental checks at the Critical Facility.

4. Consider installation of improved temperature measuring instrumentation including provision for calibration prior to core physics tests.

5. Investigate bank calibration techniques for determining the bank worth over its full length of travel.

6. Rod A calibrations as a function of various B-10 poison loadings. This would be similar to the rod C and rod 3 calibrations performed in ZPE-1 and could be performed at the Alco Critical Facility in Schenectady. This experiment would aid in separating the burnup, bank position, and temperature effects on the rod A calibrations obtained at the SM-1.

7. Maintain an accurate log book of all maintenance, calibration, and adjustments on the Δt integrator.

8. Scintillation spectra and scans should be taken periodically of the reference element S-52 for the purpose of evaluating the fuel burnout distribution in the SM-1 Core I, and to permit correlation of previous gamma scanning data obtained at the end of Core I life.

9. Provide a training program for the military operating crews to explain the reasons for various measurements, the techniques used and the need for strict adherence to the test procedures.

10. It would also be desirable to investigate the possibility of inserting a new BF_3 chamber into one of the instrument wells during the source multiplication experiment. This chamber would be removed during other reactor operations and saved strictly for use during source multiplication experiments in order to try to improve this data over core life.

Future Work

This concludes the experimental work on SM-1 Core I and SM-1 Rearranged and Spiked Core I. The summary report of all physics measurements performed has been completed and will be issued in the next quarter.(2)

FIVE ROD BANK POSITION AS A FUNCTION OF ENERGY RELEASE
AT VARIOUS CORE CONDITIONS
SM-I CORE I
SM-I REARRANGED AND SPIKED CORE I

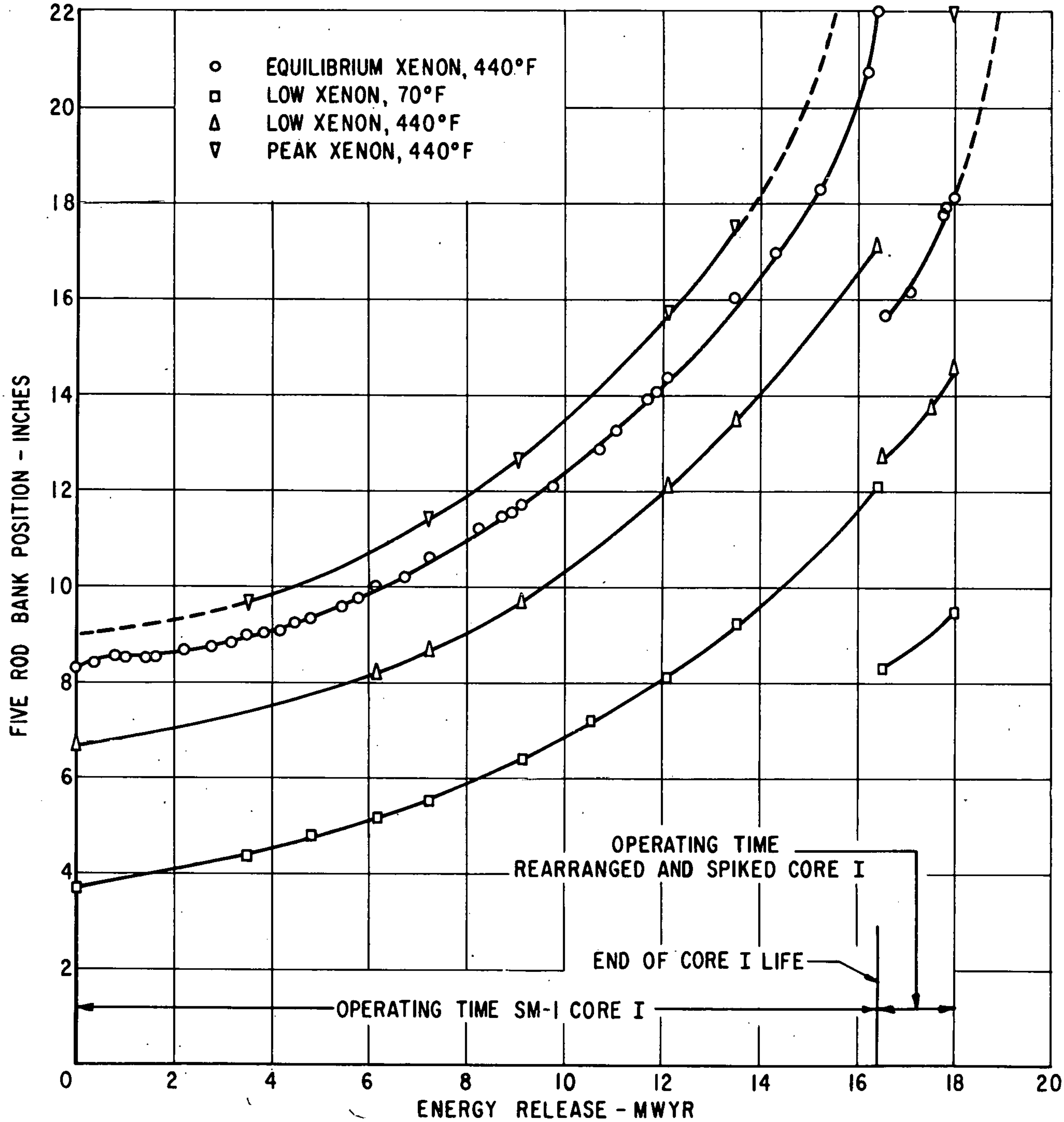

FIGURE 2.2

ITEM 2.3 - ANALYSIS OF PHYSICS MEASUREMENTS

Task Definition and Objectives

Prepare a summary report of physics analysis and comparison to experiment for SM-1 Core I. The items to be covered will be selection of best fast and thermal neutron analytical model, core characteristics at startup, as a function of burnup and core lifetime. This report will conclude the analytical development of SM-1 Core I and comparison with experimental data and demonstrate that similar replacement cores can be predicted with adequate accuracy.

Summary of Second-Half Results

The BOBCAT Code for the IBM-650 was completed and issued.(4)

Review of Analytical Models

The selection of the best analytical model for the SM-1 core was based on comparison of the results of various models with known experimental core reactivities. Thermal constants are obtained by employment of the P_3* theory, using cross sections averaged over a Maxwell-Boltzmann distribution evaluated with a hardened thermal MUFT-III code**. The effects of various cross-section files and slowing down models (used to calculate fast group constants) on core reactivities and neutron ages were analyzed. In addition a brief review of the effect of the analytical model used to calculate thermal group constants was made.

It has been found that the use of P_3 theory with Maxwell-Boltzmann hardened cross sections offers a good compromise between calculated and experimental core reactivity.

The MUFT-III files§ used in the past by Alco were compared to the newly available MUFT-V files¶. In addition the use of the P-1 SG approximation was compared to the straight P-1 approximation. Various critical experiment core reactivities were calculated and compared to measured values. It was found that best agreement was obtained using the MUFT-III fast files and P-1 slowing down model. The chief difference between the MUFT-III file and the MUFT-V file is the use of a higher eipithermal absorption cross section for Fe. In the MUFT-III file the absorption cross section of Fe is adjusted to give the measured resonance integral of 2.1 barns. The results obtained are presented in Table 2.1.

* Byrne, B. J. "Two-Dimensional P_3 Calculation for APPR-Type Fixed Fuel Elements" AP Note No. 96, Alco Products, Inc. Feb. 14, 1958.
** Rosen, S. S. Supplement to MUFT-III Code" AP Note No. 90, Dec. 6, 1957.
§ Bobe, P. E., (Editor),"Interim Report of Nuclear Analysis Performed on SM-2 Core and Vessel Sept. 1, 1958 Dec. 31, 1959,"APAE-65, May 27, 1960.
¶ Henry, A. F. "54 Group Library for P-1 Program",TM-224,April 1960.

TABLE 2.1
REACTIVITIES OF SM-1 CORE I

Cold Conditions	Exp* (%ρ)	MUFT-III P-1(%ρ)	MUFT-III P-1 SG(%ρ)	MUFT-V P-1(%ρ)	MUFT-V P-1 SG(%ρ)
Cold clean (68°F)	17.0	16.67	15.00	19.13	17.69
Hot clean (440°F)	12.1	12.87	10.87	16.00	14.00

* Based on an effective delayed neutron fraction of 0.0078 and cold (68°F) experimental five rod bank worth curve.

The measured values were reported in dollars; these were converted into reactivity by the following equation.

$$\rho = 1 - e^{-\beta_{eff} I_{ex}}$$

where I_{ex} is the excess reactivity (Dollars), and β_{eff} is the effective delayed neutron fraction. It is seen that the cold and hot reactivities are accurate to within about 0.3%ρ and 0.8%ρ respectively, using MUFT-III files with the straight P-1 approximation. The measured and calculated cold-to-hot reactivity changes were 4.9% ρ and 3.8% ρ respectively.

Core Burnup Calculations

The total lifetime of the SM-1 core was recalculated by use of the CANDLE-2* IBM-704 code. The final result was 17.4 MWYR compared to a measured value of 16.4 MWYR. An initial decrease in core reactivity and a subsequent slight increase were detected and attributed to Sm^{149} buildup. The Sm^{149} buildup and reactivity vs. burnup is shown in Fig. 2.3.

Xenon Reactivity Over Core Life

A literature search and review of the basic analytical model used in xenon reactivity calculations was made. The result was a revised thermal microscopic cross-section of 2.3 - 2.6 x 10^6 barns as well as reduction in the non-uniform spatial distribution factor, α, from 1.30 to 1.134. Fig. 2-4 shows the comparison of calculated and measured reactivity worth of equilibrium and peak xenon over the life of the core. At 12.1 MWYR detailed calculations of the xenon transient were performed and compared to measurement (see Fig. 2.5).

* Marlowe, O.T. and Ombrellars, P.A., "CANDLE-A One-Dimensional Few Group Depletion Code for the IBM-704", Addendum 1 - CANDLE-2, WAPD TM-53 Addendum 1, October 1957.

Buildup of U-236

The CANDLE-2 code was used to calculate U-235 buildup where a capture cross-section of 4 barns was employed for the U-236. At 10.5 MWYR the calculated average atom density is 1.8×10^{19} atoms/cm^3. Comparison to experiment can be made whenever data becomes available.

The central rod worth was determined using the VALPROD** code, as a function of the composition of the core. The rod worth for a boron-stainless steel poisoned, and a stainless steel poisoned core were calculated. An integral central rod worth increase of the stainless steel over the boron-stainless steel core was found to be 0.54% ρ while the measured value is 0.75% ρ.

Temperature Coefficient

A simple model for predicting reactivity temperature coefficient has been developed and applied to SM-1 Core I. The calculated temperature coefficient was within the experimental error of the measured values and showed a slight increase from -3.2¢/°F at 0 MWYR to -3.4¢/°F at 16.4 MWYR (see Fig. 2.6).

Special Analysis Items

The following analysis was completed by Dr. R. L. Murray, Consultant: Effective delayed neutron fraction, core lifetime, xenon poisoning, temperature coefficient, flux distribution and rod bank calibration. The basis of the studies was a mathematical model that accounted for simultaneous fuel and boron consumption and the motion of the control rod bank. The fast leakage and the epithermal features of the core were incorporated while the thermal leakage was ignored. This system is amenable to desk calculator work and, even if the results are not as accurate as those of more sophisticated and elaborate techniques, it provides a better insight of the physical content.

The ratio of β_{eff}/β, computed for several stages of burnup, has been found to exhibit a minimum at approximately 13.6 MWYR but the value is always close to 1.20, therefore with $\beta = 0.0065$* we obtain $\beta_{eff} = 0.0078$.

The calculated lifetime utilizing these methods was 15.6 MWYR and the equilibrium xenon worth varies from 2.42% ρ at 0 MWYR to 1.20% ρ at 15.6 MWYR.

The temperature coefficient was found to decrease slightly with burnup from -3.5¢/°F at 0 MWYR to -2.9¢/°F at 15.6 MWYR.

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* ANL 5800 Reactor Physics Constants - Argonne National Laboratory.

** Oby, P. V. "Modified Two-group Multiregion Calculation Using the VALPROD Code for the IBM 650", AP Note 24 Revised, ALCO Products, Inc., August 14, 1957.

Conclusions

1. The use of MUFT-III files with the straight P-1 approximation together with the thermal properties calculated by P_3 theory, using cross sections averaged over a hardened Maxwell-Boltzmann distribution, offers the best agreement with measurements. The cold-to-hot reactivity change for SM-1 is underestimated by as much as $1\% \rho$.

2. Excellent agreement between calculation and experiment on xenon reactivity worth and xenon transients can be obtained if the proper xenon cross-sections are used.

3. The U-236 buildup as a function of burnup is nearly linear, as would be expected, due to the low absorption cross-section of U-236.

Recommendations

1. Analysis should be performed to determine the effect of using four-group rather than two-group theory on nuclear predictions.

2. Experimental measurements of the microscopic intracell thermal flux distribution should be made (in a core with a comparable hardening index) and these used as a basis for deciding whether or not it would be advantageous to employ the variation of thermal spectrum with position, and higher order approximations, to the transport equation to obtain more accurate predictions of the thermal flux distribution.

3. Further analysis is required to verify some of the assumptions made such as the constant thermal absorption cross-section with burnup, and the constants of the rod thermal absorption with burnup.

Future Work

A final analysis report[5] will be issued in the next period to complete the work on this item.

EXCESS REACTIVITY AS A FUNCTION OF ENERGY RELEASE
SM-I CORE I-440°F. AT EQUILIBRIUM XENON

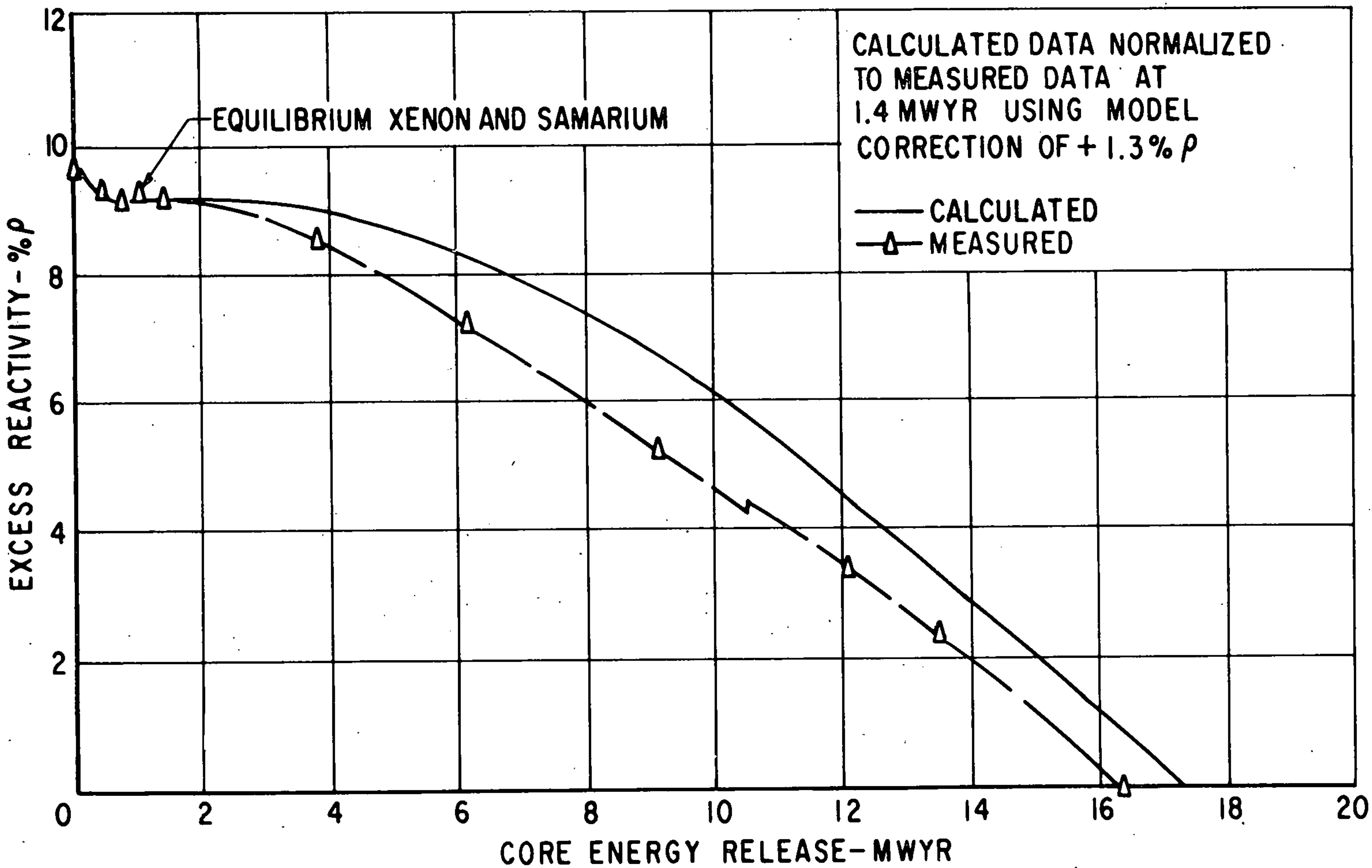

FIGURE 2.3

SM-I CORE I XENON REACTIVITY VS. LIFETIME

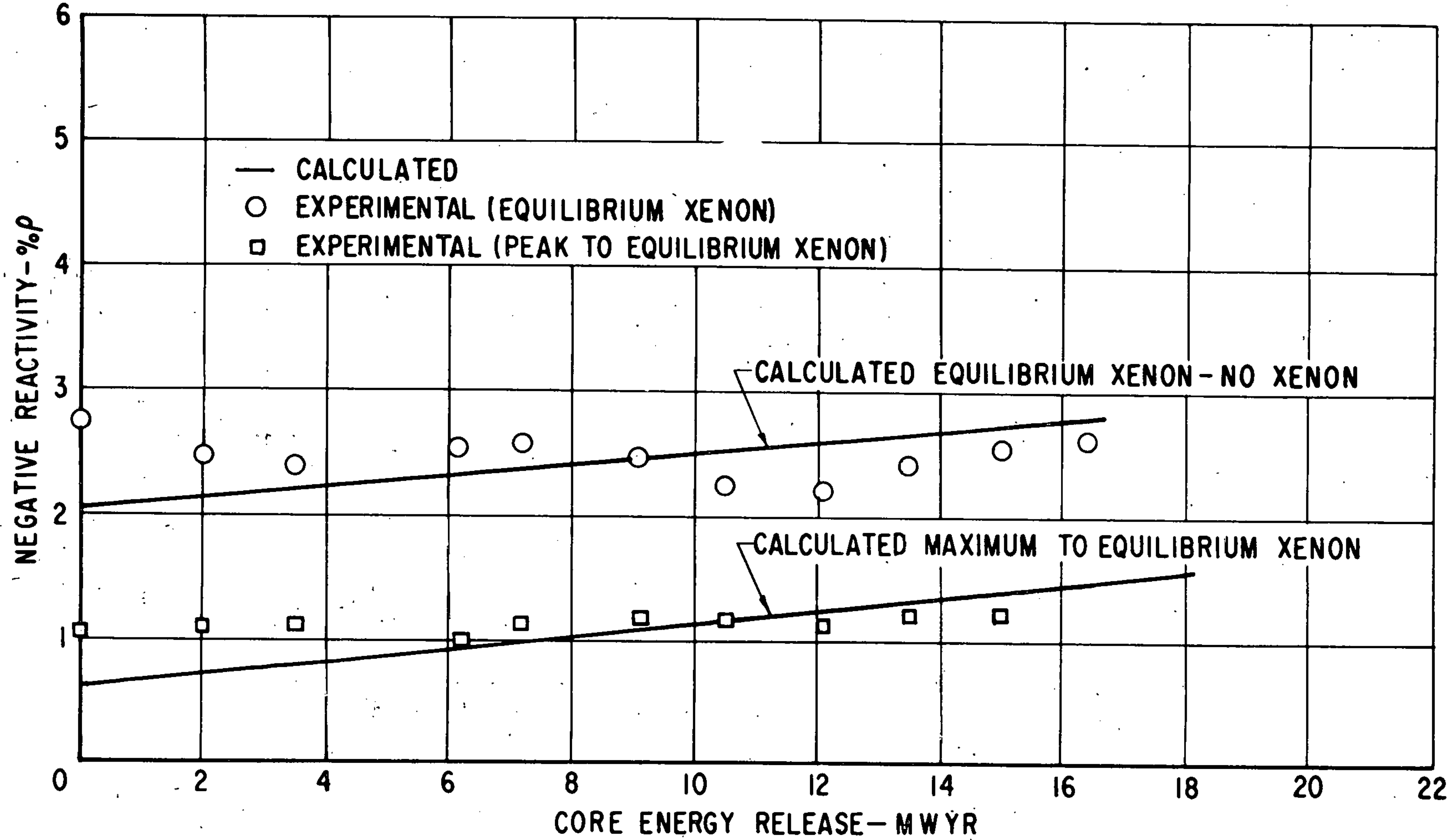

FIGURE 2.4

TRANSIENT XENON REACTIVITY IN SM-1 CORE-I AT 12 MWYRS

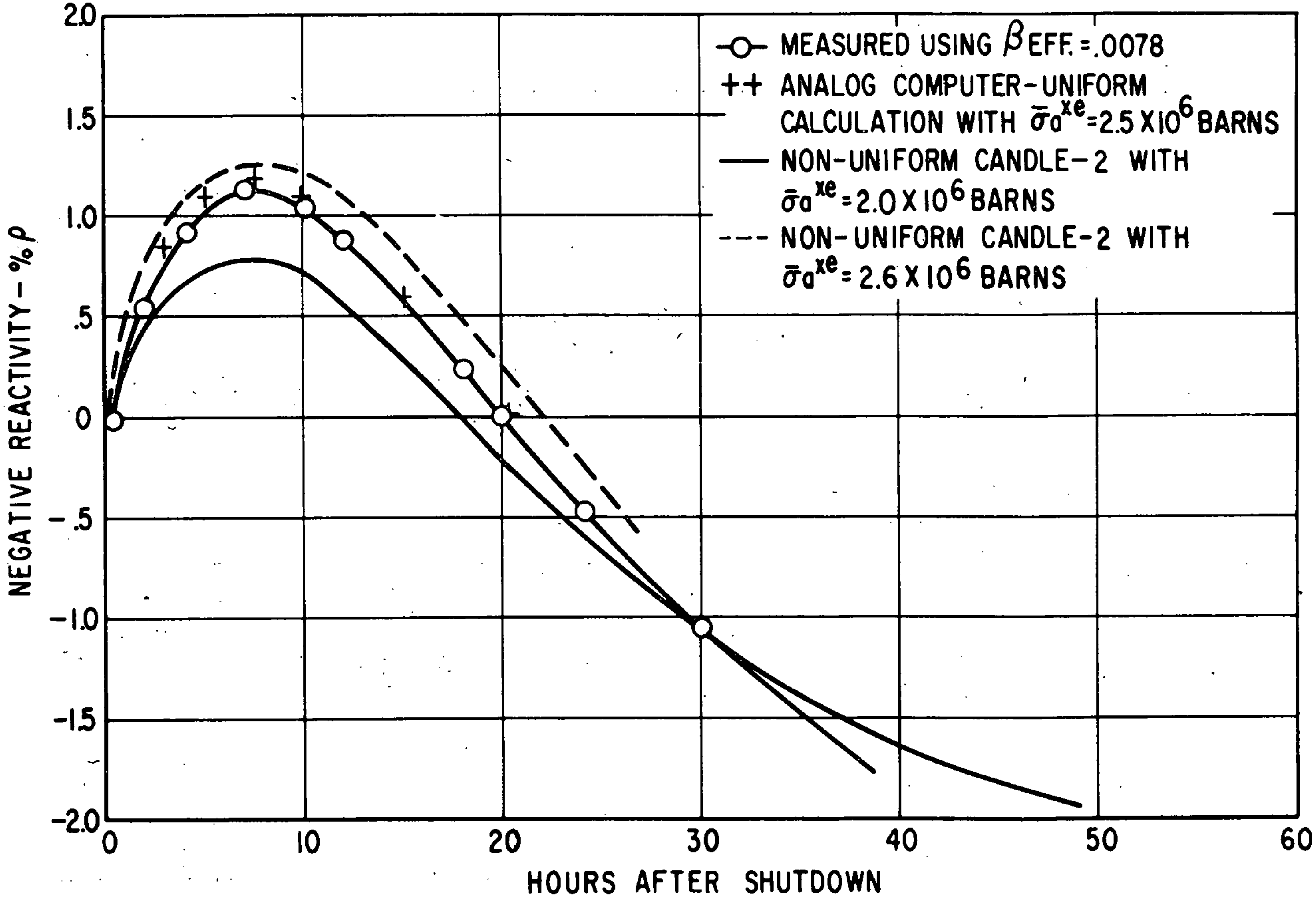

FIGURE 2.5

TEMPERATURE COEFFICIENT AT 440°F. AS A FUNCTION OF ENERGY RELEASE

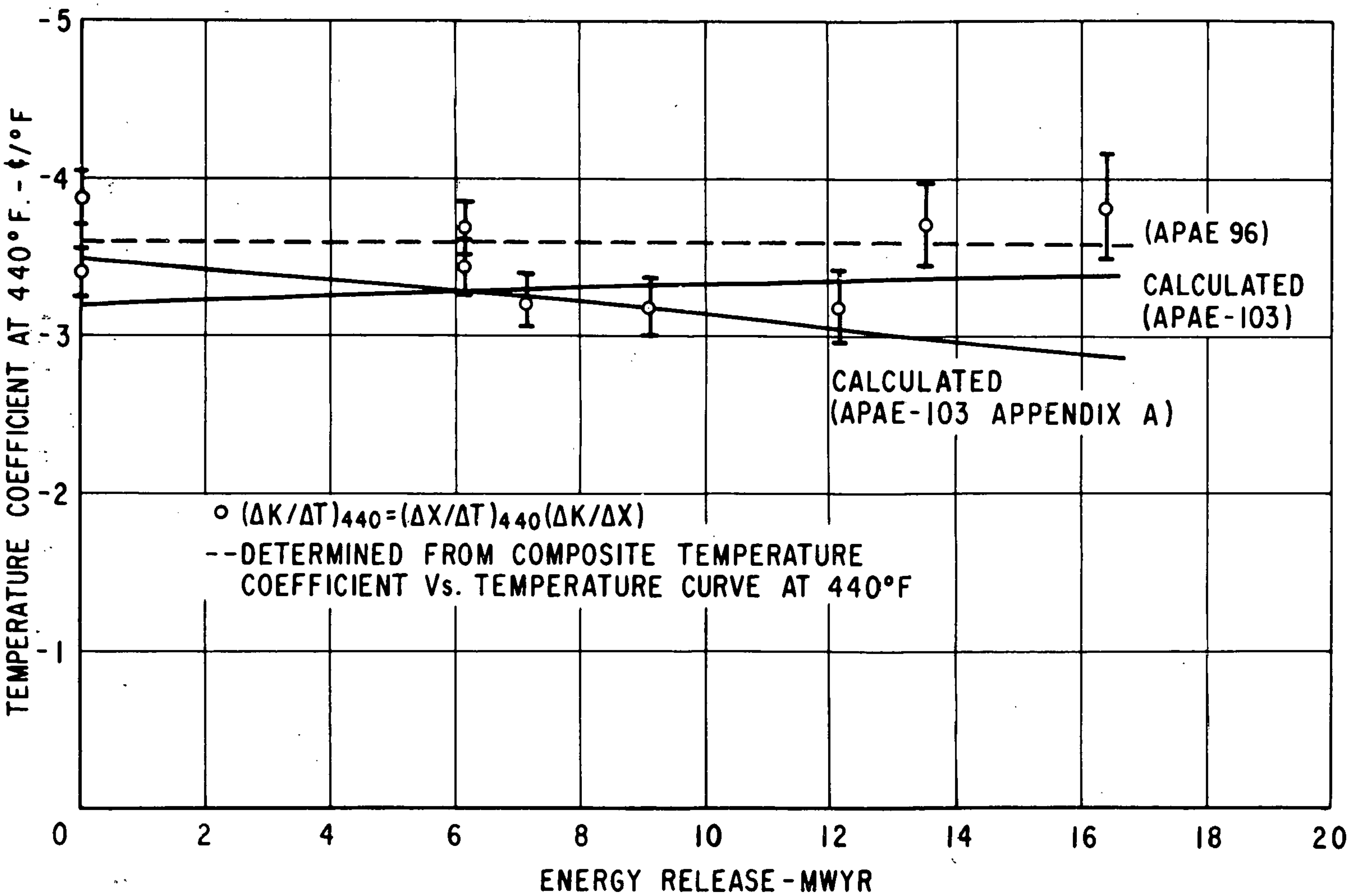

FIGURE 2.6

ITEMS 2.4 AND 2.8 - SM-1 CORE II WITH SPECIAL COMPONENTS - PHYSICS

Task Definition and Objectives

The first objective is to load the fresh Core II fuel elements into the SM-1 plant in a safe and orderly manner and with minimum time delay. All pertinent nuclear hazards involved in these operations are to be considered and provisions made for adequate precautions either by direction or reference to existing SM-1 procedures.

The second objective is to perform various core physics measurements and nuclear analysis on the SM-1 Core II with special components for the purpose of establishing the various core parameters at the beginning of SM-1 Core II life. These data, plus that obtained from SM-1 Core I, will form a basis for evaluation of the effects that the special components in the SM-1 Core II have upon core performance and life expectancy.

Summary of Second-Half Results

A detailed report(6) on the zero power experiments and analysis of SM-1 Core II with Special Components was issued.

The SM-1 Core II was satisfactorily installed during this period. The work and results were reported previously in the June Progress Report. *

A second set of SM-1 Core II Loading Procedures were issued to cover the removal of the two SM-1 Core I high burnup elements and the insertion of two replacement fuel elements during the October 1961 shutdown.

Upon completion of the SM-1 Core II fuel loading the TPA-300 test schedule was initiated. Several preliminary results have been reported in the June progress report.

Reduction and analysis of the startup data has been completed and will be included in the SM-1 Core II Startup Test Report which is to be issued in the next quarter.(7)

Figure 2.7 presents the xenon buildup and decay as a function of time after shutdown. The data is plotted relative to equilibrium xenon. The data shows very close agreement to that reported early in Core I life. **

A five rod bank calibration curve was obtained employing the period technique and is presented in Fig. 2.8.

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* AP Note 379.

** SM-1 (APPR-1) Research and Development Program Interim Report on Core Measurements, Task No. VII, APAE Memo No. 178, March 1, 1959.

Startup calibrations were performed during each shutdown period. These measurements indicate that the startup channel operating charactersitics drift significantly. However, operation is considered satisfactory so long as the startup channels are calibrated prior to all reactor startups.

Conclusions

1. SM-1 Core II with special components was satisfactorily loaded and tested and is operating according to ZPE predictions.

2. For all practical purposes, there appears to be no change in the reactivity effects of xenon or temperature as a result of the core change-over.

3. There is a distinct change in the five rod bank calibration as a result of the core change-over. This change is attributed to two effects:

   a. Utilization of integral flux suppressors at the top of the control rod fuel elements of the SM-1 Core II, which in effect extends the absorbing region of the control rod assembly compared to the SM-1 Core I and consequently displaces the rod calibration curve slightly.

   b. Substitution of two SM-1 Core I high burnup elements and two SM-2 low burnup elements in place of four regular SM-1 Core II elements, which alters the neutron flux distribution inside the Core II as compared to the SM-1 Core I.

Recommendations

1. A more detailed test program should be scheduled to investigate the change in rod worths as a result of the special components in the Core II. It is suggested that this program be fit in with the schedule for removing the SM-1 Core I high burnup elements and include calibrations of Rod A and Rod B individually with and without the various special elements. The difference in the rod calibrations thus obtained would then be used to determine the effect of the special components in Core II.

2. It is recommended that the startup channels be calibrated prior to every reactor startup in order to insure that sub-critical neutron multiplication observed is not being over-shadowed by high gamma fields. It is also suggested that action on the installation of a lifting mechanisms for the startup channels be accelerated in order to minimize the factors causing uncertainties in the startup channel operations.

3. A major effort should be placed on modifying or improving Test A-311 (Temperature Coefficient). In general, the differential temperature coefficient of reactivity appears to have the largest uncertainty of the

core physics data and is one of the most important core parameters when considering core lifetime and control. It is also very important in the design of second and third generation field plants which operate at higher temperatures.

Future Work

1. Because of uncertainties in the metallurgical condition of the two SM-1 Core I high burnup fuel elements, they will be removed from Core II during the October shutdown. Two new Core II elements will replace them.

2. Core Physics Procedure TPA-323, "Five Rod Bank Position as a Function of Energy Release" will be performed routinely without interfering with normal plant operation. Special nuclear testing will be proposed to evaluate the core reactivity effects associated with the special components in Core II and to be initiated during the October shutdown.

3. This work will continue in FY-62 as Subtask 2.1.

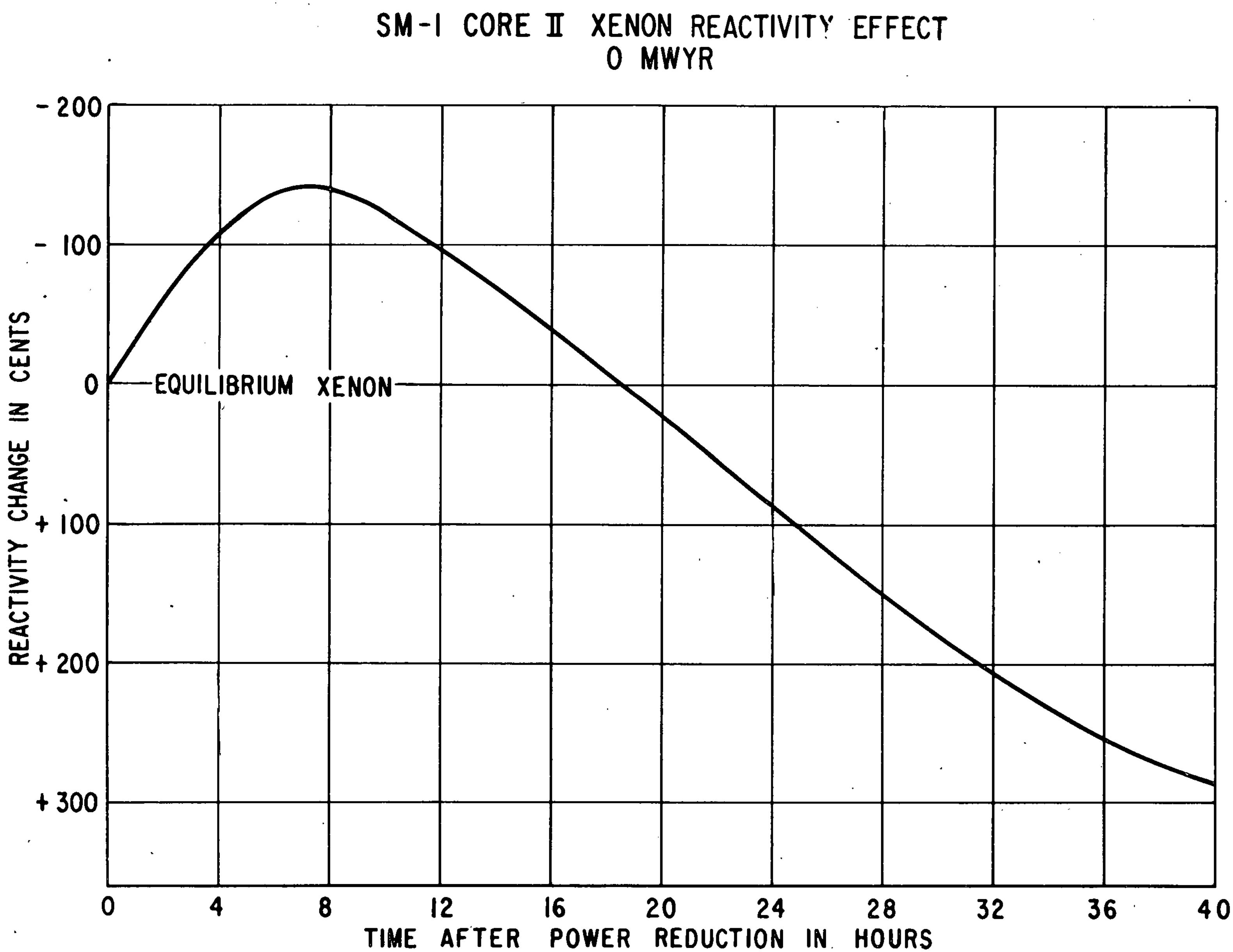
SM-1 CORE II XENON REACTIVITY EFFECT
0 MWYR
REACTIVITY CHANGE IN CENTS
-200
-100
0
+100
+200
+300
EQUILIBRIUM XENON
0
4
8
12
16
20
24
28
32
36
40
TIME AFTER POWER REDUCTION IN HOURS

FIGURE 2.7

SM-I CORE II FIVE ROD BANK CALIBRATION
0 MWYR

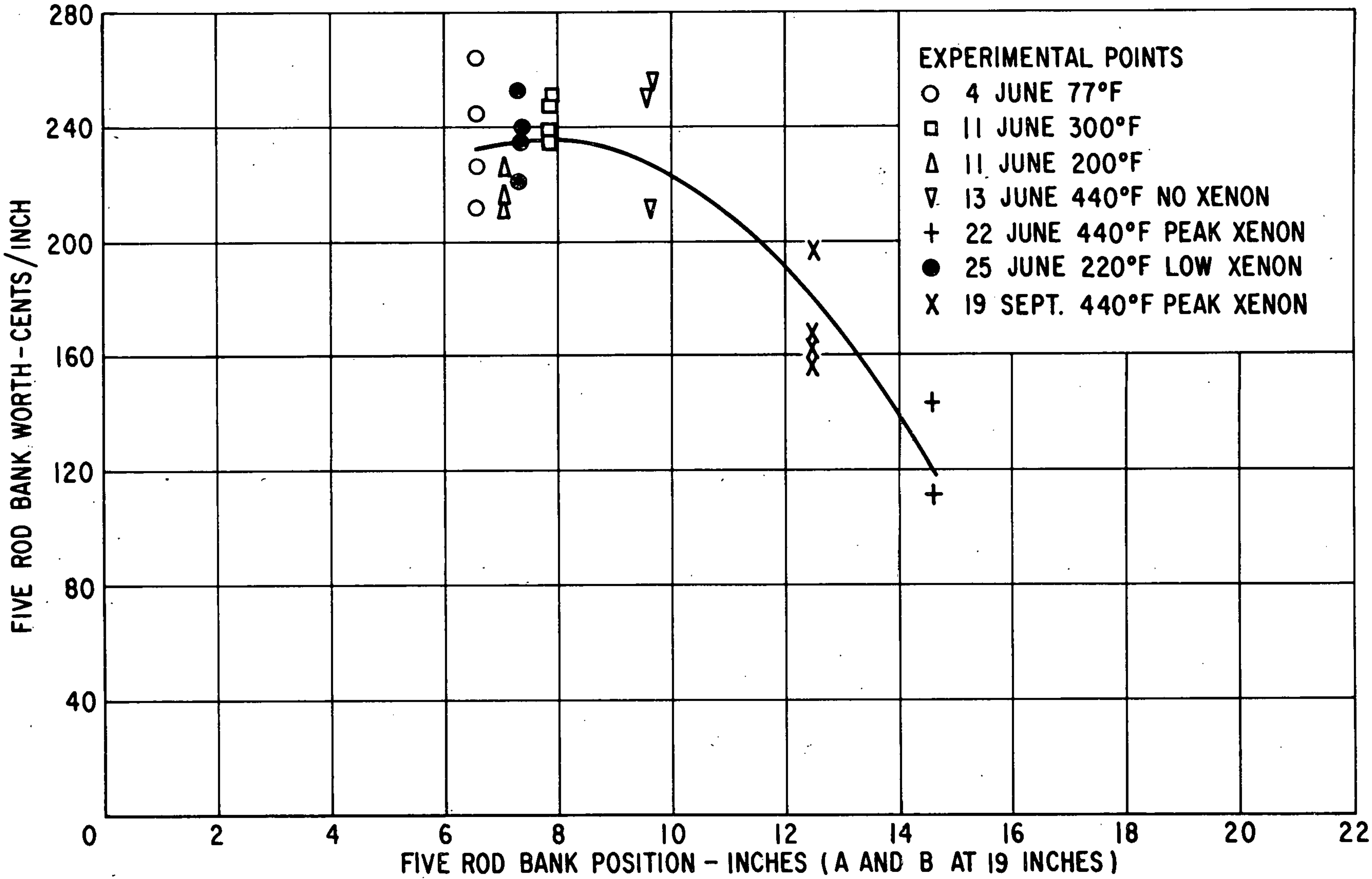

FIGURE 2.8

SM-1 CORE II WITH SPECIAL COMPONENTS - THERMAL

Task Definition and Objectives

The purpose of Task XIV is to measure various temperatures and flow rates directly within the operating core of the SM-1 reactor, developing such instrumentation, associated equipment, and fabricating techniques as are necessary. Provision will be made for inducing local core boiling by restricting coolant flow through an instrumented control rod. Observed temperatures and flow readings are to be compared with corresponding design values.

This task was previously funded under Contract AT(30-3)-326, which expired June 30, 1961. Present funding of this task is under Contract AT(30-1)-2639 as part of Task 2. 0.

Summary of Work Completed Under Contract AT(30-3)-326

1. A design was completed for providing in-core instrumentation for the SM-1 reactor.

2. The instrumentation assembly, including penetrated gasket, instrumented table top, instrumented stationary fuel element and instrumented control rod assembly was fabricated and is presently stored at Alco Products, Inc., Schenectady.

3. Various supporting equipment such as vapor container penetrations, purge system, read-out instrumentation has been designed and purchased.

Summary of Second-Half Results (Period July 1 to September 30)

1. A hazards analysis(8) of the Task XIV Penetrated Gasket has been completed and issued, evaluating the effects of a postulated failure of the as-built gasket.

2. Replies* to NPFO and TEB comments on AP Note 279,(9) Temperature and Flow Measurements - Installation, Operation and Test Procedure, have been completed and forwarded to NYOO. A rough draft of the revised procedure has been completed and is presently being reviewed by NPFO. The revised procedure will be issued in the next qaurter.

3. A review of the Task XIV purge system has been completed to determine what changes will be required to provide automatic control of the system in the eventuality of SM-1 training operation with Task XIV installed.

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* Letter M. H. Dixon to I. M. Adler, AE-90 Task 2, Task XIV, 21 September 1961.

4. A report[10] has also been issued on the effect of local boiling of SM-1.

Conclusions

1. Results of the hazards analysis indicate that no accident beyond the safe containment capacity of the SM-1 plant can be postulated as a result of braze failure in the penetrated gasket, and with close attention to the system it is not hazardous to operate with the gasket.

2. As a result of the Task XIV purge system review, it is evident that manual control of purge system pressure may be cumbersome during training operations. The system will be modified to include constant pressure differential control for purge system regulation.

Future Work

1. A complete hazards analysis of the 600 series tests and the related test procedures will be issued in the next period.

2. Task XIV instrumentation is scheduled for installation in April 1962.

3. Future work on this task will come under FY-62 Subtask 2.5.

ITEMS 2.5 AND 2.9 - SM-1A CORE I

Task Definition and Objectives

Measure the physics characteristics of SM-1A Core I and the thermal and hydraulic characteristics of the SM-1A Core I and primary system. The physics measurements will be used as the basis for replacement core development. The thermal and hydraulic measurements will be used to verify original design methods and provide a basis for replacement core development and investigation of off-design performance.

Summary of Second-Half Results

The "Test Schedule and Test Procedures for SM-1A Nuclear Power Plant at Startup"[11] was issued. Various comments and suggested revisions were incorporated into this document following a review by the AEC and the SM-1A resident engineer.

The test program described in detail in AP Note 304 Revised has been postponed until after plant acceptance.

An IBM code for calculating pressure drop and flow distribution[12] has been issued.

Recommendations

The special SM-1A Core I nuclear test program should be performed at the earliest possible date following initial plant startup. In order to provide a complete series of control rod calibrations, critical bank positions, and core reactivity effects under varying core conditions and to permit a thorough nuclear analysis of the core performance at the high thermal rating of the SM-1A. These tests provide nuclear data useful in establishing operating core parameters which are factored into the replacement core designs, and when combined with end of life measurements, provide parameter studies as a function of core life.

Future Work

1. Following initial plant startup, routine measurements of critical bank positions under various core conditions (TP B-323) will be initiated to provide data for monitoring the SM-1A Core I performance. These tests will not interfere with the normal plant operations.

2. Reduction and analysis of test data will be initiated.

3. This work is scheduled as Subtask 2.3 in FY-62.

ITEM 2.6 - PM-2A CORE I - PHYSICS

Task Definition and Objectives

Perform physics measurements during startup of PM-2A Core I at Camp Century, Greenland, in order to establish important core parameters. The measurements are to include bank position, stuck rod critical positions, rod calibrations, xenon reactivity, and temperature coefficient. These measurements, when analyzed, will provide a basis for the design of higher performance replacement cores.

Summary of Second Half Results

Core physics tests performed on the PM-2A have been evaluated and typical data reported in the Mid-Year, April, May and June progress reports. Qualitative and summary conclusions are itemized below while the more detailed and quantitative results are to be included in a combined report[13] with thermal and hydraulic, shielding, and radiochemistry data (Items 2.10, 4.4, 5.1 and 5.2).

Conclusions

1. The PM-2A BF_3 startup channel performance is satisfactory; however, in the event of large voltage or gain drifts or significant changes in counter characteristics due to radiation, and/or temperature effects, the calibration range of voltage control is not entirely adequate. With optimum voltage and gain settings, the startup channel response is approximately 2 - 5 cps with cold clean core conditions.

2. Based upon measurements obtained following the acceptance test, it is estimated that the startup channel count rate will be approximately 25 cps following full power operations of a duration long enough to bring the Ba^{140} - La^{140} fission product pair to equilibrium concentration.

3. Due to the increased beryllium block size in the PM-2A plant, the photo neutron emission rate from that application is approximately four times greater than the similar SM-1 installation.

4. Rod calibrations indicate the integral worth of the center rod is approximately $11.40 and that of an eccentric control rod approximately $6.85 under cold clean conditions.

5. The temperature coefficient agrees favorably with previous measurements at SM-1. It ranged from -0.75 ¢/°F at 100° to -3.00 ¢/°F at 400°F; above 400°F, the temperature coefficient increased sharply to more than 5.0 ¢/°F at 500°F.

6. The pressure coefficient was measured at an average core temperature of 198°F and is worth +0.011 cents/psi, agreeing favorably with previous SM-1 data.

7. Stuck rod measurements illustrated the ability to maintain criticality on any single rod under cold clean conditions. The critical position of the center rod was 16.29 in., that of an eccentric rod was 19.50 in.

Recommendations

1. Startup channel calibrations should be performed prior to all reactor startup to insure proper operation and also to provide a record of any tendency for the calibrations to drift. If a significant drift tendency is observed, action should be initiated t extend the voltage and gain range of adjustment.

2. Additional data should be obtained on temperature coefficient in an attempt to improve the accuracy of the startup test data. This forms the basis (along with SM-1 information) for evaluating the effect of high coolant temperature on system performance and core life.

3. Routine measurements of the five rod bank position as a function of energy release should be obtained on site and forwarded to Alco.

4. A special test program and appropriate procedures should be prepared and a series of PM-2A Core I midlife core physics experiments (C-300 series) performed to provide data for evaluation of Core I performance as a function of core life since this plant operates at higher temperature and pressure than previous plants and employs a unique core and control rod configuration.

Future Work

1. A test report covering the PM-2A startup will be issued[13].

2. Appropriate test procedures for routine testing of Core I will be submitted for the use of military personnel on site. These tests will cover shutdown neutron source evaluation (C-322), five rod bank position as a function of energy release (C-323), and stuck rod measurements (C-326).

3. Data reduction and analysis of routine and special measurements will be continued.

4. This work will continue as Subtask 2. 4 in FY-62.

ITEM 2. 10 - PM-2A CORE I - THERMAL

Task Definition and Objectives

Measure the steady state and transient performance of the PM-2A primary system, in terms of pressures, temperatures, flow rates, power and liquid leads, during (1) plant heat-up, (2) full power steady state operation, (3) load transients, and (4) loss of primary coolant flow.

Summary of Second-Half Results

1. The data from the following four tests, performed during the first half year, were analyzed:

   C-601 Load Transient Test
   C-602 Primary System Thermal Survey (Steady State)
   C-603 Primary System Heat-Up
   C-604 Decay Heat Removal Test

As a result of this program, some information is available on the initial performance of this plant that has never been developed on previous APPR-type plants.

2. Certain aspects of plant design or instrumentation which warrant modification, or in some instances merely better calibration, have been identified as a result of this program. Some other design details have been identified, which although not warranting change in the PM-2A, should be changed in subsequent plants.

3. In some instances the tests did not fulfill the objectives of this program because of either:

   a. Non-performance of plant instruments, such as due to inadequacy of the facilities for on-site calibration.

   b. Plant instrumentation not adequate for particular objectives of these tests, and the requirement that this program should not delay the plant in assuming camp load.

   c. The lack of stable power for instruments, immediately following a reactor scram, due to equipment limitations at the time the tests were run.

4. As a result of 2(a) and 2(b), no usable, direct experimental thermal measurement of primary loop power was obtained for comparison with secondary system output. Figure 2. 9 shows the discrepancy. The performance figures developed are therefore based on secondary system measurements, and calculated incidental heat flows.

5. The final data on power requirements for pressurizer heat-up shows a higher margin of installed over required capacity. This is seen by comparing the terminal value of the bottom trace of Fig. 2. 10 (110 kw-hr) with the maximum possible heating in this period -235 kw-hr.

6. Simultaneous, instantaneous readings of the flowmeters for steam and feedwater frequently show large discrepancies - up to 10% even after corrections for fluid densities and the effect of secondary system blowdown. This difficulty can be overcome by selective use of data or by obtaining averaged or integrated readings.

7. In-plant data taken during transient test TPC-601 is summarized in Table 2. 2 and compared with analog data for comparable load changes. When test temperature values are adjusted for the instrument lag the analog data is in close agreement with test data except for the positive pressure change. The analog peak positive pressure changes are $\sim$2 times test while the peak negative pressure changes are in the same order of magnitude.

TABLE 2.2
COMPARISON OF PEAK VALUES OF VARIABLES FROM LOAD TRANSIENT TEST WITH ANALOG CALCULATION VALUES

Load Change	Temp. Change (^{o}F)			Pressure Change (psia)		Volume Change	
	Analog	Adjusted*	Test	Analog	Test	Analog	Test
Test Conditions							
600 to 0 KW	+ 9.5	+9.0	+5.4	+175	+120	+0.77	+0.58
760 to 320 KW	+ 3.2	+3.3	+2.0	+ 52	+ 12	+0.23	+0.30
320 to 675 KW	- 2.0	-3.8	-2.3	- 55	- 69	-0.18	-0.31
0 to 750 KW	- 8.0	-8.0	-4.8	- 78	- 85	-0.68	-0.75
Design Conditions							
100% to 5%	+ 8.5	-	-	+142	-	+0.60	-
0 to 100%	-11.0	-	-	-130	-	-0.95	-

* Value of test adjusted for instrument lag.

This is the same result experienced between SM-1 plant and SM-1 analog simulation. This indicates the analog model is conservative for loss of load transients, while for increase in load transients the model is a good approximation.

8. From the decay heat removal loop test (TP-C-604) a loop velocity of 2 ft/sec and coil performance of 1,350,000 Btu/hr are estimated. The tests show a decrease in primary coolant temperature of 35^{o} during seven minutes. This amounts to about 250 to 300^{o} per hour, which is excessive. The apparent trend, however, is toward a decreased rate after seven minutes. The duration of the test was not sufficient to predict the operation after 10 minutes.

9. The complete report on these tests has been prepared.

Conclusions

1. There are no difficulties associated with bringing the primary system up to operating pressure and temperature. See Fig. 2.10.

2. The fact that primary loop flow rate is 15.4% higher than originally called for (see Fig. 2.11) provides an opportunity to modify operating pressures for improved plant performance or increased life of the reactor pressure vessel as governed by the NDT.

3. Further work is needed on the calibrations of the five temperature measuring elements in the primary loop and their circuits, using more complete calibration equipment than has been available at the site, in order to obtain useful engineering data from them.

4. The flowmeters for steam and feedwater and the feedwater flow controller have very different response rates, resulting in apparent flow unbalance across the steam generator at any given instance. This could be an indication of slight hunting on the part of the feedwater system at this low power level.

5. The results of the transient tests are consistent with, and of the same order of magnitude as, the analog simulation results. Temperature and volume changes are within the accuracy of the instrumentation. During loss of load transients the analog values are conservative by a factor of 2 compared to the test values. This is due to the use, in the analog simulation, of the conservative assumption of adiabatic compression of the steam pocket in the pressurizer, instead of accounting for condensation of steam on the vessel water and water interface. During negative pressure surges (increasing load transients) test pressures are of the same order of magnitude as the analog pressure surge values. Therefore the analog can be used as a good prediction of plant responses to load perturbations.

The decay heat loop tests show the existing coil and auxiliary loop is capable of cooling the primary system about 35^{o}F in seven minutes. This in terms of one hour is approximately 250^{o} to 300^{o} which is excessive. Comparison with the analog indicates that the loop is operating as predicted, however, the duration of the tests was not conclusive enough to predict operation of the loop beyond 10 minutes after pump failure.

Recommendations

1. A study should be made to select the optimum method for taking advantage of the excess of actual primary coolant flow rate over required. For instance this could call for use of lower primary system pressures, or higher temperatures in both systems in order to obtain thermal efficiencies exceeding design values. The current large difference between maximum possible camp load and plant design load should also be taken advantage of in this study.

2. The primary loop temperature instrumentation should be calibrated on site in the operating range with high temperature baths.

3. Test TP-C-601 should be rerun at a later date when full design load is available or at such time approximately 75% of design load is available.

4. Test TP-C-604 should be rerun for a duration of 30 minutes to 2 hours. However, before this can be accomplished the cooldown rate needs to be established with suitable proportions.

Future Work

1. The complete report on plant startup tests will be published in APAE No. 92[13] to complete this item.

FIGURE 2.9
COMPARISON OF POWER MEASUREMENTS
FULL COMPRESSOR LOAD, FEB 10-15, 1961.

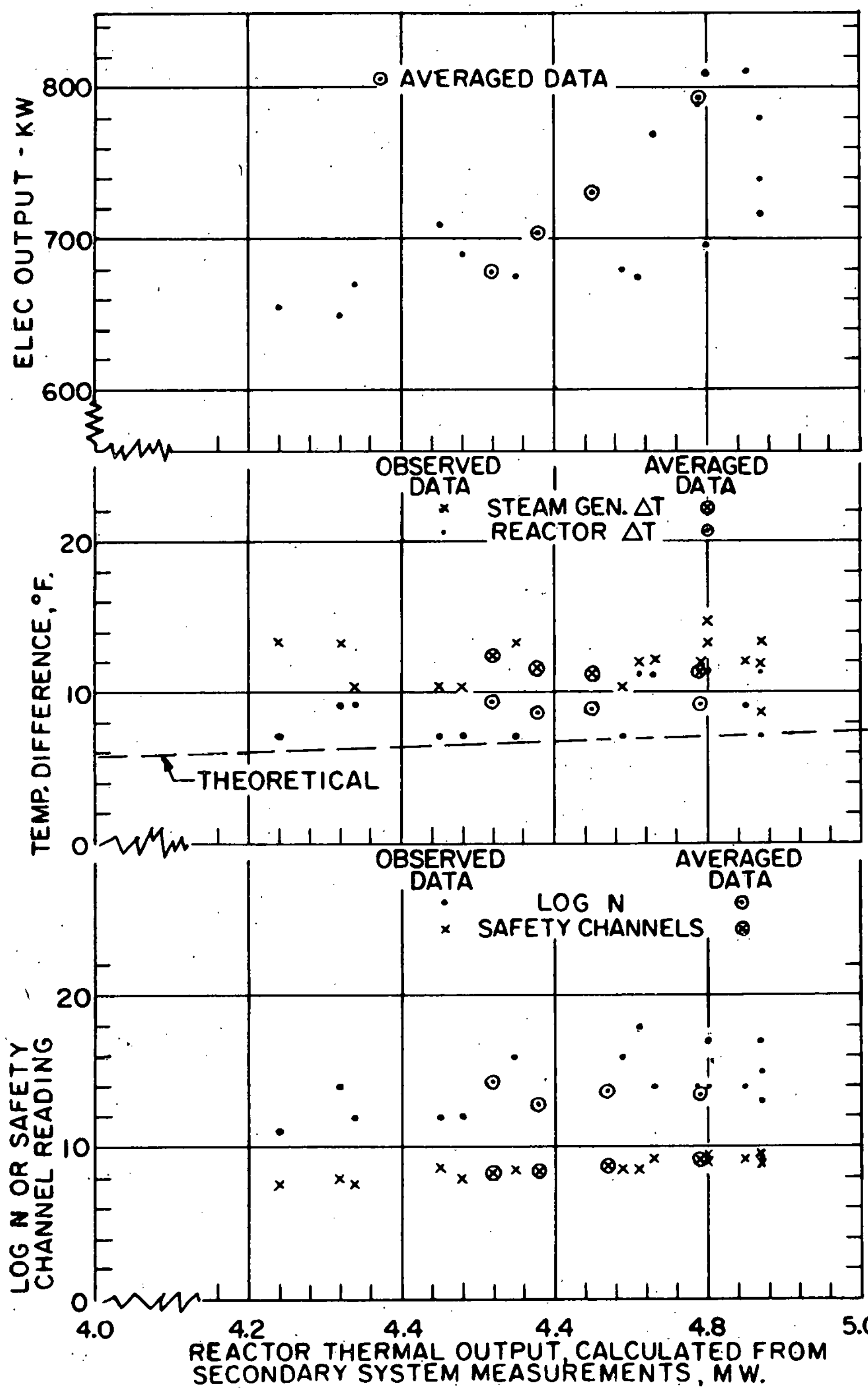


FIGURE 2.10
GRAPHIC LOG OF PRIMARY SYSTEM HEAT-UP AFTER 400 HOUR TEST MARCH 4-5, 1961, 1606 TO 0324 HRS.

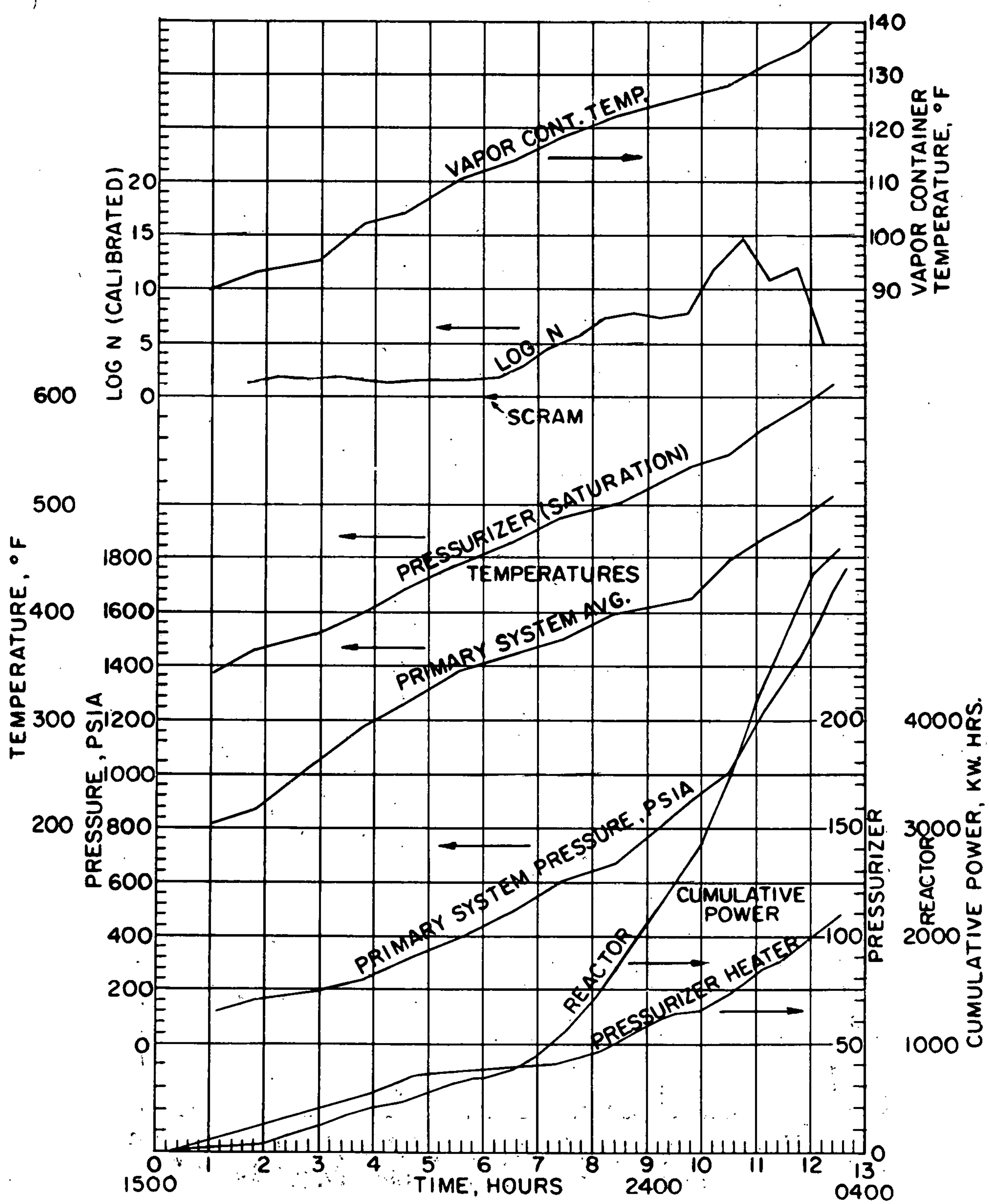
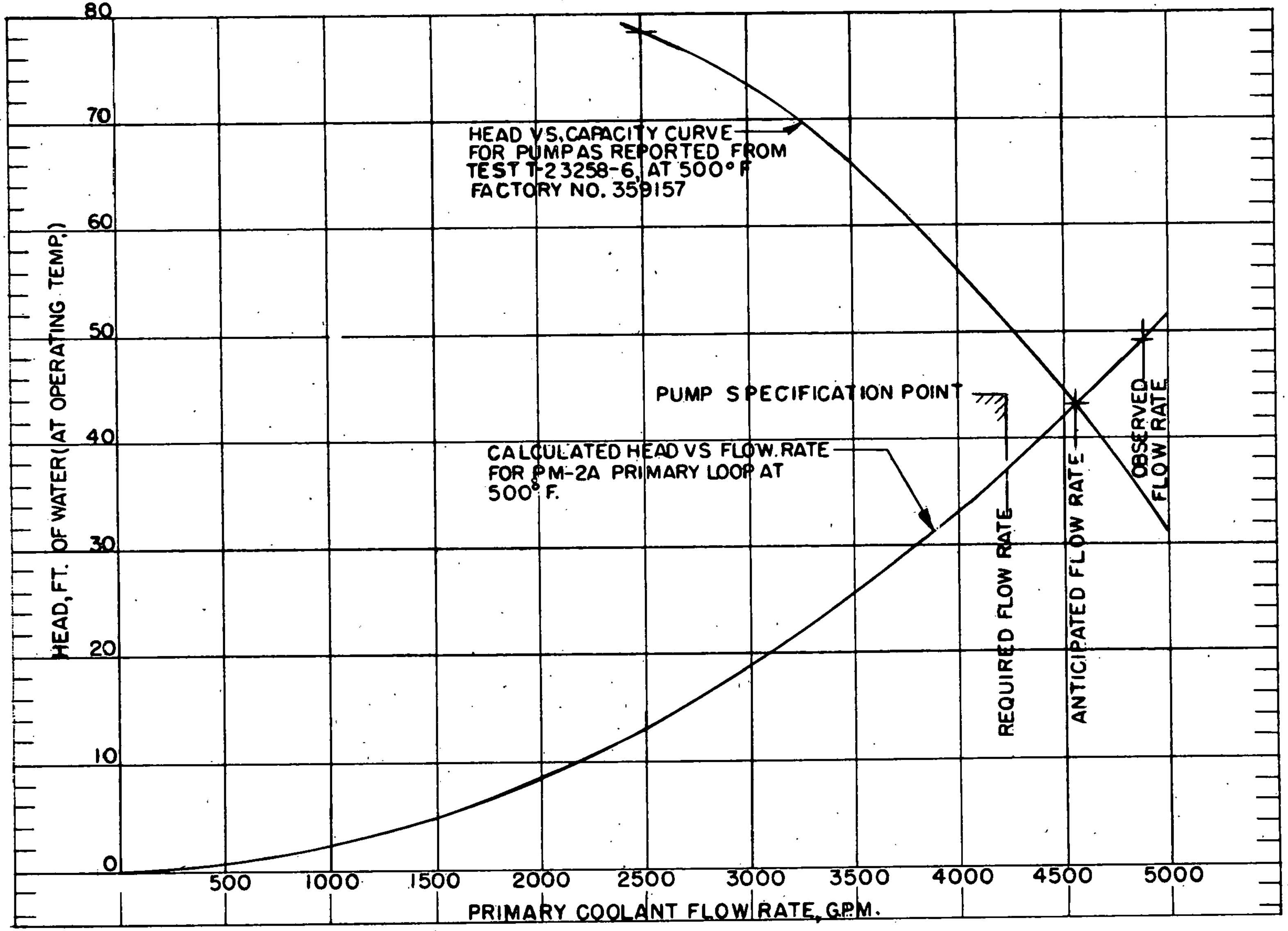


FIG. 2.11

HEAD VS. FLOW CHARACTERISTICS OF PRIMARY LOOP & CIRCULATING PUMP.



ITEM 2.7 - WIRE SCANNER EVALUATION

Task Definition and Objectives

A wire scanner for use at the SM-1 facility is to be evaluated, conceptually designed, and specifications prepared. A comprehensive review of the present status of similar efforts on other reactor projects is to be made with the objective of establishing a flexible and optimum design which will permit rapid neutron flux mapping of hot power reactor cores.

Summary of Second-Half Results

An evaluation of a wire scanner along with conceptual design considerations were completed and incorporated into a final report.[14] Various conceptual methods for instrumenting the fuel elements are discussed, listing the advantages, limitations, and hazards associated with each of these methods. Specifications for the components comprising the wire scanner are included in the report as an appendix.

Conclusions

The following conclusions were drawn:

1. Utilization of the wire scanner with the manual instrumenting method allows measurement of the neutron flux distributions and neutron spectral analysis only at room temperature, with no coolant flow pressure, in regions of low flux only. This can be accomplished only during extended periods of shutdown that last at least 14 working days if one quadrant of the SM-1 is to be mapped at one position per element. In addition, the positions of most importance could not be mapped using this method.

2. Utilization of the automatic wire scanner and the automatic instrumenting method would allow rapid measurement of neutron flux distributions and neutron spectral analysis in preselected elements of the core at any time without a significant effect on the operation of the reactor. This method would supply the most desirable type of information.

3. The neutron flux measurements that would be achieved using the automatic instrumenting method are of value in furthering the state of the art in reactor core design and understanding. These are:

   a. Fuel element corner thermal flux peaking as a function of core life.

   b. Fast and thermal neutron flux vs. core life and burnup through fuel elements.

   c. Shift in power distributions in elements as a function of core life and burnup.

d. Effect on axial power distributions due to xenon transients.

e. Change in neutron energy spectrum as a function of core burnup and spatial distribution.

f. Provide possible insight, in conjunction with appropriate machine calculations, into the problem of rod programming and/or fuel management for longer core life.

4. Commercially available equipment will meet the specifications established for all detection, amplifying, scaling, recording, and data processing equipment.

5. Special design and fabrication attention must be provided for fuel elements to be instrumented, upper core support plate, and pressure vessel cover.

6. The approximate cost of a fully automated wire scanner and instrumenting system including design and fabrication of a new pressure vessel cover and a new core support plate is $150,000 based on preliminary vendor responses.

Recommendations

The following recommendations were made:

1. Considering the value of the data received using the automatic instrumenting method compared to data that would be received using the manual instrumenting method, the location of the wire detectors in the elements and the operating conditions of the reactor as parameters of investigation, it is recommended that consideration be given only to the automatic type of instrumenting system.

2. Consideration of the automatic wire scanner and the basic wire scanner in comparison to the savings in manhours vs. initial cost dictates the recommendation to consider only the automatic wire scanner for use at the SM-1.

3. It is further recommended that an integrated automatic instrumenting system and automatic wire scanner be let for bid in order to ascertain more realistic cost estimates.

4. An immediate recommendation is the design and fabrication of a new SM-1 core support structure to accommodate the automatic wire scanning system.

Future Work

No work is scheduled on this task for the next report period.

BIBLIOGRAPHY - TASK 2

1. AP Note-357, "Nuclear Technology Data for SM-1, SM-1A, PM-2A, SM-2," June 12, 1961.

2. APAE-96, "Summary Report of Physics Measurements on SM-1 Core I," to be issued.

3. APAE Memo-281, "PWR Research and Development Program Test Report - Gamma Scanning Spent SM-1, Core I Fuel Elements: Test 318," April 6, 1961.

4. APAE Memo-289, "BOBCAT (Program #56) - Code Preparation Analysis on the IBM 650," August 9, 1961.

5. APAE-103, "Summary Report of Analysis of Physics Measurements Performed on SM-1 Core I, to be issued.

6. APAE-85, "Experiments and Analysis for SM-1 Core II with Special Components," April 26, 1961.

7. AP Note-385, SM-1 Core II Startup Test Report, to be issued.

8. APAE-79, Supplement No. 1, "Hazards Evaluation of the SM-1 Penetrated Gasket," September 8, 1961.

9. AP Note-349, "Temperature and Flow Measurements - Installation, Operation and Test Procedure," June 30, 1961

10. APAE-88, "Investigation of Local Boiling of SM-1," June 20, 1961.

11. AP Note-304, Revised, "Test Schedule and Test Procedures for SM-1A Nuclear Power Plant at Startup," May 28, 1961.

12. AP Note-343, "Program 85 - An IBM Code for Calculating Pressure Drop and Flow Distribution," April 28, 1961.

13. APAE-92, "Startup and Initial Testing of the PM-2A Plant," to be issued.

14. APAE Memo-296, "Evaluation of Wire Scanner for SM-1," to be issued November 1961.

TASK 3 - REPLACEMENT CORE DEVELOPMENT

Task Engineer - J. G. Gallagher

Task Definition and Objectives

Provide improved replacement core having longer life, improved reliability and safety and lower cost beginning with development stages and progressing through procurement specifications.

Introduction

The development of improved replacement cores for the SM-1, SM-1A and PM-2A is based on the fact that flat plate stainless steel UO_2 dispersion fuel elements can incorporate a high content of fuel and burnable poison in a relatively thin fuel plate. In addition, nature has endowed stainless steel with the property whereby it can be substituted for water in small quantities with a net decrease in neutron age. This fact coupled with the ability of the metallurgist to incorporate high wt % UO_2 and burnable poison in stainless steel results in a system of high performance capability. The technology of flat plate stainless steel fuel elements 30 mils in thickness and containing 26 wt % UO_2 and approximately 0.1% natural boron as boron carbide has been successfully demonstrated by the burnout of SM-1 Core I. The ability of flat plate fuel elements to be made with increased wt % UO_2 (up to 40 wt %) and high wt % of natural boron (up to 0.5 wt %) provides a technological basis for the design of replacement cores for Army reactor plants with improved performance.

The development of replacement cores is facilitated by the designation of core types. Table 3.1 lists the four core types presently under development in this program. Type 1 cores are now in operation at SM-1 and PM-2A and a Type 1 core is presently at Fort Greely awaiting startup of SM-1A. A Type 2 core has been completed for PM-2A Core II and one is in fabrication for SM-1A Core II. The replacement core program in the last year has been devoted to the finalization of Type 3 core specifications for the three plants. The status of this development is discussed in the subtasks which follow.

To facilitate long range core development planning, a document(1) was prepared covering status of technology, problems (metallurgical, physics, and thermal) by core type, plant core schedule and development program.

Table 3.1

CORE TYPE AND CHARACTERISTICS

DESCRIPTION	1	2	2'	3	5
Plate thickness, mils	30	30	30	40	70
Fuel	Geneva UO_2	Spherical UO_2	Spherical UO_2	Spherical UO_2	Spherical UO_2 or UN
Loading in matrix, w/o UO_2	26	26	~26	25	38 or 34
Poison type	B_4C	B_4C	B_4C or ZrB_2	ZrB_2	?
Loading in matrix, w/o B	0.08	0.09	~ 0.17	0.17	~ 0.42
Cladding, SS type	304	347	347	347	?
Assembly method	braze	braze	braze	weld	?
Loading in 45 elements, kg U-235	22.5	22.5	22.5	36.3	~119
Loading in 45 elements, gms B-10	15.8	17.8	33.8	56.8	~295
Energy release (440°F), MWYR	16.4	16.4	16.4	28	~115
Avg. a/o U-235 burnup, in core	36	36	~36	39	~ 48
Max. a/o U-235 burnup, in core *	~54	~54	~54	~59	~ 72
Avg. a/o burnup in matrix	2.1	2.1	~ 2.1	2.2	~ 4.2
Atom a/o U-235 in matrix	5.5	5.5	~ 5.5	5.3	7.9
Atom a/o B-10 in matrix	0.09	0.09	~ 0.17	0.19	~ 0.46

* Maximum burnup based on ratio of maximum to average burnup of 1.5 measured at ORNL on SM-1 Core I elements (preliminary)

ITEM 3.1 - CRITICAL EXPERIMENTS FOR USE OF TYPE 3 ELEMENTS IN SM-1, SM-1A and PM-2A

Task Definition and Objectives

Evaluate the use of Type 3 (SM-2) fuel elements in SM-1, SM-1A and PM-2A by series of critical experiments conducted in the exact core configurations of the field plants under startup core conditions. SM-2 mockup fuel elements are to be utilized throughout the experimental program. Data is to be obtained on:

1. Comprehensive power distribution to determine core average and location and magnitude of local power peaks.
2. Material worth measurements.
3. Initial critical control rod bank positions.
4. Control rod calibrations.
5. Critical rod configurations.
6. Temperature coefficients.

Summary of Second-Half Results

The objectives of the experimental program outlined above were obtained for SM-2 (Type 3) mockup elements substituted into SM-1, SM-1A and PM-2A configurations. A complete task report(2) will be issued in the next quarter. A summary of measurements and results is presented below.

The SM-1 configuration with SM-2 mockup elements contains 36.4 kg of U-235 and an estimated 67.9 gm B-10. The initial seven rod bank critical position at 68^{o}F was 8.04 in.; the five rod bank with rods A and B fully withdrawn was 6.94 in. The worth of a stainless steel skirt surrounding the core was -46 cents. The average measured material coefficients for U-235 and B-10 were 0.141 cents/gm and 23.76 cents/gm, respectively. Operation at 440^{o}F and 1200 psi causes a loss of core reactivity of 7.3 dollars, leaving an excess reactivity available of approximately 10.1 dollars. The temperature coefficient at 440^{o} was approximately -3.9 cents/^{o}F. The seven and five rod (A and B fully withdrawn) critical bank positions at operating conditions were 11.31 in. and 10.45 in., respectively. Power distribution measurements indicate peak internal power generation of 4.19 times the core average occurred near the center of the core while the minimum of 0.10 of the core average occurred at the top edge of the core. Maximum cell average power generation in the active core was 1.32 times the core average while the minimum was 0.75 of the core average. Various critical rod configurations were determined and tabulated.

The SM-1A configuration is almost identical to the SM-1; therefore only a limited experimental program was carried out. The seven and five rod (rods A and B at 19 in.) critical bank position at 68°F was 8.12 in. and 6.90 in., respectively. The worth of the stainless steel skirt that surrounds the core was -40 cents. Peak internal power generation was 4.30 times the core average and occurred in the center of the core. Minimum power of 0.10 of the core average occurred at the edge of the core. Maximum and minimum average power generation per fuel cell in the active core were 1.33 and 0.66 times the core average, respectively.

The PM-2A configuration with SM-2 mockup elements contains 30.048 kg U-235 and an estimated 56 gm of B-10. The initial five rod bank position at 68°F was 8.56 in. The worth of the stainless steel skirt was -51 cents. The measured average U-235 and B-10 reactivity coefficients in the core were 0.154 cents/gm and 30.19 cents/gm, respectively. Control rods 1 and 2 were calibrated over their entire length of travel with the three rod bank at 8.447 in. Critical rod configuration measurements indicated that at the beginning of life, criticality cannot be attained by full withdrawal of any single control rod. Extensive power mappings of one quadrant of the symmetrical core were made. Peak internal power and minimum power occurring respectively at the center and edge of the core were 3.67 and 0.10 times the core average. Highest cell average power generation in the active core was 1.30 times the core average while the lowest was 0.80 of the core average.

Conclusions

1. Measurements of power distribution,material worth coefficient, bank positions, stuck rod positions, and temperature coefficient were completed using SM-2 Core I mockup plates in the exact startup configuration of SM-1, SM-1A and PM-2A.

2. The measurements will provide the information required to perform the nuclear analysis of the Type 3 cores at startup in the SM-1 and SM-1A configuration.

3. The measurements indicated that SM-2 Core I boron loading resulted in inadequate excess reactivity for Type 3 elements in the PM-2A configuration. As a consequence of this the boron loading for Type 3 cores in the PM-2A configuration must be reduced below that present in SM-2 Core I.

Recommendations

1. Additional samples of boron bearing Mylar tape from the SM-2 Core I mockup should be analyzed chemically to reduce the uncertainty in regard to boron content of the mockup cores.

2. Improved experimental techniques such as the use of dysprosium foils should be employed to provide a direct indication of the thermal flux variation in mockup cores employing burnable poison in the form of Mylar on the surface of fuel plate.

3. Initial mockup test to support the utilization of a PM-2A size (37-element) Type 3 core as SM-1 Core III.

Future Work

1. Midlife mockup experiments of Type 3 cores in SM-1, SM-1A and PM-2A configurations will be erected at the Alco Critical Facility. Measurements will be made of the critical bank positions, and stuck rod positions. Individual and control rod bank calculations will be performed.

2. Related work will continue in FY-62 under Subtask 3.5.

ITEM 3.2 - ANALYSIS OF POWER DISTRIBUTION, CORE BURNOUT, AND ROD OPERATION USING SM-2 FUEL ELEMENTS IN THE SM-1, SM-1A AND PM-2A

Task Definition and Objectives

Determine the nuclear characteristics of replacement cores for the SM-1, SM-1A and PM-2A utilizing Type 3 SM-2 fuel elements. The analysis will include power distributions, core burnout behavior, and rod operation and has as its objective the safe operation of SM-2 elements in all plants.

Summary of Second-Half Results

Comparisons between the measured and analytical power distributions for the SM-1, SM-1A and PM-2A cores at 68°F with Type 3 mockup elements have been completed. Correction factors (i. e. F_{θ} , F_T, and F_T) to account for the differences between the calculated and measured data have been developed from these comparisons. Final calculations of the most adverse power generation factors and radial and axial power distributions for each of the cores have been completed. Core burnout or lifetime calculations for the SM-1, SM-1A and PM-2A cores containing elements with SM-2 Core I uranium and boron loadings[3] have been initiated and are near completion. Control rod operation considerations were completed.

Because they define the hottest spots in the core, the parameters Q $(\Delta\theta)$ and F_θ are most significant.

Q $(\Delta\theta)$ is the power generation rate at the hottest spot in any element in the hot, reference core (i. e. Q $(\Delta\theta)$ = P $(x, y)_{\text{max calc.}}$ F_θ).

F_θ, the correction factor for Q $(\Delta\theta)$ is defined as:

$$\frac{\text{P(x, yz) measured at hottest spot in experimental element}}{\text{P (measured to a core average of 1. 0)}}$$

$$F_\theta = \frac{\text{P (Z) max calc P (x, y) calculated at hottest spot in experimental element}}{\text{P (calculated to a core average of 1. 0)}}$$

The value, P(x, y) calculated, is obtained from the IBM-704 PDQ-3 code. This power peak value is usually underestimated by 10 to 12 percent when calculated by the PDQ-3, however the axial power value, P $(Z)_{\text{max calc.}}$, is usually overestimated by 10 to 20 percent. The general range of values obtained for F in the SM-1 and SM-1A with Type 3 mockup elements was 0. 90 to 1. 10. In the PM-2A with Type 3 mockup elements the general range for F_θ was 1. 00 to 1. 10.

Table 3. 2 lists the values of the power generation factor at 0 MWYR and midlife for the hottest element in each core. These values were obtained from "no xenon" cases and are assumed to be conservative for that reason.

TABLE 3. 2
RADIAL POWER GENERATION FACTORS FOR THE SM-1, SM-1A AND PM-2A
(Hottest Elements; 0 MWYR and Maximum Values)

Core	Element	F_θ	$Q(\Delta\theta)_o$	$Q(\Delta T)_o$	$Q(\Delta T)_o$	$Q(\Delta\theta)_B$	$Q(\Delta T)_B$
SM-1	31	. 99	1. 98	1. 47	. 87	2. 09	1. 55
SM-1A	16	. 97	1. 97	1. 21	. 85	2. 27	1. 40
PM-2A	44	1. 00	1. 77	1. 51	. 68	2. 05	1. 75

The maximum-to-average axial power ratios with integral flux suppressors included were calculated to be 1. 91, 1. 91, and 1. 43 for the SM-1, SM-1A and PM-2A cores respectively. The most adverse maximum-to-average power generation in each of the cores is 3. 99, 4. 34, and 3. 92 respectively.

Lifetime calculations using the CANDLE-2 IBM-704 code for the SM-1, SM-1A and PM-2A core utilizing SM-2 Core I, Type 3 elements have been initiated. The core lifetime of a Type 3 core in an SM-1 plant is 31.5 MWYR. Calculations for the SM-1A and PM-2A applications are continuing. SM-1 lifetime calculations were made using a seven rod control rod bank.

The major considerations in choosing the number of control rods to employ as the bank are (1) the effect of the rod bank on the axial power distribution and (2) stuck rod considerations. Early recommendations have indicated that a 5-rod bank produces a somewhat lower axial maximum than that due to operation with a 7-rod bank. From stuck rod configurations it is preferable to always have all rods partially inserted into the reactor core. On the basis of the advantage from the stuck rod standpoint a recommendation is made that SM-1 and SM-1A be operated with a 7-rod bank when Type 3 cores are installed.

Conclusions

1. Final calculations of the most adverse power distribution using Type 3 elements with SM-2 Core I loadings for the SM-1, SM-1A and PM-2A have been completed. These calculations included correction factors based on comparisons of calculated and measured power distributions in mockup cores of each reactor.

2. The initial lifetime calculations of Type 3 cores in SM-1 have been completed, and work is continuing for Type 3 cores in the SM-1A and PM-2A plants.

3. A seven rod bank was chosen for operation in the SM-1 and SM-1A cores primarily on stuck rod considerations.

Recommendations

1. A detailed analysis of rod programming and fuel management schemes should be made in order to improve the power distribution and lifetime of the various cores.

2. Two-dimensional burnup codes should be employed in order to more accurately predict the variation of power distribution and reactivity with core life.

Future Work

1. Determine loading for the midlife mockup experiments for Type 3 cores in SM-1, SM-1A and PM-2A configuration to yield the same reactivity expected for the operating core at midlife.

2. Analyze the initial and midlife experiments to prepare final U-235 and B-10 loadings for Type 3 cores in SM-1, SM-1A and PM-2A.

3. Related work will continue as Subtask 3.6 in FY-62.

ITEM 3.3 - STEADY STATE AND TRANSIENT THERMAL BEHAVIOR OF SM-1, SM-1A AND PM-2A USING SM-2 CORE I FUEL ELEMENTS

Task Definition and Objectives

Predict the steady state and transient thermal characteristics of SM-1, SM-1A and PM-2A cores using SM-2 Core I fuel elements. Establish the feasibility of the use of SM-2 elements in replacement cores for these plants to lengthen core life.

Summary of Second-Half Results

The final steady state and transient thermal analysis has been completed on the SM-1, SM-1A and PM-2A Cores with SM-2 elements. The limitations imposed upon the preliminary thermal analysis(3) were removed by the following:

1. The analytical predictions(4) and measured nuclear power distributions(2) have been improved.

2. Channel-to-channel flow distribution was established as a result of single element flow testing.(5) Final maldistribution factors used in the analysis were obtained from this testing.

3. Improved and more realistic transient results were obtained by the use of the transient code ART-02. Previous analyses treated the transients as a quasi-steady state problem.

The steady state analysis was conducted using the IBM-704 Code STDY-3* which performs a complete steady state parallel channel analysis resulting in enthalpy, temperature rise and quality calculations for the nominal channel, and enthalpy, steam quality, plate surface temperature, bulk water temperature, meat centerline temperature and burnout ratios for the hot channel. The transient analysis was conducted using the IBM-704 Code ART-02** which performs a similar hot channel analysis to determine reactor safety during the loss of flow transient under consideration. A burnout index was established for each element in their respective cores in order to select for analysis the elements with the highest ratio of power generation to available flow. If these elements prove safe during steady state operation with a minimum DNBR (departure from nucleate boiling) of 2.0, or during a loss of flow transient with a minimum DNBR of 1.5 then the entire core under analysis will be thermally safe.

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* Pyle, R.S., "STDY-3, "A Program for the Thermal Analysis of a Pressurized Water Nuclear Reactor During Steady State Operation." WAPD-TM-213, June 1960.

** Myer, J. E., Smith, R. B., Gilbard, H. G., George, D. G., and Peterson, W. D., "ART-02, A Program for the Treatment of Reactor Thermal Transients on the IBM-704," WAPD-TM-156, November 1959.

A final report[6] on the above analysis has been written and will give the details of the analysis summarized here. A tabulation of the important steady state and transient results is presented in Tables 3.3 and 3.4. Figures 3.1, 3.2, 3.3 show the variation of plate surface temperature for the SM-2 plate at design and scram power levels.

TABLE 3.3
STEADY STATE ANALYSIS

A. SM-1 Core with SM-2 Elements - Reactor Power 10.77 MW

Element*	Max. Surface Temp. °F	Max. Bulk Temp. °F	Min. DNBR	Quality	Local** Boiling
55	576.3	469.1	7.560	0	0
61	575.6	524.0	8.013	0	7

Scram Power Level Setting 13.0 MW

Element*	Max. Surface Temp. °F	Max. Bulk Temp. °F	Min. DNBR	Quality	Local** Boiling
55	576.8	473.2	6.239	0	0
61	576.1	538.6	6.567	0	5

B. SM-1A Core with SM-2 Elements - Reactor Power 20.2 MW

Element	Max. Surface Temp. °F	Max. Bulk Temp. °F	Min. DNBR	Quality	Local Boiling
33	578.5	473.9	3.675	0	0
72	577.7	473.0	4.156	0	0

Scram Power Level Setting - 24.2 MW

Element	Max. Surface Temp. °F	Max. Bulk Temp. °F	Min. DNBR	Quality	Local Boiling
33	579.1	481.1	3.043	0	0
72	578.2	480.0	3.452	0	0

C. PM-2A Core with SM-2 Elements - Reactor Power - 10.0 MW

Element	Max. Surface Temp. °F	Max. Bulk Temp. °F	Min. DNBR	Quality	Local Boiling
44	610.3	545.0	8.632	0	0
55	600.0	539.9	9.209	0	0

Scram Power Level Setting 12.0 MW

Element	Max. Surface Temp. °F	Max. Bulk Temp. °F	Min. DNBR	Quality	Local Boiling
44	620.7	556.4	7.172	0	0
55	615.2	543.0	7.677	0	0

* Element positions found in APAE Memo-291.

** Axial increment at which nucleate boiling begins, measured in inches from bottom of the element.

TABLE 3.4
TRANSIENT ANALYSIS - SUMMARY OF RESULTS OF LOSS OF FLOW ANALYSIS

A. SM-1 Core with SM-2 Elements - Reactor Power 10.77 MW

Element	Coastdown Parameter b=	Min. DNBR	Transient Min. DNBR
55	4.0	8.630	8.630
61	4.0	8.088	8.080

Scram Power Level Setting 13.0 MW

55	2.2	6.999	6.968
55	4.0	6.999	6.940
55	4.0	6.999	6.976
61	2.2	6.334	6.330

B. SM-1A Core with SM-2 Elements - Reactor Power 20.2 MW

Element	Coastdown Parameter b=	Min. DNBR t = 0 sec.	Transient Min. DNBR
55	4.0	4.156	4.107
72	4.0	4.945	4.940

Scram Power Level Setting 24.2 MW

55	4.0	3.331	3.280
72	4.0	4.029	3.920

C. PM-2A Core with SM-2 Elements - Reactor Power 10.0 MW

Element	Coastdown Parameter b=	Min. DNBR t = 0 sec.	Transient Min. DNBR
44	4.0	6.579	6.579
55	4.0	7.231	7.231

Scram Power Level Setting 12.0 MW

44		5.411	5.380
55		6.008	6.008

Conclusions

1. The final steady state analysis indicates that at design power and at scram power, cores composed of SM-2 elements in the SM-1A and PM-2A can operate with adequate thermal safety. Based on present core nuclear data there is no extensive local boiling in either of these cores and DNBR's are well above 2.0 at design power.

2. The SM-1 core with SM-2 elements shows local boiling in stationary elements adjacent to the core reflector. The results of single element flow testing indicating channel-to-channel flow maldistribution as high as 55% for elements in this region. This maldistribution contributes to the inception of local boiling in these channels. A conical type diffuser fix was developed in the single element flow test program and proved to reduce the maldistribution to 23%. This improvement in flow maldistribution is expected to reduce local boiling in the hot channels. Since DNBR's are well above 2.0 for both design and scram conditions, the SM-1 can operate with thermal safety.

3. The final transient analysis indicates that at design power and scram power DNBR's for the most critical elements in the SM-1, SM-1A and PM-2A are above the design minimum of 1.5. These results are based on present core nuclear data and indicate that the entire reactor will operate safely during loss of flow transients.

Recommendations

1. Performance of laboratory measurements of DNBR for exact SM and PM flow conditions and geometry.

2. Installation of a flow fix in SM-1 Type 3 stationary elements presenting poor channel-to-channel flow distribution.

3. Re-analysis of the steady state and transient analysis for the SM-1 elements utilizing improved flow distribution factors.

4. A complete re-analysis of all steady state and transient work utilizing the newest codes such as TITE* and XITE**.

Future Work

Perform a complete steady state and transient thermal analysis of a 37-element Type 3 Core in SM-1. This will continue in FY-62 under Subtask 3.7.

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* Miller, R.I., Pyle, R.S., "TITE - A Digital Program for the Prediction of Two-Dimensional, Two-Phase Hydrodynamics," WAPD-TM-240.

** Rose, R.P., Pyle, R.S., "XITE-1 Program," WAPD-R (J)-73.

PLATE SURFACE TEMPERATURE OF SM-2 ELEMENTS
IN SM-1 CORE ELEMENT IN CORE POSITION #55

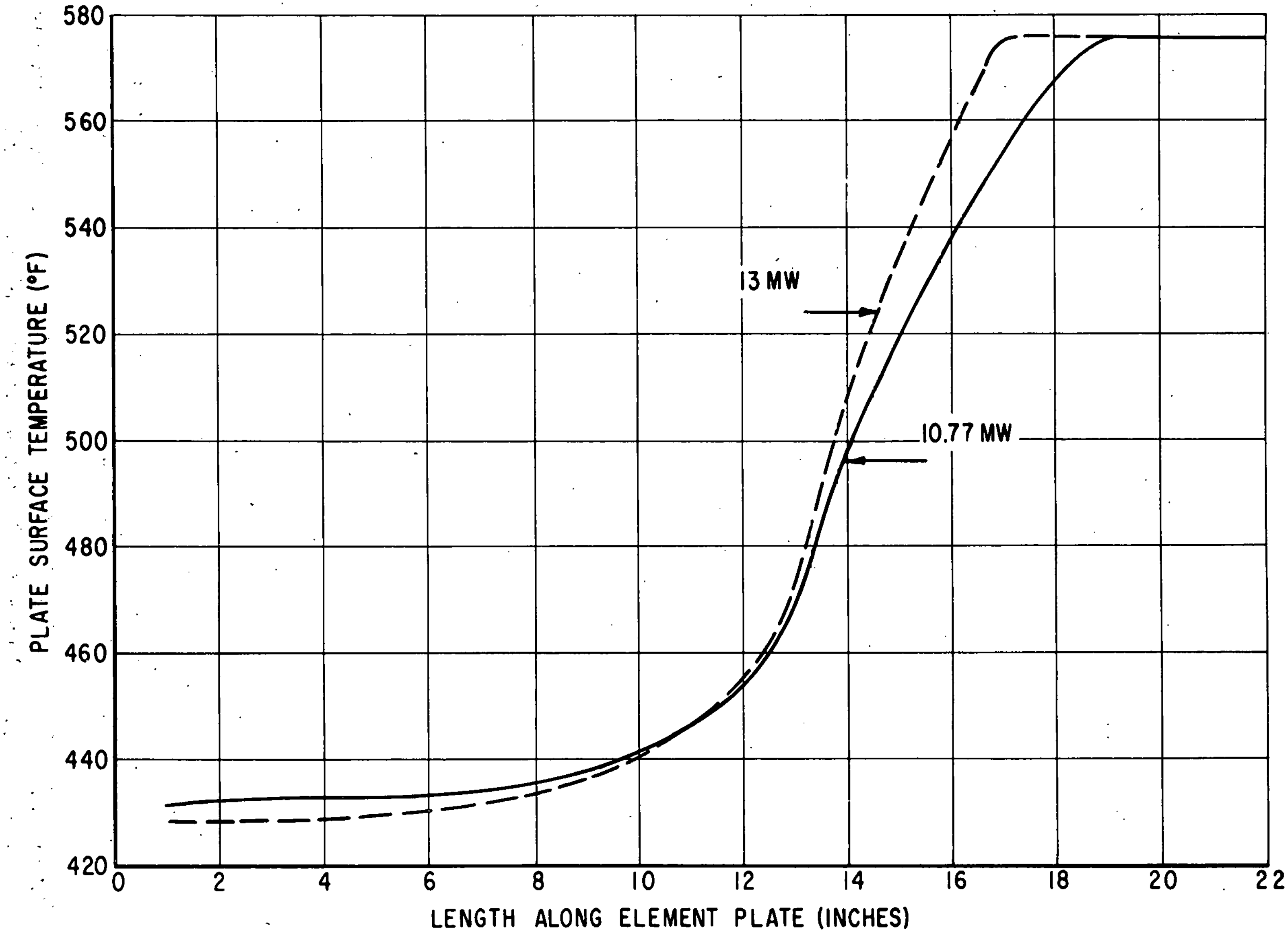

FIGURE 3.1

PLATE SURFACE TEMPERATURE OF SM-2 ELEMENTS IN
SM-IA CORE ELEMENT IN CORE POSITION # 33

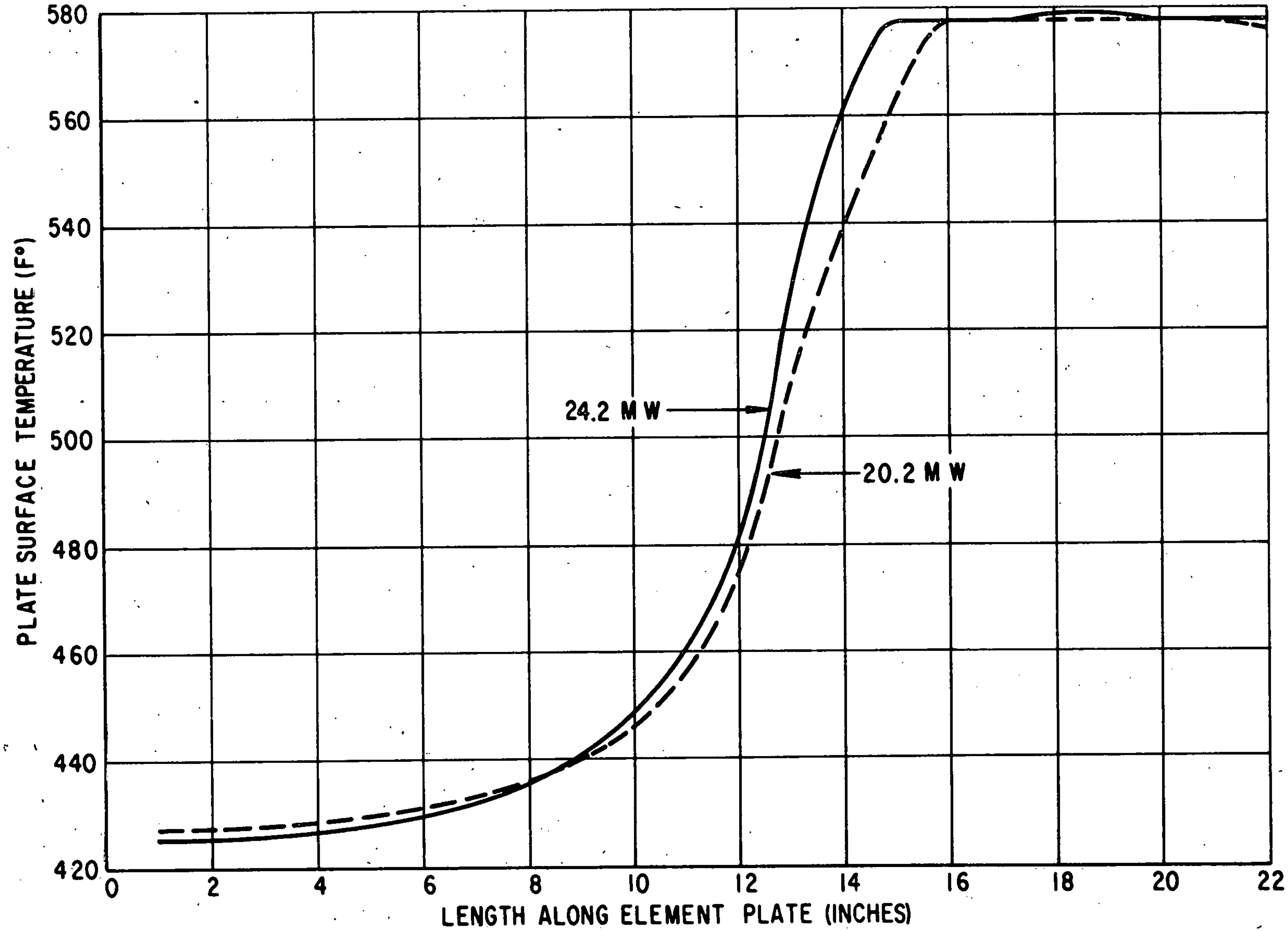

FIGURE 3.2

PLATE SURFACE TEMPERATURE OF SM-2 ELEMENTS IN
PM-2A CORE ELEMENT IN CORE POSITION #44

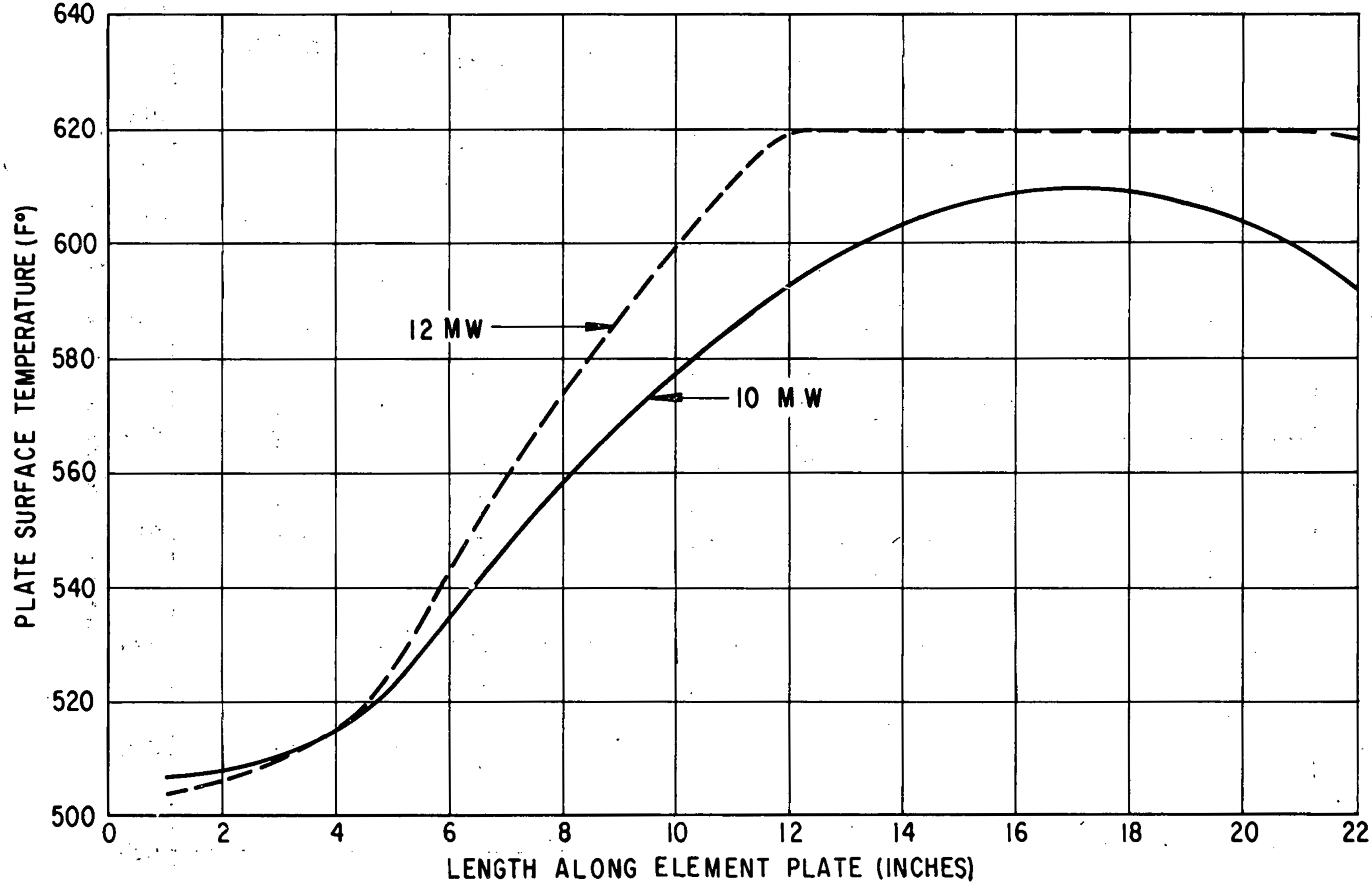

FIGURE 3.3

ITEM 3.4 - SINGLE ELEMENT FLOW TEST OF SM-2 FUEL ELEMENTS FOR SM-1, SM-1A and PM-2A

Task Definition and Objectives

Measure channel-to-channel flow distribution within SM-2 stationary and control rod fuel elements modified to fit SM-1, SM-1A and PM-2A core support structures. This data provided the flow maldistribution used in calculating hot channel factors.

Summary of Second-Half Results

All test work has been completed and a final test report has been written. This final report[5] presents complete results of all test work accomplished in this program.

A brief summary of the more important test results is tabulated in Table 3.5. The test report should be consulted for actual channel-to-channel flow distribution.

TABLE 3.5
SUMMARY RESULTS - SINGLE ELEMENT FLOW TESTS - SM-2 ELEMENTS IN SM-1, SM-1A, PM-2A

Core	Component	Orifice Size-inch	Design Flow Rate-gpm	% Deviation from average channel velocity at design flowrate +%	-%
SM-1	Fixed Element	1.19	43.66-46.57	44	55
SM-1	Fixed Element	2.02	106.13	6.8	8.8
SM-1A	Fixed Element	1.68	127	6.1	6.7
PM-2A	Fixed Element	1.83	113	6.2	11.1
SM-1	Control Rod	-	100	23.5	27.8
SM-1A	Control Rod	-	147	13.2	22.8
PM-2A	Control Rod	-	110	12.0	21.2

Conclusions

Results of testing SM-2 stationary elements in mockups of the SM-1A and PM-2A core support structures indicate that the flow maldistribution with these elements is within the -12% factor utilized in the preliminary thermal analysis[3]. The orifice plate is located on the outlet end of the fuel element in SM-1A and PM-2A.

Results of testing an SM-2 stationary element for SM-1 indicate a maldistribution factor much higher than anticipated in the elements with small orifices. The orifice plate is located at the inlet end of the fuel element in SM-1. Results of testing with the maximum orifice show that no bad flow maldistribution exists there. The effect of the intermediate orifices upon flow maldistribution was not investigated because of the limited scope of the program. An investigation was made to determine whether the poor flow distribution with the minimum orifice could be improved by modifying the fuel element inlet end box. As a result of this investigation, two different element modifications were tested: a conical flow diffuser and a perforated plate. The results of these tests show a definite improvement. Details of these two modifications and test results are presented in the final test report[5].

Testing SM-2 control rod fuel elements in SM-1, SM-1A and PM-2A control rod assemblies indicated that flow distribution within the fuel element is not within the $\pm$ 12% maldistribution used in the preliminary analysis.

Recommendations

1. An investigation should be made to determine the effect of the two proposed element modifications on elements other than the minimum orificed element and to determine the feasibility of using such a flow improvement device in all future elements for the SM-1.

2. A method for improving flow distribution within the control rod fuel element should be designed and tested.

3. An investigation should be made to measure local flow velocities and velocity profiles within a single SM or PM channel to provide fundamental information for use in thermal analysis.

4. Initiate a test program to determine the effect on flow distribution of non-central positioning of the control rod fuel element within the control rod tube.

Future Work

None.

ITEM 3.5 - METALLURGICAL DEVELOPMENT (ETR IRRADIATION PROGRAM)

Task Definition and Objectives

The basic objective is to demonstrate that the SM-2 stainless steel - UO_2 dispersion fuel plates (Type 3) will satisfactorily resist irradiation damage. This is being done by irradiating specimens in ETR capsules to 50-90 a/o U-235 burnup, and by irradiating full size test elements in SM-1. Post-irradiation examination will then be made of the specimens and test fuel elements. SM-1 Core I fuel elements (Type 1) will be irradiated to high levels in SM-1 Core II and will be examined after irradiation to give further basic irradiation data.

Summary of Second-Half Results

The irradiation of miniature SM-2 fuel plates in high flux capsules in the ETR was originally part of the SM-2 program. * The specimens are miniature fuel plates 1 in. x 1/2 in. x 0. 040 in. fabricated by the same procedures as full size plates. **.

The six high burnup ETR instrumented capsules were inserted during the period June-September 1960. Each capsuble contained four instrumented SM-2 specimens heated by six Kanthol heaters of 2-1/4 kw capacity plus one or two uninstrumented specimens. The specimens were heated in order to simulate acual operating temperatures. Four thermocouples are in the instrumented area, one adjacent to each specimen in order that accurate records of sample temperature could be maintained.

During the report period, irradiation continued on all six capsules up to July 10, 1961, at which time capsules 32-6 and 32-8 were removed. Irradiation on the remaining four capsules continued through September 30, 1961, with removal of capsule 32-9 scheduled for October 2, 1961. Capsules 32-5, -7 and -10 are scheduled for removal at the end of cycle #41 on November 13, 1961. Current status is presented in Tables 3. 6 and 3. 7.

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* BMI-1516, Irradiation Capsule Study for SM-2 Reference and Alternate Dispersion Fuels.

** BMI-1528, Fabrication and Irradiation of SM-2 Core Materials.

TABLE 3.6
STATUS OF CAPSULES AS OF SEPTEMBER 30, 1961

Capsule	Full Power Days 9-30-61	Peak Flux ($x10^{14}$)	Predicted Burnup 9-30-61	Thermo-couples	Heaters	Present Surface Temp. (^{o}F)	Predicted Burnup at Removal
32-5	150	3.5	65	4 OK	6 OK 4.3 KW	550-650	69
32-6	197	2.7	65	-	-	-	65
32-7	186	3.4	69	2 OK	Out	400-450	71
32-8	197	3.0	67	-	-	-	67
32-9	198	5.1	89	4 OK	Out	335-450	89
32-10	159	3.2	63	3 OK	3 OK 6.8 KW	550-650	65

TABLE 3.7
NUMBER OF MINIATURE SPECIMENS OF THE VARIOUS CORE TYPES TO BE FOUND IN THE CAPSULES, ACCORDING TO BURNUP LEVEL

Specimen Type	Burnup Levels				
	up to 40%	65-67%	65-71%	90%	Total
1 ref.	1	2	1	1	5
1 alt.	3	0	1	0	4
2 ref.	0	0	0	0	0
2 alt.	8	3	2	1	14
3 ref.	6	1	4	1	12
3 alt.	9	3	6	2	20
*4 ref.	0	0	2	0	2
*4 alt.	0	2	2	0	4
Totals	27	11	18	5	61

* Similar to Type 5, but 0.040 in. thick instead of 0.070 in.

Periodic inspection of regular and special core components being irradiated in the SM-1 is also part of this program. Two SM-2 Type stationary fuel elements, originally inserted in the SM-1 Core I Spiked and Rearranged were reinserted in SM-1 Core II, where they have experienced 5 a/o U-235 burnup. One element contained 18 reference SM-2 fuel plates, each with 26 w/o UO_2 and ZrB_2 poison in 0. 040 in. thick type 347 stainless steel plates. The second element contains two 0. 040 in. thick fuel plates each of 9 different concepts as shown in Table 3. 8.

TABLE 3. 8
PLATE DESCRIPTION AND FUEL TYPE OF A SPECIAL SM-2 STATIONARY FUEL ELEMENT

Plate Type	Fuel Type	Core Type	Poison
SM-2 Reference	Spherical UO_2	3	ZrB_2
SM-2 Boron in frame only	Spherical UO_2	4	Boron in frame
SM-2 Green core	Spherical UO_2	4	ZrB_2
SM-2 Geneva oxide	Geneva Oxide	3	ZrB_2
SM-2 CbB_2 poison	Spherical UO_2	3	CbB_2
SM-2 UN fuel	UN	4	ZrB_2
SM-2 Coated ZrB_2	Spherical	3	ZrB_2
SM-2 UN and coated ZrB_2	UN	4	Coated ZrB_2
SM-1 (except . 040" thick)	Geneva Ozide	1	B_4C

Prior to reinsertion, each of these fuel elements received a thorough visual examination by periscopic means. No defects were noted.

The europium oxide absorbers were also examined prior to reinsertion and were found free of visual defects.

Three SM-1 Core I stationary fuel elements were also visually examined. These elements had experienced 18 MWYR operation in SM-1 Core I and in SM-1 Core I Rearranged and Spiked. No visual defects were noted. Two elements, #80 and 81, were selected for reinsertion in SM-1 Core II.

The majority of the remaining SM-1 Core I fuel elements, discharged at end of spiked core life, were examined in the spent fuel pit during July 1961. Thirty fuel elements were visually examined. No positive indication of a defect was found.

ORNL started the complete examination of element #79 during this period. This element was removed from SM-1 Core I at the end of normal core life, after 16. 4 MWYR. It was found that intergranular clad cracking, first noted on a small scale on element #72 examined after 10. 8 MWYR*, had greatly increased, and that

* ORNL-2907, "Army Package Power Reactor Project Annual Progress Report For Period Ending January 31, 1960."

transgranular cracking had been initiated. Matrix stability, however, remained good. Cracking of the clad was most prevalent in high burnup areas on the longitudinal centerline of the fuel plate.

The complete significance of this examination is not yet known. A decision has been made to remove the two remaining SM-1 Core I fuel elements (#80 and 81) from the reactor during the October shutdown and to ship one to ORNL and store the other at the site. The examination of element #79 will be completed. The element at ORNL will be examined and a decision made whether to reinsert the other element in SM-1 Core II.

Results and Conclusions

1. Performance of the ETR instrumented capsules continues to be satisfactory.

2. The two SM-2 fuel elements placed in SM-1 Core I at 16.4 MWYR were examined at 18.0 MWYR, found satisfactory, and were reinserted with Core II. Further irradiation in SM-1 Core II has been started.

3. Selected components from SM-1 Core I were visually examined periscopically and appeared to be satisfactory. The SM-1 Core II europium oxide absorbers inserted in Core II at 10 MWYR and two Core I fuel elements were examined and found satisfactory, and were reinserted in Core II for further irradiation. The two high burnup SM-1 Core I fuel elements were visually inspected and were reinserted.

4. Visual examination of thirty spent SM-1 Core I elements revealed no discernible defects.

5. Complete metallurgical examination of SM-1 Core I element S-79 removed at 16.4 MWYR at ORNL revealed extensive intergranular and transgranular clad cracking but no matrix cracking after 58a/o U-235 total burnup. As a result the remaining two core I elements in SM-1 Core II (#80 and 81) will be removed. One of these elements may be reinserted at a later date, after examination of elements 79 and 80 or 81 has been completed.

Recommendations

1. Re-examination of the two SM-1 Core I fuel elements to be removed from the reactor in October to determine if it is possible to observe on site the clad cracking found on metallurgical examination of the similar element #79. If the standard periscopic equipment available on site is not suitable for this examination, investigate possible equipment for such and on site examination, so that if similar clad cracking occurs in Core II it may be found expeditiously.

Future Work

1. Continued irradiation of ETR capsules.

2. Hot cell examination of specimens contained in the ETR capsules, results of which will be factored into Type 3 core fabrication specifications.

3. Initiation of further core development (Type 5) and irradiation testing of developed specimens.

4. Continued irradiation of the two SM-2 fuel elements in SM-1 Core II.

5. Further testing and analysis of the results of the hot cell examination of SM-1 Core I element #79, and 80 or 81.

6. Visual examination of SM-1 Core II special components when the vessel is opened.

7. Work related to this item will be carried on in FY-62 in Subtasks 3.1, 3.3 and 3.4.

ITEM 3.6 - METALLURGICAL STUDIES

Task Definition and Objectives

Conduct metallurgical studies necessary for the development of replacement cores. Included will be:

1. Economic analysis of the use of advanced cores in SM-1, SM-1A, and PM-2A.

2. Correlation of available data on stainless steel - UO_2 dispersion fuel irradiations.

3. Development of an alternate absorber having low cost and good reliability.

Summary of Second-Half Results

A cost survey(7) was conducted for development of replacement cores of the latest proven type for SM and PM type reactors. Major cost reduction is possible by using Type 3 cores (40 mil fuel plates, 25 w/o UO_2, welded assembly) in reactors currently using Types 1 or 2. Other significant savings are possible by multiple core reprocessing, relaxation of cobalt and tantalum limits in Type 347 stainless steel, and multiple core procurement. Minor savings may be realized by quantity procurement of UO_2 and special Type 347 stainless steel. Table 3.9 is a summation of core costs for various types of procurement.

TABLE 3.9

CORE COSTS BY TYPE AND SINGLE VS. MULTIPLE PROCUREMENT

Item of Cost	Type 1 and 2 Cores One Core	Type 1 and 2 Cores Four Cores	Type 3 Cores One Core	Type 3 Cores Four Cores	Type 3 Cores One Core	Type 3 Cores Four Cores
UF_6 to UO_2	15,000	54,000	23,500	86,000	23,500	86,000
Low Impurity 347	-	-	-	-	15,000	60,000
Relaxed Impurity 347	5,000	20,000	5,000	20,000	-	-
Core Fabrication*	175,000	590,000	175,000	590,000	175,000	590,000
Core Reprocessing	61,200	76,500	61,200	76,500	61,200	76,500
Total	$256,200	$740,500	$264,700	$772,500	$274,700	$812,500
Available MWYR**	16	64	34	136	34	136
$/MWYR	$ 16,000	11,600	7,800	5,700	8,100	5,950

* Includes liaison.

** Approximate, in SM-1 or SM-1A.

Costs were generated in April-May 1961. While costs may change from time to time, the economic relationships will remain valid.

The prediction of limiting burnup for the SM-2 system (designated as Type 3) may be approached from several viewpoints.

One approach is experimental, relying on a data extrapolation technique. This method indicates that matrix fracture at U-235 burnup of 82%, the maximum predicted for Type 3 cores, is unlikely. Limited data at such high burnup levels prevents closer analysis at this time.

A failure model developed at BMI was found not to be useful for the present study. This is because the model depends on knowing a detailed history of time, temperature and burnup of a specimen irradiated to failure. Predictions of burnup for a specimen of the same specifications under different conditions could then be made. The required data are not generally available.

An ORNL model has been developed based on a failure criterion of plastic yield. Given the required material properties and other data this model may be used to predict failure burnup. For an assumed set of data a prediction of 151% burnup is made for the SM-2 fuel element.

The matrix material has a tendency to become brittle as a result of radiation damage. A new mathematical model has been recently developed in which the postulated failure mechanism is fracture of an elastic matrix under the combined action of thermal stress and the pressure caused by fission gas accumulated in the fuel particle cavities. Because of the complexity and uncertainties of the actual mode of failure many restrictive assumptions are incorporated in the investigation of the model. Predicted burnup may therefore be questionable. For the set of data corresponding to that assumed for the ORNL model the failure burnup is predicted to be 149%.

A report summarizing the results of the first three techniques has been prepared.[8] The Alco model is the subject of a report to be issued shortly.[9]

Development of Alternate Absorber

The alternate absorber under development in this program is silver-cadmium-indium mechanically clad with stainless steel.

Samples of the selected alternate absorber material, silver-cadmium-indium (70-25-5 w/o) was ordered from the low bidder. Two thicknesses, 0.056 in. and 0.096 in., were procured.

The initial analysis of the gamma heating problem indicated that maximum temperature in 0.096 in. thick Ag-Cd-In could reach 1343°F, assuming a worse case of a 10-mil air gap on one side of the absorber slab, between the slab and the 0.030 in. thick stainless steel can. For this reason thinner material (0.056 in.) was also procured. The analysis further indicated that the maximum temperature would decrease markedly as the air gap or gaps decreased. It was intended that

0.050 in. thick stainless steel would be used with the 0.056 in. thick absorber material, and 0.030 in. with 0.096 in., so that the total absorber thickness would be constant.

As previously reported, the initial 0.030 in. thick inside absorber box "hour-glass" at the middle during the automatic TIG welding procedure. Therefore, the welding jig was modified to incorporate an expanding mandrel in the center, to prevent such a recurrence. The first internal absorber box fabricated with the modified jig was found to be within 0.005 in. in all dimensions, and did not hourglass. This box was 0.050 in. thick. Slabs of 0.056 in. Ag-Cd-In were placed on this box, and the external box was welded. Tolerances after welding the second box were $\pm$ 0.008 in. maximum, indicating that air gaps of 0.003 in. are the most likely to be found. Concurrently, further thermal analysis was performed on both thickness absorbers, with single and double air gaps up to 0.010 in. An available IBM-704 code was used for this work. The results indicate that with maximum expected air gaps, 0.003 in. the temperature will not exceed 1000^{o}F. Complete results of the thermal analysis will be available in October.

The first completed 0.056 in. type absorber was inserted in the Alco Critical Assembly and its worth was determined relative to a SM-1 Core I boron absorber. The relative worth was found to be -60 cents. Details of this test, and of the test of other Ag-Cd-In and Ag-Cd rods, will be reported at conclusion of the work.

Conclusions

1. An economic analysis indicates that core cost per MWYR may be cut nearly in half by use of Type 3 cores in SM-1, SM-1A and PM-2A. Multiple core procurement, core reprocessing and relaxation of Type 347 impurity levels will reduce core costs still further.

2. A failure analysis of the UO_2 stainless steel matrix was made using an elastic model. At the low temperature (630^{o}F) used in the analysis, and for the assumed material properties and dispersion characteristics, it was found that matrix failure is not expected. It was also found that fuel plate failure is unlikely, based on extrapolation of existing irradiation data up to the 82% U-235 burnup level anticipated for Type 3 cores.

3. A dimensionally satisfactory Ag-Cd-In absorber has been fabricated. Nuclear testing revealed that this first absorber, which had a 0.056 in. Ag-Cd-In section, was worth 60 cents less than a standard SM-1 Core I B-10 absorber. Thermal analysis of the Ag-Cd-In absorber indicates that if the air gaps are less than 0.003 in. the effect of gamma heating is not serious.

Recommendations

1. The economic analysis of core fabrication costs should be reevaluated at periodic intervals.

2. Further effort should be made to relax some of the assumptions that are made in the analytical models for failure of matrix elements. Further data are needed to improve the predictions for either the present models or improved models. Since only matrix cracking was assumed in the present investigation an investigation of cladding failure should be initiated.

Future Work

1. No future economic analysis of core fabricating costs is scheduled.

2. The elastic model of failure will be reported in an APAE report.

3. Further welding development, nuclear testing and thermal analysis of the Ag-Cd-In absorber will be performed leading to the insertion of a test absorber in SM-1 Core II. This effort will continue as Subtask 3.2 in FY-62.

BIBLIOGRAPHY - TASK 3

1. AP Note-355, "Ten Year Core Development Program for AEC - Army Reactors," June 5, 1961.

2. APAE-100, "Nuclear Measurements for Use of Type 3 Fuel Elements in SM-1, SM-1A and PM-2A Configurations," to be issued.

3. APAE Memo-291, "Interim Report on the Use of SM-2 Elements in SM-1, SM-1A and PM-2A Cores," June 20, 1961.

4. APAE-104, "Final Report on the Nuclear Analysis of SM-2 Elements in the SM-1, SM-1A and PM-2A Cores," to be issued.

5. APAE Memo-297, "Single Element Flow Tests for Type 3 (SM-2) Fuel Elements in SM-1, SM-1A, PM-2A Cores," to be issued November 1961.

6. APAE-105, "Final Report on the Thermal Analysis of SM-2 Elements in the SM-1, SM-1A and PM-2A Core," to be issued.

7. APAE Memo-295, "Economic Analysis of SM and PM Type Reactor Cores," October 5, 1961.

8. APAE Memo-294, "Determination of Maximum Permissible Irradiation of SM-2 Fuel Plates," to be issued.

9. APAE Memo-298, "Failure Analysis of Dispersion Fuel Elements Based on Matrix Crackings, to be issued.

TASK 4 - PRIMARY SYSTEM AND COMPONENT PERFORMANCE

Task Engineer - J. P. Tully

ITEM 4.3 - PM-2 STEAM GENERATOR MOISTURE CARRYOVER TEST AND ANALYSIS

Definition and Objectives

PWR type plants experience some moisture carryover from the steam generator during an increasing load transient. Based on analog computer analysis, the PM-2A moisture carryover is within design limits; however, only limited experimental or actual data have been taken to confirm the theoretical analysis. Item 4.3 will determine the means to measure the moisture in steam from the PM-2A steam generator during an increasing load transient. Following selection of a measuring device the test is to be designed, equipment procured and installed, the test conducted, and the results analyzed.

Summary of Second-Half Results

1. A detailed review of the task philosophy resulted in acceptance of the proposed method of test but indicated need for revision of mechanical design of the test equipment.

2. Revised detailed drawings of the calorimeter and its components, instrumentation circuitry to include parts list, and plant installation of the test equipment have been completed but require checking and approval.

3. Test specifications were prepared and are being reviewed prior to submittal to the Army for approval.

Conclusions

None.

Future Work

Upon customer approval of the test specifications:

1. Issue test, calibration and equipment installation procedures.

2. Procure all test equipment.

3. Assemble and checkout equipment in the laboratory.

4. Ship equipment to site and install.

5. Calibrate the instrumentation and conduct the test.

6. Reduce data and issue report.

The above work will be done under Subtask 7.1 of the FY-62 program.

ITEM 4.4 - PRIMARY SHIELDING MEASUREMENTS

Task Definition and Objectives

Perform an experimental evaluation of existing shields at the SM-1 and PM-2A. The data obtained will serve to verify design methods and provide a sound basis for safe plant operation and improvement of shielding at these and future plants.

Summary of Second-Half Results

SM-1

A summary of tests performed is outlined below. A data report for these tests is in preparation.

Test A-401-Primary Shielding Measurements - Neutron Flux

This test consisted of a set of thermal neutron flux measurements between the shield rings and in the upper shield tank during reactor operation. Activation of bare and cadmium covered gold foils was used in the experimental technique. The test will provide an extension of the results reported in APAE-35* to permit evaluation of the axial variation in the fluxes.

Test A-402 - Primary Shielding Measurements - Gamma Flux

Gamma doses were measured between the shield rings and in the upper shield tank by exposure of gamma monitor films. The radial attenuation curves are similar to measurements reported in APAE-35* but the dose rates are higher by a factor of five or six. Due to uncertainty in the power level calibration for these runs, this factor may be in error by as much as a factor of two. The dose rate measurements would then be higher than previously measured by about a factor of three. Figure 4.1 shows the radial gamma dose rate distributions in the shield rings at various elevations

* Rosen, S. S., "Shielding Experiment and Analyses, Task VI," APAE-35 and -35 Supplement 1, October 15, 1958.

measured from the core midplane. Axial gamma traverses are plotted in Fig. 4.2.

Test A-403 - Neutron Flux in the Instrument Wells

Test A-404 - Gamma Flux in the Instrument Wells

Foil activation measurements and gamma sensitive film exposures were made in instrument well A. These data, obtained during a 20-min run at 5 kw power level, are shown in Fig. 4.3 and 4.4. The gamma traverse does not show a peak just below the core centerline as does the background curve. This may be an error in the reading from the single film badge at -12 in.

Test A-406 - Neutron Measurements in the Rod Drive Pit

Bare and cadmium covered gold foils were used to measure the thermal neutron flux.

Test A-407 - Gamma Measurements in the Rod Drive Pit

A gamma survey meter provided measurements for mapping the rod drive pit after shutdown.

Radiation Surveys were made:

a. prior to Core I removal and cleanup

b. after Core I removal

c. after flushing and cleanup of the bottom of the reactor vessel

d. after completion of a 2-day full power run on Core II.

The background radiation level and dose rates at most of the surveyed points did not change significantly after core removal and cleanup. However, an increase of 50 to 100% in the radiation level at the cracks on the side and bottom of the water box was noted. Probing of the high dose rate area was more complete after cleanup of the reactor vessel, because of the unexpectedly high radiation levels; however, readings of this magnitude should not have been overlooked previously. The source of this radiation has not been identified, but it seems possible that some crud penetrated along the rod drive seal during the flushing and cleanup operation.

Test A-408 - Dose Rates During Spent Fuel Transfers

Gamma dose rates from spent fuel elements as a function of water shielding thickness are shown in Fig. 4.5.

PM-2A

Initial shielding measurements at PM-2A are included in the following tests.

Test C-401 - Neutron Flux Measurements

This test was accomplished by exposure of bare and cadmium covered gold foils and sulphur pellets. Neutron flux measurements were made on the primary shield tank, on the vapor container, in the rod drive area, in the instrument wells, and on the upper refueling platform.

Test C-402 - Gamma Flux Measurements

This test was performed by exposure of monitor films and use of survey meters. Locations of measurements were essentially those specified in Test C-401. Dose rate measurements made at various locations in the plant are listed in Table 4.1. Figure 4.6 identifies the dose point locations.

A data report and analysis will be included in the PM-2A startup report, APAE-92.

Conclusions

Those tests performed will provide data for experimental evaluation of the shielding at the SM-1 and PM-2A. PM-2A data reduction indicates that there are no major defects or problems in the shielding as modified. Measured radiation levels show that an individual can work a normal 84-hr week in the overall plant, excluding the reactor trench, and remain within the allowable radiation dose rates recommended by the AEC.

Recommendations

1. The data obtained in all the tests should be analyzed.

2. Additional testing necessary to assure safe operation or develop improved shield designs for the Army Nuclear Power Program should be performed.

Future Work

1. A data report for SM-1 shielding tests will be published and test results will be analyzed.

2. Analysis of PM-2A shielding test data will be completed and published.

3. Future work will be performed under Subtask 5.3 of the FY-62 program.

TABLE 4.1
PM-2A DOSE RATES AT FULL POWER (EXTRAPOLATED) - Mr/hr

Position	F. P.	Position	F. P.
1	.04	21	6000
2	.038	22	7900
3	.038	23	65
4	.038	24	750
5	.34	25	910
6	.11	26	65
7	.08	27	45
8	.06	28	.06
9	1.25	29	.05
10	4.2	30	.05
11	.9	31	.045
12	15	32	.045
13	75	33	.04
14	144	34	.04
15	2460	35	.04
16	3600	36	
17	3000	37	138
18	4500	38	10500
19	1.5		
20	7200		

Survey - Jordan AGB-OK-SR

Date - 2/7/61 and 2/10/61

Power Level - 0.18 and 12 Log N

GAMMA RADIATION DOSE RATE IN REACTOR SHIELD 10 JUNE 1961
CORRECTED TO 10 MW POWER (TEST A-402)

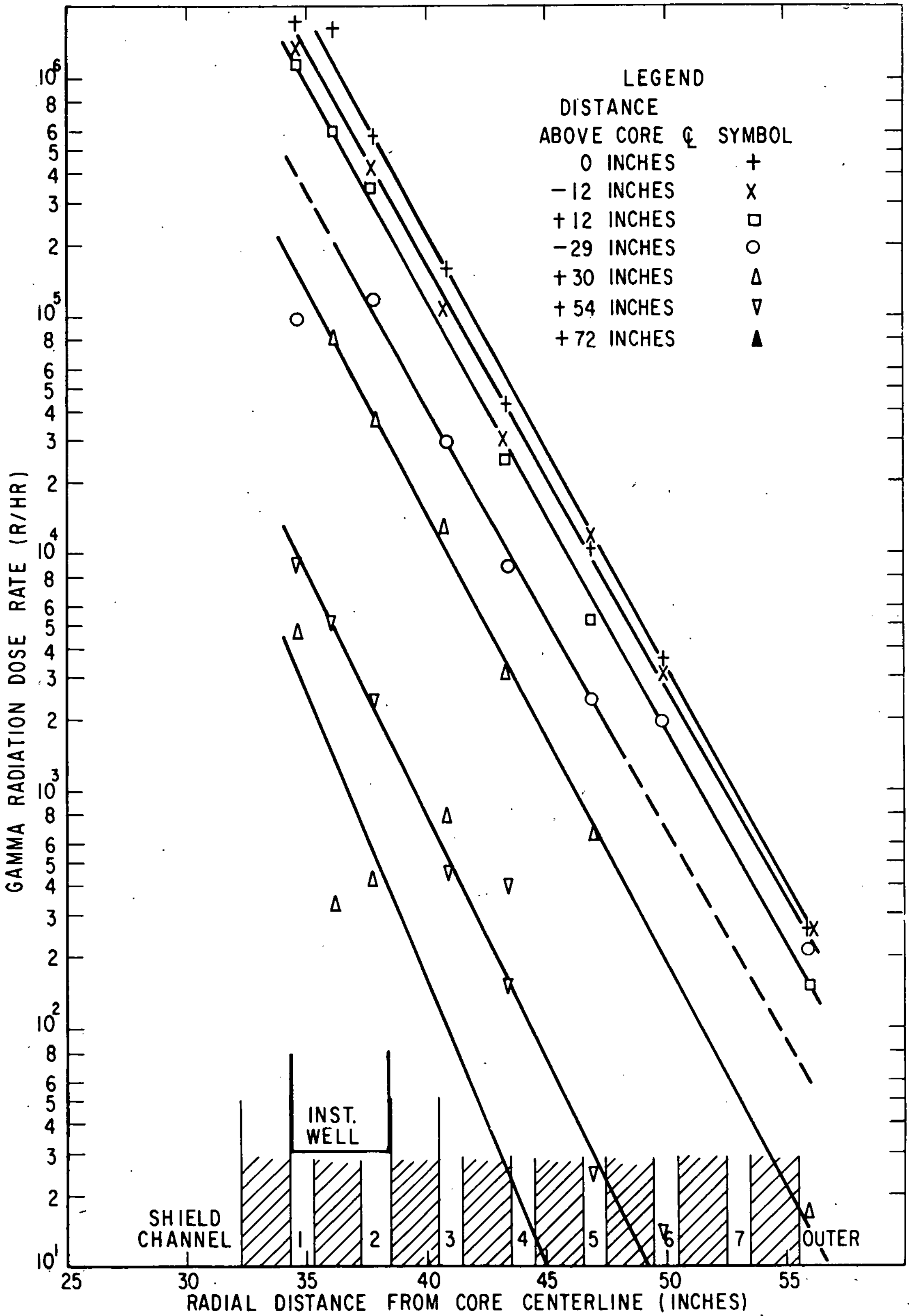


FIGURE 4.1

GAMMA RADIATION DOSE RATE IN REACTOR SHIELD
CHANNELS 1 TO 6 AND WELL "A" 10 JUNE 1961
CORRECTED TO 10 MW POWER (TEST A-402)

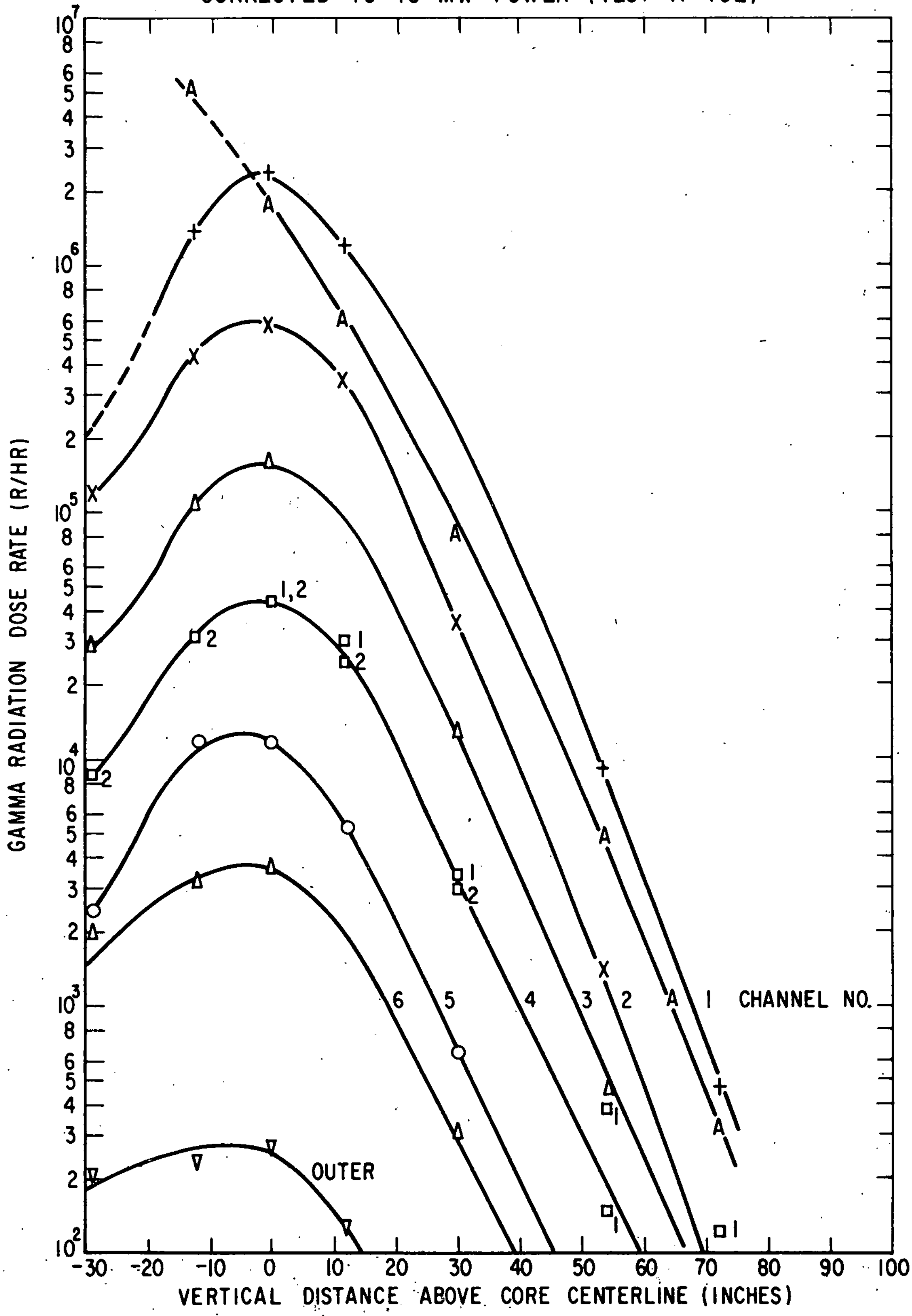


FIGURE 4.2

RELATIVE NEUTRON FLUX INSTRUMENT WELL "A"
10 JUNE 1961 (TEST A-403)

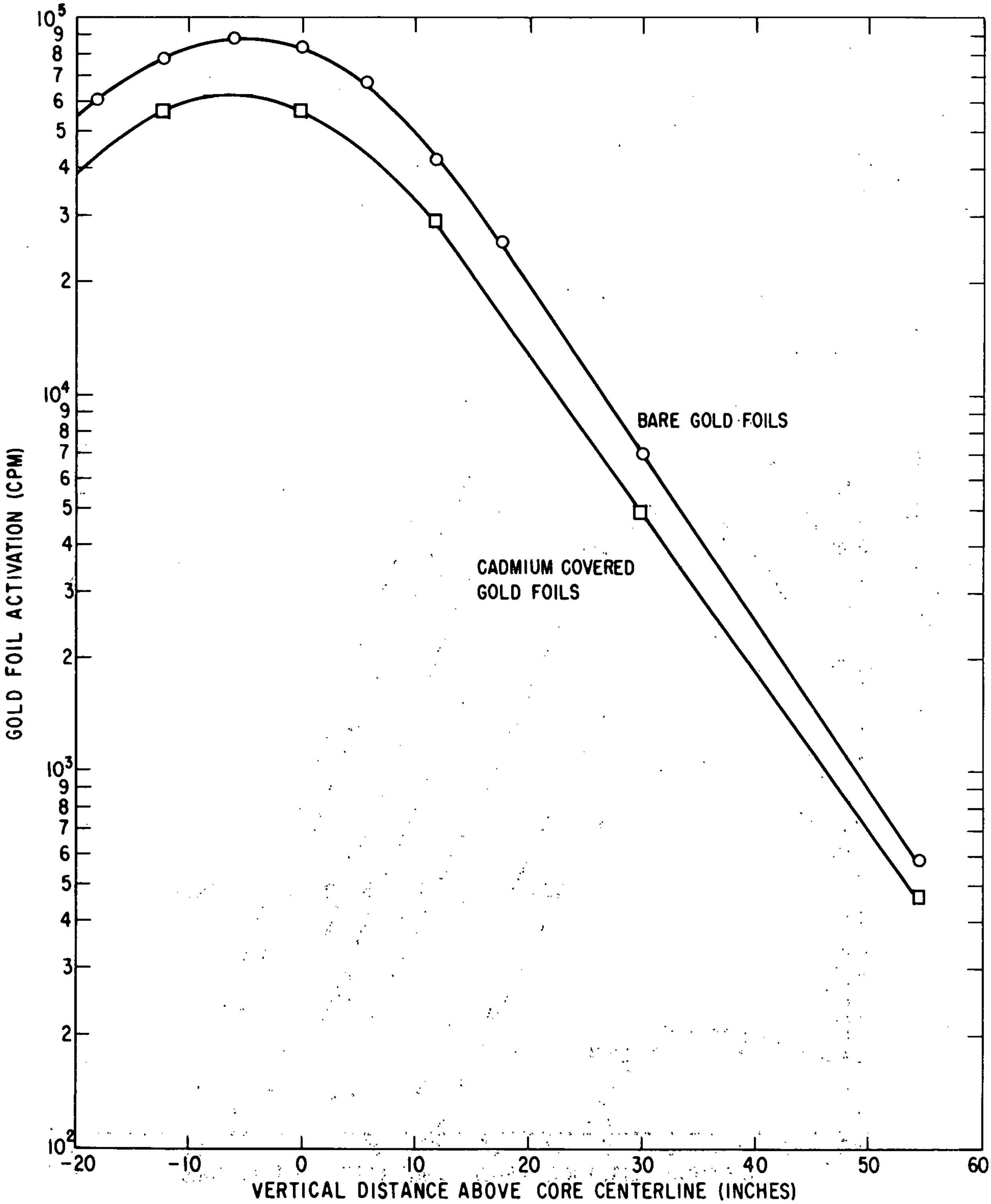


FIGURE 4.3

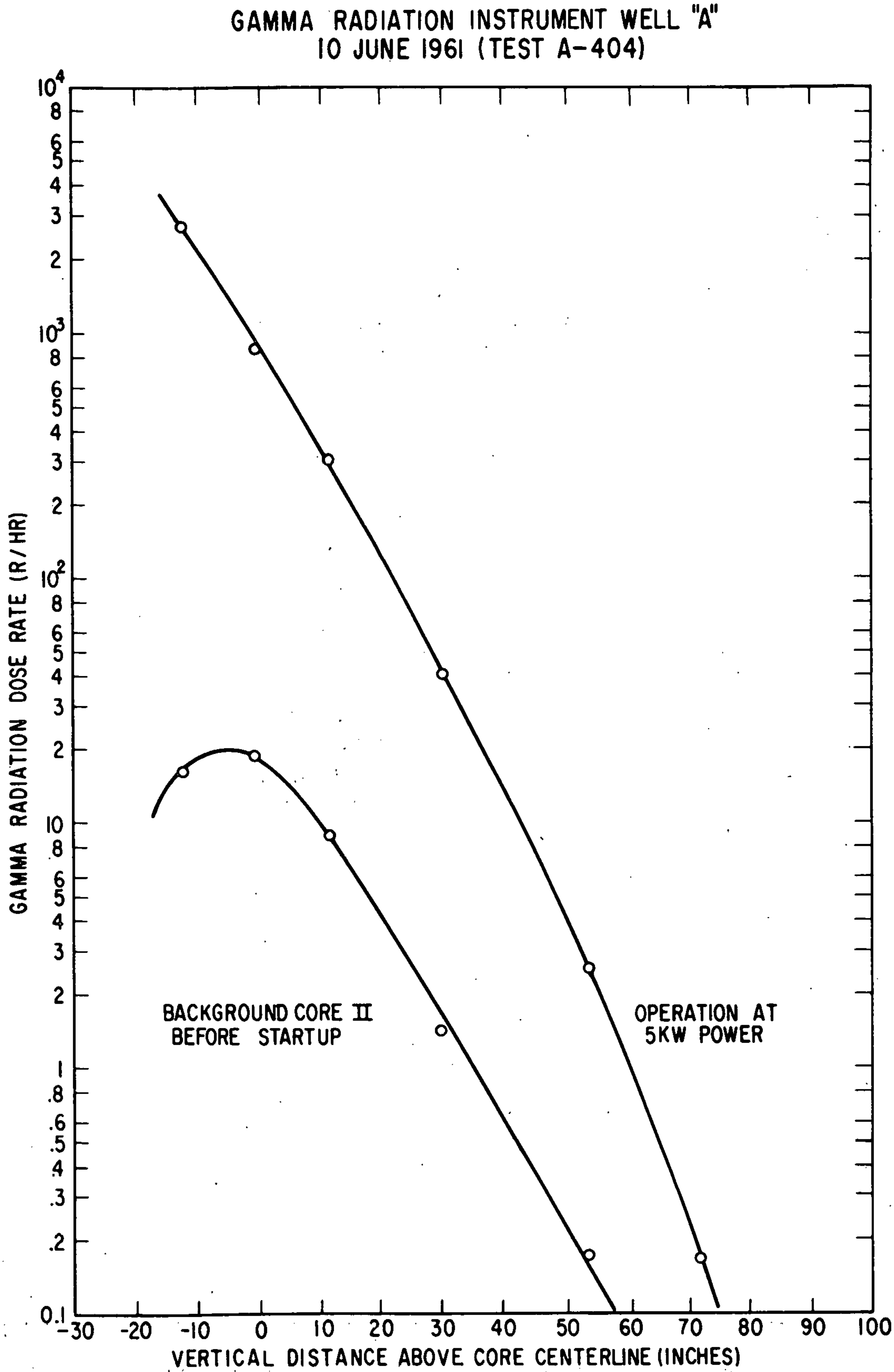

GAMMA RADIATION INSTRUMENT WELL "A"
10 JUNE 1961 (TEST A-404)
GAMMA RADIATION DOSE RATE (R/HR)
VERTICAL DISTANCE ABOVE CORE CENTERLINE (INCHES)
BACKGROUND CORE II
BEFORE STARTUP
OPERATION AT
5KW POWER

FIGURE 4.4

DOSE RATES FROM SPENT FUEL ELEMENTS

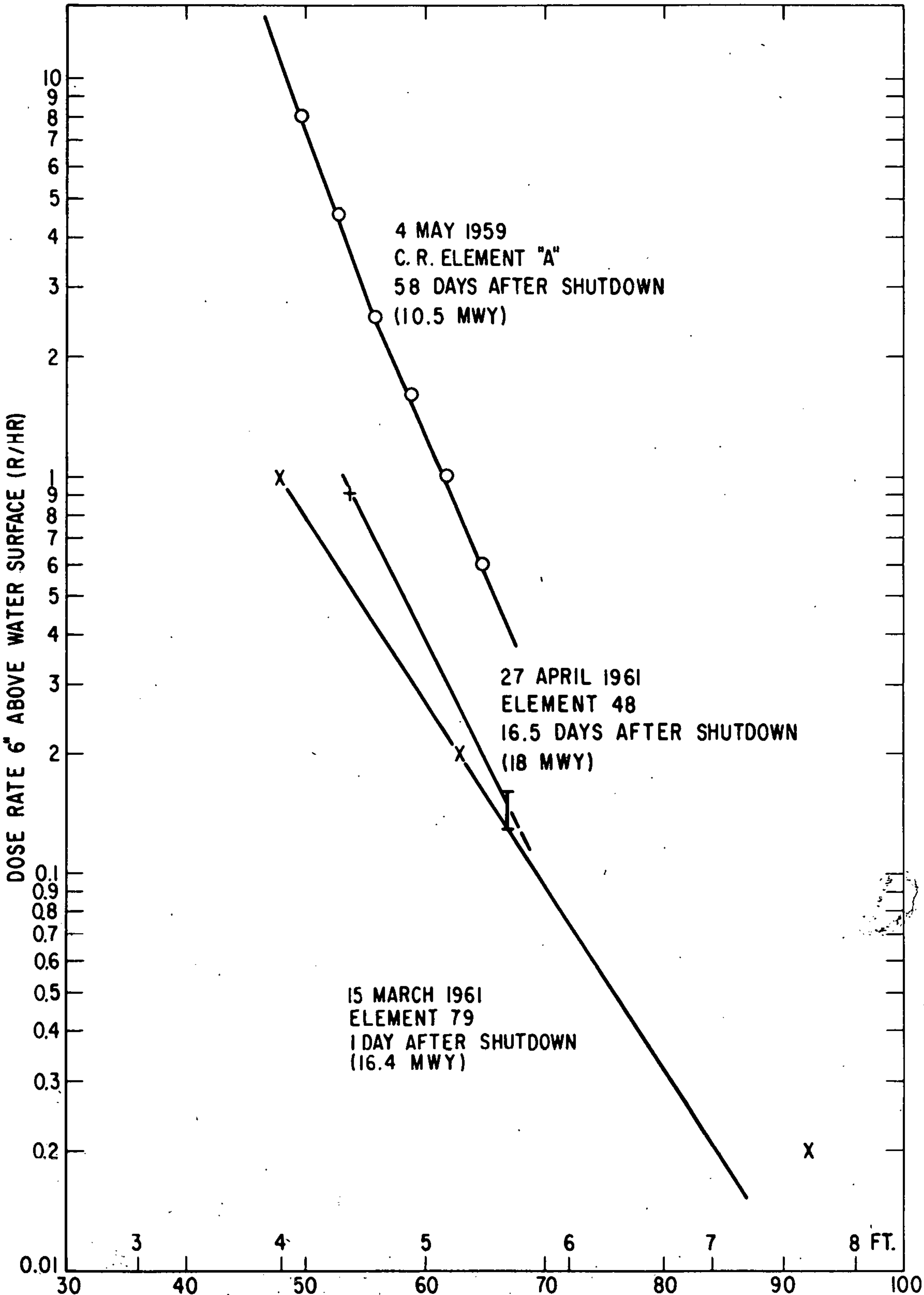
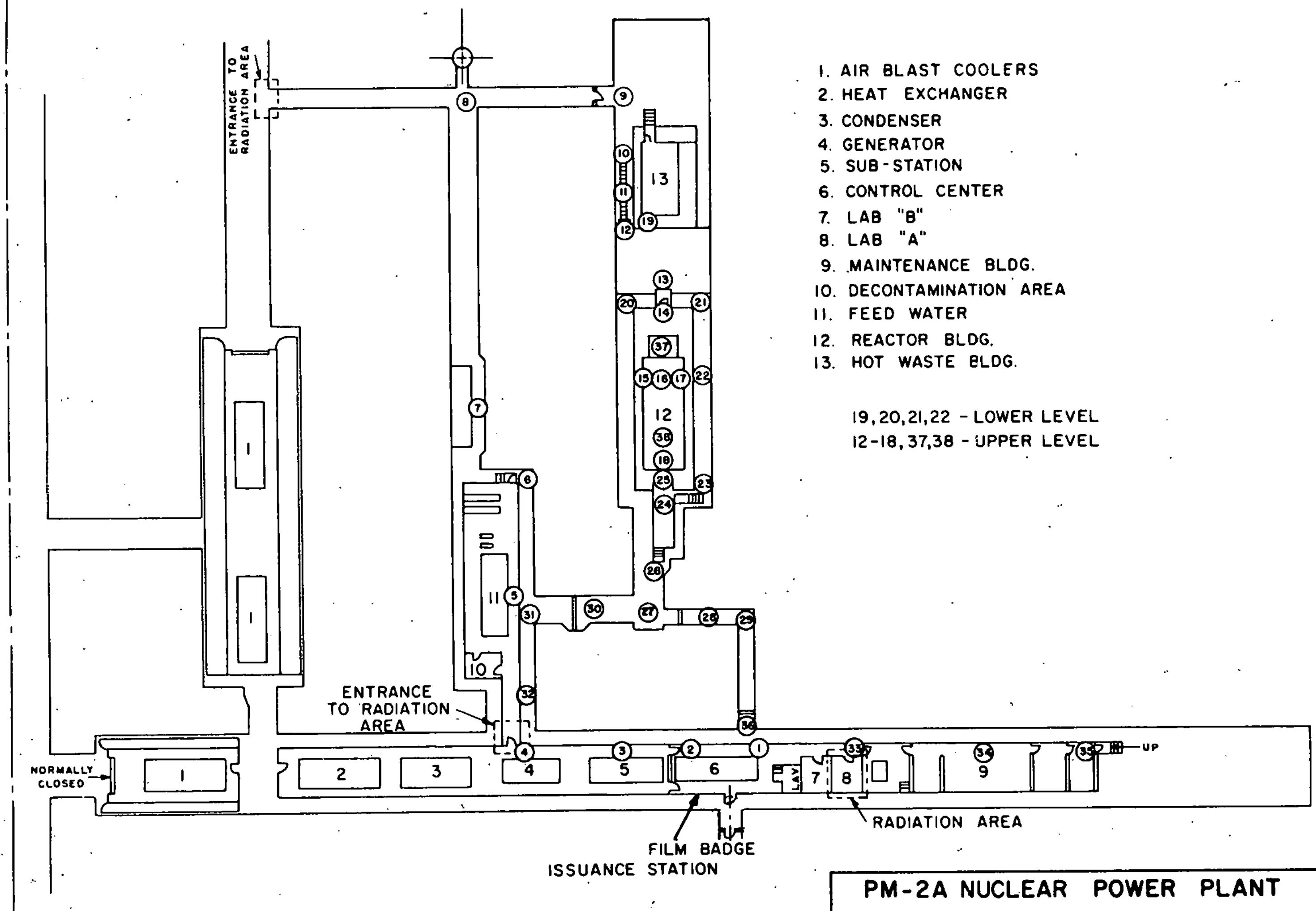


FIGURE 4.5

FIGURE 4.6



PM-2A NUCLEAR POWER PLANT
PLOT PLAN
RADIATION AREA LOCATION

TASK 5 - PRIMARY SYSTEM ACTIVITY CONTROL

Task Engineer - C. A. Bergmann

Task Definition and Objective

Task 5 is concerned with chemical and radiochemical aspects of primary coolant activity. Work is being performed in two major areas: Item 5.1 - Activity Buildup and Coolant Chemistry, and Item 5.2 - Fission Product Studies. The long term objectives of this task are to determine the means for control, prediction, and reduction of primary system activity. The activity may originate from the formation and release of induced radioactive materials and fission products. Sampling in support of Task 5 is carried out under the 100 and 200 series test procedures. Work performed under each item during the last six months was divided into these areas:

Item 5.1

1. Radiochemical and chemical analyses of induced activity of the SM-1 and PM-2A coolant and crud samples. Evaluation of the data by statistical and mathematical techniques in order to better understand and predict buildup of activity in present and future reactors.

2. Development of chemical methods for control of activity buildup. These methods include studies of coolant additives in the laboratory and test operation of the SM-1 with high pH coolant.

3. Determination of a radiochemical analysis procedure for induced and fission product europium in the SM-1 coolant.

Item 5.2

Radiochemical analyses and evaluation of data for fission product activities from SM-1 and PM-2A coolant and crud samples.

Summary of Second-Half Results

Item 5.1

Chemical and radiochemical analyses for induced nuclides of six crud and coolant samples from the SM-1 and PM-2A were performed. In addition, crud from the lower portion of the SM-1 reactor was analyzed. The detailed results of the SM-1 and PM-2A coolant and crud samples are reported in the April, May, June, and July-August, 1961 progress reports. Additional data obtained from the PM-2A test procedures is listed and evaluated in APAE-92. Individual 500-ml samples of SM-1 coolant were filtered through various sizes (5.0 to 0.01 microns) of Millipore filters (TP-A217) at the Schenectady laboratories. The activity of the material on the filters and the activity of the filtered water was determined as cpm/ml of water. Work was started on the deposits removed from metal coupon samples taken from the SM-1 in April, 1961.

In order to compare the activity of the crud removed from the lower portion of the SM-1 reactor pressure vessel to that of "normal" coolant crud, the percent distribution of induced nuclides in the crud was calculated. The results are listed below:

Sample	Sample Date	EFPH	Percent Distribution Co^{60}	Co^{58}	Fe^{59}	Cr^{51}	Mn^{54}
Primary Crud	11/1/60	13,958	71	16	2.6	0.6	10
Primary Crud	11/3/60	13,958	69	20	2.8	0.6	7.4
Primary Crud	1/4/61	14,647	50	25	9.5	8.7	7.0
Primary Crud	1/18/61	14,761	41	29	12	13	5.6
Crud from lower Portion of	5/3/61	15,209	79	14	5.4	1.5	1.0
Pressure Vessel	5/3/61	15,209	87	7	3.4	1.2	1.2

In general, the data reflect the SM-1 plant operating schedule in that the percent contribution of Co^{60} activity to the total induced activity in the crud decreases with increased power output. This can be seen by comparison of the data obtained from the November samples to that found from the January samples. The reactor had essentially been down from April, 1960 to November, 1960 and thus the shorter-lived nuclides had decayed away during this period, causing the long-lived Co^{60} activity to contribute about 70 percent to the total activity level. Previous to and during the January sampling dates, the reactor had operated at an average power level of about 40%, and the activity of the shorter-lived nuclides increased at a faster rate than that of Co^{60} activity. Thus, the contribution of Co^{60} activity to the total decreased. The high percent of Co^{60} activity in the May 3 samples (those removed from the lower portion of the pressure vessel) can be partially due to decay and low power operations. From January 18, 1961 to April 14, 1961, the average power level of the reactor was about 20%. From April 15th to May 3rd the reactor was down. In addition, the crud from the lower portion of the pressure vessel is probably older than typical coolant crud. For the above reasons, a high percent of Co^{60} activity in the crud is not unexpected.

In order to obtain an idea of the origin of significant sources of activity in the PM-2A coolant, theoretical ratios of the Co^{58} total coolant activity to the Co^{60} activity were calculated by two methods. In the first method, it was assumed that coolant activity originated only from corrosion release of activated in-flux components and core cladding. * The calculated Co^{58}/Co^{60} ratio for this case was 960. In the second case, it was assumed that coolant activity was due to an equal

* APAE No. 51, "SM-1 R&D Activity Buildup Program Task I Final Report - Feb. 1958 to June 1959," August 10, 1959.

combination of corrosion release of in-flux components and a release of corrosion products originating from out-of-flux components deposited and activated in-flux. * This calculated Co^{58}/Co^{60} ratio was 37.5.

From the two PM-2A coolant samples, the Co^{58}/Co^{60} ratios found from the data were 7.3 and 32.4. Thus, it appears that a significant amount of induced coolant activity in the PM-2A is due to release of out-of-flux corrosion products deposited and activated in-flux. If this is the case, the use of extremely low cobalt impurity stainless steel in-core may not be justified when a considerable amount of Co^{60} activity arises from "normal" cobalt impurity stainless in out-of-core areas. It should be pointed out that the above statements and calculations are based on very limited data, and that considerably more information on PM-2A coolant activity is required before definite conclusions can be made.

Test Request, Test Specifications and Test Procedure A-220, "Determination of Mn^{56}, Cs^{138} and Ba^{139} Activities in the Coolant," were submitted for approval. Data obtained from this test will be useful in checking release rates of induced and fission product nuclides. Since the nuclides that will be analyzed for are all short-lived (2.6 hr or less half-life), the reactor only has to be at power for 24 hr in order to obtain valid equilibrium data.

In the Mid-Year Summary Report (APAE-86) it was noted that specific activity and crud levels in the primary coolant increased markedly following thermal and certain chemical shocks to the primary system. The increase was attributed to high oxygen levels and/or thermal shock since it was reasoned that either parameter could be responsible for weakening the bonding of the in-flux core deposits. These deposits being in-core would be highly activated and thus would increase the circulating crud activity. It appears the reasoning was valid. A considerable amount of crud was found to be deposited in the inlet edges of the fuel plates in element #79 which was removed at the end of core life. This buildup is shown by comparison of Fig. 5.1 (a) and 5.1(b), which shows the inlet and outlet edges of the fuel plates in the fuel element. The buildup of crud is at the high flux end of the element. ORNL analysis of fuel element 79 end boxes showed 30 times more crud in the inlet than in the outlet. ** Conservative calculations based on crud thickness measurements from the photograph of the fuel element indicate that at least one-third of the total in-flux crud is located in the entrance deposits on the fuel plates.

Examination of Fig. 5.1(a) shows indications of a recent release of part of the deposit. Release of the activated deposit would result in distribution of the crud throughout the primary system. The in-core deposits are probably due to impingement, and because of thickness would be expected to be subject to spalling through thermal or chemical shock.

Further refinements in the gross deposition rate of crud were made. From the deposition rate of crud the release rate may also be determined. From knowledge

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* AP Note-239, "Effect of an Inconel Steam Generator on Deposited Activity and Water Treatment of the PM-2A," March 25, 1960.

** Personal communication, R. D. Robertson, Alco Products, Inc., July 28, 1961.

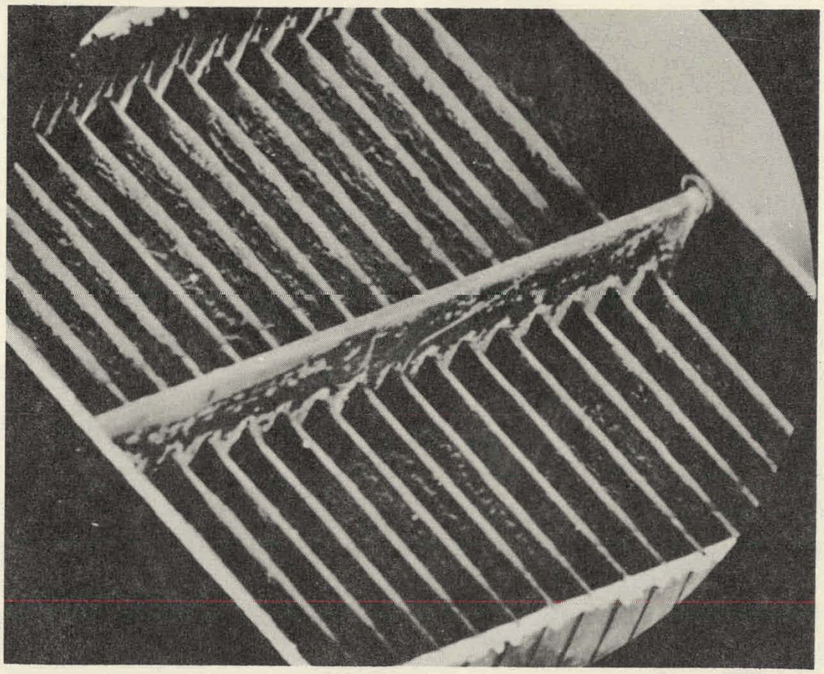

Figure 5. 1 (a) - Photograph of Inlet Edges of Fuel Plates in Element No. 79.

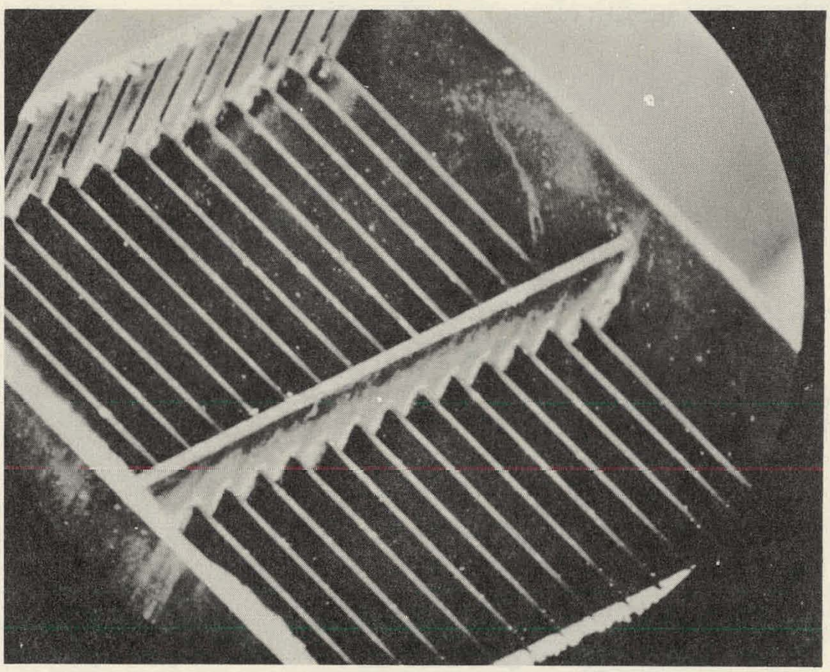

Figure 5. 1 (b) - Photograph of Outlet Edges of Fuel Plates in Element No. 79.

of these two rates, the equilibrium crud level can be determined. In order to determine the gross deposition rate on SM-1 primary system surfaces, data obtained from 27 coupons, which had been inserted in the purification line, were studied. The deposition rate was based on the use of Fe^{59} as a tracer of crud. The deposition rate as a function of exposure hours of the coupons is shown in Fig. 5. 2. It is seen that the rate is high for fresh surfaces but after about 3000 hr decreases to an approximately constant value of 1.90×10^{-2} mg/dm^2/hr. This value corresponds to a release rate of 2.35×10^{-2} mg/dm^2/hr, and an equilibrium crud level of about 0. 07 ppm. This crud level is in line with values measured at the SM-1. Using these deposition and release rates, the Fe^{59} activity in the baffle plate deposits was calculated as 4.04×10^4 dpm/mg. This level of activity is below the limit of detection. Since no Fe^{59} activity was detected in the baffle plate deposits, the values for the deposition and release rates appear to be realistic.

Work was begun on equations to predict when each of the three reactors would need to be decontaminated. Seven simultaneous linear differential equations describing activity buildup, * were programmed for the analog computer. Deposition probabilities were determined for three of the four nuclides of interest (Co^{60}, Co^{58}, and Mn^{54}) from SM-1 operating data and analysis of baffle plate samples obtained after 17 months operation of SM-1. Curves of the total activity of Co^{60}, Co^{58} and Mn^{54} nuclides in the SM-1 coolant vs. time were graphically integrated to the time when the baffle plate samples were removed. The decay of Co^{58} and Mn^{54} were accounted for. The decay of Co^{60} during this period would not be significant. These numbers were compared with nuclide activity in the system at that time, assuming the same activity throughout the system as on the baffle plate, to determine the effectiveness of the purification system.

It was found that 33% of the Co^{60}, 16% of the Co^{58}, and 85% of the Mn^{54} activities were removed. The activity of Fe^{59} was below the level of detection on the baffle plate so that a similar calculation for this nuclide was not made. The release probability for each nuclide is being varied in the analog program so as to fit the activity buildup data obtained from SM-1.

The study of methods for improved activity control techniques was continued. A literature review of presently available or contemplated methods was issued.[1] In summary, it appears that two general methods of activity control and reduction are feasible. Both involve control of coolant chemistry in such a way that; (1) the major portion of active material remains in-core, or (2) deposition is inhibited and better transport and removal by the purification system occurs. No commercially available equipment satisfies the above requirements. At present, studies of methods for activity reduction are being pursued along two avenues: (a) bench studies to develop additive for further investigation, and (b) in-plant (SM-1) testing of promising techniques.

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* APAE-77, "Long-Lived Induced Activity Buildup in the SM-1 During Core I Lifetime.

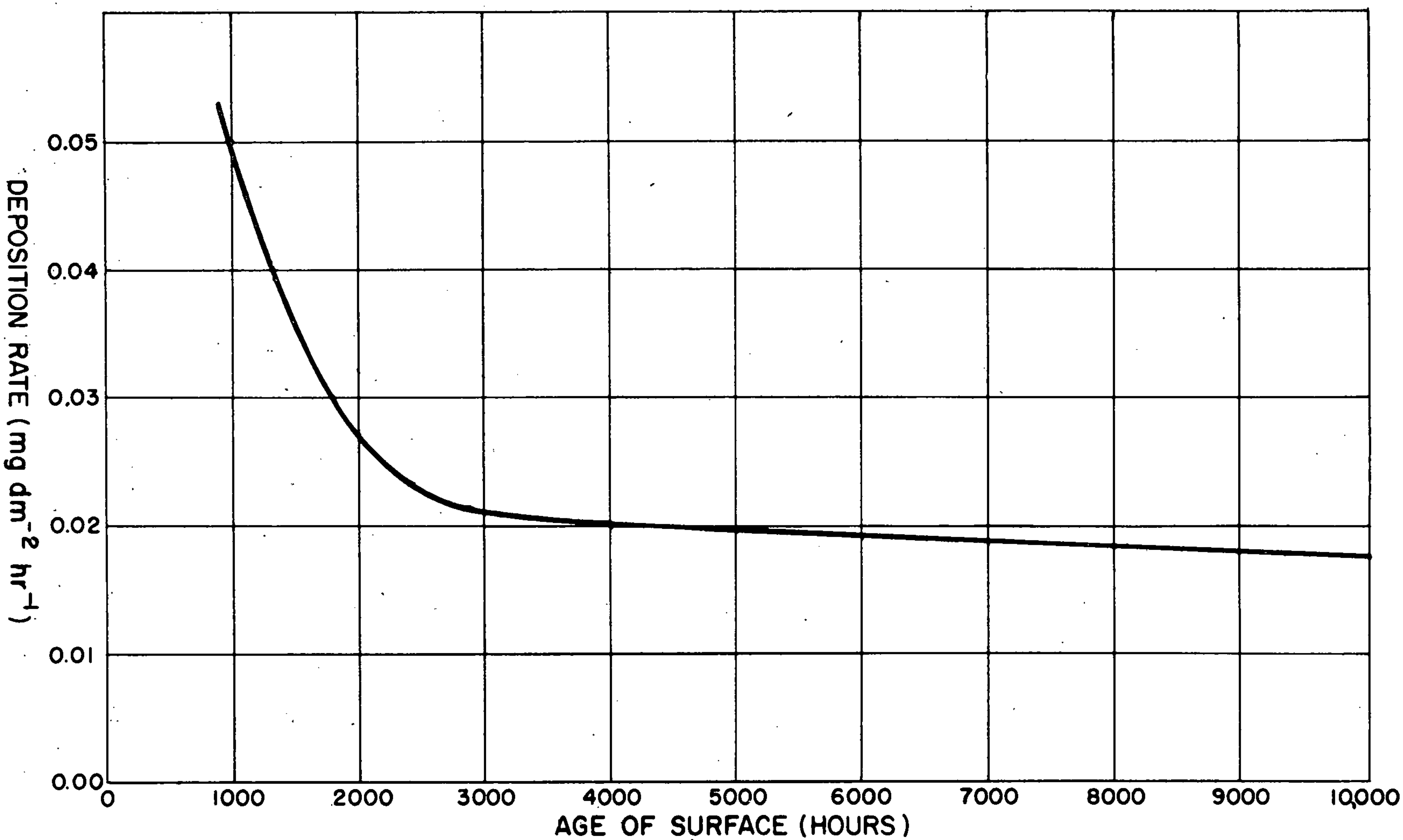

FIGURE 5.2

RATE OF DEPOSITION OF MAGNETITE (CORROSION PRODUCTS) ON STAINLESS STEEL (304) AT 430° F IN NEUTRAL HYDROGENATED WATER

Under item (a) preliminary bench studies, aimed at setting up valid criteria for selecting promising additives, were begun. Synthetic samples of non-radioactive crud were made, under an inert atmosphere, by precipitation of the chloride salts using ammonium and lithium hydroxides. The hydroxides were converted to the oxide form (magnetite) in an autoclave. The physical and chemical characteristics of the synthetic crud samples were then determined. Additional tests are required before any definite conclusions can be made.

Under item (b) the testing of promising techniques includes operation of the SM-1 at high pH (TP-A112) and the improvement in activity control with stricter oxygen control (TR-A113). Test A112 is scheduled to begin with plant startup around November 15, 1961. Installation of equipment necessary for the high pH test is proceeding on schedule. In conjunction with the high pH test and also those tests concerned with demineralizer activity measurement (A214, A215 and A216), Test Request and Specifications A221 were written. This test involves sampling of a small by-pass stream of the purification flow by means of a mixed bed demineralizer. The equipment for performing this test is already installed at the SM-1 and thus little effort is required to obtain the samples.

Test Request and Specifications for measurement of the improvement in activity control with improved oxygen control were issued for NYOO approval (TR and TS A113). This test is planned to provide better oxygen and hydrogen control at the SM-1. It is also planned to run two tests in which oxygen will be deliberately added to the coolant in order to determine whether crud bursts are due to high oxygen levels, or thermal shock to the fuel elements.

A radiochemical method for the analysis for europium in reactor coolants was developed. Basically, the method involves the separation of europium and other trivalent rare earth activities from other fission product and induced nuclides by fluoride precipitation. The europium is separated from the other rare earths by reduction to the +2 state with zinc dust followed by precipitation of the unreduced rare earths as hydroxides. Europium in the +2 state is not precipitated as the hydroxide. The europium is then counted using a G-M tube and scaler. One run was made at Schenectady on an SM-1 coolant and crud sample. However, only the long-lived europium nuclides remained and the levels were almost below the limit of detection. It is planned to have the procedure checked out at the SM-1 so that short-lived europium fission products will also be included. However, due to the low fission yield of the europium isotopes, and barring a significant fuel leak, it appears that if europium is found in the coolant it most probably is due to an absorber leak.

Item 5.2

Radiochemical analyses for fission products of coolant and crud samples taken from the SM-1 and PM-2A in January, February, March and May, 1961 were performed. Results of the analyses are given in the April, June, and July-August, 1961 progress reports. Work on the analyses for fission products in the deposits removed from certain SM-1 coupons was started. The possibility of diffusion of I_2 through polyethylene bottles was investigated by obtaining samples of untreated primary coolant shipped in glass and polyethylene bottles and analyzing for I^{131}. Evaluation of gross fission product iodine data from the PM-2A was performed. The variance in the gross iodine levels was not consistent with length of reactor operation. For example, I_2 levels were lowest during a run of about 205 hr at constant power, and were highest during a run of 67 hr at constant power. The gross fission product iodine found in the PM-2A after two months of operation were about 100-1000 times less than those found in the SM-1 after 18 months operation. A half-life of 2.6 hr was found for the short-lived activity in the PM-2A coolant and crud. This value agrees with the half-life of short-lived activities found in the SM-1. The primary contribution to the short-lived activity is probably due to fission products and Mn^{56}. Listing of the fission product data obtained to date is given in APAE-92.

Evaluation of the Cs^{137} and Sr^{90} activity levels found in the SM-1 was performed. Figure 5.3 shows the increase in Cs^{137} and Sr^{90} activity in the SM-1 coolant during Core I and spiked core operation. The increase corroborates earlier conclusions*,** that a defective element, or elements, was present in the SM-1. The earlier conclusions were based on fission product iodine data obtained during special tests. If it is assumed that cesium and strontium are found primarily in the filtered coolant only, and not in the crud, then their levels in the coolant are governed only by the rate of purification removal. (Analysis of filtered coolant and crud for these nuclides show that this assumption is valid; insignificant amount of Cs^{137} and Sr^{90} are found in the crud). Furthermore, if the production of long-lived fission products is due only to recoil from uranium surface and/or bulk impurity contamination, their concentrations in the coolant should be constant. As Fig. 5.3 shows, the concentrations of the long-lived fission products had increased, and thus, it was concluded that a defect had occurred in one or more fuel plate. As further evidence of fuel element defects, it was noted that gross fission product iodine levels decreased from 20,000 cpm during Core I spiked core operation, to about 8000 cpm during Core II operation, for comparable periods of reactor operation. These iodine levels were measured by passing a coolant sample through a cation exchange resin and counting the effluent on a gamma spectrometer. The spectrometer was biased so as to detect those gamma photons emitted by iodine only.

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* APAE-44, "SM-1 (APPR-1) R&D Program Final Report on Short-Lived and Fission Product Activity in SM-1 Primary Coolant, Task III," March 10, 1959.

** APAE-76, "Fission Product Activity in SM-1 Core I Primary System and Surface Contamination on SM-1 Type Fuel Elements - Task XVIV Phases 2 and 3," February 28, 1961.

BUILDUP OF TWO LONG-LIVED
FISSION PRODUCTS IN
SM-I PRIMARY COOLANT

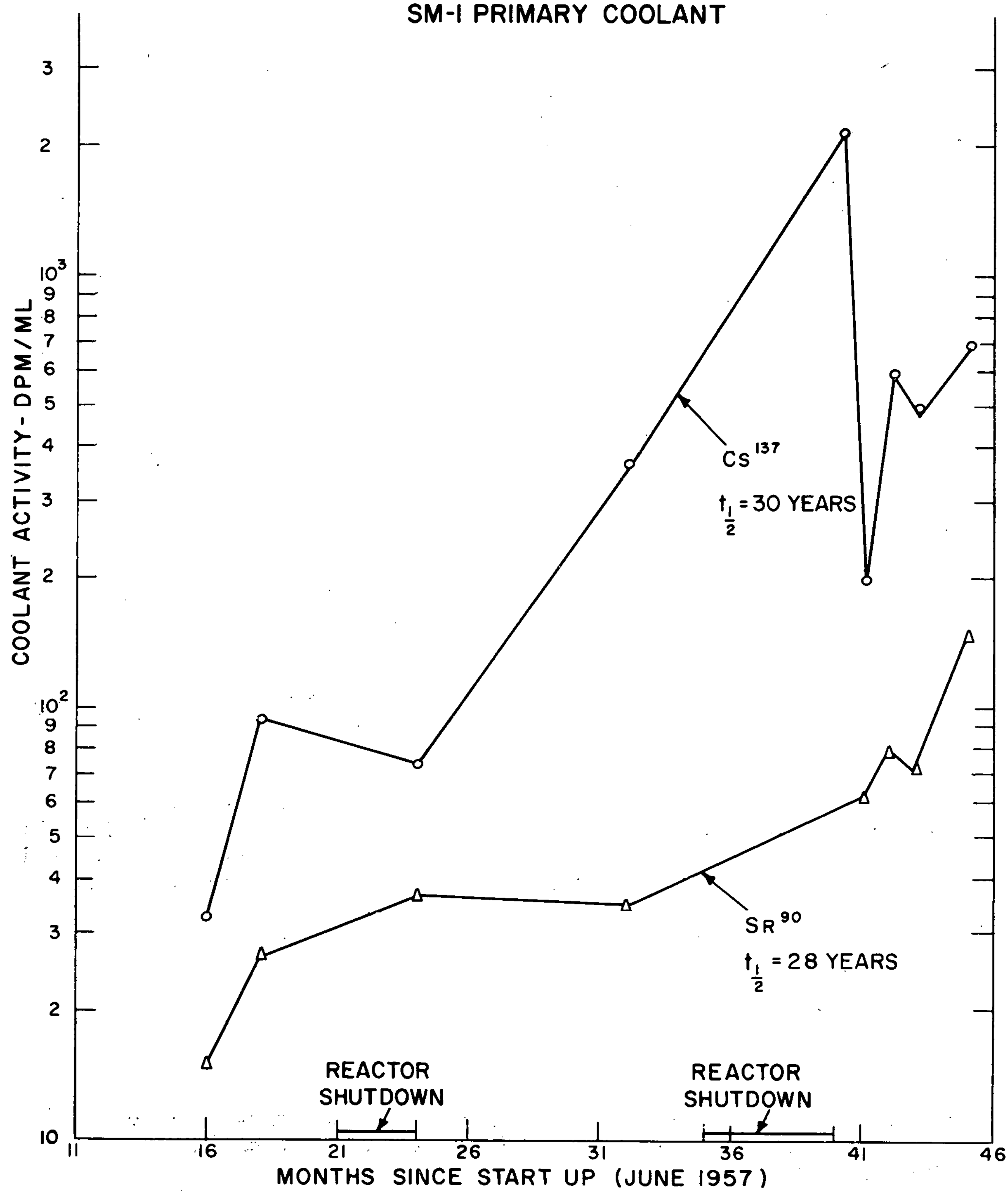


FIGURE 5.3

Conclusions

Item 5.1

1. The results of filtering coolant samples through various sizes of Millipore filters showed that all filters, with the exception of the smallest three pore sizes (0.10, 0.05 and 0.01 microns), retained approximately the same amount of activity. This indicates that nearly all particles have an effective diameter of 5 microns or greater or that all filters are equally effective in removing particulate matter regardless of filter pore size. The results of the filter test also suggest that a major portion of the coolant activity is either in the colloidal, ionic, or very small particulate, state.

2. Analysis of preliminary data from the PM-2A showed that a significant amount of induced activity originates from release of out-of-flux corrosion products deposited and activated in-flux. If this is the case, the use of extremely low cobalt impurity stainless steel in-core may not be justified when a considerable amount of Co^{60} activity arises from "normal" (about 0.1% cobalt) cobalt impurity stainless steel in out-of-core areas.

3. Examination of fuel element number 79 removed from the SM-1 showed that a considerable amount of crud had built up on the inlet edge of the fuel plates. It is estimated that about one-third of the total in-flux crud is located on the inlet edges of fuel plates.

4. The gross deposition rate of crud calculated from Fe^{59} activity found on coupon samples is in agreement with that observed at the SM-1. Analysis of activity removed by the purification system showed that not all nuclides are removed at the same rate, and since most of the Mn^{54} activity is in the non-filterable portion of the coolant, the results suggest that non-filterables are more readily removed from the coolant than the filterables.

5. Results of a literature survey indicated that two general methods of activity control and reduction are feasible. Both methods involve control of coolant chemistry in such a way that (1) the major portions of active material remains in-core, or (2) deposition on out-of-core surfaces is inhibited and better transport and removal by the purification system is effected. Under item (1), it was concluded that operation of the SM-1 at high pH coolant would reduce the rate of activity transport and thus keep the major portion of the activity in-core. In addition, stricter oxygen control through better hydrogen control should reduce the transport rate. Under item (2), the results of laboratory studies to date are not sufficient to allow definite conclusions.

6. A radiochemical analysis for europium made on one SM-1 coolant and crud sample, showed that the method developed was workable. In addition, the europium can be counted on a G-M tube and scaler. This type of equipment

is available at the reactor sites. Since the analysis was performed in Schenectady, only the long-lived europium activities were present. In order to obtain an idea of the sensitivity of the method, the analysis should be performed at the SM-1, so that short-lived europium fission products will also be included.

Item 5. 2

1. The iodine analyses of samples in polyethylene and glass bottles showed that the concentration of I^{131} was the same in both shipping containers. Thus, it appears that iodine does not diffuse out of a polyethylene bottle. Evaluation of PM-2A gross fission product iodine data showed that after two months of reactor operation, the iodine levels in the PM-2A were a factor of 100-1000 times less than those found in the SM-1 after 18 months operation. A direct comparison cannot be made, since the method of iodine analysis used at the PM-2A, was not in use at the SM-1 after it had operated for 2 months. The levels found in the PM-2A,however, should not pose any operational problem, since the SM-1 has operated at levels considerably greater, with little effect on reactor operations.

2. Evaluation of the Cs^{137} and Sr^{90} activity levels found in the SM-1 corroborates earlier tentative conclusions that a defective element, or elements was present in the SM-1 during Core I and spiked Core I operation. Since the levels of these fission products kept increasing with time of reactor operation, surface contamination and/or bulk impurity contamination is not responsible for a significant amount of fission products. If these two mechanisms were the source of fission products, the levels would remain constant. The results of gross fission product iodine monitoring during Core I and Core II also show that one or more defective element was present during Core I operation.

Recommendations

1. Sampling of SM-1 coolant, crud and metal coupons, as well as coolant and crud samples from the PM-2A and SM-1A, should be continued. Radiochemical analysis should be made for induced and certain fission product nuclides. At the SM-1, the analysis for Mn^{56}, Ba^{139} and Cs^{138} should also be performed. Since these nuclides have half-lives of less than 3 hr, the sampling will have to be done at the SM-1. Sampling of a small by-pass stream of the SM-1 purification flow by means of a mixed bed demineralizer should be done.

2. A determination of the best method for gross iodine analysis (Test Procedure A207) should be done in Schenectady with the use of I^{131} tracers. Since SM-1 operation is intermittent, it is felt that the use of a standard tracer will give better results than relying on the primary coolant as a constant source of I^{131}.

3. SM-1 operation at high pH should be started as soon as possible (TP-A112). Data from the test should be evaluated. If the results show that high pH operation is effective in controlling activity transport, high pH operation of other reactors should be considered.

4. Bench testing to determine the effects of various additives on crud characteristics should be continued. Use should be made of small autoclaves and dialysis membranes for determining which portion of the coolant (i. e. filterables or non-filterables) contribute significantly toward activity deposition. If suitable additives are found, experiments using a small high temperature, high pressure loop designed to simulate reactor conditions should be carried out.

5. The radiochemical procedure for euorpium analyses in the coolant should be tested at the SM-1. Testing is required to determine the contribution of the short-lived europium isotopes to the gross europium activity. After an analysis has been made, evaluation of the data and the sensitivity of the method should be performed.

6. Work on analytical expressions for predicting need for activity control and decontamination times at the reactors should be continued. Since certain constants in the expressions are dependent upon experimental data, evaluation of data from the reactors should also be continued. In addition, data evaluation is required to give insight into the basic mechanisms of activity transport, provide basis for methods of activity reduction, and supply data for health physics and waste disposal programs.

Future Work

1. Radiochemical and chemical analyses of induced and fission product activities of SM-1, SM-1A and PM-2A samples will be performed. Samples which show unusually low concentrations of elements will be analyzed using an emission spectrograph and special analyses made for foreign elements. Additional analyses of samples will be made to determine in which fraction of the coolant the activity is concentrated. A determination of the best method for I_2 analysis using an I^{131} standard as a tracer, will be made.

2. The SM-1 will be operated with high pH coolant. Evaluation of coolant data to determine the effect of high pH operation on activity reduction will start.

3. Laboratory testing of coolant additives will continue to determine the effects of the additives on the physical and chemical characteristics of simulated non-radioactive crud. It is hoped to find two types of additives (1) those that inhibit crud deposition, or (2) those that provide for more efficient removal by the purification system. In addition studies of activity transport mechanisms will be started using small autoclaves and tracers.

4. Work on equations to predict decontamination times and the need for activity control at the three reactors will continue. Coolant data obtained previously will be used to determine certain constants required in the equations. Other constants will be determined empirically by using an analog computer, and fitting the results to SM-1 experimental data. After the constants have been determined, analog runs will be made to calculate the expected radiation levels at the SM-1A and PM-2A.

BIBLIOGRAPHY - TASK 5

1. AP Note - 342, "A Review of Radioactivity Control Problems in Pressurized Water Reactors," June 1, 1961.

TASK 6 - ANALYSIS OF REACTOR PRESSURE VESSELS DURING OPERATING LIFE

Task Engineer - D. W. McLaughlin

Overall Task Definition and Objectives

1. To assure the continued safe operation of the SM-1 and to develop safe operating procedures for other field plants. Operating limits based upon irradiation damage and a detailed stress analysis must be established for each reactor vessel.

2. To provide actual reactor flux data for verification of design calculations.

3. To provide materials data which will supplement the data obtained in the NRL program.

4. To provide irradiation test data from power reactors not available to NRL aimed at both improving fundamental knowledge of irradiation damage and forming a sound basis for future design and operation of carbon or low alloy steel reactor vessels.

ITEM 6.2 - FLUX MEASUREMENTS IN SM-1, SM-1A, PM-2A

Task Definition and Objectives

The objective of this task is to provide on-site readings of fast flux dosage on the pressure vessels during operation, so that irradiation damage data obtained from test reactors may be more accurately evaluated, and to provide a correlation with calculated values.

Summary of Second-Half Results

The first flux monitor capsule containing cobalt and sulfur detectors inserted in SM-1 rearranged Core I was removed after being irradiated for approximately six months. Data on this capsule has been reported and is currently awaiting analysis by NRL. A second flux monitoring capsule containing four threshold detectors (cobalt, cadmium shielded cobalt, sulfur and nickel) was inserted in SM-1 Core II in April 1961 and is now under irradiation.

The flux monitoring system for the SM-1A internal vessel wall was designed by Alco, manufactured by NRL to Alco Drawing No. AEL-679 Rev. B, installed in the SM-1A, and is currently awaiting reactor startup. NRL has requested that these flux monitors remain in the SM-1A for 30 days of reactor operation at a high steady operating load. Approval has been obtained by Capt. Meeken of ARM to shut down the reactor and remove the monitors after the desired exposure. The monitors are iron, nickel and cobalt wires with intermittent cadmium shielding on the cobalt wire.

The interior of the PM-2A vessel was not found accessible for flux monitors.

Conclusions

None.

Recommendations

None.

Future Work

1. Remove and analyze the flux monitoring capsule in SM-1 Core II during the October shutdown.

2. Initiate irradiation of the SM-1A flux monitoring system upon startup of the SM-1A plant.

ITEM 6.3 - ACCELERATED TESTS IN LITR AND MTR OF SM-1, SM-1A, AND PM-2A REACTOR VESSEL MATERIAL

Task Definition and Objective

Provide radiation damage data on the materials of the three reactor vessels in advance of actual materials changes in the vessels themselves so that safe operating limits may be established to compensate for the changed mechanical properties.

Summary of Second-Half Results

Accelerated tests have been conducted in the MTR and LITR on SM-1, SM-1A and PM-2A reactor vessel materials to determine the transition temperature characteristics of each material and its relation to both integrated neutron flux and irradiation temperature. Test capsules containing samples of A-212B (SM-1), A-350-LF-1 modified (SM-1A), and A-350 LF-3 (PM-2A) material were irradiated "cold" ($\sim 200^{o}F$) and "hot" (at reactor operating temperature). Transition temperature shifts observed for these materials are tabulated in Table 6.1.

TABLE 6.1
CHANGES IN TRANSITION TEMPERATURE FOR IRRADIATED REACTOR VESSEL MATERIALS

Material		Irradiation Data		
Reactor	Type	Irrad. Temp.	NVT > 1 Mev.	ΔTT(^{o}F)
SM-1	A-212B	300^{o}F	1.3×10^{19}	260
SM-1	A-212B	275^{o}F	2.1×10^{19}	290
SM-1	A-212B	200^{o}F	2.5×10^{19}	295
SM-1	A-212B	430^{o}F	2.6×10^{19}	305
SM-1	A-212B	510^{o}F	2.2×10^{19}	210
SM-1A	A-350LF-1	200^{o}F	2.9×10^{19}	365
SM-1A	(Modified)	430^{o}F	3.1×10^{19}	395
PM-2A	A-350LF-3	275^{o}F	2.5×10^{19}	385
PM-2A	A-350LF-3	510^{o}F	2.7×10^{19}	320

One-cycle annealing studies were initiated on irradiated Army reactor vessel materials to determine the effect of elevated annealing temperatures on the recovery of the initial properties of these steels. The annealing studies to date have dealt primarily with the recovery of notch-ductility characteristics of Army pressure vessel steels as shown in Table 6.2.

Two-cycle "cold" irradiation-annealing studies are underway for the SM-1 and PM-2A vessel materials irradiated at $\sim 3 \times 10^{19}$ nvt. This two-cycle annealing at 750^{o}F will essentially complete all "cold" irradiation-annealing studies on the Army vessel steels. Future N-cycle annealing studies will be on "hot" irradiated materials only. Testing of the SM-1 material is also considered ecomplete since fluxes higher than 1-1.5 x 10^{19} nvt are not anticipated over a 20-year period. Cyclic irradiation annealing studies will therefore be directed toward the PM-2A and SM-1A materials only.

TABLE 6.2
ANNEALING DATA ON IRRDIATED REACTOR VESSEL MATERIALS

Material		Irradiation Data		Annealing Data*	
Reactor	Type	Irrad. Temp.	NVT > 1 Mev	Anneal. Temp.	% Recovery in TT
SM-1	A-212B	200°F	2.5×10^{19}	600°F	65%
SM-1	A-212B	430°F	2.6×10^{19}	600°F	26%
SM-1	A-212B	250°F	2.2×10^{19}	650°F	74%
SM-1	A-212B	250°F	2.2×10^{19}	750°F	95%
SM-1A	A-350 LF-1	200°F	2.9×10^{19}	600°F	74%
SM-1A	(Modified)	430°F	3.1×10^{19}	600°F	58%

* Annealing Time = 168 hr.

In addition to transition temperature determinations on the SM-1, SM-1A and PM-2A vessel materials, tensile specimens of these steels were prepared by NRL from Alco-supplied material (SM-1A and PM-2A) and irradiated to 3×10^{19} nvt in the MTR at temperatures of 200°F and at reactor operating temperatures (430°F for the SM-1 and SM-1A and 510°F for the PM-2A). An additional capsule containing drop weight, charpy, and tensile specimens of PM-2A material (A-350 LF-3) was also irradiated at 200°F in the MTR to approximately 5×10^{19} nvt and is currently awaiting testing. Elevated temperature (510°F) irradiation of a test capsule is also in progress in the LITR. An integrated neutron flux of 5×10^{19} nvt is anticipated for this capsule which contains a total of 80 tensile and charpy specimens of PM-2A materials. All PM-2A testing will be done first, followed by the SM-1A and SM-1 respectively.

Conclusions

1. Analysis of the above data indicates that the transition temperature shift is not significantly affected by operating temperatures up to 430°F. At 510°F, however, a 50° to 60° smaller transition temperature shift is encountered, as observed for the PM-2A reactor vessel material (A-350 LF-3) irradiated to 3×10^{19} nvt. This reduction in damage is similar to that previously reported by NRL on vessel materials irradiated at 550° to 575°F.

2. A preliminary study on the SM-1 which is similar in design to the SM-1A, has indicated that it appears feasible to anneal the SM-1 vessel with hot gases at 750°F. For the PM-2A vessel material, both temperatures of

750^oF and 600^oF will be investigated since it may be possible to sufficiently anneal this vessel using the reactor core and coolant (at 600^oF) as the heat source.

Recommendations

None.

Future Work

1. These irradiation-annealing tests will continue on the PM-2A and SM-1A materials for as many cycles as are necessary to insure a safe 20-yr service life for each plant.

2. A 750^oF anneal will be used on the SM-1A material since it has been shown that at this temperature, a major portion of the pre-irradiated material properties can be recovered.

3. Related work will continue in FY-62 under Subtasks 6.1 and 6.2.

ITEM 6.4 - DETERMINATION OF INCREASE IN TRANSITION TEMPERATURE OF SUB-SIZE SM-1 VESSEL SPECIMENS IRRADIATED IN SM-1 REACTOR DURING CORE I LIFE

Task Definition and Objectives

Determine the extent of embrittlement incurred during Core I life of the SM-1 pressure vessel, utilizing sub-size Izod impact specimens of actual vessel material installed when the reactor was built.

Summary of Second-Half Results

One capsule of the sub-size Izod impact specimens of SM-1 reactor vessel material irradiated during SM-1 Core I life (16.4 MWYR) was removed from the reactor, sent to ORNL and tested. The results of these tests are given in Table 6.3.

Conclusions

None.

Recommendations

None.

TABLE 6.3
CAPSULE SPECIMEN DATA TAKEN FROM IRRADIATED SM-1 REACTOR VESSEL MATERIAL AFTER 16.4 MWYR

	Notch Position	Sub-Size Izod Δ TT (°F)	Charpy Equiv. ΔTT (°F)*	NRL Indicated NVT**
Bottom	6	118°F (1)	145°F (1)	4.6×10^{18} (1)
	5	138°F	165°F	5.8×10^{18}
	4	99°F	125°F	4.2×10^{18}
	3	77°F	100°F	3.65×10^{18}
	2	52°F	75°F	2.9×10^{18}
Top	1	36°F	60°F	2.65×10^{18}

* Charpy Equivalent Δ TT = Sub-size Izod Δ TT + 25°F (per R. Bergen - ORNL)

** Estimated from NRL, NVT vs. Δ TT curve.

(1) Specimen Notch 6 (bottom) data questionable - showed evidence of overheating.

Future Work

Publication of a final report on the above data will complete this work. Related work will continue in FY-62 under Subtasks 6.1 and 6.2.

ITEM 6.5 - LIMITED SAMPLE TESTING IN THE SM-1A REACTOR

Task Definition and Objective

Provide radiation damage data on SM-1A vessel material under the nuclear and physical environment of the SM-1A pressure vessel.

Summary of Second-Half Results

A total of eight capsules were prepared by NRL for irradiation above the SM-1A core. These capsules contain Charpy-V and tensile specimens of A-201, A-212B, A-350 LF-1 (modified), A-350 LF-3 and HY-80 type steels. Four of the capsules have been inserted in the SM-1A core support ring, and the other four test capsules have been loaded in a "clothes hangar" assembly hung on two of the four reactor vessel tie rods. These capsules are currently awaiting irradiation.

Conclusions

None.

Recommendations

None.

Future Work

Upon initiation of SM-1A reactor operations, these capsules will be irradiated and returned to NRL for evaluation.

Related work will continue in FY-62 under Subtasks 6.1 and 6.2.

ITEM 6.6 - TESTING OF MATERIALS IN TWO FUEL ELEMENT POSITIONS IN SM-1 CORE II

Task Definition and Objective

The objective of this task is to provide accelerated embrittlement data on the three operating reactor vessel materials at integrated fast fluxes equivalent to full 20-yr operation. Five additional actual or potential vessel materials are also included.

Summary of Second-Half Results

Two dummy fuel elements containing encapsulated test specimens of various pressure vessel steels were fabricated and inserted in fuel element positions #12 and #72 of SM-1 Core II. Irradiation of these two test elements is currently in progress, and will continue until at least April 1962 for one element, and to the end of SM-1 Core II life for the other element.

Conclusions

None.

Recommendations

No need is evident at the present time for additional experiments of this type in the SM-1A Core I. The only NDT testing to be done in SM-1A involves the surveillance capsules discussed in Item 6.5.

Future Work

Continue irradiation of test specimen in SM-1 Core II under Subtask 6.4 in FY-62.

ITEM 6.7 - DETERMINATION OF VESSEL WALL DOSAGE

Task Definition and Objective

The task objective is to obtain the best estimate of the above 1 Mev flux on the SM-1, SM-1A, and PM-2A reactor vessels through analysis and measurements at Alco's Critical Facility.

Summary of Second-Half Results

An experimental and theoretical analysis of the integrated fast neutron flux in the SM-1 has been issued.[1]

Data reduction and analysis of the experimental data obtained during the SM-1 mockup portion of the program have been completed. Data on the neutron flux incident on the SM-1 pressure vessel wall at the 3.375 in. axial plane has been obtained for the thermal and epithermal energies, and for energies above 2.9, 5.3, and 8.6 Mev.

Over one-half the PM-2A core configuration was mocked up using reject SM-1 elements and fuel plates. The remainder of the core was loaded with SM-2 type elements. In order to maintain experimental integrity, the control rod bank in the SM-1 mocked up section of the core were set at the cold clean critical bank position of the PM-2A and criticality was attained using the eccentric control rod in the SM-2 section of the mockup.

Experiments to determine the relative power distribution in one quadrant of the symmetrical PM-2A core mockup have been completed. The data reduction is complete and the core average has been computed. These data are utilized for purposes of power normalization.

The experimental program to determine the neutron dose rate on the PM-2A pressure vessel wall utilizing the PM-2A mockup has been completed.

Analytical calculations for the SM-1, SM-1A, and PM-2A neutron spectrum and distributions have been completed using two calculational models. These results were being correlated to experimental measurements during this report period.

Experimental Results

Neutron activation experiments were completed in SM-1 and PM-2A core and vessel mockups under cold clean core conditions. Activation foils of sulfur, aluminum, bare and cadmium covered gold, U-238, and as B^{10} shielded U-238 were irradiated to establish the neutron intensity and approximate neutron energy spectra in core regions between the core centerline and the pressure vessel wall. The U-238 foils were not available for use during the SM-1 tests. The complete results of these tests will be reported(2) in the next half. From these data, an estimate of the integrated fast neutron flux incident on the SM-1 and PM-2A pressure vessel walls will be obtained.

Results of fast neutron flux measurements for the SM-1 have been reported previously in the June Progress Report.

Power distribution measurements have been obtained for the PM-2A mockup core and will provide a power normalization factor for the fast flux measurements. These data also provide useful information on power distribution in the previously unmapped PM-2A core for the core analyst. The power distribution was determined by making a detailed flux-map study of one quadrant of the core. The flux-map was obtained by instrumenting three fuel plates of each element in the quadrant with fission foils. The plates were instrumented in such a manner that an average activity for each fuel cell relative to a common standard could be calculated. The relative average cell activities are then normalized to yield a core average of unity and reported in this manner. Figure 6. 1 shows the PM-2A core mockup and the power distribution normalized to a core average of unity.

Calculation of the fast neutron flux per unit power (1 watt thermal) has not been completed. However, Fig. 6. 2 and 6. 3 illustrate the thermal and intermediate (4. 9 ev) radial flux distributions and the radial distribution of fast neutron flux (above 2. 9 Mev). The data presented are preliminary and subject to further analysis.

An experimental program plan for the use of threshold detectors in the form of nickel, iron, cobalt, and cadmium shielded cobalt to measure the neutron flux at the SM-1A pressure vessel inner surface during reactor operation was prepared and submitted. Provisions were made so that the experiment may be performed at any time during core life.

Analytical Results

In order to properly assess the structural damage within a reactor, it is necessary that a detailed knowledge of the spatial and energy dependence of the neutron flux be known. Accordingly, two calculational models are being used to determine the neutron spectrum and distribution: (1) PDQ, a two-dimensional code which gives fluxes at any desired point within the reactor, and (2) PIMG-2, a one-dimensional code which gives the energy dependent flux within the reactor.

The results of the calculations at critical positions in the SM-1, SM-1A and PM-2A are given below:

The fast flux (E > 1 Mev) at various positions of importance in the SM-1 are tabulated below.

Location	Nvt ($neuts/cm^2$) (20 Yr Period, 60% l. f.)	Nvt per MWYR ($neuts/cm^2$)	Nvt to Date** ($neuts/cm^2$)
Pressure Vessel*	1.42×10^{19}	1.18×10^{17}	2.28×10^{18}
Nozzle Weld	4.73×10^{17}	3.94×10^{15}	7.60×10^{16}
Support Ring	2.27×10^{18}	1.89×10^{16}	3.65×10^{17}

* Value in axial plane of maximum exposure.
** Based upon 19.3 MWYR total energy release, as of September 25, 1961.

The fast flux (E > 1 Mev) at various positions of importance in the SM-1A are tabulated blow.

Location	Nvt ($neuts/cm^2$) (20 Yr Period, 60% l. f.)	Nvt per MWYR ($neuts/cm^2$)	Nvt to Date** ($neuts/cm^2$)
Pressure Vessel*	8.05×10^{19}	3.35×10^{17}	-
Nozzle Weld	8.05×10^{18}	3.35×10^{16}	-
Support Ring	1.29×10^{19}	5.38×10^{16}	-

* Value in axial plane of maximum exposure.

** Note operational as of September 1961.

The fast flux (E > 1 Mev) at various positions of importance in the PM-2A are tabulated below.

Location	Nvt (neuts/cm^2) (20 Yr Period, 60% l. f.)	Nvt per MWYR (neuts/cm^2)	Nvt to Date** (neuts/cm^2)
Pressure Vessel*	3.2×10^{20}	2.67×10^{18}	5.33×10^{18}
Upper Nozzle Weld	8.96×10^{19}	7.47×10^{17}	1.49×10^{18}
Decay Cooling Nozzle	1.07×10^{20}	8.92×10^{17}	1.78×10^{18}
Vessel Flange Weld	5.6×10^{19}	4.67×10^{17}	9.34×10^{17}

* Value in axial plane of maximum exposure.
** Based upon 2 MWYR of total energy release.

Conclusions

The threshold and thermal detectors being used at Alco's Critical Facility have proven adequate to measure the fast and slow portions of the flux spectrum of the SM-1 and PM-2A reactors out to the pressure vessel. The analytical calculations of the fast flux on the pressure vessel using the various calculational models have proven consistent with one another when the variations in techniques were taken into account.

Recommendations

1. The entire program should be expanded, if possible, to allow a greater number of calculations and measurements to be performed. This would decrease the probable error in the final values published.

2. Additional methods of measuring the neutron flux in the Key range should be developed.

3. Definite consideration should be given to investigating possible methods of measuring the neutron flux spectrum and distribution on the SM-1 pressure vessel as a function of core lifetime. This would indicate the change in neutron spectrum during an operational core life.

Future Work

1. In the next half, a report summarizing the experimental work to date will be published.[2] This report will correlate the experimental data with analytical predictions.

2. Program planning, selection of resonance detectors, neutron detector and core mockup materials procurement, and a general review of the techniques of resonance activation measurements are to be completed.

3. Threshold and resonance foil activations are to be initiated in core mockups simulating operating conditions in the SM-1, SM-1A, and PM-2A plants.

4. This work will continue under Subtasks 6. 7 and 6. 8 in FY-62.

		13 0.789	14 0.737	15 0.789		
	22 1.128	23 1.092	24 0.378 1	25 1.092	26 1.128	
31 0.849	32 1.178	33 1.531	34 1.405	35 1.531	36 1.178	37 0.849
41 0.804	42 0.378 3	43 1.423	44 0.483 4	45 1.423	46 0.378 5	47 0.804
51 0.849	52 1.178	53 1.531	54 1.405	55 1.531	56 1.178	57 0.849
	62 1.128	63 1.092	64 0.378 2	65 1.092	66 1.128	
		73 0.789	74 0.737	75 0.789		

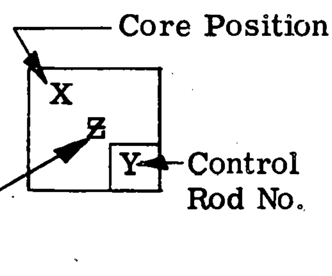


Figure 6.1 - PM-2A Core Configuration and Normalized Power Distribution

RADIAL THERMAL AND INTERMEDIATE (4.9 EV) FLUX DISTRIBUTION PM-2A

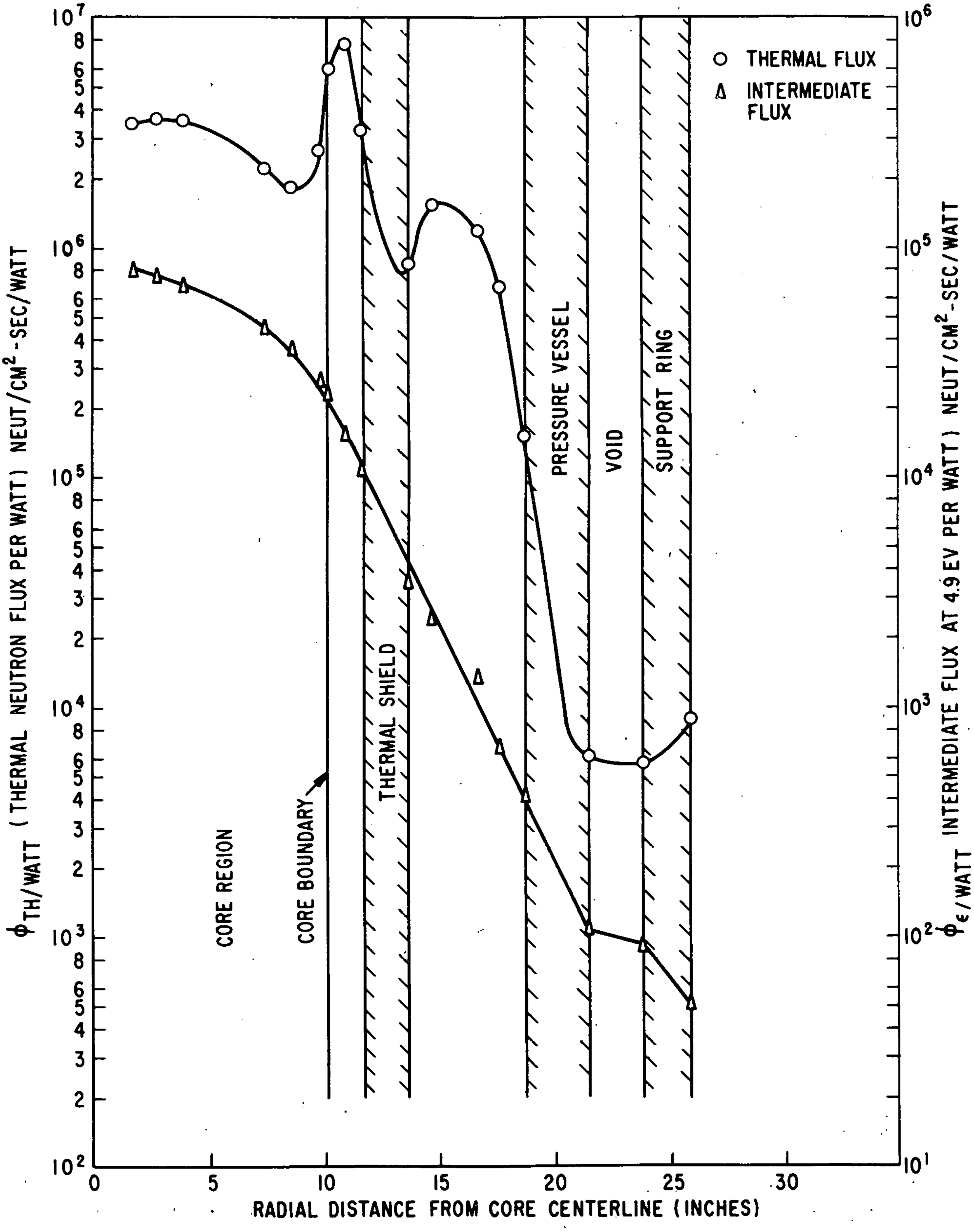


FIGURE 6.2

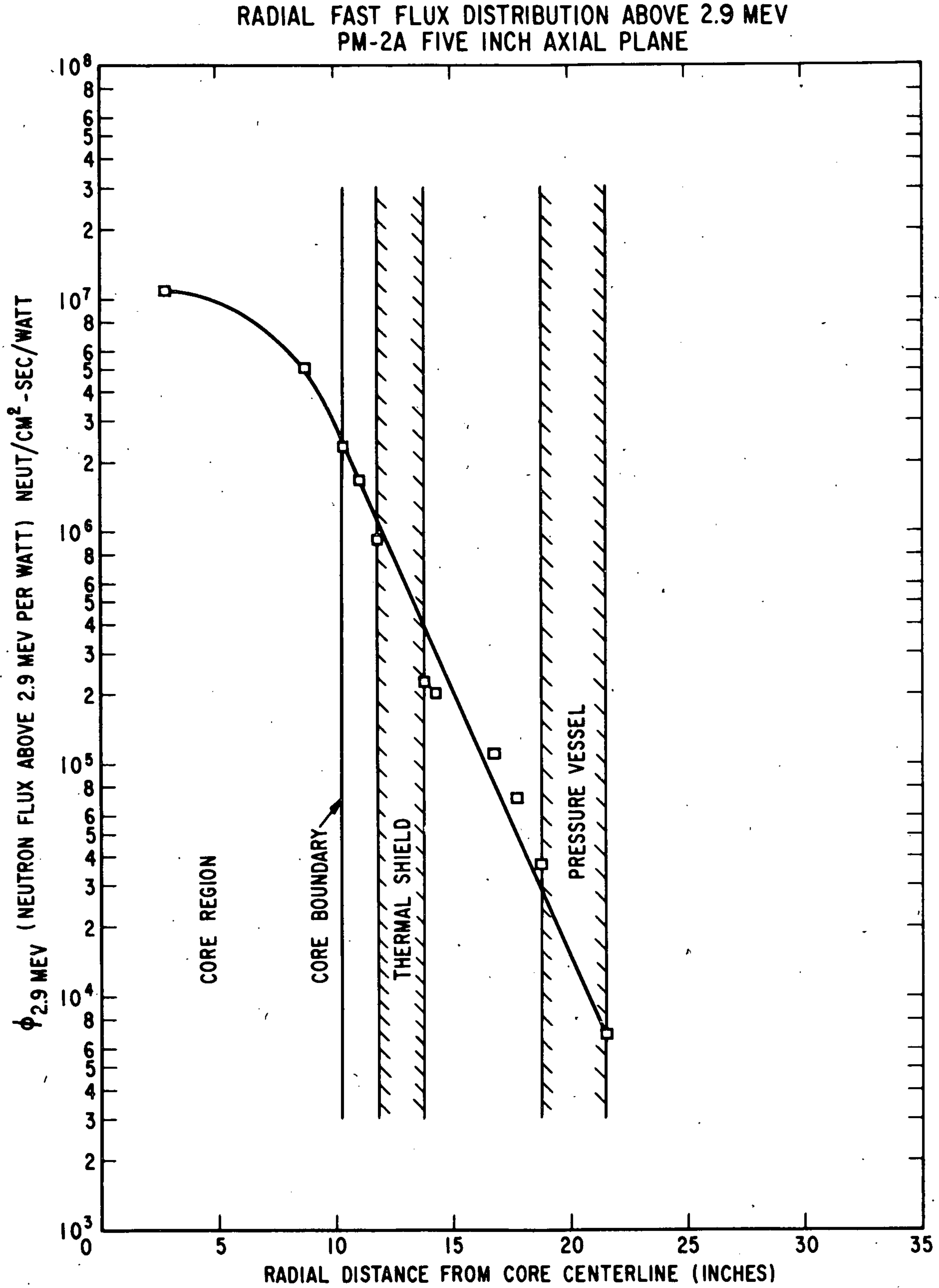

RADIAL FAST FLUX DISTRIBUTION ABOVE 2.9 MEV
PM-2A FIVE INCH AXIAL PLANE
$\phi_{2.9\,MEV}$ (NEUTRON FLUX ABOVE 2.9 MEV PER WATT) NEUT/CM^2-SEC/WATT
RADIAL DISTANCE FROM CORE CENTERLINE (INCHES)
CORE REGION
CORE BOUNDARY
THERMAL SHIELD
PRESSURE VESSEL

FIGURE 6.3

ITEM 6.8 - RADIATION DAMAGE TO METALS REVIEW

Task Definition and Objectives

Design criteria will be developed to guide the design of future reactor vessels and provide a means for determining safe operating limits for existing vessels. These criteria must provide a firm basis for the design and operation of irradiated structures and be consistent with sound engineering practices.

Summary of Second-Half Results

The task was completed with publication of the design criteria as a report(3). The criteria were as follows:

> When the operating temperature of the structure is below NDT +60° (the FTE), the maximum principal stress in the vessel at a point may not exceed 18 percent of the yield stress of the material at that point. The stresses to be considered are those due to pressure, thermal, and mechanical loads, applied singly or in combination, and include secondary stresses due to discontinuities and stress concentrations. Welding stresses are included only for non-stress-relieved vessels. In those areas of the vessel which are of simple, readily analyzed geometry and are free of flaws, the allowable stress is determined from the irradiated yield stress. In other areas of the vessel such as nozzle openings and welds, where stresses are more difficult to define and there is a higher probability that flaws exist, the allowable stress must be determined from the lower, unirradiated yield stress.

A general discussion of brittle fracture and a point-by-point derivation and evaluation of the criteria were included.

Conclusions

Above the NDT +60°, there is no danger of brittle fracture and stresses are limited only by the requirements of applicable design codes.

Recommendations

Efforts should be made to secure formal recognition of the criteria by professional engineering groups such as ASME.

Future Work

Work is complete on this item.

ITEM 6.9 - STRESS ANALYSIS OF REACTOR VESSELS

Task Definition and Objectives

To perform a detailed stress analysis of the SM-1, SM-1A and PM-2A reactor vessels, going beyond the scope of ASME code design calculations. To define safe operating limits for these vessels through a study of problems associated with the long time operation of reactor vessels: radiation damage to vessel material, and fatigue failure due to plastic strain cycling.

Summary of Second-Half Results

Results of the FY-61 program and the planned FY-62 radiation damage program were summarized in a report[4] prepared in response to an AEC questionnaire on radiation damage in the Army reactor vessels.

Safe operating limits have been established for the SM-1, SM-1A and PM-2A from a consideration of radiation damage. The operating limit curves for each reactor are shown in APAE-107. A separate study on the effect of irradiation on the service life of the SM-1A vessel[5] was also completed.

The second-half results are summarized below for each of the reactor vessels.

SM-1A Reactor Vessel

Temperature distributions in the SM-1A cover, flange and stud region were computed for steady state and 50^{o}/hr transient conditions as described below. Both the steady state thermal stresses and transient thermal stresses were completed. These were obtained by performing a discontinuity analysis for each condition using free body thermal displacements and rotations based on the average temperature of each piece of the cover and flange.

Results of Steady State Temperature Distribution Analysis, SM-1A

A. Method of Analysis

In support of thermal stress analysis of the SM-1A vessel and cover, steady state temperature distributions were calculated using the ROC code.

The physical geometry of the vessels and covers were reduced to rectangular geometry, each region having uniform properties. Estimates of boundary temperatures and convective film heat transfer coefficients were made. The problems were run in r, Z geometry.

For conservatism, the cross section analyzed passed through the center of the cover studs. In r, Z geometry this cross section is rotated to determine region volumes and integrated region temperatures. The heat flow paths between the bolts will reduce the severity of the temperature gradients somewhat.

B. Boundary Conditions

The bulk water temperature inside the vessel is at an average temperature of 433°F and natural convection is employed as the mode of heat transfer to determine the heat transfer film coefficient. The water surrounding the vessel cover is assumed to be at a bulk temperature of 150°F. The air outside of the vessel wall is also assumed at 150°F. Two further boundary conditions were employed for the water surrounding the nuts on the flange bolt circle (sections 36 and 37) and the water between the cover vessel wall (section 50). In these sections, the bulk water temperature was assumed to be at 175°F. A final boundary condition was chosen for the water on section 24 which was assumed to be at 350°F. Natural convection was assumed to be the governing mode of heat transfer in all the determinations of film coefficient.

The water in the gap between the bolt and cover was assumed to be at 200°F and for the steam condition at 250°F - 300°F. Conduction was assumed to be the mode of heat transfer in the gap. This gap was divided into four sections as seen in Fig. 1 - Sections 45-48.

The effect of gamma heat generation on temperature was investigated and found to be negligible, the greatest difference being only 0.6°F. A further variable in this analysis was the effect of steam or water on the space between the stud and vessel cover. (Sections 45 and 46 in attached Fig. 1).

The following summarizes the boundary conditions for this problem.

TABLE 6.4
SUMMARY OF BOUNDARY VALUES

Region Number	Location	Bulk Temp.	Heat Transfer Coefficient Btu/Hr-Ft²F
2	Water above reactor vessel cover	150°F	64.0
3	Air alongside of reactor vessel	150°F	1.0
4	Water inside reactor vessel	433°F	340
24	Water between cover & vessel, rear gasket (inside)	350°F	230
36	Water above bolt-vertical	175°F	80
37	Water above bolt - horizontal	175°F	60
50	Water between cover & vessel, near gasket (outside)	175°F	75

The following table lists the materials and properties used in this analysis.

TABLE 6.5
MATERIALS AND PROPERTIES

Part	Material	Btu/hr-ft^{o}F
Vessel body	A-350LF1	26.0
Vessel clad	A-304	10.3
Vessel insulation	Fiberglass	.04
Cover	A-350LF1	25.5
Cover Clad	A-304	10.3
Cover Insulation	Fiberglass	.04
Nut	A-403	9.2
Bolt	A-403	14.5
Gasket	A-304	10.00

C. Results

Figure 6.4 shows a sketch of the vessel and cover, the outline of the various regions and the region average temperatures as calculated by the ROC Code.

The solution indicates severe temperature gradients across the vessel flange and across the bolt hole in the cover ring.

Results of Transient Thermal Analysis, SM-1A

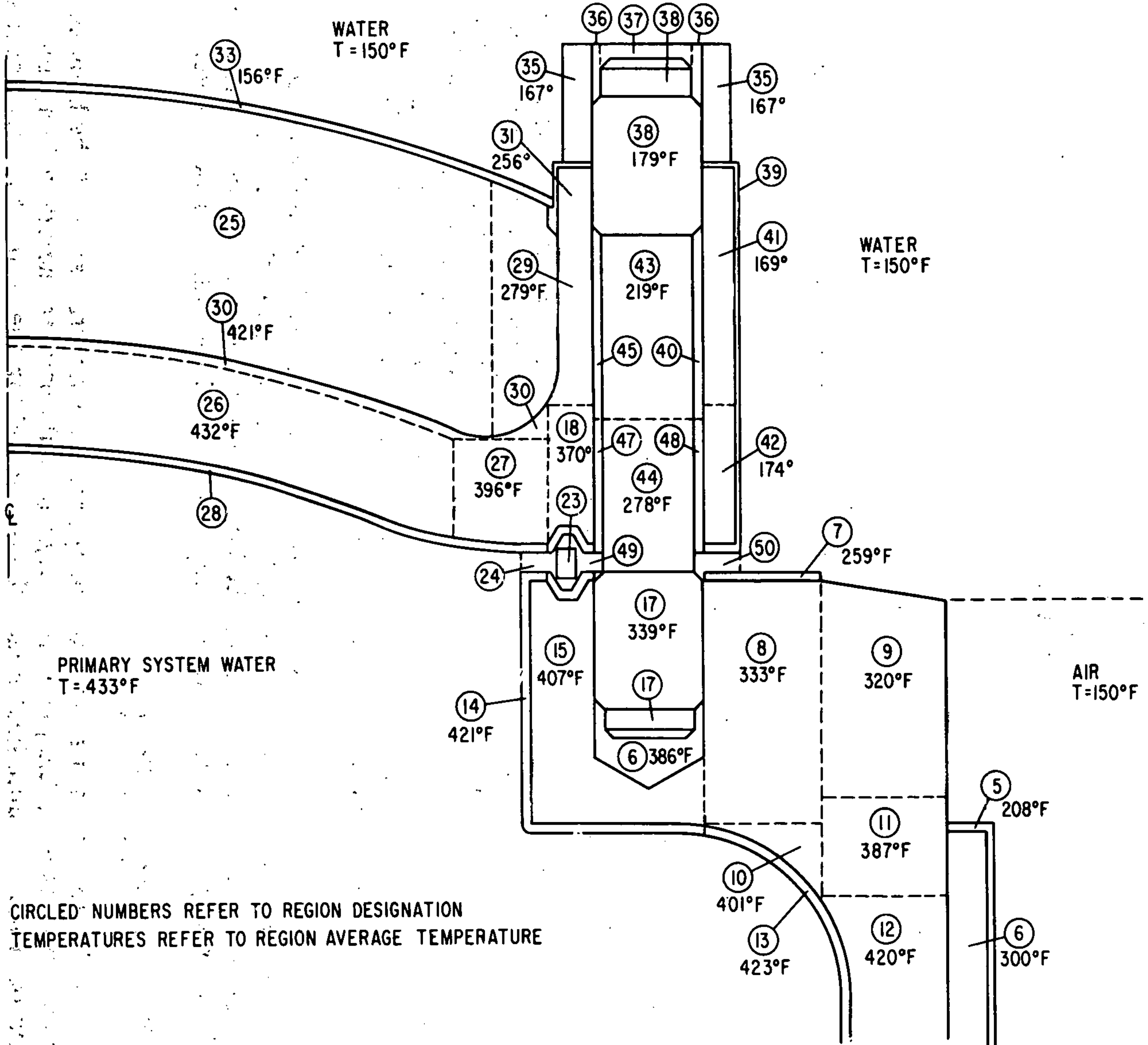
A. Method of Analysis

The most severe temperature differences and thermal stresses will occur during a heatup of the primary system. Consequently both of the reactors investigated were analyzed starting at a uniform metal and water temperature of 100^{o}F. A fifty degree per hour transient was imposed on the reactor systems. The transient Tiger II code was requested to provide temperature distributions at intermediate times between start of heatup and full system operating conditions in order to determine the point of worst temperature distribution.

The physical geometry of the transient problems are broken down into rectangular geometry to facilitate code input. The cross section of the covers and vessels was again taken through the bolt. As this code computes on the basis of slab geometry with constant depth, the results will be highly conservative.

B. Boundary Conditions

In order to determine the time-temperature characteristics of the boundary fluids for this reactor, the problem analyzed would have to include all physical regions out to an absolute heat sink. To do this within the limit of 25 surface nodes,



CIRCLED NUMBERS REFER TO REGION DESIGNATION
TEMPERATURES REFER TO REGION AVERAGE TEMPERATURE

SM-1A VESSEL AND COVER STEADY STATE TEMPERATURE DISTRIBUTION

FIGURE 6.4

the detail picture of the vessel and cover would have been lost. Therefore it was decided to hold all fluid boundaries to a constant 100°F. This unfortunately introduces more conservatism into the answer by creating heat sinks adjacent to the reactor thus forcing reactor vessel surface temperature down.

Table 6.5 indicates the boundary values for this problem. This code calculates film coefficients as a function of temperature.

TABLE 6.5
BOUNDARY VALUES AT WORST TEMPERATURE

Area	Fluid	Fluid Temp. °F	Direction of Heat Removal	Distribution Heat Transfer Coefficient Btu/hr-ft^2°F
Inside Vessel	Water	433	Horiz. & Vert.	340
Above Cover Insulation	Water	100	Horizontal	24
Along side of Cover Flange	Water	100	Vertical	126
Along top of Vessel Flange	Air	100	Horizontal	1.5
Along side of Vessel	Air	100	Vertical	1.2
Along vessel wall insulation	Air	100	Vertical	0.6

The following table lists the materials and the properties used in this analysis.

TABLE 6.6
MATERIALS AND PROPERTIES

Part	Material	Heat Capacity Btu/lb°F	Density lb/in^3	Productivity Btu/hr-ft°F
Vessel & Cover	A-350	.128	.282	22.50
Insulation	Fiberglass	.190	.089	0.04
Nut & Bolt	A-403	.128	.283	14.50
Boundary	Water	1.000	.0360	.385

Analysis of the data indicated that the most severe temperature gradients occurred just at the point where the primary system water reached its operating temperature of 433°F. This occurred at 400 min after the beginning of a 50°/hr heatup.

C. Results

Figure 6.5 shows a sketch of the SM-1A vessel and cover, an outline of the regions as established for this analysis and the average temperatures for each region at the point of the worst temperature distribution from a thermal stress standpoint.

PM-2A Reactor Vessel

The discontinuity analysis of the cover, flange and stud region for the conditions of stud load and stud load plus pressure was completed. The temperature distributions in the cover, flange and stud regions and the thermal stresses in the cover, flange and stud region for both steady-state and transient conditions are in process.

SM-1 Reactor Vessel

The analysis of the SM-1 cover, flange and stud region, being performed by a consultant, has been completed. A summary report will be issued in November.

Conclusions

A draft of a summary report of SM-1A vessel stress analysis has been completed. Stresses at all critical areas in the vessel are summarized. The maximum vessel stress, 44,300 psi, occurs during the steady state condition 3.11 in. above the junction of the lower transition head and the vessel shell. The maximum stress in the cover, 43,190 psi, occurs at the junction of the insulation cover and the cover ring, during the steady state thermal condition. The maximum total stress in the studs, during the transient temperature condition, is 32,700 psi. This is comprised of 29,000 psi simple tension and 3700 psi bending stress.

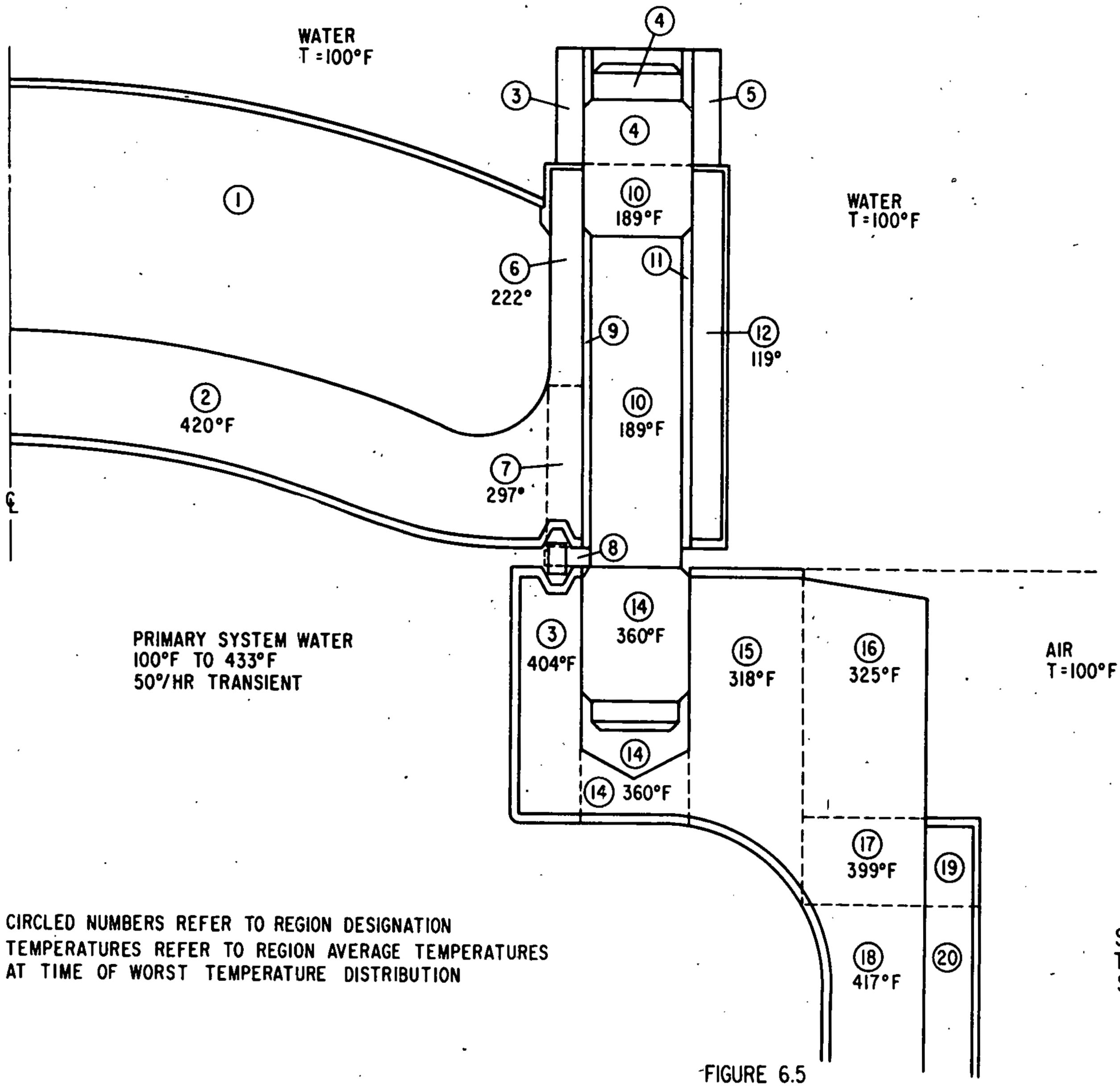
Future Work

1. The summary reports(6)(7) on the SM-1A and the SM-1 vessel stress analysis will be published during November.

2. The transient thermal stress analysis work on the PM-2A will be completed and a summary report of the vessel stresses will be published in December.

3. The results of the stress analysis of all three vessels will be applied in making plastic strain cycling studies of each vessel.

4. Related work will continue under Subtask 6.9 and 6.10 in FY-62.



CIRCLED NUMBERS REFER TO REGION DESIGNATION
TEMPERATURES REFER TO REGION AVERAGE TEMPERATURES
AT TIME OF WORST TEMPERATURE DISTRIBUTION

SM-IA VESSEL AND COVER TRANSIENT
TEMPERATURE AT TIME WHEN PRIMARY
SYSTEM WATER JUST REACHES 433°F

FIGURE 6.5

BIBLIOGRAPHY - TASK 6

1. AP Note-371, "Experimental & Theoretical Analysis of the Integrated Fast Neutron Flux in the SM-1," July 21, 1961.

2. APAE-95, "Fast Neutron Flux Measurements in SM-1 and PM-2A Core and Vessel Mockups," to be issued.

3. APAE-98, "Design Critiera for Irradiated Vessels - Task 6.8 Summary Report," September 29, 1961.

4. APAE-107, "Effect of Radiation Damage on SM-1, SM-1A and PM-2A, Reactor Vessels," October 14, 1961.

5. AP Note-351, "Effect of Radiation Damage on the Service Life of the SM-1A Reactor Vessel," May 29, 1961.

6. APAE Memo-302, "SM-1 Reactor Cover and Flange Stress Analysis," to be issued.

7. APAE Memo-303, "Stress Analysis of the SM-1A Reactor Vessel," to be issued.

TASK 7 - SECONDARY SYSTEM PERFORMANCE

Task Engineer - J. A. Barrett

Task Definition and Objective

Recent trends in the Military Nuclear Power Program place rigid specifications on the quality of electrical power produced. To aid in understanding the problem of meeting these specifications, a research and development program was initiated at the SM-1 plant. This program involved modifying, instrumenting, and performing tests at SM-1 to obtain the transient response of selected parameters to sudden and severe electrical and steam load change.

This task was originally performed as Task XII under Contract AT(30-3)-326 and has been continued under Contract AT(30-1)-2639 since July 1, 1961. Previous to July 1, 1961, progress of this task was made a part of the AT(30-3)-326 progress reports, and did not appear in the AT(30-1)-2639 progress reports.

Summary of Second-Half Results

1. The report, "SM-1 Transient Analysis by Analog Computer Methods," was issued as APAE Memo-290.[1]

2. The draft of a final report describing the Task XII instrumentation status as of June 30, 1961 was written.[2]

3. The instrumentation was 90% calibrated and checked with simulated process signals. Calibration procedures and final drawings were brought up to date.

4. Task XII equipment has been transferred to the Army.

Conclusions

None.

Recommendations

None.

Future Work

1. The final Task XII report will be issued.

2. Efforts will be made to get hazards approved for Test A700 or the work will be rescoped so it can be performed during normal training runs.

3. The steam dump line design will be evaluated to determine if any additional installations are required to provide an acceptable system.

4. Instrumentation already calibrated will be rechecked and troubleshooting done.

5. Test runs will be obtained during normal plant training cycle.

BIBLIOGRAPHY - TASK 7

1. APAE Memo-290, "Task XII Analytical Report - SM-1 Transient Analysis by Analog Computer Methods," May 26, 1961.

2. APAE-93, "Task XII - SM-1 Plant Response and System Performance - Termination Report," to be issued.

TASK 8 - WASTE DISPOSAL

Task Engineer - P. J. Kelleher

Task Definition and Objective:

The objective of Task 8 is to design, fabricate and test a prototype liquid waste processing system capable of processing all liquid waste generated at the SM-1 plant.

Summary of Second-Half Results

The work during the second-half has resulted in two general courses: the design and assembly of the prototype skid, and the laboratory investigations into the effectiveness of evaporation, chemical precipitation, and ion-exchange in processing liquid wastes.

A. Design and Assembly

The design of the skid was completed. The major design drawings include a flow diagram, skid structure drawing, equipment and piping layout, panel layout, and electrical diagram. The skid was designed to be a self-contained unit having its own housing. The house is divided into two sections by the operating panel. The 8' x 16' area in front of the panel is the control room and the 8' x 16' area behind the panel houses the processing equipment. Shielding requirements for operator safety were determined and shielding designed for a maximum dose rate of 1 mr/hr at the panel. The skid structure has been assembled. All major processing units have been installed. The wiring and piping of components have begun.

B. Laboratory Investigations

A number of tests on a laboratory scale were made to estimate the effectiveness of the unit operations on the skid to process various liquid wastes.

1. Laboratory Wastes. Raw laboratory wastes investigated in the Alco Radiochemistry Laboratory were subjected to addition of precipitating agents, pH adjustment, distillation and distillate ion-exchange. The initial activity of the lab wastes was in the order of 5 to 7 x 10^{-2} μc/cc. Addition of precipitating agents and pH adjustment to about 8.5 resulted in approximately the same final supernate activity of about 1.8 x 10^{-2} μc/cc or a D. F. in this case of about 3 to 4. Distillation of supernate resulted in activities whose counts were somewhere in the order of background. (Background is about 50 cpm.) No appreciable counts above background were detectable in the distillate after having been passed through cation and anion resin beds. The numerical values of final

activities are questionable because of the response of the available counting equipment in the $10^{-7} \mu c/cc$ range.

2. Laundry Wastes. Laundry wastes were treated at Ft. Belvoir by distillation and distillate ion exchange. The initial activity of the laundry wastes was in the order of $8.5 \times 10^{-4} \mu c/cc$. Distillation of raw waste brought the activities to the background level. Ion exchange of distillate resulted in no counts above background.

3. Primary Wastes. Primary waste (Ft. Belvoir) having a specific activity of $2 \times 10^{-3} \mu c/cc$ was subjected to distillation. Distillate activities were not detectable.

4. Decontamination Wastes. Decontamination wastes supernate (from decontamination of stainless steel tubing exposed to SM-1 blowdown) were subjected to ion exchange. The initial activity was about $6 \times 10^{-5} \mu c/cc$. Of the resins tested, the Rohm and Haas XE78, a strongly basic quaternary ammonium anion exchange resin supplied with at least 80% of the exchange sites in the hydroxide form, was most effective. No counts above background were detectable for the effluent.

Conclusions

The laboratory work accomplished to date gives a rough indication of anticipated results using the prototype processing system decontamination. The previous section gives an idea of the approaches to treatment of actual wastes. Only the actual prototype testing will give a design of an ultimate waste treating system. Although the lab work indicates chemical treatment and preliminary solids removal followed by distillation and ion exchange are tenable steps in waste processing, the operation of the skid may prove the process can be shortened, perhaps to only distillation and ion exchange.

It appears, however, from the data that the unit operations incorporated in the skid design can handle any type of liquid waste and decontaminate it safely to MPC levels.

Future Work

The assembly and testing of the skid must be completed. The skid is to be tested at Schenectady using non-radioactive solution prior to shipment to SM-1 where available wastes will be processed.

TASK 9 - INSTRUMENTATION AND CONTROL

Task Engineer - J. A. Barrett

Task Definition and Objectives

The remote location of the small military nuclear power plants and the desire for operation with a minimum of personnel require maximum reliability in the design and application of instrumentation and control systems. The objective of this task is to design and specify reliable systems and equipment for thorough testing at SM-1 prior to use in the field plants, and in the design of future plants. A standard instrumentation and control design for small nuclear power plants is also an objective of this task. A standard design will facilitate training of personnel, reduce construction costs, and simplify the maintenance program.

Safety will be given prime consideration to insure that the design and application of instrumentation and control systems will not jeopardize safe plant operation.

Summary of Second-Half Results

An installation design was drawn up and a proportional counter lifting mechanism was purchased. The preliminary test and installation procedures were written and forwarded for comments and approval. Comments received from the Ft. Belvoir personnel and TEB have been incorporated in the test and installation procedures. Components of the installation design have been purchased.

The design utilizes the existing pull box and conduit of the present SM-1 BF_3 chamber installation, thus eliminating shield tank contamination during installation.

The lift mechanism was mechanically and electrically checked for performance, calibrated, and adjusted for maximum travel of 6 ft prior to shipment to SM-1 site.

Prior to the termination of work on SM-1 re-instrumentation, the following was accomplished:

SM-1 Feedwater Control System

A review of the existing feedwater control system, and feedwater control systems used in other nuclear power plants was completed. A preliminary system design for the new feedwater control system was sketched as well as a critique.

SM-1 Pressure Control System

A review of the existing pressure control system was completed.

SM-1 Pressurizer Level Control System

This design was not initiated.

SM-1 Scram and Alarm System

This design was not initiated.

No progress is reported on the evaluation of the Bendix solid state instrumentation.

Conclusions

None to date.

Recommendations

It is advisable not to overrun the adjusted 6 ft limit due to a welding restriction in the chamber well, approximately 8-1/2 ft above the bottom of the well.

Future Work

1. The BF_3 lift mechanism may be installed during the October, 1961 planned shutdown. Evaluation of the lift mechanism by testing will be initiated. Testing will consist of calibration of the neutron detector and electronic instrumentation, determining the effects of flexing and radiation on signal cable characteristics, completion of lift mechanism operating check, determination of startup channel response at various positions and reactor power levels, and issuing a report. This work will continue as Subtask 9. 2 in FY-62.

2. Plans will be formulated to allow Bendix to repair and/or modify the existing solid state instrumentation design in order to incorporate the latest technical advances. Neutron detectors, cables, and accessories will be specified and procured. The solid state instrumentation will be electrically calibrated and tested at Alco, Schenectady, and a complete nuclear calibration will be conducted in the Alco Criticality Facility. This work will continue as Sutbask 9. 1 in FY-62.