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EBR-II PROJECT
ARGONNE NATIONAL LABORATORY
Argonne, Illinois - Idaho Falls, Idaho

January 1982

~~Released for announcement in ATF. Distribution limited to participants in the LMFBR program. Others request from office of INPO, NE, DOEW.~~

ANL/EBR--117

DE82 013130

Distribution Category:
Liquid Metal Fast
Breeder Reactors
(UC-79T)

ANL/EBR-117

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ABSTRACT

Experimental Breeder Reactor No. 2 (EBR-II) is an unmoderated, sodium-cooled reactor with a design power of 62.5 MWt. The primary cooling system is a pool type. EBR-II (including the adjacent Fuel Cycle Facility) was completed in 1963 at a cost of $32 500 000--after five years of construction. The early operation of the reactor successfully demonstrated the feasibility of a sodium-cooled fast breeder reactor operating as an integrated reactor, power plant, and fuel-processing facility.

In 1965, the role of EBR-II was reoriented from a demonstration plant to an irradiation facility. Many changes have been made and are continuing to be made to increase the usefulness of EBR-II. Mild transient-overpower tests are scheduled to start in 1982.

A review of EBR-II's operating history reveals a plant that has demonstrated high availability, stable and safe operating characteristics, and excellent performance of sodium components. Levels of radiation exposure to the operating and maintenance workers have been low, and fission-gas releases to the atmosphere have been minimal. Driver-fuel performance has been excellent. Repairability of radioactive sodium components has been successfully demonstrated a number of times. Some of the more significant achievements are as follows:

- A plant capacity factor averaging 62.5% from calendar years 1970 through 1980.
- Generation of a total of 1 279 971 MWh electrical from initial startup through July 1981.
- No water-to-sodium leaks in the steam generators over the life of the plant.
- No failures or incidents where serious in-core or out-of-core consequences have resulted.

- Safe control and detection of cover-gas and sodium contamination.

Efforts are under way to qualify EBR-II for performing a variety of in-core tests associated with the LMFBR program on operational-reliability testing (ORT). These tests will be initially focused on determining the reliability and ability of fast-breeder-reactor (FBR) fuel and blanket elements to withstand a variety of off-normal conditions. The off-normal tests are to simulate conditions of transient behavior ranging from reactivity inputs of 0.1 to 10 ¢/s that will produce overpowers from 15% to perhaps as high as 100% on test failed and unfailed fuel elements.

I. INTRODUCTION

Experimental Breeder Reactor No. 2 (EBR-II) is an experimental liquid-metal fast breeder reactor located at the Idaho National Engineering Laboratory. EBR-II was originally designed and operated as an engineering facility to demonstrate the feasibility of fast reactors for central-station power-plant application. It also demonstrated the feasibility of closed-cycle fuel reprocessing.

EBR-II has operated safely and satisfactorily for 17 years. The initial operation was very conservative. During the early years, EBR-II experienced operational problems and component failures that reduced the plant factor; however, all failed components were repaired successfully. After the initial operational difficulties were solved, steps were taken to increase the plant availability and the usefulness of EBR-II. Many plant modifications have been made to the original facility.

In the last 10 years, no failures have occurred in major components, and no significant unscheduled outages have occurred. A very good plant capacity factor has been achieved during this period, reaching as high as 77.1% in calendar year 1980. A rigorous monitoring and inspection program has been helpful in achieving the high availability and operability of EBR-II.

The life expectancy of the plant is not clearly defined, but there is no apparent limitation due to any single component or situation that will limit the life. The major mechanical components have demonstrated durability and reliability while operating in high-temperature sodium.

This report summarizes the more important difficulties encountered during the 17 years of operation and also describes the significant achievements at EBR-II.

II. PLANT DESCRIPTION

EBR-II consists of an unmoderated, heterogeneous, sodium-cooled reactor with a designed thermal power outage of 62.5 MW, an intermediate closed loop of secondary sodium coolant, and a steam plant that produces 20 MW of electrical power through a conventional turbine-generator.[1,2] It also incorporates a Hot Fuel Examination Facility (HFEF) to examine fuel elements, to assemble experiments for irradiation in the reactor, and to examine irradiated experiments. Part of the HFEF (HFEF/South) was originally the Fuel Cycle Facility (FCF), which was designed to reprocess spent fuel from EBR-II.

The main parts of the EBR-II plant are the reactor, sodium boiler building, power plant, and the HFEF (South and North), as shown in Fig. 1. The reactor building is a steel containment vessel that houses the reactor and the primary sodium system. EBR-II uses the pool-type concept shown in Fig. 2, in which the reactor, major primary-system components and piping, and much of the fuel-handling equipment are submerged in a large, double-walled tank containing over 341 m^3 of 473°C sodium. The two main centrifugal primary pumps, each rated at 0.347 m^3/s, take suction from the bulk sodium. This sodium is circulated in a single pass through the reactor, through the single outlet pipe to the intermediate heat exchanger (IHX), and back to the bulk sodium. A dc electromagnetic pump, rated at 0.032 m^3/s, is in the outlet pipe. The pump operates continuously for the specific purpose of removing decay heat if both main primary pumps should become inoperative.

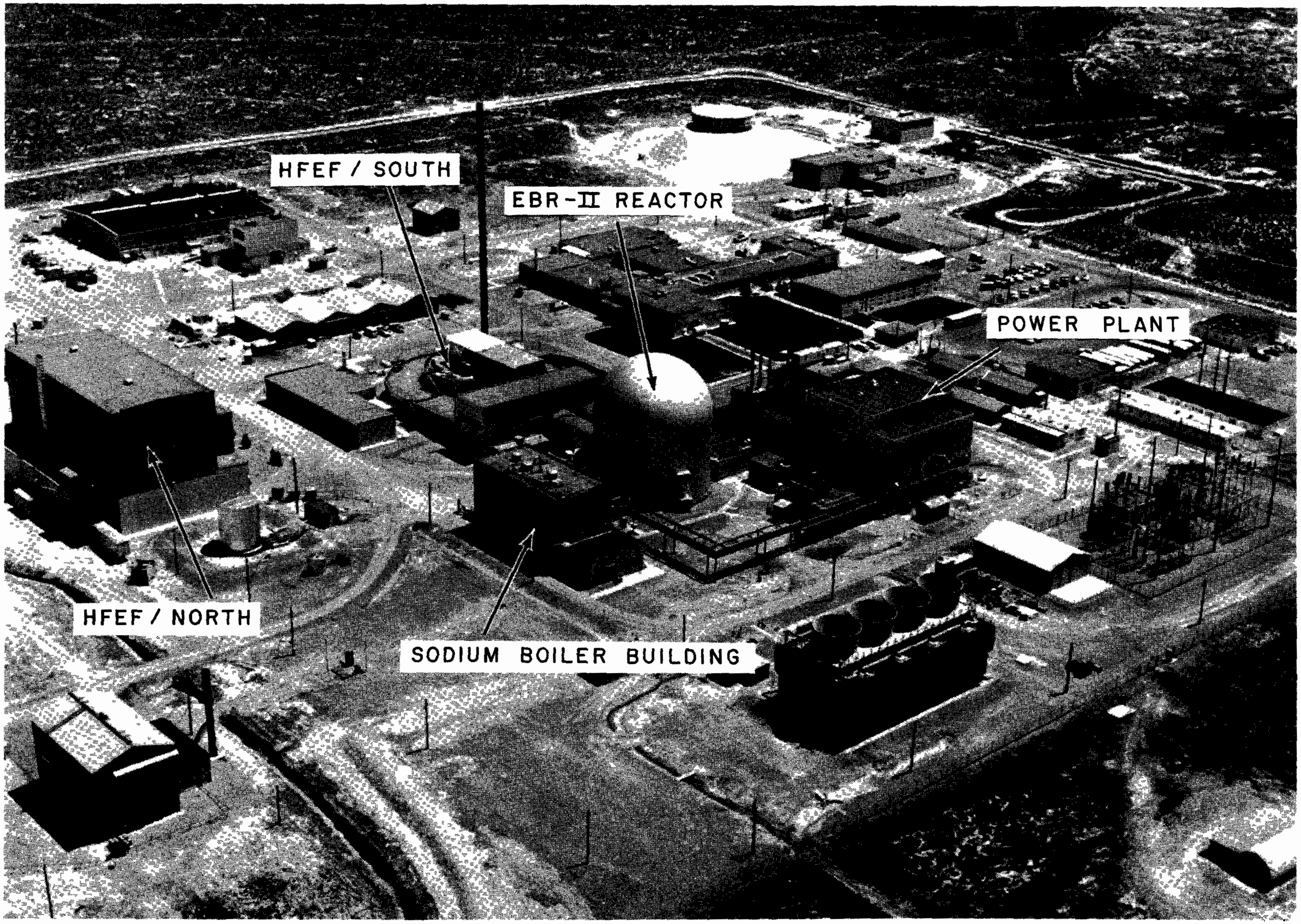

Fig. 1. Argonne-West Site at Idaho National Engineering Laboratory

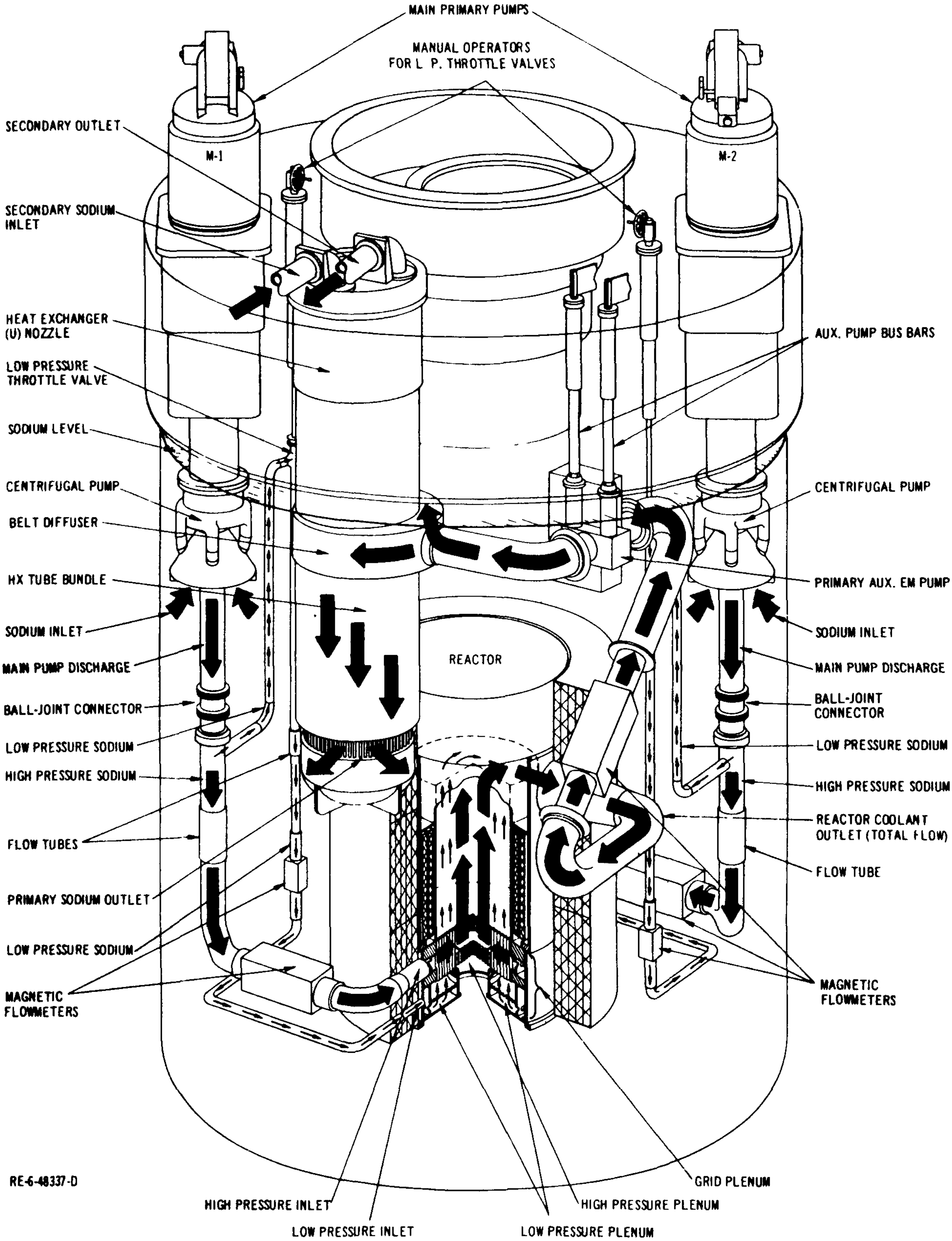

Fig. 2. EBR-II Primary Tank and Contents, Showing Flow Paths of Primary Sodium

The secondary system and the sodium-to-water steam-generating equipment are in the sodium boiling building. Nonradioactive secondary sodium is pumped to this building, where it passes through two superheaters and seven evaporators and goes to a surge tank. The surge tank is the high point in the secondary system. It provides a constant head for the single ac linear-induction electromagnetic secondary pump and provides a free surface between sodium and argon for escape of entrained gases. Sodium is pumped from the surge tank back to the IHX in the primary tank. Nominal flow rate in the secondary system is 0.353 m^3/s.

Superheated steam is supplied to the power-plant turbine generator, which produces about 20 MW of electricity for the 138-kV power loop. The power plant is conventional in design.

The reactor was designed with 12 control rods. Three of the control-rod positions have been converted to in-core testing facilities. A fourth control-rod drive has been deactivated, and the in-core position it served is now used as an irradiation facility designed to allow accumulation of burnup on experiments before they are placed under one of the in-core testing facilities for run-beyond-cladding-breach (RBCB) or transient testing. Any one of the remaining eight fueled control rods can be used for reactor control; all control rods are used for scram. One of the eight fueled control rods can be selectively dropped at power for reactor kinetics experiments and measurement of reactivity feedback coefficients.

Two fueled safety rods provide additional removable reactivity during reactor operation and also provide shutdown reactivity during fuel handling. The control and safety rods are similar to standard driver subassemblies but contain only two-thirds the number of fuel elements. The standard driver subassemblies contain uranium-fissium metallic fuel that is sodium-bonded to stainless steel jackets.

Two types of fuel handling are done at EBR-II. Refueling of the reactor consists of transferring subassemblies between the reactor vessel and a subassembly storage basket (submerged in the sodium in the primary tank). The second type consists of transferring subassemblies between the storage basket and the HFEF and is done during reactor operation.

Part of the HFEF (HFEF/South) was originally operated as a process plant in which irradiated fuel elements and blanket material removed from EBR-II were disassembled and processed by pyrometallurgical methods to remove fission-product contamination. The fuel was then reconstituted, refabricated into new fuel elements, and assembled into new subassemblies. The facility has been enlarged and consists primarily of argon-atmosphere cells where reactor subassemblies and irradiation experiments are assembled, disassembled, and inspected. It is no longer used to reprocess EBR-II fueled subassemblies.

III. OPERATING HISTORY

Construction of EBR-II began in 1957 and was completed in five years. The total cost of the project was $32 500 000, including the cost of the FCF. Sodium filling and preoperational testing were done during the late stages of construction. Preoperational testing was conducted in several steps, beginning with the dry-critical experiments, followed by wet-critical experiments, and completed with the approach to power.

Initial operation of the reactor began when dry criticality (no sodium in the primary tank) was reached in October 1961.[3] The dry-critical experiments were then performed to determine the neutronic characteristics of the reactor core. The critical mass for dry criticality was 230.26 kg of ^{235}U, contained in 87 subassemblies including 12 control rods and two safety rods.

After the dry-critical experiments, a thorough proof-testing and final checkout of the fuel-handling system was made before the primary tank was filled with sodium. Rotation of the large plug became increasingly difficult during this checkout, and later examination revealed severe corrosion of the copper seal blades. To correct the problem, both rotating plugs were removed from the primary-tank cover, and the copper blades were replaced with similar stainless steel blades.

The secondary and primary systems were filled with sodium in early 1963. Immediately after the sodium filling of the primary tank, the wet-critical experiments were performed.[4] Wet criticality was reached with 181.2 kg of ^{235}U in 70 subassemblies.

Further operations were then delayed because of mechanical failure of both primary pumps. This early failure of the pumps, after less than 200 h ' of operation, was believed to be caused by bowing of the pump shafts.[5] Both pumps were removed from the primary tank, repaired, and reinstalled.

Shortly after the primary pumps were repaired, a sodium leak occurred in the pump duct of the main secondary electromagnetic pump. The pump was disassembled, repaired, and reinstalled by June 1964.

The approach to power started in July 1964 and consisted of a series of stepwise power increases.[6] During the approach, the first reprocessed and recycled fuel from the FCF was inserted into the reactor. A four-month delay occurred between October 1964 and February 1965 when a ball bushing failed in a vertically reciprocating oscillator rod that had been temporarily installed in the reactor to measure the transfer function. During removal of the oscillator rod from the reactor, a routine inspection revealed a pinhole lead (water to air) in a tube-to-tube-sheet weld of one of the steam generators. The faulty weld was repaired with the steam generator in place and required about one month.

The approach-to-power program was completed in March 1965, when the reactor power was raised to 45 MWt, with gross electrical power of 14 MW being produced in the plant.

Throughout the early years of operation, the thermal and neutronic performance of the reactor was very close to what was predicted. The oscillator rod, before its failure, was used to measure the dynamic behavior of the reactor The results of these measurements, in which the transfer function was measured under various conditions of power and flow, demonstrated that the reactor was stable and that a prompt negative feedback existed. This prompt feedback was believed to be caused primarily by the radial and axial expansion of the metallic fuel and coolant.

In addition to the transfer-function measurements, the power coefficient of reactivity was determined while the reactor power was being increased and decreased. Major importance was placed on the use of these measurements as standards to indicate possible changes in reactor kinetics. In summary, from September 1961 to March 1965, EBR-II was successful in reaching the initial design goals of (1) demonstrating the operation of a sodium-cooled fast reactor as a central-station power plant, and (2) demonstrating the operation of an integral fuel-processing facility in which irradiated fuel elements were reprocessed and recycled to the reactor.

In May 1965, EBR-II began its role as an irradiation test facility with the insertion of the first experimental subassembly. Shortly after this time, an ambitious program was launched to irradiate and test a large number of fuels and structural materials. The transition to an irradiation facility required significant changes and improvements of the reactor. These changes were accomplished over a number of years. The major ones included (1) improvement of core driver fuel, (2) power increase to 62.5 MWt, (3) enlargement of the core size from 61 to 91 and finally to 127 subassemblies to allow accommodation of a large number of experimental-irradiation subassemblies, (4) replacement of the depleted-uranium blanket with a stainless steel reflector to improve the irradiation capabilities of the core, and (5) reduction in the number of fueled control rods to eight, by the introduction of high-worth control rods, to allow installation of special irradiation facilities.

In February 1966, the transfer arm was removed from the primary tank and repaired. The locking-pin actuating arm (used to lock subassemblies in the transfer arm during fuel handling) had become very difficult to operate. Since reinstallation of the transfer arm, no further difficulty has been experienced with it.

During August 1966, a new oscillator rod with a rotary drive was installed in the reactor. Slow manual rotation of the oscillator rod revealed a slight rub at one point at reactor powers above 30 MWt. When the rotational speed was increased, the rubbing disappeared. The difficulty was believed to be due to thermal distortion of the oscillator rod. Operation of the oscillator rod was, therefore, limited to power levels no higher than 30 MWt. Subsequently, a rod-drop method using a special stainless steel control rod was developed for continuing the stability-monitoring program for powers above 30 MWt. The stainless steel control rod was of low reactivity worth and could be selectively scrammed from the reactor. This scramming caused a step loss of reactivity from which reactivity-feedback measurements could be made. Results from the rod-drop method were comparable with the measurements of the oscillator rod. Later, the rotary oscillator rod was removed from the reactor.

After completion of the early irradiation program, the reactor power was raised to 50 MWt in August 1968. The 50-MWt operation was an interim step in the program to operate EBR-II at its design power of 62.5 MWt.

Operation at 50 MWt was satisfactory and as predicted. Before the power was increased again, it was necessary to evaluate the effect that the increased temperature at 65.2 MWt would have on driver-fuel swelling, because an increased irradiation-swelling rate of the driver fuel could possibly restrict the maximum permissible burnup at higher temperatures. To provide advance information on the fuel behavior, a fuel-surveillance program was initiated with special subassemblies to simulate 62.5-MWt reactor operation.

Upon successful completion of the fuel-surveillance study, the reactor power was increased in steps from 50 to 56 to 62.5 MWt in September 1969. The first advantage of reactor operation at 62.5 MWt was that the irradiation capabilities were improved by the increase in neutron flux and temperatures. Second, operation at full design power proved the correctness of the original design criteria for 62.5 MWt.

Operation continued at full power, except for routine maintenance shutdowns and reactor refueling, until November 1970. Further operation was prohibited until repairs could be made to both the IHX and one of the primary pumps.[7,8] The IHX was repaired without removing it from the primary tank. Two support clips holding a drain tube had become loose, and this allowed the drain tube to bang against the side of the IHX inlet pipe. The drain tube was removed. The primary pump had experienced binding because of a buildup of sodium and sodium oxide on the pump shaft. The pump was removed and repaired along with the work on the IHX.

A four-month delay was required for repairs to the pump and the IHX. Operation was resumed and has continued to the present except for scheduled shutdowns.

In 1974, one of the two superheaters began to show degraded performance when a reduction in outlet steam temperature occurred. The reduced efficiency of the superheater caused no operational problems, however, and only a very small, undetectable reduction in electrical output. To determine the cause of the abnormal performance, the superheater was removed from the system in April 1981 for destructive examination. It was replaced by an evaporator module that was removed from the steam system in 1980 and converted to a superheater.

Recent performance of EBR-II has been excellent. Operation has not, however, been completely without difficulty. Most of the problems have been minor and have been caused by instrumentation failures rather than mechanical failures. Sticking of the rotating plugs has often been experienced. To minimize the sticking, inspection and cleaning holes were drilled for both plugs. The seals for the plugs are now routinely inspected and cleaned three times a year.

Significant improvements have been made to the secondary-sodium and steam systems by (1) provision for early detection of sodium-to-water leaks in steam generators (by installation of hydrogen-meter leak detectors) and (2) provision of a system for rapidly dumping the water from the steam generators.

In 1975, an extensive program was started to upgrade the EBR-II facility to permit operation of the reactor for extended periods with breached fuel. Before then, extended operation with breached fuel would have led to unacceptable radiation levels in the reactor building because of a high leak rate of the primary-tank cover gas. The major changes made were (1) installation and successful operation of a cover-gas cleanup system (CGCS) that removes xenon and krypton from the cover gas by means of a cryogenic distillation process and (2) reduction of the leak rate of the cover gas to the reactor building. With the completion of these changes, the RBCB program was started in June 1977. A cesium trap was installed in the primary purification system in March 1978 and had (through March 1981) removed an estimated 345 Ci (12.8 TBq) of cesium from the primary coolant.

EBR-II has served as a steady-state irradiation facility from 1965 to present. It is anticipated that FFTF will soon become fully qualified to assume this steady-state role for the LMFBR program. Consequently, with the concluding of EBR-II's steady-state program, the EBR-II Project is now qualifying the reactor as a test-bed facility for performing a test program of a more severe nature. These tests will focus on the effect of the more probable upset conditions that might occur in a commercial LMFBR. Present identified concerns include:

a. Effects on fuel and blanket elements of a mild overpower transient terminated by reactor safety instrumentation, e.g., 15% overpower with a reactivity ramp of 1 ¢/s.

b. Effect of more-severe transients terminated by delayed reactor scram, e.g., 60-100% overpower with a 10-¢/s ramp rate.

c. Effect on reliability and lifetime of fuel operating within two temperature regimes, e.g., the change in operating temperature of a fueled subassembly moved to a different location in the reactor core or the change in operating temperature of the fuel upon loss of one of the three coolant loops of a commercial-size LMFBR.

d. Reliability of operating with failed fuel elements during the above conditions as well as steady-state.

e. Loss of pump power and establishment of convective flow.

The above concerns are being formulated in terms of specific tests under a program known as ORT (operational-reliability testing). The RBCB tests, which are part of the ORT program, are under way, and the first of the transient tests is scheduled for mid-1982.

To perform these tests, the EBR-II reactor system, including driver and blanket subassemblies, must be qualified to sustain transients of 0.1-10 ¢/s of reactivity insertion over a temperature range equivalent to a power change of 24 to 62.5 MWt. With this capability and by a judicious choice of irradiation positions in the core, core-loading arrangements, and reactor-flow adjustments, it is possible to precondition test fuels at prescribed steady-state conditions and then subject them to the required transients without exceeding the design limits of the EBR-II reactor. The strategy for performing these functions while continuing the remaining steady-state experimental program has been formulated and suggests that the ORT program can be integrated into EBR-II over the next four to five years.

The Project's experience in detecting, identifying, and then operating with failed fuel under a variety of failure modes already qualifies the reactor as a very useful test bed for extending the experimental program on RBCB to more-severe operating conditions. To this end, special in-core facilities have been designed and are already in use.

Past tests on natural circulation of the reactor coolant under upset conditions such as loss of primary-pumping power have suggested the need for a more ambitious program for whole-plant modeling. Therefore, a series of whole-plant natural-circulation tests with EBR-II have been developed. These tests will begin in early 1983 and are scheduled to be completed within two years.

The first core-qualifying tests were done in January 1980, and the second were performed in February-March 1981. The final tests are scheduled for the fall of 1982.

The ORT program, as now identified, is scheduled to begin in the fall of 1982 and extend at least through mid-1986. It is expected that, as the tests proceed, additional programs could be integrated into the program to make use of the unique capabilities of EBR-II.

A few of the significant operational achievements at EBR-II are:

- Operation of the plant with minimal release of fission gases to the environment. During 1978, only 150 Ci (5.55 TBq) were released to the atmosphere.
- Low level of radiation exposure to the operating and maintenance personnel. Table I shows the levels received during recent years. (The data were provided by C. E. Holson.)
- Seventeen years of successful experience with sodium-to-water steam generators (evaporators and superheaters) without sodium-to-water leaks.
- Seventeen years of experience with sodium components such as pumps, IHX, and fuel-handling equipment, with only minor and repairable problems.
- Achievement of annual plant capacity factors (since 1970) that compare favorable with the best performance of commercial light-water power plants. The plant capacity factor averaged 62.5% between 1970 and 1980.
- Demonstrated that radioactive sodium components can be maintained and repaired by straightforward techniques, with relatively simple equipment, and without undue hazard to personnel.
- Development and demonstration of techniques and equipment for fuel-failure identification and detection, for characterizing and monitoring fast-reactor kinetic behavior, and for monitoring and controlling sodium impurities.

TABLE I. Exposure Histories for EBR-II Personnel: 1970-1980

	Number of People	Individual High Exposure, mRem	Individual Low Exposure, mRem	Total ManRem	Average mRem
Operators					
1970	51	200	0	1.100	21
1971	48	440	0	3.210	66
1972	52	600	0	2.600	50
1973	60	320	0	3.275	55
1974	58	495	0	5.435	92
1975	65	475	0	6.675	103
1976	67	455	0	6.953	104
1977	69	510	0	10.047[a]	183
1978	59	298	0	8.662	147
1979	66	409	0	9.188	139
1980	58	267	0	5.239	90
I&C Instrumentation Personnel					
1970	13	500	0	1.810	139
1971	18	740	0	4.120	228
1972	18	675	0	6.150	341
1973	14	350	0	2.290	163
1974	26	1330	0	6.840	263
1975	31	430	0	4.470	144
1976	34	625	0	5.980	176
1977	33	543	0	3.982	121
1978	29	589	0	3.847	133
1979	26	384	0	3.980	153
1980	23	619	0	2.362	103
Maintenance Personnel					
1970	5	1095	250	2.955	591
1971	5	1420	315	4.500	900
1972	4	965	230	2.400	600
1973	3	665	0	0.945	315
1974	5	715	75	1.660	332
1975	6	1595	75	6.070	1011
1976	6	1210	120	5.160	860
1977	6	1224	667	5.398	900
1978	10	979	113	5.314	531
1979	16	960	30	6.290	393
1980	17	540	0	3.286	193

[a]Operators assisted maintenance personnel in removal of the primary cold trap in 1977.
Conversion factor: 1 mRem = 10 μSv.

- Achievement of increased drive-fuel performance (burnup limit of 8 at.%) while maintaining high reliability.
- Significant operational experience with various fuel types (oxides, carbides, metal) irradiated in breached elements in RBCB testing.
- Generation of 1 279 971 MWh electrical from initial startup through July 1981.

IV. ORGANIZATION

The EBR-II Project is managed by a Project Director and consists of four departments; Operations, Engineering, Fuels and Materials, and Analysis. Each department is headed by an Associate Director, who reports to the Project Director. As of July 31, 1981, total personnel was about 270.

The Operations Department is diagramed in Fig. 3. The department consists of five sections: Plant Operations, Training and Procedures, Core Surveillance, Critical Systems Maintenance, and Experiment Coordination; it also has a Planning and Scheduling group.

The Plant Operations Section provides the personnel for around-the-clock operation of the EBR-II plant. Four crews work normal 8-h shifts on rotating schedule. Each crew consists of a shift supervisor, an alternate shift supervisor, a foreman, and 10 operators. Support personnel for each operating crew include a radiological-safety technician, a coolant-chemistry specialist, and at least two maintenance people.

The other departments--Engineering, Fuels and Materials, and Analysis--provide engineering and technical talent to support the ongoing EBR-II operations and programs. In addition, these departments provide a cadre of talents to develop new programs for EBR-II and to transfer technology to other LMFBR facilities.

V. REACTOR OPERATION, MAINTENANCE, AND SCHEDULING

A stated objective of the EBR-II Project is "to operate EBR-II with the highest practical plant capacity and plant availability factors consistent with plant safety, project funding, and the requirements of the EBR-II experimental programs." More specifically, the Project's goal is to operate EBR-II with a capacity factor greater than 60% and an availability factor greater than 70%. Scheduling of reactor operations is therefore based on the stated goals, the need for periodic preventive maintenance and plant modifications, and the occurrence of plant-controlling experiments and other unavoidable delays.

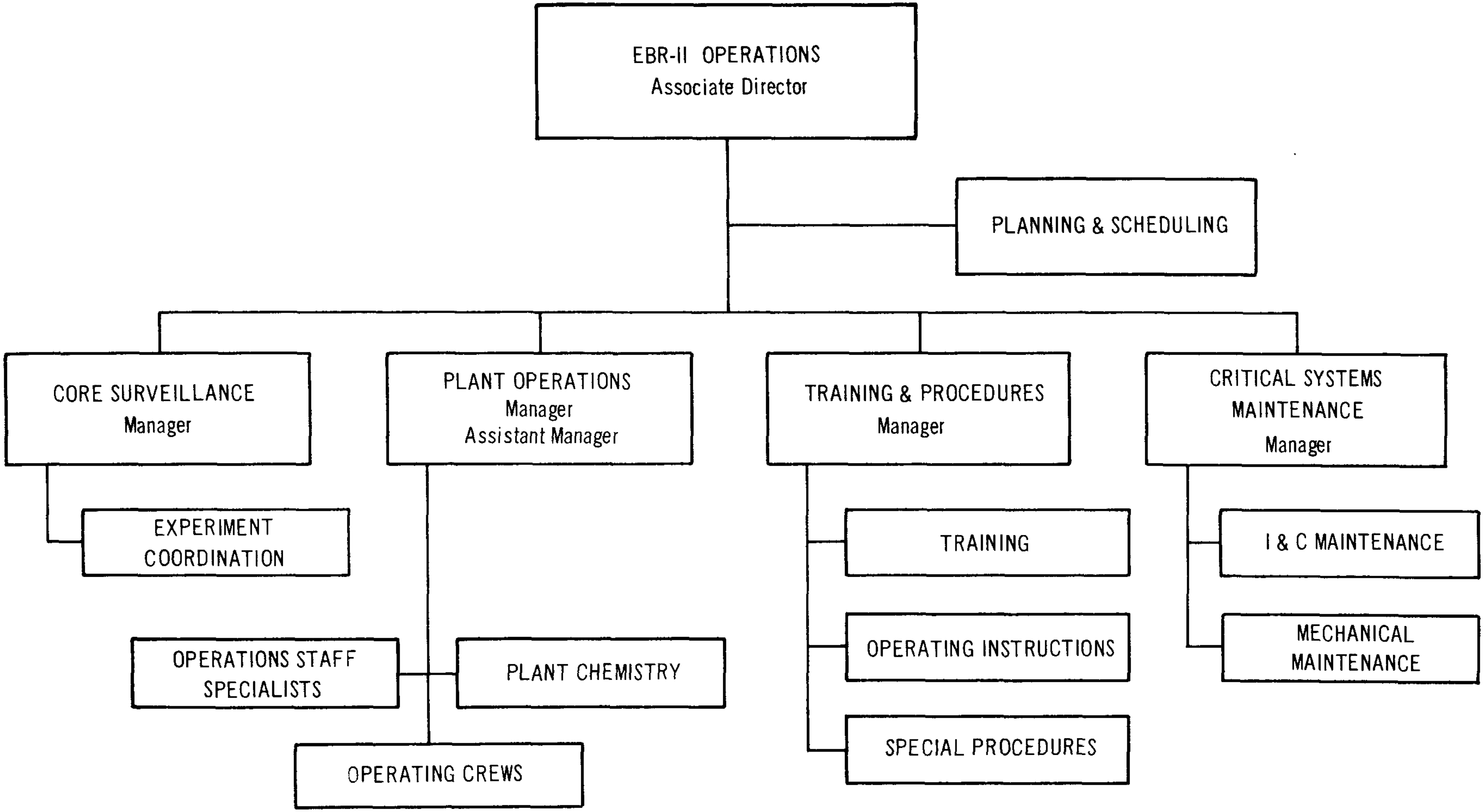

Fig. 3. Organization Chart for EBR-II
Operations Department

The length of a reactor run is generally 2700 MWd thermal. This length was selected because it provides the optimum utilization of fuel. Between-run shutdowns average about seven days, but can vary from five to 10 days, depending on the required maintenance and calibration and on the refueling and core-surveillance requirements. Additionally, one major maintenance outage of 30 to 45 days is scheduled annually. Our experience shows that by following this schedule, plant capacity factors of 65 to 75% are attainable. Table II (next page) shows the plant factors since 1965.

During its history, EBR-II has achieved a respectable operating record. In particular, EBR-II especially demonstrated reliability and maintainability by achieving plant capacity factors in 1974-1980 that are excellent for a research reactor. The plant capacity factor is not as high as would be expected for a commercial nuclear power plant. It has been affected primarily by the irradiations program and to a lesser extent by spurious reactor scrams and operational difficulties.

A. Scheduling

A working projection of reactor operation, plant shutdowns, and major maintenance and modifications for the current fiscal year and the next fiscal year is updated and issued to EBR-II Project personnel and experimental users about every 60 days. Issued at the same time is a projection of plant capacity for the current fiscal year, and superimposed on this, the record of plant capacity attained.

Reactor planning and scheduling meetings are held weekly to provide EBR-II Project management with current information on reactor operations, maintenance, procedures, and engineering activities. This exchange of information has helped to identify scheduling conflicts and material-supply problems long before they affect reactor operations, and has led to a general increase in plant capacity and availability.

Several weeks before a reactor shutdown, a detailed maintenance and modification schedule is prepared. Figure 4 (p. 19) shows a typical example. The schedule shows major plant conditions during the shutdown and also gives the expected schedule for the required maintenance and modifications. This schedule is then reviewed by various EBR-II Project groups for potential conflicts and is made final one week before the shutdown.

TABLE II. EBR-II Plant Capacity Factor: 1965-1980

Year	Plant Capacity Factor, %
1965	26.4
1966	43.0
1967	20.1
1968	41.8
1969	42.4
1970	57.9
1971	39.1
1972	46.9
1973	49.9
1974	58.7
1975	66.1
1976	76.9
1977	71.5
1978	72.8
1979	71.1
1980	77.1

B. Scheduling of Annual Long Outage

An extended maintenance outage is scheduled once a year. This is required for maintenance and modification activities that require unusual plant conditions and that cannot be accomplished during reactor operation or the between-run shutdowns. Typical activities of this type are the changeout of control-rod and safety-rod thimbles, maintenance or modification of the steam or secondary-sodium systems that requires both systems to be drained, and maintenance or modification of components in the primary tank.

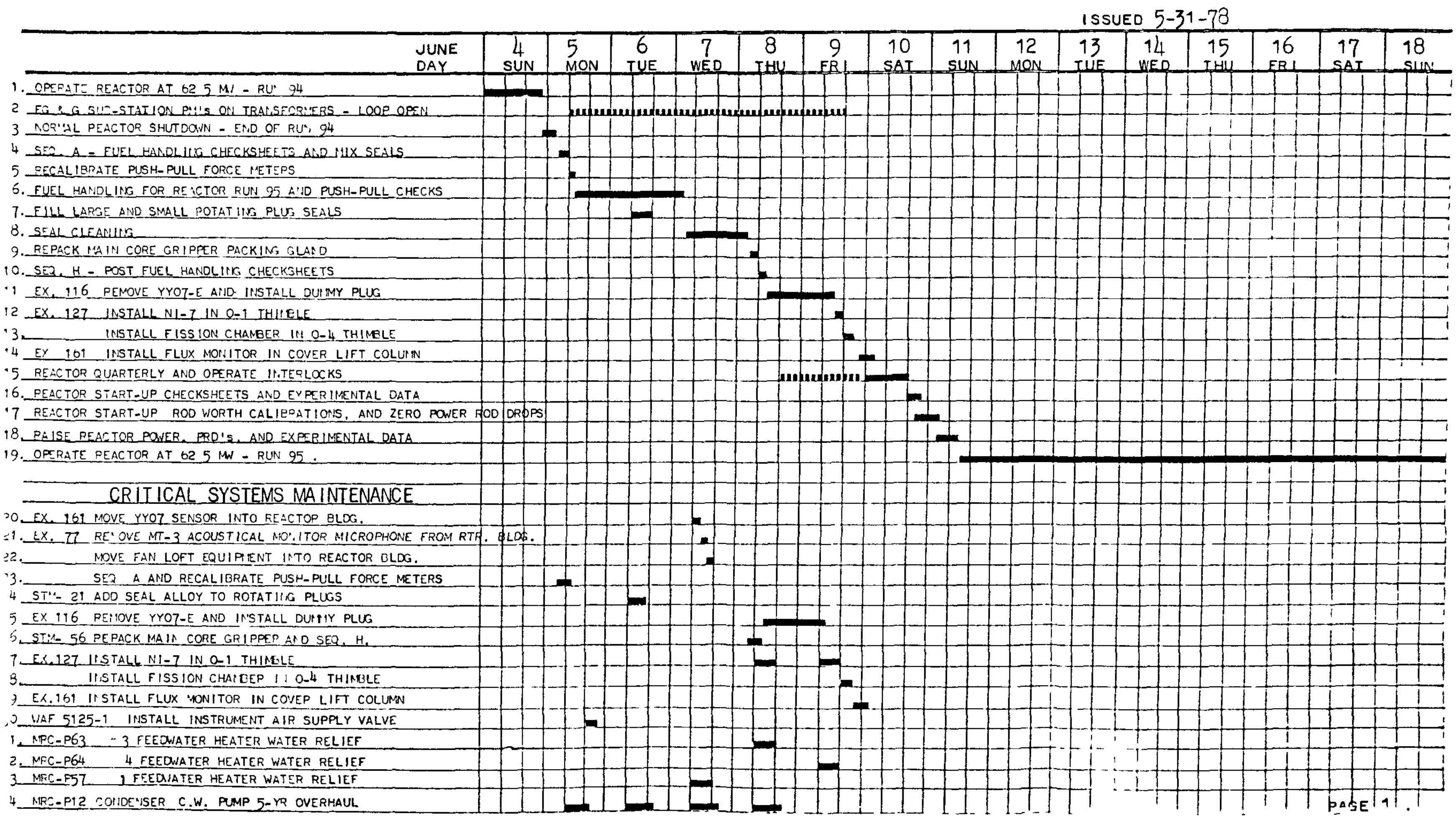

Fig. 4. Example of Maintenance and Modification Schedule

Planning for this shutdown begins about nine months before its expected starting date. Major activities to be accomplished are outlined, and the required plant conditions are established. The initial schedule is further detailed at monthly meetings; Fig. 5 is an example of this schedule.

The benefits from a scheduling process such as this are optimum utilization of personnel, minimum feasible plant downtime, and identification of problem areas early enough to correct for them before they lessen plant availability.

C. Recovery from Shutdown Activities

All maintenance and modification activity is closely followed by Operations personnel who are specialists in particular areas of the plant systems (for example, electrical systems, power-plant systems, or primary-tank components). As each activity is completed, the affected system or component is operationally tested. For major plant modifications, the authorizing documents are required to contain acceptance and checkout criteria.

During the recovery from an annual long maintenance outage, the steam system is filled with water and leak-checked at 38°C. Additionally, any sections of it that were modified or penetrated by cutting, etc., are hydrostatically tested.

The final system checks for operation and availability are covered by prestartup checksheets for required systems and by interlock checks, which are performed by maintenance personnel from the EBR-II Instrumentation and Control Section.

During the initial startup following a maintenance period, the shift operators pay special attention to any systems that were involved in shutdown activities.

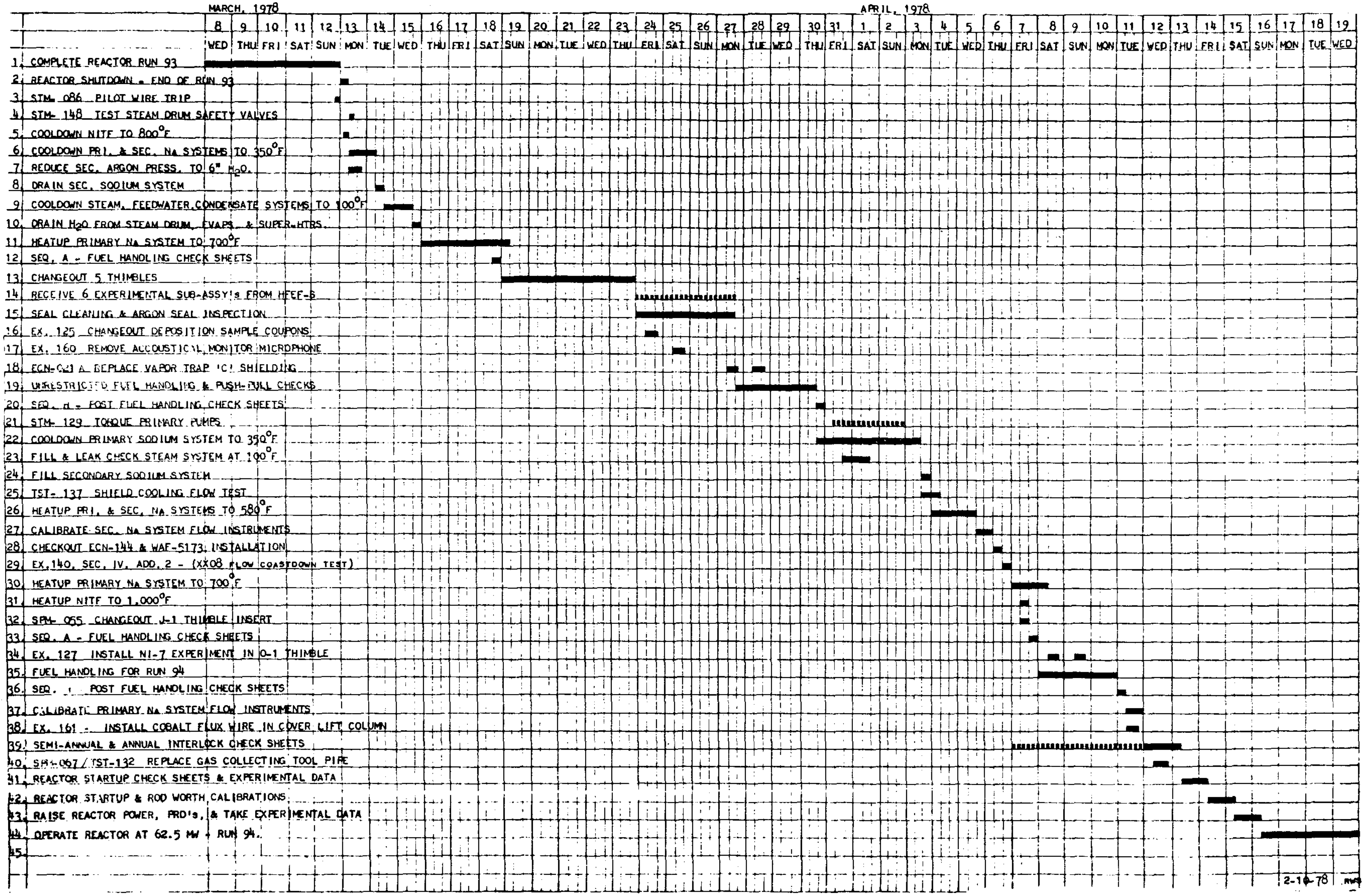

Fig. 5. Example of Maintenance and Modification Schedule for Annual Long Shutdown

VI. OPERATING EXPERIENCE AND COMPONENT PROBLEMS

The major mechanical components in the primary-sodium, secondary-sodium, and fuel-handling systems have demonstrated durability and reliability during 17 years of operation. No major or minor nuclear incidents have been experienced, and no plant shutdown from equipment failure has exceeded four months. Most of the failures occurred during the early years and were due to design deficiencies. Successful repairs and modifications were made on the failed components.

The performance of the primary system and reactor has been very good. Throughout its operating history, the reactor has been highly stable and readily controllable. An ever-present and predictable negative power coefficient guarantees kinetic stability under all foreseeable operating conditions.

The mechanical stability of the core components has been excellent. No indication of vibration or coolant-flow blockage has been observed during the years of operation, nor has any indication of gas entrainment been detected in the coolant. The postirradiation examinations conducted on discharged core components (subassemblies, control- and safety-rod thimbles, etc.) have shown no evidence of wear, fretting, ratcheting, vibration, or abrasion.

A. Fuel-handling System and Core Components

The fuel-handling system has been used extensively.[9] As of July 31, 1981, the system had made more than 23 000 separate fuel transfers. The unique design of this system, which provides for very rapid refueling of the reactor, results in efficient use of the reactor and permits interim storage within the primary tank. Most of the fuel-handling components are original equipment and have operated with only minor difficulties. Some design changes were required after initial checkout, and a number of changes were made as operating experience was gained; in general, however, the system has performed very well.

Fuel-handling and primary components have been removed from the primary tank routinely and without significant problems. Since the primary tank was filled with sodium, the core gripper has been removed four times, the transfer arm has been removed once, control-rod drives have been replaced 17 times, and primary pumps have been removed and reinstalled three times. Mechanical failures have occurred 11 times in control-rod drives: three times from gripper-jaw failures, and eight times from sealing-bellows failures in the drive shafts. Seven of the original 12 control-rod drives have operated perfectly for over 17 years.

Buildup of sodium and sodium oxide in the cover-gas space has also been responsible for problems with fuel-handling components. The sodium and sodium oxide builds up in the clearances and causes binding. Corrective action in such cases generally consists of enlarging clearances and/or improving sodium-draining provisions to reduce the oxide buildup to a minimum. Routine removal, cleaning, and reinstallation is another alternative used to eliminate untimely problems due to the oxide buildup.

Several other problems have occurred during fuel handling, but these were minor and readily overcome. Two subassembly upper adapters were inadvertently twisted during fuel handling in the core. These subassemblies were removed from the primary tank with little difficulty by normal procedures. In addition, one control-rod thimble was damaged slightly in 1967, and another in 1977. The damaged thimbles were removed by normal fuel-handling procedures and equipment.

A subassembly was bent in the storage basket in 1978. The damage occurred when the subassembly bound up on a fission-gas-collecting tool, and the storage basket was subsequently rotated and raised. A special retrieval tool was fabricated and used to remove the bent subassembly from the storage basket. To prevent a recurrence of this incident, the gas-collecting tool has been deactivated and raised to a position where it cannot interfere with any subassemblies in the storage basket.

During the last five years, high forces have been required occasionally to remove and reinsert stainless steel reflector subassemblies in rows 8-10. The primary causes of the high forces have been reflector-subassembly bow because of the large temperature and flux gradients across the reflectors and the irradiation-induced swelling of the hex cans. The reflector subassemblies are now rotated 180° on a regular basis to allow them to reach their full irradiation lifetime based on swelling limitations. This rotation has reduced subassembly bow and has greatly reduced the frequency that high forces are required to remove or reinsert the reflector subassemblies. (This subject is discussed in more detail in Sec. IX.)

B. Rotating-plug Seals

The seals of the rotating shield plugs have been a continuing source of difficulty since the system became operational.[10,11] The large rotating plug is mounted in the primary-tank cover, with its vertical axis aligned with the vertical axis of the reactor core. The small rotating plug is positioned off-center within the large plug. Around the outside of each plug is a dip ring or blade that dips into a seal trough.

The 315-mm-deep tin-bismuth eutectic alloy in the trough must be molten during fuel handling to permit plug rotation. When the reactor is operating, the alloy must be half molten, with the lower portion molten for a gas seal and the surface frozen to prevent the seal from being displaced if the cover-gas pressure becomes abnormally high in the primary tank.

A copper ring was originally used at the bottom of the blade to provide an even temperature distribution in the lower part of the seal. Before the filling of the primary tank with sodium and during initial checkout of the fuel-handling system, plug rotation became increasingly difficult.[6] Both plugs were finally removed when rotation became impossible. The copper rings were found to be badly eroded, especially near the heaters, and at some heater locations the ring was completely broken.[12] This condition had caused severe binding between the blade and the trough wall. Replacement of the copper rings with stainless steel rings corrected the erosion problem, and heat distribution was not noticeably affected.

After the primary tank was filled with sodium, difficulty with plug rotation again increased. Additional heat was usually used to enable the plugs to be moved, because it became harder to obtain proper seal temperatures. The time needed to obtain rotation steadily increased. Finally, almost a full day of seal melting was required before the plugs could be moved, and manual force had to be routinely applied to the large plug to achieve free rotation.

Inspection of the air side of the seals showed that considerable oxidation of the tin-bismuth alloy had taken place. A dry, black, powdery oxide was found on the top of the seal alloy on the air side of each seal. To improve the access to the air side of the seals, an 18-mm-dia hole was drilled in 1966 through the steel and high-density concrete of each of the shield plugs over the outer annuli of the seals. Initial attempts were then made to clean the seals through these holes.

The first operation performed through the new access holes was a vacuum-cleaning process to remove as much dross or oxide material as possible from the surface of the alloy. This vacuuming process and exploratory probing of the alloy in the outer annuli of the troughs showed the presence of moderately compacted deposits of oxide-like material extending about 200 mm down in the trough and, except for a few gaps, apparently extending around the trough circumference.

The solid areas were difficult to penetrate. Finally, a 13-mm-dia steel tube was hammered through one of these areas, and when the tube was removed from the alloy, the solid substance remained in the tube and could then be removed by hitting the end of the tube. This procedure was soon adopted as the initial seal-cleaning method, and the air side of each plug seal was cleaned in this manner.

When the cleaning was completed, it was evident that a large quantity of loose oxide remained in the seal alloy and should be removed. Temperature profiles of the seal alloy were taken through the new access holes, and the trough thermocouples were found to be indicating low temperatures. Oxide deposition appeared to be insulating the thermocouples from the alloy, and an attempt was made to brush off the oxide with a steel brush. That effort was partly successful, and some of the temperatures began to be indicated more accurately. More important, though, was the discovery that the oxide stuck to the steel brush when it was removed from the alloy. The brush-cleaning technique evolved from that incident. Cleaning was then accomplished by rotating the plug in 0.5° increments, inserting and removing the brush at least once at each increment, and cleaning the oxide material from the brush. After the initial cleaning and refilling of the seal troughs, plug rotation became much easier.

Although crude and time-consuming, a program of periodic brush-cleanin of the seals proved effective and provided relatively trouble-free plug rotation. The brush-cleaning was used until November 1972, when a 76-mm-dia access hole was drilled through the plug-support structure to the air side of the seal for the large plug. The method of cleaning was changed to a more direct means of skimming the large-plug seal with specially designed tools.

In April 1973, a similar access hole was drilled in the small plug to the air side of the seal. The same skimming technique is now used on the small-plug seal. The new access holes also permitted scraping of the seal ring, which had oxide accumulations as thick as 7.9 mm sticking to it, and provided a means of removing these scrapings from the seal. The new seal-cleaning technique reduced tremendously the amount of seal-alloy material that had to be added after each cleaning and considerably reduced the time required for cleaning the seal.

Even though the air sides of the seals were being maintained in good condition, periodic sticking of the large plug continued and became progressively worse. To achieve free rotation initially, the seal has required heating to elevated temperatures (200 to 300°C) and application of a manual force of up to 36 000 N to the large plug. In April 1975, a 76-mm-dia hole was drilled through the large plug; this provided better access to the argon side of the seal.

Observations through the access hole revealed that the argon side of the seal contained large accumulations of materials, but it was immediately seen that the large plug was sticking because of large accumulations of material in the plug annulus, as shown in Fig. 6. The annulus is the clearance between the large-rotating-plug wall and the rotating-plug support structure and opens directly to the argon blanket of the primary tank.

Samples of this observed material contained 40% sodium; the remainder was mostly seal-alloy material that had apparently spilled over the inside of the seal wall during seal-filling operations. The melting point of the material was about 400°C, and 100-140 kg of material was estimated to be in the annulus. Most of the material was removed during the long maintenance shutdown in March 1976. A glovebox was fabricated to prevent escape of radioactive argon to the reactor-building atmosphere during the cleaning

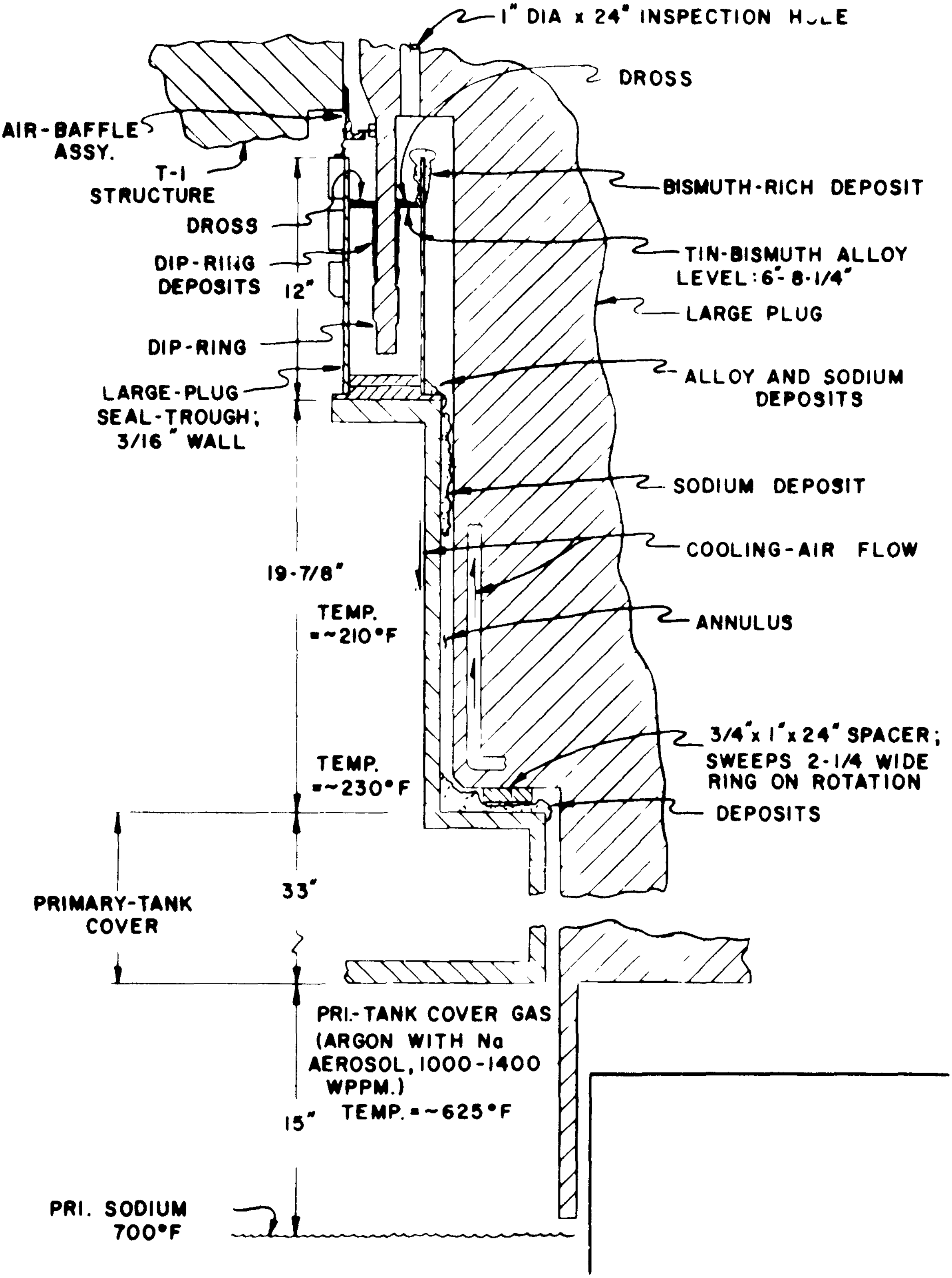

Fig. 6. Cross Section of Seal of Large Rotating Plug, Showing Location of Deposits Related to Argon Side of Seal.
Conversion factors: °F = 1.8°C + 32; 1 ft = 0.3048 m; 1 in. = 25.4 mm.

operation and to prevent ignition of the material as it was removed. Various cleaning methods were developed, the most successful being a clamshell-digger device that was applied through the glovebox. After the annulus was cleaned, free rotation of the large rotating plug was again achieved.

A similar access hole was drilled into the small-plug-annulus area during the long spring shutdown of 1977. Annual inspections of the small-plug annulus have shown it remaining relatively clear of any material buildup similar to that observed in the large-plug annulus. However, we had to repeat the cleaning of the large-plug annulus in 1979 and 1981. It appears that the annulus of the large plug must be cleaned every two or three years. Accumulations have not formed as rapidly in the small-plug annulus, and cleaning of that annulus is not expected for the foreseeable future.

A program of periodic cleaning of the seals (air and argon sides) and the large-plug annulus should provide relatively trouble-free plug operation in the future. The main problem associated with the EBR-II rotating seals has been lack of access in the initial design to allow cleaning and maintenance. Any future design of a seal of this type should include provisions for access for periodic maintenance and cleaning of the seals.

C. Major Primary- and Secondary-sodium Components

The only major problems that have been encountered with the primary-tank components were the initial binding of both of the primary pumps and the loose drain tube in the IHX. These were briefly discussed in Sec. III.

Only one incident has occurred during the many interchanges of components within the primary tank. An instrumentation plug was being reinserted when a sodium-water reaction occurred in the hole for the plug. The plug contained instrumentation for measuring the reactor outlet temperature and pressure. A small amount of sodium was expelled around the plug shaft and up through the hole. Investigation showed that the plug had not been adequately dried after cleaning, and the residual alcohol-and-water cleaning solution reacted with the sodium. Damage to equipment by the expelled sodium was insignificant, and contamination was minor. The plug was not damaged. Since this incident, stricter rules have been made to ensure adequate drying of each component before installation into the primary tank.

The secondary system has been essentially trouble-free except for an early failure (fatigue cracking due to a design deficiency) of the main secondary-sodium pump duct, a pinhole leak (due to a faulty weld) of water to atmosphere in one of the sodium-to-water steam generators, and a sodium leak of 300 L during maintenance activities. The piping in the secondary system has never experienced a weld or pipe failure. However, several ring-joint flanges have had minor sodium leaks. These leaks were detected by flange leak detectors or during maintenance inspection. Only ordinary problems have been experienced with conventional components such as pumps and valves.

In 1974, one of the two superheaters began to show a degradation in performance when a reduction in outlet steam temperature occurred. The two EBR-II superheaters are identical except for the type of bonding used during fabrication of the duplex tubing. One superheater has metallurgically bonded tubes. The superheater that has shown the anomalous behavior has mechanically bonded tubes. The abnormal performance of the one superheater is thought to be caused by a slight separation of some of the duplex tubes. To allow destructive examination of the failing superheater, it was removed from the steam system in the spring of 1981. It was replaced with a converted evaporator that has metallurgically bonded tubes. The evaporator was removed from the steam system early in 1980 for the conversion process. While the evaporator was being removed, each of its tubes was ultrasonically tested for integrity. The inspection revealed that the tubes were in excellent condition after 17 years of service. No flaws or cracks were found.

A flow orifice was installed in the secondary-sodium supply header to mitigate the effect of flow unbalance on the individual evaporator units. Flow measurements (by pulsed-neutron activation) were taken before and after the evaporator removal to verify that the distribution of secondary-sodium flow was as predicted. The steam and secondary-sodium systems have operated with seven evaporators instead of the original eight without degradation in performance.

Overall operation of the steam-generator system continued to be excellent. Replacement of the failing superheater with the converted evaporator restored the steam system to full design capabilities.

D. Power-plant Components and Systems

The power plant is essentially a conventional steam plant. Performance of the system has been very good, and the power plant has seldom been responsible for loss of plant availability.

No steam leaks have occurred in the main steam piping. Most of the problems have been the usual ones experienced with conventional components. Pump and valve failures have caused most of the breakdowns. A few minor leaks were discovered in the piping and welds in the blowdown and desuperheating systems. These systems were readily repaired with minimum loss of time.

In August 1979, a tube leak was discovered in the No. 4 feedwater heater (which operates at 1250 psig, or 8.62 Pa). The initial leak was repaired by driving a steel expanding plug into each end of the tube. Since the initial tube failure, 10 more tubes failed and were plugged. Up to 30 tubes can be plugged without affecting plant operation. Because the repeated tube failures caused several interruptions of plant operation, the No. 4 feedwater heater was replaced in early 1981.

E. Instrumentation

Throughout the operating history of EBR-II, a number of nonreplaceable sensing devices for primary-sodium flow, pressure, and temperature have failed. Redundancy was provided for all important parameters, and the loss of the instrumentation has not severely affected the capability to monitor the key parameters. However, additional failures could eventually affect plant operation if all monitoring capabilities for any important parameter, such as flow or core ΔT, were completely lost.

During the past 17 years, continuous efforts have been made to upgrade the quality of accessible instruments. Much of the original instrumentation has been replaced with instruments designed in the 1970's. Significant improvements have been made in the sodium-level and -pressure indicators and the piping heaters used at EBR-II. Although the performance of the original nuclear instrumentation was always satisfactory, three wide-range nuclear channels were installed in 1975 to reduce the total number of nuclear channels and to simplify the plant protection system (PPS). The old system consisting of nine separate detectors and their instrumentation was removed.

1. Flow Instrumentation

Magnetic flowmeters were installed in the high-pressure-plenum and low-pressure-plenum inlet lines from each primary pump and in the reactor outlet piping (see Fig. 2). These were chosen because of their simplicity and linear output for a wide range of flow rates.

At the time of installation. the state of the art required that the magnetic flowmeters be calibrated in place. To provide this capability and to increase system reliability, venturi flow tubes were installed in series with each of the magnetic flowmeters. All of the venturi flow tubes have failed except the one in the reactor outlet piping. This single remaining flow tube is used to check the calibration of the magnetic flowmeters and to monitor total flow as input to the reactor shutdown system (RSS).

Three of five original magnetic flowmeters have now failed. The two magnetic flowmeters still operating are those for primary pump No. 2 high-pressure and low-pressure supply to the reactor inlet plenum.

Alternative methods of monitoring reactor flows and providing flow-related inputs to the PPS are being developed as part of contingency planning for further failures in original flow instrumentation.

2. Temperature Instrumentation

Numerous thermocouples and some resistance thermometers were installed in the EBR-II primary system to monitor component and sodium temperatures. The resistance thermometers were installed in areas where high accuracy was desired; however, most of the temperature sensors were thermocouples because of their lower cost and easier installation. Among the components monitored are the primary pumps, the IHX (inlet and outlet), the instrument thimbles, selected subassembly outlets, the bulk sodium, and the primary-tank walls.

Many of the thermocouples and all of the 10 original resistance thermometers have failed. Because of the redundancy used, however, these failures have caused a problem only in monitoring one parameter--coolant outlet temperature. Originally, five thermocouples and two resistance thermometers were available in the outlet piping. Now only one thermocouple is operable. That thermocouple is being used as both a monitor for reactor coolant outlet temperature and as one input to a derived signal for reactor ΔT. A temperature and pressure monitoring probe that has been placed in the reactor coolant outlet plenum is being evaluated as a backup to the one remaining coolant-outlet-temperature thermocouple.

3. Sodium-pressure Sensors

Sixteen pressure sensors were installed in the primary system. These sensors transmit primary-sodium pressure by means of a capillary tube filled with NaK (22% sodium-78% potassium) to a pressure transmitter outside the primary tank. Of the 16 installed pressure sensors, only three are still operable. Two of these are at the discharge of the primary pumps and can be replaced, if necessary, whenever the primary pumps are removed. The third is in the reactor outlet plenum.

Of the 13 failed sensors, nine were associated with the reactor-vessel inlet and outlet coolant pressures, and the other four were in the primary-pump high-pressure- and low-pressure-plenum discharge lines. None of these is replaceable.

4. Sodium-level Sensors

Two types of sodium-level sensors were originally used in the primary and secondary systems. Resistance-type probes were installed in the primary tank, secondary storage tank, and secondary surge tank. A pressure-type sensor that included a temperature compensator was also installed in the primary tank to measure the static pressure head of the sodium. The resistance probes are still used for the main level indication in the secondary-sodium storage tank and as backup in the primary tank. The probes have several inherent problems and shortcomings. They indicate level only in stepwise increments of about 100 mm, as displayed by a series of lights. Readings have frequently been erroneous because of insulation breakdown in the probes or bridging of sodium and sodium oxide from one probe to another.

The pressure sensor in the primary tank did not give the accuracy or reliability desired for a continuous level indicator and was replaced by a float device in 1969. The float device was developed at EBR-II.[13] It consists of a partially submerged buoyancy cylinder hanging from a force transducer. This device has operated successfully since installation and has demonstrated an accuracy of ± 6.0 mm over a measuring range of 508 mm. The resistance probe in the secondary surge tank has been replaced by an induction-type level probe. This probe has been stable and accurate. A similar induction-type level probe has recently been installed in the primary-purification surge tank.

5. Improvement of Nuclear Instrumentation

In the spring of 1975, three wide-range nuclear channels were installed in EBR-II. These channels replaced the nine channels previously used to monitor the startup-range, intermediate-range, and power-range flux signals and to provide the reactivity-protection input associated with the PPS. The new channels give improved reactivity protection and flux monitoring by providing (a) neutron-flux signals that are unaffected by gamma-flux levels, (b) 1.5 decades of overlap between the startup-range and intermediate-range flux signals, and (c) a period indication that is unaffected by the transition from the startup range to the intermediate flux range. The wide-range nuclear channels have been trouble-free and have not caused any spurious reactor scrams.

Each new channel consists of a ^{235}U-lined fission chamber, a preamplifier, a 10-decade log-power drawer, a linear-power drawer, and associated equipment. The fission chambers, shielded by Boral cans, have a response range of approximately 2 nv to 2 x 10^{10} nv. The signal and high-voltage cables are separate, shielded coaxial cables. All other components are outside the primary tank.

6. Sodium-system Trace Heating

Auxiliary heating of sodium piping is done in two ways: induction trace heating and resistance trace heating. The main secondary system is heated by induction heating coils wrapped on the outside of the thermal insulation. This system has been virtually trouble-free and extremely satisfactory. Because the wire is on the outside of the insulation, repairs are simple to perform.

The resistance trace heating, which requires the wrapping of heating tape or elements directly onto the pipes, valves, etc., has not been quite as satisfactory. The original heaters were primarily glass-covered tape-type heaters, which after a short period of service, became quite brittle and would fail if disturbed. Since the insulation must be removed and reinstalled for heater replacement or repair, considerable time is required for such work. Replacement heaters for the tape-type heaters have been primarily "wrap-on" mineral-insulated, sheathed heaters. The lifetime of these heaters is very good; there have been very few failures of the heating elements.

VII. CONTROL AND MONITORING OF SODIUM AND COVER-GAS PURITY

The ability to maintain low levels of impurities in the EBR-II sodium and cover-gas systems has been a major positive factor in safe and reliable plant operation. A rigorous program of purity monitoring and control is used t ensure that operating limits for more than 25 chemical and radioactive impurities are not exceeded. Some of more important efforts in sodium and cover-gas technology are described below.

A. Sodium Purification

Purity of the EBR-II primary and secondary sodium is maintained by essentially continuous cold trapping of a small side steam of sodium at a temperature of 115-125°C. Experience has shown that continuous operation of the cold traps is the most effective method for removing oxygen and hydrogen to the desired levels.[14] Tritium is also removed by coprecipitation with the hydrogen, and to a lesser extent, by ion exchange with hydrogen in the cold trap. Radioisotopes such as ^{131}I and ^{137}Cs are not effectively trapped; however, there is evidence that ^{137}Cs as well as ^{54}Mn, ^{60}Co, ^{65}Zn, ^{124}Sb, ^{125}Sb, ^{134}Cs, and ^{182}Ta tend to concentrate in the primary cold trap.

Five cold traps have been used in the primary sodium system. The first trap was installed temporarily to provide cleanup after the initial sodium fill, and was in service for only about a month in 1963. Each of the next three traps was progressively smaller. The second trap had a sodium volume of about 1890 L and was in service from April 1963 to June

1968. The third trap had a sodium volume of about 1135 L and was in service from November 1968 to November 1977.

The second and third traps were packed with stainless steel mesh having a density of 16 kg/m^3. Neither of these traps completely plugged in service. Each was removed during modifications to the purification system. Contact radiation levels on the third trap, removed in November 1977, were as high as 8 R/h.

The fourth primary trap was a spare secondary cold trap. It had a sodium volume of about 570 L and was packed with stainless steel mesh having a density of 32 kg/m^3. This trap was integrally shielded with about 25 mm of lead to reduce radiation levels in the sodium-purification room. It was in service from March 1978 to October 1980. It showed signs of plugging within two months, and was essentially plugged at shutdown in October 1980. The brief lifetime of this trap compared to previous primary traps is attributed to its smaller size and the higher-density mesh.

The fifth primary cold trap was installed in November 1980. It is the same size as the fourth trap and is integrally shielded. However, it is a quasi-meshless design that contains mesh only in the upper half of the center section of the trap. It is intended that impurities will normally be trapped in the meshless zone, and that deposition in the wire mesh will occur only when source rates are abnormally high.

The new, quasi-meshless cold trap has reduced oxygen to an equilibrium level corresponding to a plugging temperature of 135-138°C. This is about 5.5°C higher than the equilibrium level attained with the mesh-filled trap. The initial efficiencies of the cold trap (measured in May 1981) for removal of oxygen, hydrogen, and tritium were 0.5, 14, and 7%, respectively.

Six cold traps have been used in the secondary sodium system. The first trap became plugged during system cleanup. The temporary trap used to purify primary sodium was then used to complete initial cleanup of secondary sodium. The third trap was operated from 1963 to December 1970. The fourth trap was operated from February 1971 through February 1977 and processed about 117 ML of sodium. The fifth trap began operation in April 1977 and was replaced with a meshless cold trap in October 1979.

The last four traps removed from the secondary system were not totally plugged, but had become difficult to operate. Pressure drops were high, and trapping effectiveness was low. This deterioration of performance was undoubtedly due to two causes: (1) the constant trapping of sodium hydride produced as hydrogen diffused into the sodium through steam-generator tubes, and (2) clean-up operations following periodic maintenance on the secondary sodium system.

The meshless cold trap installed in the secondary sodium system in 1979 is similar to the previous mesh-filled traps except for deletion of the mesh and addition of stilling baffles. Efficiency of the meshless cold trap is much lower than that of the mesh-filled trap (15% vs 90% for removal of hydrogen). However, the rate of impurity removal depends on the product of efficiency and flow rate. Since the meshless trap is not susceptible to plugging, high flow rates can be maintained to provide acceptable impurity control. The meshless trap has twice successfully purified the secondary sodium after severe contamination during maintenance activities, once in March 1980 and again in May 1981. It has also held the equilibrium hydrogen concentration during reactor operation to about 100 ppb, compared with 90 ppb for the mesh-filled traps. The meshless trap is expected to have a high capacity for sodium impurities and to require replacement less often than the mesh-filled trap.

B. Monitoring of Sodium Purity

A program to characterize and monitor impurities in EBR-II sodium began in 1967 following the discovery of copper deposits in the primary plugging meter.[15,16] The source of the copper was unclad copper bus bars of the primary auxiliary pump, which were immersed in the sodium.

Some of the important discoveries and conclusions resulting from this characterization program are listed below:

- The discovery of bismuth and tin and their activation products showed that the eutectic tin-bismuth alloy was entering the primary sodium from the seal troughs of the rotating plugs.

- Lead was discovered in the primary sodium, but not in the secondary sodium. The source of the lead has not been identified.

- It was discovered that ^{131}I, ^{137}Cs, ^{54}Mn, and ^{124}Sb segregate from sodium to surfaces, but also that ^{113}Sn, ^{117m}Sn, and ^{110m}Ag remain distributed in the sodium as sodium temperature is reduced toward the freezing point.

- Source rates of hydrogen and oxygen into the sodium have been measured. Hydrogen enters the secondary sodium by diffusion through steam-generator tubes. Oxygen and hydrogen enter the primary sodium as water and air contamination during fuel handling.

- The principal plugging agent in the primary sodium has not been firmly identified. The principal plugging agent in the secondary sodium is sodium hydride.

- The principal activation products in the primary sodium (excluding ^{22}Na and ^{24}Na) are those of tin, which enters the sodium from the rotating-plug seals. The principal activated corrosion product is ^{54}Mn. The principal fission product is ^{137}Cs.

- The principal source of tritium in EBR-II is ternary fission.

- Plutonium-239 has been detected only three times in the primary sodium despite numerous fuel-cladding ruptures.

C. Cesium Trap

The EBR-II primary purification system was modified in March 1978 to provide a trap for removal of ^{137}Cs from the primary sodium. The cesium trap is shown in Fig. 7. Its dimensions, including a lead shield, are approximately 900 x 900 x 600 mm. The packing material is a glassy carbon manufactured from a polymer foam and has the trade name Reticulated Vitreous Carbon (RVC). The trap contains approximately 0.011 m^3 of RVC having a surface area of approximately 370 m^2. A 35-μm sintered stainless steel filter prevents sodium washout of the RVC.

The cesium trap is between the economizer and crystallizer of the primary cold trap. The normal operating temperature at this location is 193°C when the cold trap is operating at 115°C. The temperature of the sodium entering the purification system from the primary tank is 371°C.

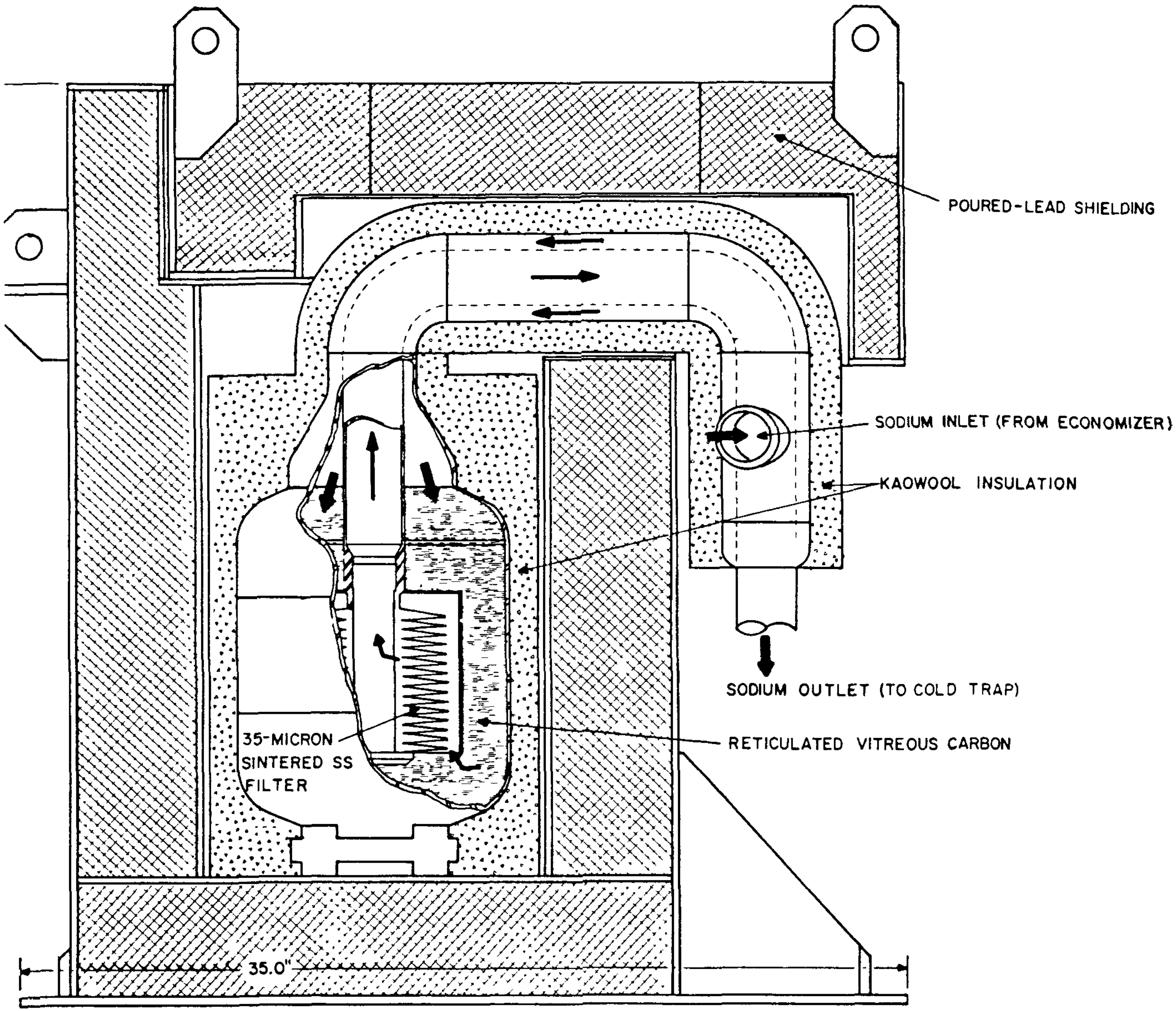

Fig. 7. Cesium Trap for Primary Sodium. Conversion factors: 1 micron = 1 μm; 1 in. = 25.4 mm.

During the initial nine days of cesium-trap operation, ^{137}Cs activity was reduced from 351 to 56 nCi/g (13.0 to 2.1 kBq/g). The total inventory of ^{137}Cs was thereby reduced from 103 to 16.5 Ci (3.81 to 0.611 TBq). Further operation of the trap reduced the total ^{137}Cs inventory to less than 10 Ci (0.37 TBq). The cesium trap is now operated on an interim basis to keep the ^{137}Cs activity in the primary system at a low level. More than 385 Ci (14.2 TBq) of ^{137}Cs had been accumulated in the trap through July 1981.

D. Cover-gas Cleanup System (CGCS)

A major and important part of EBR-II's irradiation program has been the testing of fuels in the run-to-cladding-breach (RTCB) program, and more recently, the irradiation of oxide-type fuels in the RBCB program. These irradiation programs have several distinct operational problems that are caused by the excessive equilibrium levels of fission-gas activity in the argon cover gas. Because of small, recurring cover-gas leaks, this increase in cover-gas activity results in a radiological problem in the reactor building. It also causes difficulty in both the detection and identification of later fuel-cladding failures. The only solution to these problems before the installation of the CGCS was to purge the cover gas by discharging it to the site suspect-exhaust system. The CGCS was proposed and installed to correct both of these problems and, at the same time, make EBR-II a "near-zero release" facility.[17]

The installation of the CGCS was completed in the spring of 1977. The CGCS removes krypton, xenon, and other condensable gases, using a cryogenic distillation process. Included in the CGCS is an on-line automatic xenon-tag trapping and analysis sytem. The CGCS allows identification of breached elements and cleanup of the resulting gas activity and residual xenon tag without interrupting reactor operations.

Figure 8 shows the major components and flow paths of the cover-gas purification portion, or main loop, of the CGCS. The basic process involves extraction of 0.0005 to 0.005 m^3/s of argon from the 8.5 m^3 of argon cover gas, processing and cleaning of the argon, and reheating of the cleansed argon before its return to the argon cover-gas system in the primary tank.

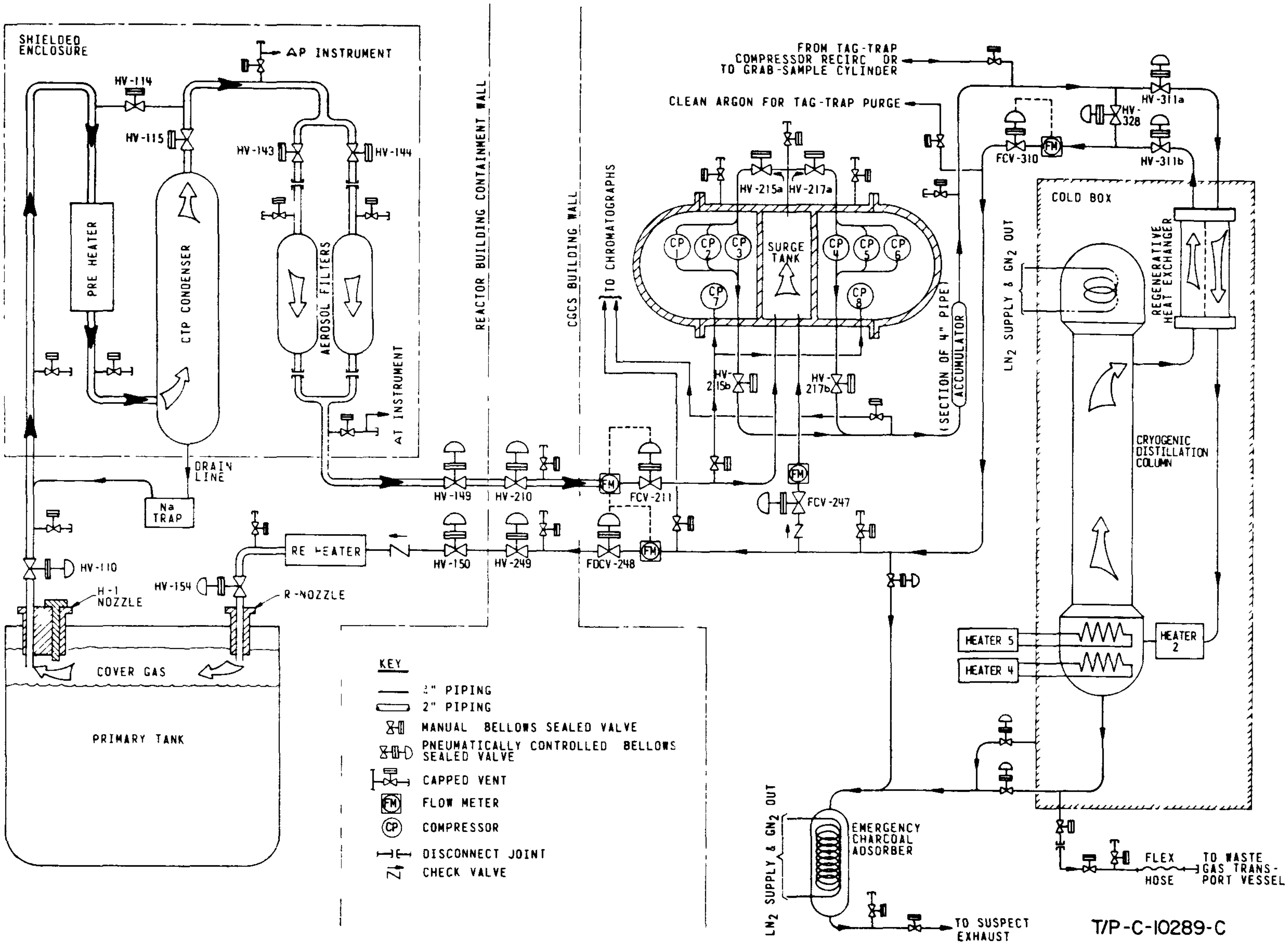

Fig. 8. Simplified Schematic of Main Loop of Cover-gas Cleanup System (CGCS)

The major steps of the purification processing are (1) heating by the preheater, (2) removal of sodium vapor and aerosols in the controlled-temperature-profile (CTP) condenser and aerosol filters, (3) cryogenic distillation at -182°C of xenons, kryptons, and other condensable impurities in the cryogenic distillation column, and (4) reheating of the cleansed argon before its return to the cover-gas system.

Two of the eight compressors (compressors CP-7 and CP-8) provide a cover-gas-sample flow to the xenon-tag trapping system. (See Fig. 9.) Xenon isotopes from the cover-gas sample are concentrated at -78°C by adsorption on activated charcoal in one of the three primary tag beds (PTB-1, -2, and -3). The sample is then transferred to one of the secondary tag beds (STB-1, -2, and -3) by heating the primary tag bed at 200°C and trapping the eluted gas on the activated charcoal in the secondary beds at -173°C. Contaminants (argon and krypton) are driven off by heating the secondary beds to -78°C, and then the concentrated xenon-isotope sample is transferred to the sample vial, where it awaits analysis by the on-line mass spectrometer. The entire tag-trapping process and analysis is controlled by a NOVA 2/10 computer.

The main loop of the CGCS became operational in June 1977. The purification efficiency of the cryogenic distillation column has been greater than 99% for xenon and krypton, and the entire main loop has provided excellent cleanup service, making EBR-II, a "near-zero release" facility.

The xenon-tag trapping system has had a number of problems, however. Numerous heaters have failed, the mass spectrometer did not exhibit adequate sensitivity, and there were problems in programming and in interfacing between the computer, the system, and the mass spectrometer. Most of the difficulties have been corrected by upgrading and modifying the system.

During 1979, the xenon-tag trapping system operated with an availability of greater than 90% and was used to correctly identify 14 subassemblies in which elements had breached. The system is exhibiting a 95-98% availability and has identified a total of 45 subassemblies containing cladding breaches.

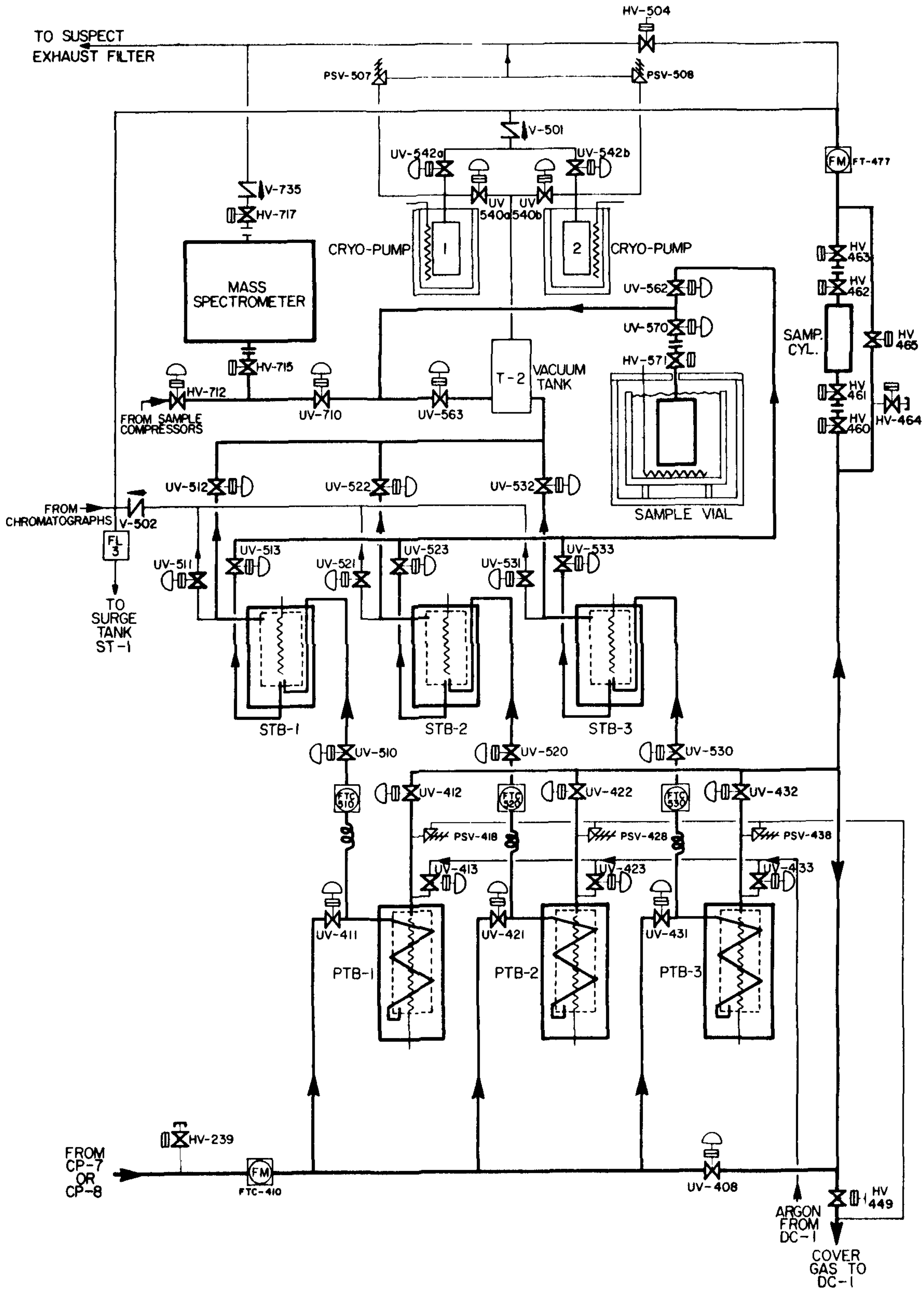

Fig. 9. CGCS Tag-trap Analysis System

The trapping system is being operated at least once a day to obtain a background sample. If a fuel-element cladding breaches, the sampling frequency is increased. The system can provide a xenon-tag analysis about every three hours with sensitivity equal to or exceeding that of the manual sampling method previously used.

E. Hydrogen-meter Leak Detectors

A water-to-sodium leak-detection system was installed in EBR-II in April 1975. The system now consists of 11 hydrogen-meter leak detectors (HMLD's) at the outlets of each of the evaporators and the two superheaters, with two additional HMLD's at the outlet manifolds of each group of evaporators.

Each HMLD unit consists of a sodium system, a nickel membrane, and a vacuum system, as shown in Fig. 10. An ion pump continually removes the hydrogen entering the vacuum system, creating an electric current that is proportional to the hydrogen flux across the nickel membrane. The HMLD's are calibrated by periodic equilibrium-pressure measurements with a vacuum gauge.

Twelve modified units were installed in 1976 and 1977 to replace the original 10 HMLD units that failed within a year after their installation.[18-20] The original failures were caused by sodium-to-vacuum leaks through stringers in the stainless steel at a butt-weld joint. The replacement units have socket-weld joints, which have eliminated this cause of failure. In May 1979, one of the modified units failed. It was replaced during the 1980 maintenance shutdown with the HMLD salvaged from the evaporator that was removed for conversion to a superheater. (See Sec. VI.D.) Subsequent examination showed that sodium had leaked into the vacuum system through an intergranular crack in the dome of the nickel membrane. In November 1980, a second modified HMLD failed. This unit was replaced during the 1981 maintenance shutdown with a newly built spare.

The EBR-II data-acquisition system (DAS) provides data storage, data display, and alarm functions for the HMLD's. Alarms are annunciated for abnormal nickel-membrane temperature, low sodium temperature, high hydrogen level, and high hydrogen rate of rise. The hydrogen alarms are set at 200 ppb H (high level) and at 500 ppb H/h (rate of rise) and are the

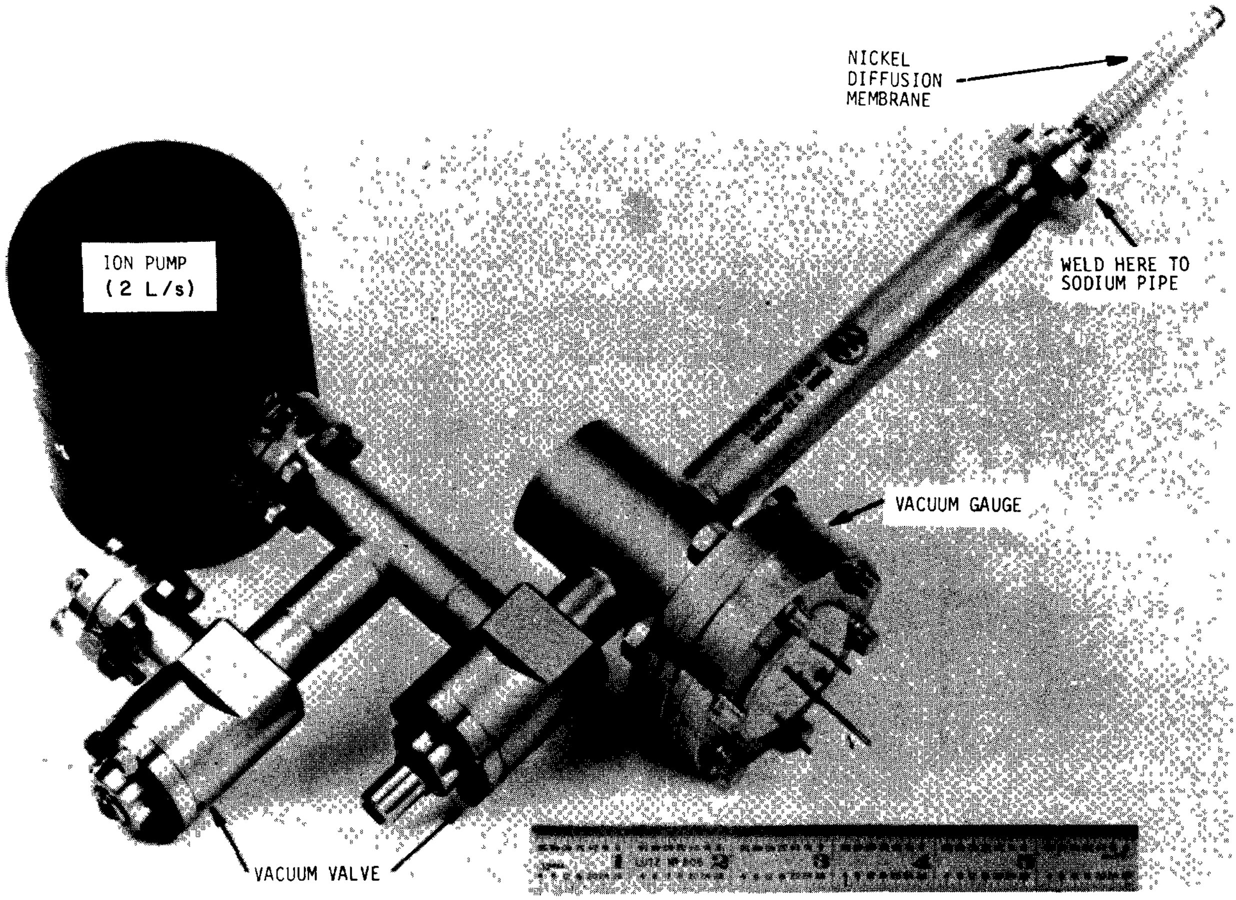

Fig. 10. In-sodium Hydrogen Meter

primary basis for detection of water-to-sodium leaks. The alarm limits correspond to a leak-rate sensitivity of about 80 μg H_2O/s.

The HMLD's have provided continuous data on background hydrogen levels, source rates, and variations related to plant chemistry. From the hydrogen data, we conclude the (1) the hydrogen level (normally about 100 ppb) is affected by reactor power level and cold-trap temperature, (2) the hydrogen source rate into the secondary sodium is 7-20 ppb H per day, depending on the hydrazine concentration on the water side, and (3) a new mesh-filled cold trap initially removes hydrogen with an efficiency of 80-90%.

VIII. FAILED-FUEL DETECTION AND IDENTIFICATION

EBR-II's irradiation program has grown and increased in complexity as new fuel types have been tested, as the RTCB program has been expanded, and as the RBCB program has been inaugurated. As experience has been gained in the area of cladding-breach characteristics, EBR-II's operational philosophy has changed to allow continued reactor operation during the identification phase of a cladding breach as long as no anomalous reactivity behavior or delayed-neutron (DN) signals are observed. The growth of the irradiation program and the evolving operational philosophy have continually presented complex and challenging problems in the areas of failed-fuel detection and identification. The following is a brief summary of EBR-II's identification techniques and detection systems.

A. Identification Techniques

Cladding breaches have been readily detectable in EBR-II by noticeable increases in the normally low activity of the argon cover gas. In addition, the type of fuel (metallic, ceramic, etc.) and bond in the breached element have often been identified by seeing which of the radioactive fission-gas isotopes is released first.[21] For example, breached sodium-bonded elements initially release ^{135m}Xe; this isotope is born from ^{135}I that escapes, with the sodium bond, ahead of stored fission gas. Occasionally, the DN monitor FERD (fuel-element rupture detector) has detected the short-lived, DN-emitting isotopes of iodine and bromine that are also released with the sodium bond. In contrast, helium-bonded oxide elements generally release all the gas isotopes together.[22]

Identification of a subassembly that contains a breached element has in the past been a long process, but one that has been improved considerably by experience. Before 1972, identification was by the systematic removal of groups of one or more suspect subassemblies, aided by such nonspecific methods as the measured $^{134}Xe/^{133}Xe$ "age" ratio of the released gas and failure statistics. Since 1972, the increasing use of unique ratios of stable isotopes of xenon (xenon "tags") in experimental elements has gradually shortened the search process from days, and sometimes weeks, to hours; often, only one suspect--the correct one--need now be removed. This improvement has meant that the number of reactor shutdowns and startups required to remove a "leaking" subassembly has dropped from the range of three to six (or more) to one or, at the most, two.[21]

Recent experience with nearly simultaneous breaches has shown that problems still exist with the xenon-tag method of identification. First, the unique ratios of the tag provided for each experiment tend to become less unique as residence time in the core increases. This is due to changes in concentration of several of the tag isotopes caused by radiation. Empirical data have been collected from which correction curves have been developed. However, many of the factors contributing to the change in tag concentrations cannot be quantified, so the correction factors tend to become more and more imprecise as burnup increases.

Second, without the ability to continuously monitor the residual concentrations of xenon-tag isotopes in the cover gas, it is very difficult to arrive at a specific tag ratio. At present, varying concentrations of residual tag are assumed on the basis of experiment, and new tag analyses are corrected to account for the residual tag gas. The current success with the CGCS tag-trap system (see Sec. VII.D) has shown that this problem can be eliminated by continuous operation of the tag-trap system, because the current xenon-tag-isotope inventory is always known.

B. Failed-fuel Detection Systems

The failed-fuel identification systems now in use at EBR-II consist of the FERD, the germanium-lithium argon-scanning systems (GLASS I and II), and the DN detector of the breached-fuel test facility (BFTF). Another DN detector is in the fuel-performance test facility (FPTF), which was installed in June 1981.

The GLASS systems consist of GE(Li) detectors coupled to multi-channel pulse-height analyzers. Isotopes monitored by the GLASS are ^{85m}Kr, ^{87}Kr, ^{88}Kr, ^{133}Xe, ^{135}Xe, ^{135m}Xe, and ^{138}Xe. Because of the excellent performance of these units, the old fission-gas monitor (FGM) and reactor cover-gas monitor (RCGM) have been retired.

The FERD system has four detection channels. Two are for use by the Operations Department; and other two are experimental and can be used as desired by the Analysis and Engineering Departments. Both operations channels and one experimental channel use seven BF_3 detectors per channel. The fourth channel now uses six ^{10}B-lined detectors. The detectors are arranged in two concentric circles, are parallel to the sodium-sample piping, and are housed in a Benelex moderator.

The present FERD system was installed in March 1978. Its newer electronics have improved gamma-discrimination capability, and the BF_3 detectors, with their new geometry, have proved much less subject to the "gamma aging" that occurred with the older system. Operating experience with the new system has been excellent.

The DN detectors of the BFTF and the FPTF are pulsed-fission counters in the upper part of the test facilities. The BFTF is designed to channel the hot sodium effluent from the subassembly beneath it through an instrumented sample probe and then return the sample flow to the reactor outlet plenum. The FPTF is similar to the BFTF except that the sodium effluent is channeled into the bulk sodium instead of to the reactor outlet plenum. The FPTF also has a programmable flow-control valve that can be used to periodically cause temperature transients on the fueled experiment beneath the facility. These facilities provide the capability to discretely monitor test subassemblies for emission of DN precursors while the rest of the core is being monitored independently with the FERD system. Cross-calibration between the DN detectors of the FERD, BFTF, and FPTF, combined with appropriate signal conditioning, allows operation with three DN-emitting subassemblies in the core.

IX. SURVEILLANCE OF REACTOR SUBASSEMBLIES

Two principal phenomena cause concern for the performance of in-core structural components: irradiation-induced swelling and irradiation-enhanced creep. These phenomena, acting independently or together, can produce permanent changes in the configuration of reactor components. In-core components such as fuel cladding, spacer wire around the cladding, subassembly internals, subassembly ducts, and near-core components such as the reactor-grid-plenum assembly are subject to both swelling and creep. Nevertheless, a review[23] summarizing EBR-II surveillance experience with a wide range of components concluded that no significant reliability or safety problems have arisen at EBR-II as a result of creep or swelling of in-core steel components.

To provide advance warning of possible difficulties because of irradiation swelling, a thorough surveillance program was implemented several years ago. This ongoing program includes theoretical modeling and compares the models against the available surveillance information. New techniques have been developed to acquire the required surveillance information. This section summarizes our surveillance experience with only subassembly ducts and is a typical example of the thorough surveillance program used at EBR-II to monitor the irradiation swelling of in-core stainless steel components.

Irradiation swelling can act in three ways on subassembly ducts. It can cause: (1) length increases of subassemblies because of its integrated effect over the length of the core; (2) bowing of subassemblies because of gradients in swelling across the ducts caused by gradients in temperature and flux, and (3) dilation of ducts in the core region.

Interactions between neighboring subassemblies may cause irradiation creep that lessens the adverse effects of swelling. These adverse effects can include reduced reactivity because of core expansion or geometry change, increased difficulty in fuel handling because of misalignment of fuel-handling equipment and duct/duct interactions, and displacement of control rods within their guide tubes.

A. Changes in Subassembly Length

Minor changes in the length of individual subassemblies do not pose a problem to the EBR-II fuel-handling equipment. However, there is a potential problem of subassemblies growing in length to the hold-down devices positioned above them in rows 1-5. The clearance between an unirradiated subassembly and its hold-down finger is about 5.1-6.3 mm.[24] If the subassembly should contact its hold-down, there is danger of deforming the subassembly top end fixture, which is grappled by the fuel-handling equipment.

Because of this possibility, the lengths of various high-fluence subassemblies were measured in the reactor, and the measured lengths were compared with lengths calculated as a function of neutron exposure. The agreement between calculated and measured values was good up to the point of contact between hold-down finger and top end fixture. The measurements indicated that some subassemblies had contacted the hold-down fingers; however, deformation of the top end fixture was limited, and the subassemblies were handled routinely. As a result of this investigation, the top end fixtures of EBR-II subassemblies were redesigned, and their length was reduced by about 4.8 mm to eliminate the problem. No further difficulties as a result of subassembly lengthening are expected under present exposure guidelines.

B. Subassembly Bow

Bow is not a primary concern for subassemblies in the EBR-II core (rows 1-7). There are two principal reasons for this: the temperature and flux profiles across the subassemblies in these rows are relatively flat, and the residence times of subassemblies in the core are relatively short. Subassemblies outside the core region (reflector and depleted-uranium blanket subassemblies) are subject to bow because they are in regions of large flux gradients and are seldom moved to new orientations. Bowing of reflector subassemblies was found to be a problem in EBR-II.[23,25-29] The observations concerning bow of those subassemblies and the corrective actions taken are summarized here.

Figure 11 plots the average force required for handling subassemblies in core row 7 of EBR-II for various runs in November 1975 through April 1979. Row 7 is the last row of the reactor core region and is bordered by the first row of the outer-blanket region. Rows 8-10 of the outer-blanket region make up the reflector and contain the stainless-steel reflector subassemblies. The remaining rows (11-15) comprise the breeder blanket and contain depleted-uranium subassemblies. Going into run 87, the fuel-handling forces were increasing rapidly. Because of the high fluence accumulations on row-8 reflector subassemblies (approaching 9×10^{26} n/m^2, all energies), it was suspected that bowing of those subassemblies might be causing the increasing fuel-handling forces in row 7. To meet the problem, two things were done simultaneously: (1) the row-8 reflector subassemblies with the highest fluence were replaced with new subassemblies, and (2) some row-9 and row-10 reflector subassemblies were rotated 180° to study the effectiveness of rotation in reducing the fuel-handling forces for bowed subassemblies. Force measurements showed that rotation reduced fuel-handling forces significantly.[30] These forces remained low through run 94. Figure 11 also shows that the replacement of row-8 reflector subassemblies by new ones reduced the fuel-handling forces in row 7.

Figures 12 and 13 are typical plots of measured physical profiles of a reflector subassembly. Figure 12 is a plot of bow, or deviation from a true vertical reference line, versus the elevation of the measurement. For orientation, the location of the EBR-II core is shown. The negative bow (deflection) observed in the core region is the result of irradiation-creep-induced relaxation of the stresses that occur as the bowing subassembly interacts at its top with its neighbors.

Figure 13 is a polar projection of the data of Fig. 12. It shows the direction of bow with respect to the core centerline of the reactor. The maximum bow measured for all reflector subassemblies has been directed away from the core centerline, which is consistent with irradiation-swelling-induced bow. Figure 14 summarizes the data obtained for all row-8 reflector subassemblies through April 1979.

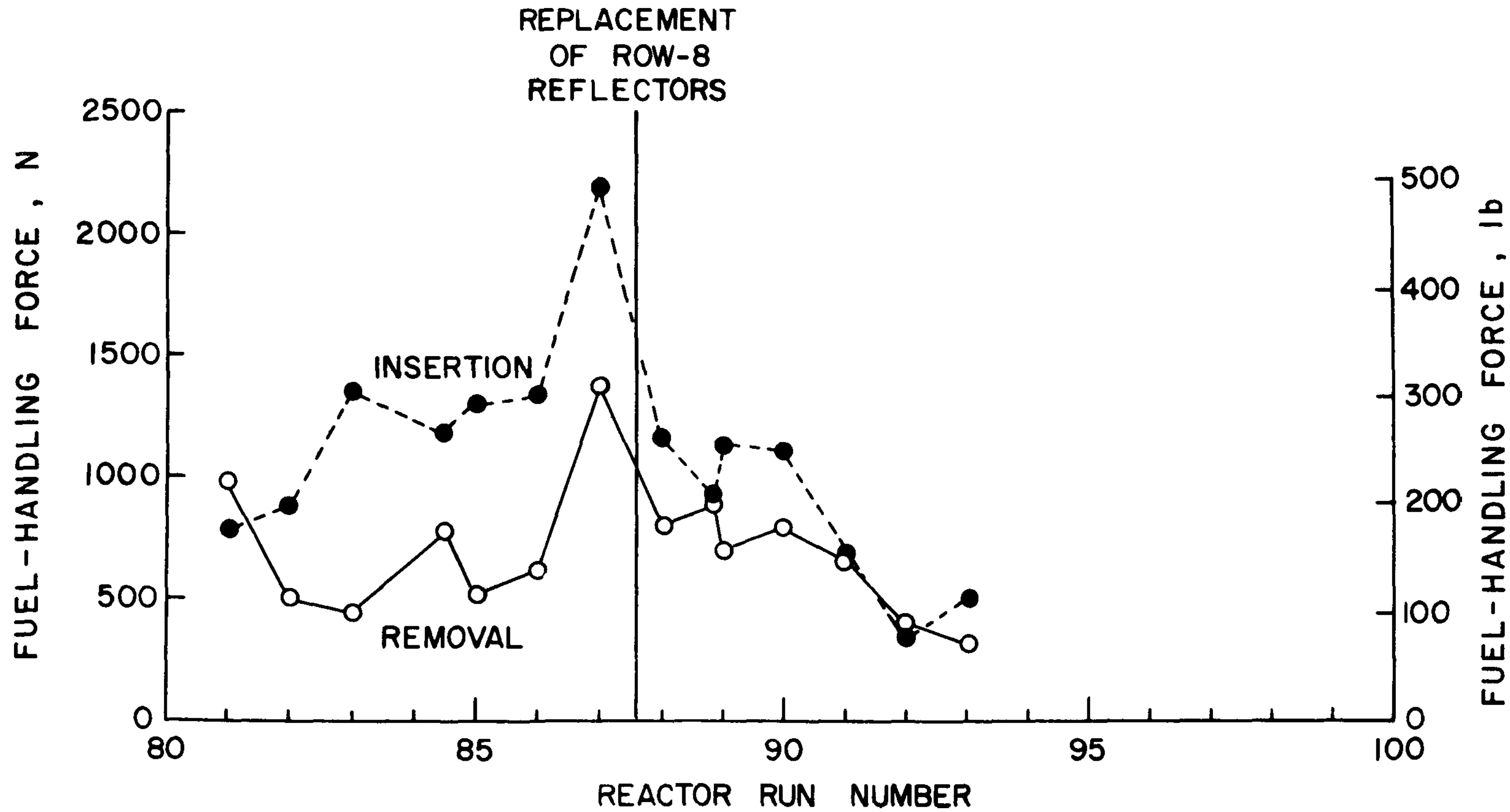

Fig. 11. Fuel-handling Forces in EBR-II Core Row 7 as a Function of Reactor Run: November 1975-April 1979

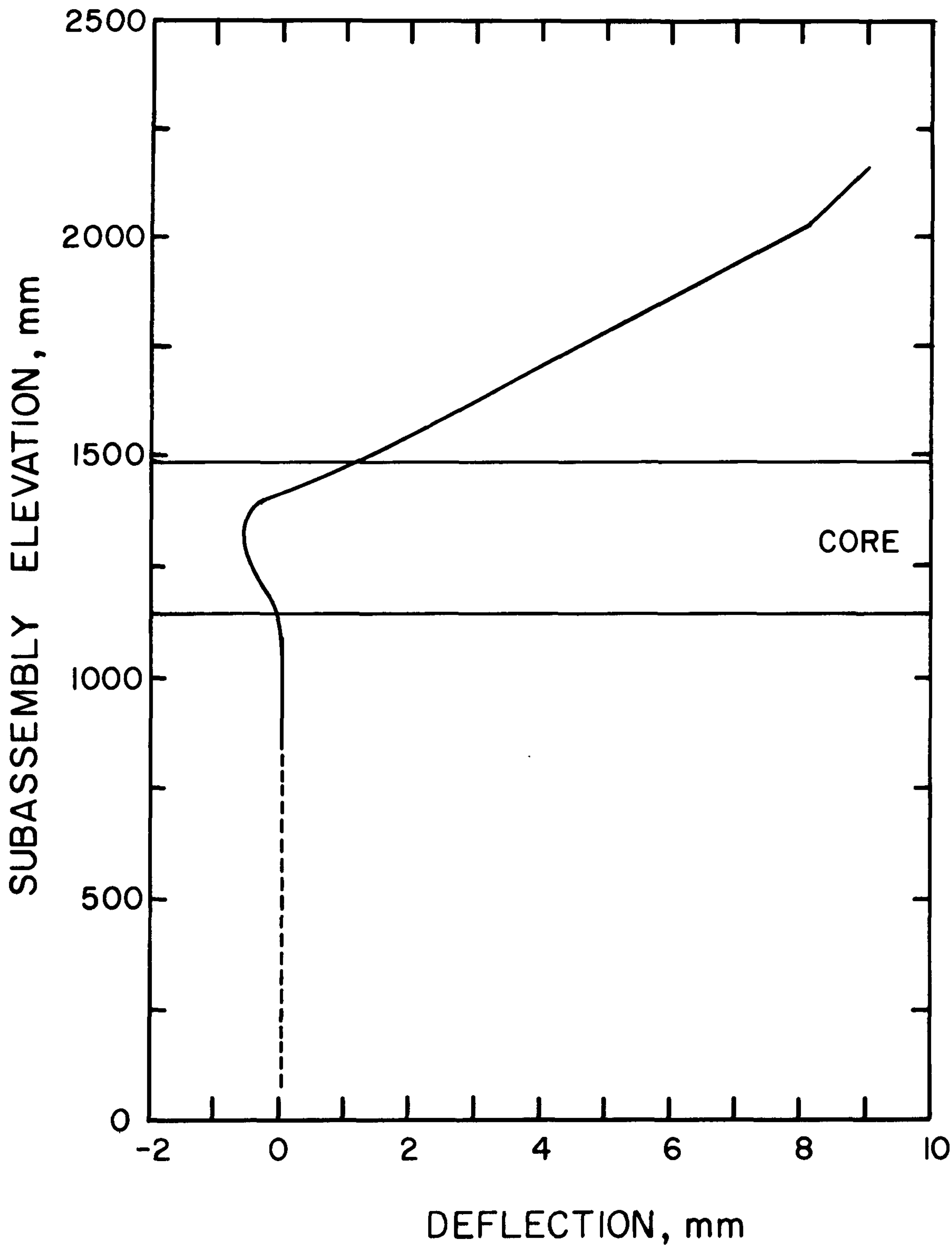

Fig. 12. Subassembly Bow as a Function of Elevation for Row-8 Reflector Subassembly U-8804D after Accumulating 10.8 x 10^{26} n/m^2 (all energies)

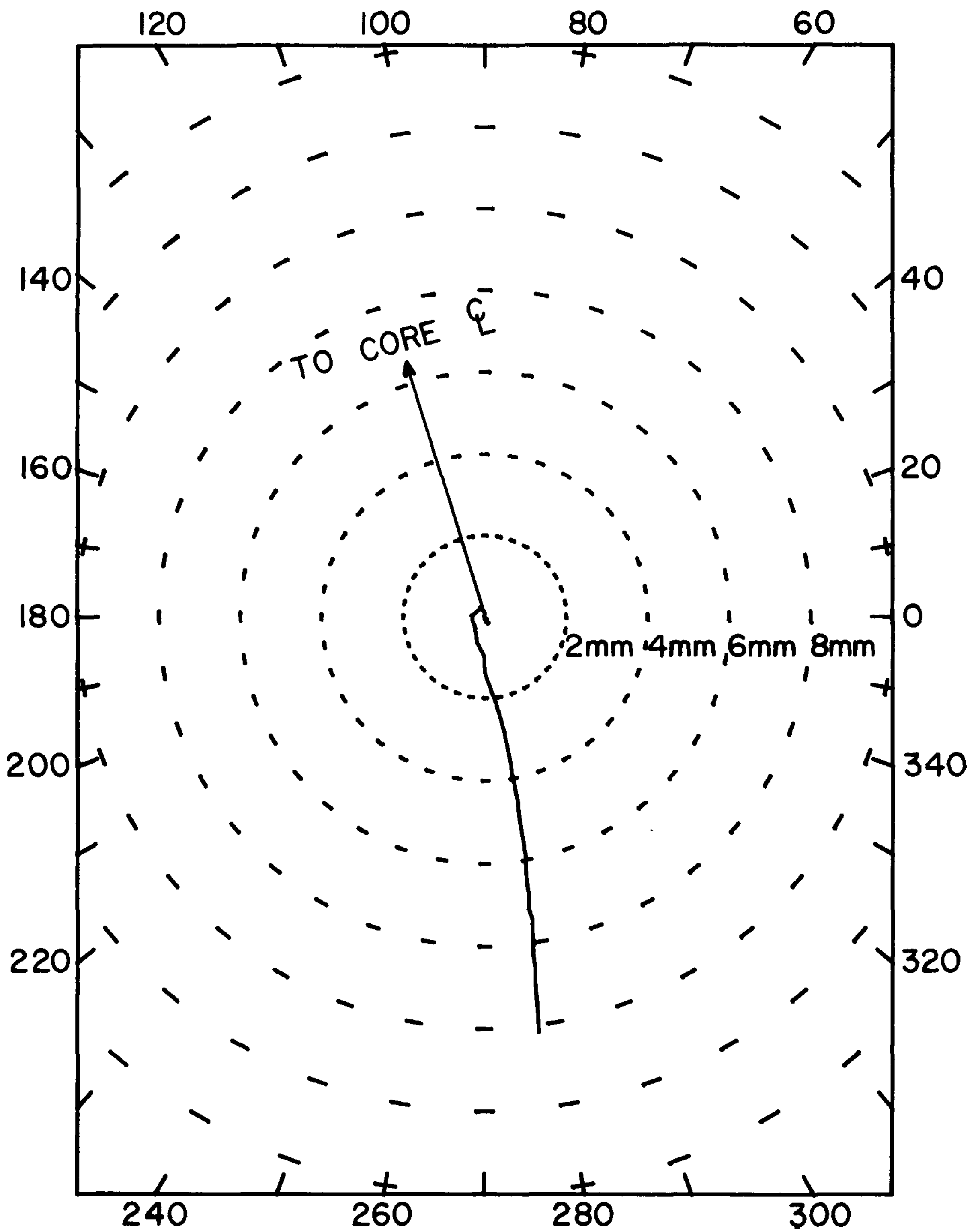

Fig. 13. Polar Projection of Bow Measured on U-8804D. Angles are referenced to subassembly index notch. Bow oriented away from core centerline is consistent with differential-swelling-induced bow.

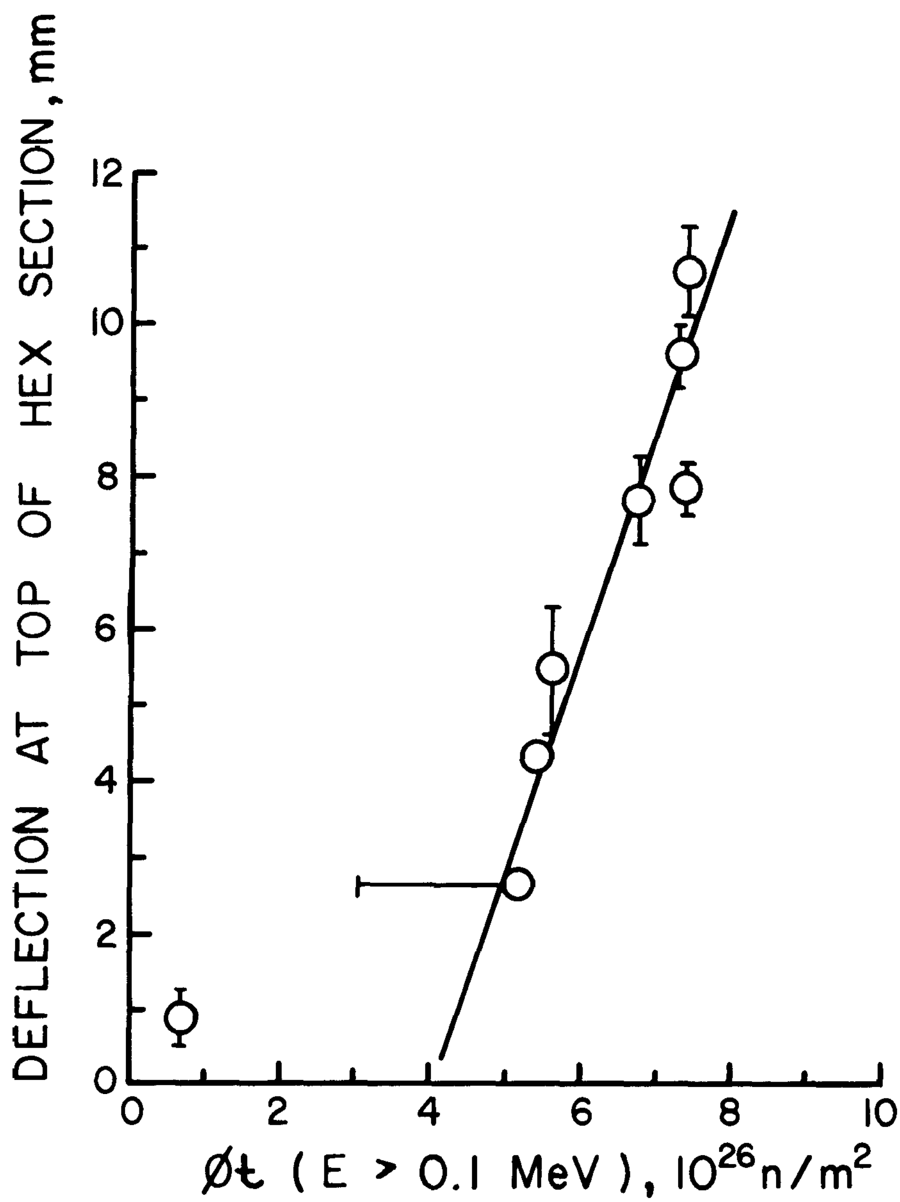

Fig. 14. Bow of Row-8 Reflectors as a Function of Neutron Exposure. All bow values are the average of three or six measurements. Bow error bars indicate one standard deviation; fluence error bar indicates fluence at which subassembly changed position in the core.

Bowing of reflector subassemblies again became a problem at the end of run 102 (August 1979). At that time, the fuel-handling forces for the reflector subassemblies in row 10 started to increase. Investigation indicated that bowing in the row was the problem. Deflections as large as 18.5 mm were encountered at fluences of 9.7 x 10^{26} n/m^2 (E>0.1 MeV). Again the maximum bow was directed away from core centerline, which is consistent with irradiation-swelling-induced bow.

The flux gradient across row-10 subassemblies is expected to be large because of the buildup of ^{239}Pu in the row-11 blanket subassemblies. To verify this flux gradient and the differential swelling of the row-10 subassemblies, immersion-density samples were taken from all six flats of a subassembly.

The results indicate up to 3% difference in the swelling between opposite flats. Calculations indicate that swelling is the cause for most of the bow observed.

As a result of these and other measurements, guidelines were established in August 1979 for subassembly rotation and changes to allow the subassemblies to reach their full irradiation lifetime based on dilation.

C. Dilation of Subassembly Ducts

Dilation of subassembly ducts can also cause problems if the ducts are allowed to expand to the point where they cannot be removed from the core. Extensive investigations into the swelling properties of Types 304L and 304 stainless steel[31] have resulted in the establishment of exposure guidelines based on dilation of subassembly ducts. These guidelines are based on the concept that ducts are free enough to move in the core so that dilations greater than the intersubassembly clearance can be accommodated.

Figure 15 shows the seven-subassembly cluster concept used for evaluating duct dilation. With the use of conservative approximations for core-midplane duct temperatures (413°C for structural subassemblies, 441°C for fueled subassemblies), the dilation of each duct in the core is calculated on a run-by-run basis. Each duct is then evaluated as the central duct in a cluster such as shown in Fig. 15. If the sum of dilations

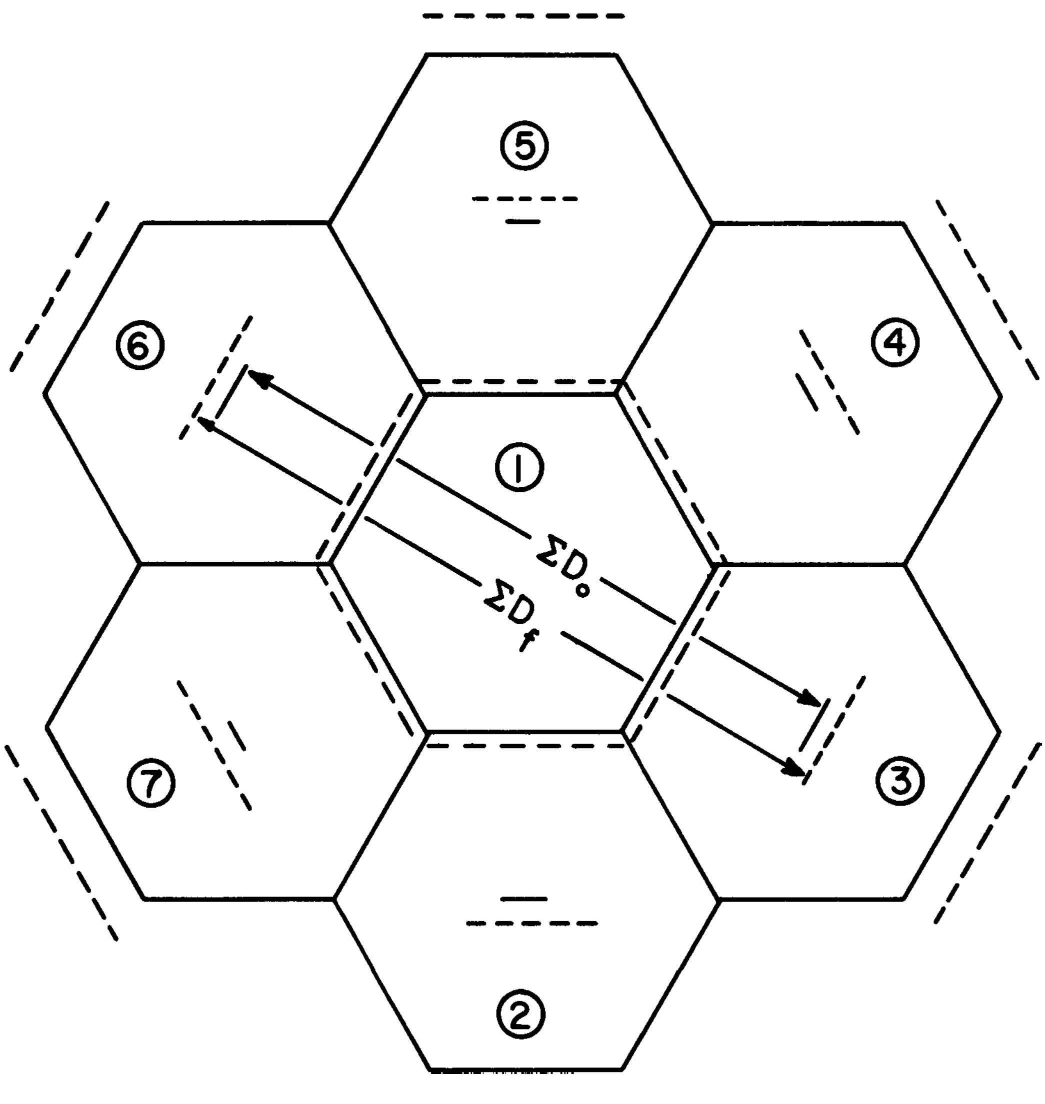

Fig. 15. Seven-subassembly Cluster Used in EBR-II Exposure Guidelines. Solid lines indicate positions of unirradiated ducts: dashed lines indicate positions of ducts after irradiation-induced dilation.

across any set of flats exceeds the sum of intersubassembly clearances, the ducts in that particular cluster are scheduled for rearrangement or removal after the run in question. Using the notation of Fig. 15, the exposure criterion is

$$D_f - D_o \geq 1.52 \text{ mm},$$

where

D_o = center-to-center spacing across flats of the cluster before irradiation,

D_f = center-to-center spacing across flats of the cluster after irradiation,

and

1.52 mm = twice the intersubassembly clearance of 0.76 mm before irradiation.

The change in spacing is also given as

$$D_f - D_o = \Delta D_1 + 0.5(\Delta D_3 + \Delta D_6),$$

where

ΔD_1 = diameter change of central ducts

and

ΔD_3 and ΔD_6 represent the diameter changes of neighboring subassemblies on opposite flats of the central subassembly.

Two additional guidelines on dilation are used. The maximum dilation allowed for any subassembly is 1.02 mm, because of possible problems with insertion and removal of the subassembly in the storage basket. Control- and safety-rod thimbles, because they are transferred directly out of the reactor without storage, have a dilation guideline of 1.14 mm.

To date, no difficulties have been experienced in handling subassemblies in the core that can be traced to subassembly dilation. Figure 16 shows data for a structural dummy subassembly, K005, which was very difficult to insert into the storage basket. The measured dilation was substantially larger than that which could be easily accommodated by the basket, as shown in the figure. Even in this case, the flexing of the hex duct allowed insertion and removal.

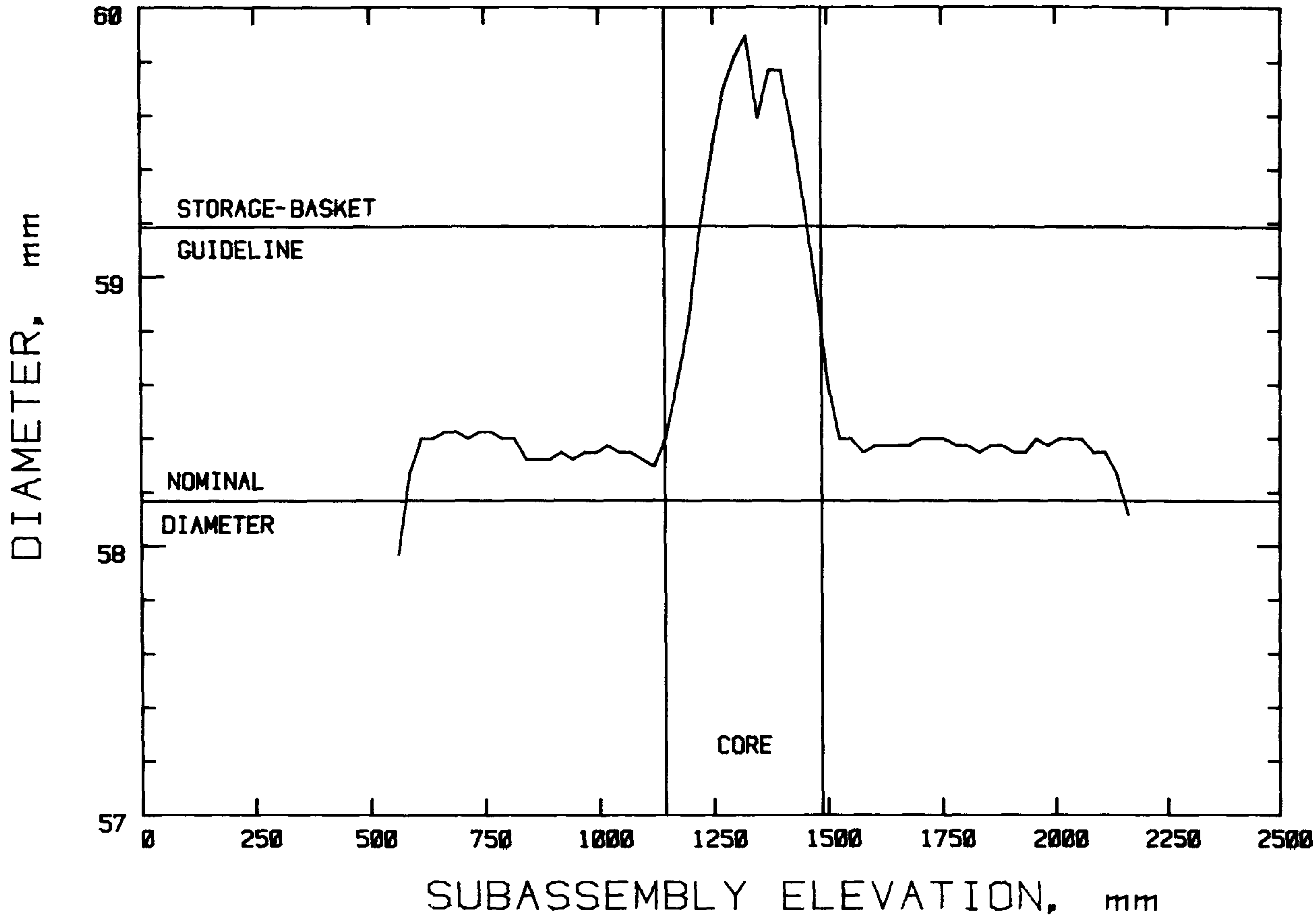

Fig. 16. Typical Across-the-flats Profile of Core Dummy Subassembly K005, which Accumulated a Fluence of 9.2 x 10^{26} n/m^2 (all energies). Nominal initial diameter and storage-basket interference limit are shown for reference.

X. NONDESTRUCTIVE INSPECTION OF EVAPORATORS

The EBR-II steam-generator system consists of a conventional steam drum, seven shell-and-tube recirculating evaporators, and two shell-and-tube superheaters. A program for inspecting the evaporator tubes was undertaken to establish current evaporator condition and to ensure the continued reliable operation of the EBR-II steam system.

The evaporator tubes inspected were fabricated from 2.25 Cr-1 Mo steel. These are duplex tubes: that is, a tube within a tube, bonded together with a nickel braze. The duplex tubes are 8.2 m long and have a bore diameter of 27 mm. The wall thickness of each individual tube is 2.4 mm, so the total wall thickness of the duplex is 4.8 mm. Figure 17 is a sketch showing the steam-outlet region, the tube sheets, and the duplex tubing in an evaporator. The duplex-tube design and the inspection-access area are visible in the figure.

The evaporator tubes were inspected ultrasonically. The time available for each inspection was limited, which dictated the use of an automatic method of handling the transducers. A mechanical drive was used to rotate a probe connected to it by a flexible metal hose. The mechanical drive rotated the probe at 30 rpm while it was being withdrawn at a rate of 0.25 mm per rotation. This procedure provided an overlapping helical path for a complete volumetric inspection.

For the 1976 inspection, an inspection probe was designed with three transducers, one at the top of the probe, one at the bottom, and one in the middle. The top and bottom transducers were potted in the probe body so as to produce 45° shear waves in the tubing. These two transducers were 6.4 mm in diameter and were fabricated from lead metaniobate. They operated at 2.25 MHz, at a repetition rate of 5000 pulses/s, and were positioned to transmit a wave clockwise and counterclockwise. The transducer in the center produced a longitudinal wave used to measure wall thickness and the quality of the braze between the tubes. This transducer was also lead metaniobate, but operated at 7.5 MHz.

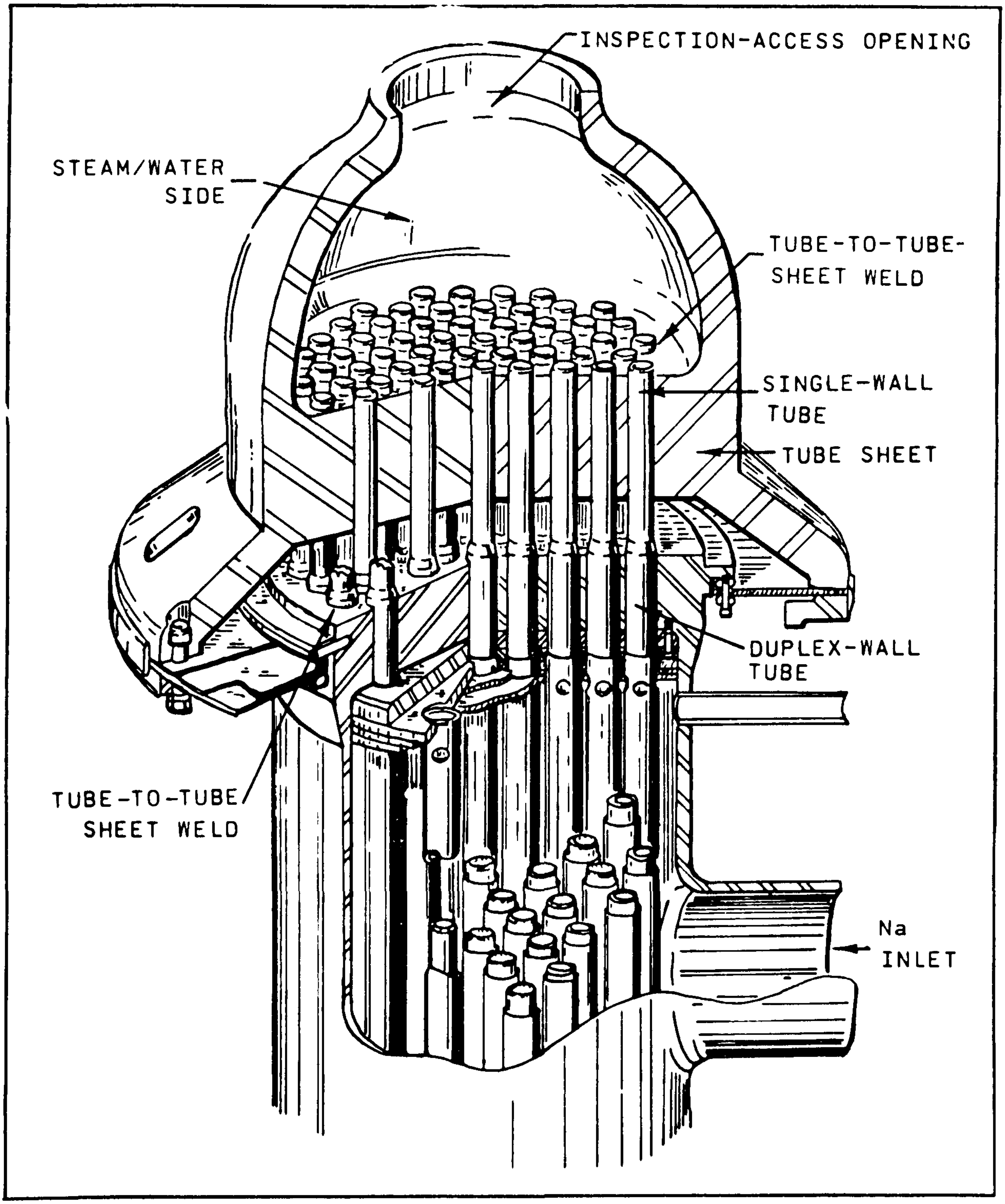

Fig. 17. EBR-II Evaporator

To check, evaluate, and calibrate this inspection system, a standard was fabricated. The material for the standard was obtained from original samples of tubing material that exhibited a good nickel-braze condition. Notches, to represent flaws, were machined by electrical discharge on the outside surface. These notches ranged from 1.6 mm long and 0.25 mm deep to 6.4 mm long and 0.76 mm deep. The notches were also angled at 22.5, 45, and 90° to the longitudinal axis of the tube. Three thickness steps were machined on the outer surface of the standard. Each step was 0.25 mm deeper than the previous step. The first step represented a wall thinning of 0.25 mm, the second step 0.50 mm, and the third step 0.76 mm from the original diameter.

Three ultrasonic inspections have been completed to date. Evaporator 702 was inspected in 1976 and 1978, and evaporator 706 in 1980. During these inspections, many tubes were visually examined with a fiber-optic borescope to an axial length of 3.3 m, the maximum length of the borescope. A small television camera was obtained in 1979, and the entire length of the tubes was examined with it during the 1980 inspection of evaporator 706. This examination revealed the presence of tightly adherent, dark-rust-colored corrosion deposits, minor amounts of scale deposits, minor surface pitting, and some of the original manufacturing draw-marks. Analysis of the corrosion product revealed magnetite (Fe_3O_4) and hematite (Fe_2O_3). Generally, the bore of the tubes appeared to be in a satisfactory condition.

Before the ultrasonic inspection of the tubes, an air-driven rotary wire brush was used to clean the tubes. In 1980, chemical cleaning was performed on evaporator 706. The corrosion-scale deposit scatters the sound beam, causing some ultrasonic noise. The wire brushing did not remove the deposit and only smoothed some of the surfaces. Chemical cleaning did not completely remove all the deposit either.

More than 200 tubes were examined for longitudinal (lengthwise) ultrasonic indications during the three inspections. For these examinations, the instrumentation was calibrated so that a 6.35-mm-long and 0.69-mm-deep notch represented 80% of full scale. Therefore, 30% of scale would indicate a depth of approximately 0.26 mm. During these three inspections of the more than 200 tubes, only 10 longitudinal indications were found with depths as deep as 0.26 mm.

Several tubes were found to contain inside-surface indications that were traced to manufacturing draw-marks. These draw-marks were less than 0.88 mm deep, with some running the full length of the tube. No other large indications were recorded.

For the 1978 examination and the subsequent examination in 1980, a new inspection probe was designed and procured that detected circumferential flaws within the tube by use of shear waves. The probe contained two transducers of the same material and frequency as the probe used for longitudinal flaw detection. Two circumferential notches were cut into the standard for calibration purposes. These notches were both 6.35 mm long; one was 0.51 mm deep, and the other was 0.76 mm deep. Eleven tubes were scanned for circumferential defects, and as expected, none was found. The tube-to-tube-sheet welds were easily detectable.

Visual and ultrasonic inspection of two evaporators did not reveal any anomalies. The units are in satisfactory condition.

XI. NONDESTRUCTIVE-INSPECTION PLAN FOR THE SECONDARY AND STEAM SYSTEMS

From construction to March 1976, no formal plan for the periodic inspection of pipe welds, hangers, and other components existed other than preventive-maintenance inspections. In 1976, a formal plant nondestructive-inspection plan was completed.

This plan provides for periodic inspections to ensure the operational readiness and mechanical integrity of the secondary-sodium and high-pressure-steam (8.6-MPa) systems. The plan classifies equipment with respect to the effect of component failure on plant operation and concentrates on inspection of the most critical components. It is also based on considerations such as physical accessibility for repair and maintenance, operating conditions, predicted component stress, state-of-the-art nondestructive-examination methods, and reasonable limits on available resources.

The EBR-II plant was constructed to the ASME "American Standard Code for Pressure Code for Pressure Piping (ASA B31.1)." This code was in existence during construction of the plant.

Five years of inspection from 1976 to 1981 have not revealed any defects caused by operation. All areas inspected were in satisfactory condition. Transition pipe welds (joining pipes of different material) are smooth and show no discontinuities. Generally, these welds were shop-made. Welds in stainless steel pipe (sodium pipe) have a smooth, high crown with little or no undercutting, generally full fusion, and minor drop-through. The welds in carbon steel pipe (steam pipe) are rough-weave and have a high crown with minor undercutting adjacent to the weld. However, there is sometimes a drop-through and/or lack of fusion in the root area of these welds.

REFERENCES

1. L. J. Koch et al., *Hazard Summary Report, Experimental Breeder Reactor II (EBR-II)*, ANL-5719, Argonne National Laboratory (May 1957).
2. L. J. Koch, W. B. Loewenstein, and H. O. Monson, *Addendum to Hazard Summary Report, Experimental Breeder Reactor II (EBR-II)*, ANL-5719 (Addendum), Argonne National Laboratory (June 1962).
3. J. K. Long et al., *EBR-II Dry Critical Experiments. Experimental Results*, ANL-6462, Argonne National Laboratory (February 1962).
4. F. S. Kirn and W. B. Loewenstein, *EBR-II Wet Critical Experiments*, ANL-6864, Argonne National Laboratory (October 1964).
5. J. R. Davis et al., *Operating Experience with Sodium Pumps at EBR-II*, ANL/EBR-027, Argonne National Laboratory (October 1970).
6. M. Novick, F. D. McGinnis, and G. K. Whitham, *EBR-I and EBR-II Operating Experience*, Proc. Am. Nucl. Soc. National Topical Meeting on Fast Reactors, Detroit, April 26-28, 1965, ANS-100, pp. 30-40 (1965).
7. B. C. Cerutti et al., *Removal and Repair of EBR-II Primary Sodium Pump No. 1*, ANL-7835, Argonne National Laboratory (May 1972).
8. H. W. Buschman, B. C. Cerutti, and A. F. Clark, *Noise Investigation and Repair of the EBR-II Intermediate Heat Exchanger*, ANL-7834, Argonne National Laboratory (August 1971).
9. B. C. Cerutti and J. B. Waldo, *Ten Years of Operating Experience with Mechanical Components of the EBR-II Fuel-handling System (1963-1972)*, ANL-7836, Argonne National Laboratory (December 1973).
10. L. P. Cooper, B. C. Cerutti, and D. W. Cissel, *Operating Experience with the EBR-II Rotating-plug Freeze Seals*, ANL-7617, Argonne National Laboratory (December 1969).
11. A. F. Clark, R. G. Nobles, and H. W. Buschman, *Operating Experience with the EBR-II Rotating-plug Freeze Seals: 1969-1975*, ANL-75-10, Argonne National Laboratory (March 1976).
12. W. E. Ruther and S. Greenberg, *Investigations of Material Compatibility Relevant to the EBR-II System--FY 1973*, ANL-8042, Argonne National Laboratory (October 1973).
13. J. B. Waldo and L. J. Christensen, *System for Measuring Sodium Level in EBR-II*, ANL-7623, Argonne National Laboratory (October 1969).

14. J. T. Holmes, C. R. F. Smith, M. M. Osterhout, and W. H. Olson, *Sodium Purification by Cold Trapping at the Experimental Breeder Reactor-II,* Nucl. Technol. *32*(3), pp. 304-314 (March 1977).
15. W. H. Olson, *Sampling and Analysis of EBR-II Sodium,* ANL-7844, Argonne National Laboratory (May 1971).
16. C. R. F. Smith and J. T. Holmes, *Purity of EBR-II Sodium, 1967-1974,* ANL-75-26, Argonne National Laboratory (December 1975).
17. L. R. Monson et al., *The EBR-II Cover-gas Cleanup System,* Proc. Int. Symposium on Design, Construction and Operating Experience of Demonstration Liquid Metal Fast Breeder Reactors, Bologna, April 10-14, 1978, pp. 465-475, IAEA, Vienna (1978).
18. Marilyn M. Osterhout, *Operational Experience with EBR-II Water-to-Sodium Leak Detection System,* Trans. Am. Nucl. Soc. *27*, p. 253 (1977).
19. Marilyn M. Osterhout, *Operation of the Water-to-Sodium Leak Detection System at the Experimental Breeder Reactor II,* Nucl. Technol. *40*(2), pp. 159-169 (September 1978).
20. M. M. Wrightson, K. McKinley, and J. T. Holmes, *EBR-II Water-to-Sodium Leak Detection System,* Proc. Int. Conf. on Liquid Metal Technology in Energy Production, Champion, Pa., May 3-6, 1976, CONF-760503-P2, pp. 508-514 (November 1976).
21. J. D. B. Lambert, F. S. Kirn, and E. R. Ebersole, *Experience with Breached Elements at EBR-II,* Trans. Am. Nucl. Soc. *27*, p. 820 (1977).
22. G. S. Brunson, *Monitoring Fission Gas in EBR-II by High-resolution Gamma Spectrometry,* Nucl. Technol. *25*(3), pp. 553-571 (March 1975).
23. L. C. Walters, *Creep and Swelling in EBR-II Reactor Components,* Trans. Am. Nucl. Soc. *28*, p. 140 (1978).
24. *Reactor Development Program Progress Report, July-August 1975,* ANL-RDP-42, p. 1.26, Argonne National Laboratory (Oct. 7, 1975).
25. *Reactor Development Program Progress Report, May-June 1977*, ANL-RDP-61, p. 1.27, Argonne National Laboratory (Aug. 11, 1977).
26. *Reactor Development Program Progress Report, July-August 1977,* ANL-RDP-62, p. 1.14, Argonne National Laboratory (Oct. 15, 1977).
27. *Reactor Development Program Progress Report, March 1978,* ANL-RDP-69, p. 1.9, Argonne National Laboratory (May 12, 1978).

28. J. A. Shields, *Irradiation-swelling-induced Bow in Fast Reactor Subassemblies,* Trans. Am. Nucl. Soc. *30*, p. 154 (1978).
29. J. E. Flinn and T. A. Kenfield, *Neutron Swelling Observations on Austenitic Stainless Steels Irradiated in EBR-II,* Proc. Workshop on Correlation of Neutron and Charged Particle Damage, Oak Ridge National Lab., June 8-10, 1976, CONF-760673, pp. 253-289, USERDA (1976).
30. *Reactor Development Program Progress Report, March 1977,* ANL-RDP-59, p. 1.15, Argonne National Laboratory (May 20, 1977).
31. J. E. Flinn, R. A. Weiner, and G. L. Hofman, *Fast-neutron Swelling of Type 304 Stainless Steels,* Trans. Am. Nucl. Soc. *28,* p. 224 (1978).