PROPOSAL FOR OPERATING EBR-II AT 62.5 MWt

Coordinated by

R. A. Cushman

EBR-II Project

Argonne National Laboratory

Argonne, Illinois - Idaho Falls, Idaho

August 1969

Work performed under the auspices of the U. S. Atomic Energy Commission

—LEGAL NOTICE—

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT	9
I. INTRODUCTION	9
II. REASONS FOR INCREASING POWER	10
III. PROCEDURE FOR ATTAINING 62.5 MWt	11
IV. CONDITIONS AT 62.5 MWt	14
A. Reactor Conditions	15
B. Plant Conditions	30
V. REACTOR SAFETY ANALYSIS	32
A. Reactor Temperatures	32
B. Reactor Stability	32
C. Power-Reactivity Decrement	43
VI. PLANT PERFORMANCE CAPABILITY	43
A. Primary System	43
1. Reactor Vessel	44
2. Intermediate Heat Exchanger	45
3. Primary-System Piping	47
4. Removal of Decay Heat	47
B. Secondary Sodium System	48
1. Secondary-sodium EM Pump	48
2. Evaporators and Superheaters	50
C. Steam System	50
1. Main Turbine	50
2. Condenser	52
3. Condensate Pumps	55
4. Feedwater Pumps	56
D. Cooling Tower	57
E. Electrical System	58
VII. FUEL PERFORMANCE CAPABILITY	60
VIII. OPERATING PROCEDURES AT 62.5 MWt	63
A. Revision of Existing Procedures	63
B. System and Component Surveillance	64
IX. CONCLUSIONS	65
ACKNOWLEDGMENTS	66
REFERENCES	67

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
1.	Typical Core Loading in Initial 62.5-MWt Operation	13
2.	Typical Core Loading in Extended Operation at 62.5 MWt	16
3.	Calculated Relative Radial Distribution of Two Projected 62.5-MWt Cores and Measured Distribution of Run-29D Core	18
4.	Maximum Temperatures at 62.5 MWt for Rows 1-6 in a 91-sub- assembly Core without Fueled Experiments	20
5.	Maximum Temperature Distribution at 62.5 MWt in Hottest Channel in Row 5 of a 91-subassembly Core without Fueled Experiments	21
6.	Maximum Temperatures at 62.5 MWt for Rows 1-6 in a 91- subassembly Core with Fueled Experiments	22
7.	Axial Temperature Distribution at 62.5 MWt in Hottest Channel in Row 5 of a 91-subassembly Core with Fueled Experiments	23
8.	Run-29C 50-MWt Rod-drop Power and Reactivity Behavior	36
9.	Typical Drop-rod Calibration Curve	38
10.	Comparison of Extrapolated and Measured Feedback for Run 29C	39
11.	Predicted Feedback at 50 MWt, Based on Run-35 Data	40
12.	Predicted Feedback at 56 MWt, Based on Run-35 Data	41
13.	Predicted Feedback at 62.5 MWt, Based on Run-35 Data	42
14.	Maximum Fuel-centerline and Coolant Temperatures	61
15.	Swelling of 70%-enriched EBR-II Fuel and of 52%-enriched EBR-II Fuel from Controlled-flow Subassemblies, as a Function of Burnup	62

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page</u>
I.	Power Distributions per Standard Driver-fuel Subassembly for Various Cores at 62.5 MWt	19
II.	Maximum Temperature at Various Fuel-element Locations in Several Cores, as Calculated without and with Uncertainty Factors	25
III.	Disposition of Experimental Subassemblies during Initial 62.5-MWt Operation	27
IV.	Effect of Power Increase on Typical Experimental Loading . .	28
V.	Heat Balances at Various Reactor Powers	31
VI.	Parameters of Plant Components at Various Reactor Powers . .	33
VII.	Row-7 and Row-8 Temperatures at 50 and 62.5 MWt	35

PROPOSAL FOR OPERATING EBR-II AT 62.5 MWt

Coordinated by

R. A. Cushman

ABSTRACT

The benefits of increasing the power level of EBR-II from 50 MWt to 62.5 MWt have been determined. A two-stage plan for increasing the power to 62.5 MWt is presented. The first stage calls for operation at 56 MWt for about two days and then 62.5 MWt for about eight days with the experimental subassemblies removed from the core; the second stage calls for continual operation with experimental subassemblies at 62.5 MWt after six months of operation at 50 MWt, which will give experimenters time to plan for routine 62.5-MWt operation. Conditions in the reactor and plant have been predicted, and no conditions are foreseen that would preclude successful operation at 62.5 MWt.

I. INTRODUCTION

The EBR-II reactor and power plant were designed for operation at 62.5 MWt.^{1,2} Until August 1968, however, the maximum operating power level was 45 MWt. In August 1968, the power level was increased to 50 MWt.

The role of EBR-II has changed since its initial operation in 1964. It has demonstrated, as it was intended to, that a liquid-metal-cooled fast breeder reactor power plant could operate in conjunction with a fuel-processing plant (the EBR-II Fuel Cycle Facility, or FCF). Its role now is to serve the overall LMFBR program as an irradiation facility and as an engineering test bed for future sodium-cooled fast breeder reactors. Its importance cannot be emphasized too strongly. There is no other facility similar to it currently operating in the United States. The authors of this report believe that EBR-II's role in the LMFBR program can be significantly enhanced by increasing the operating power level of the reactor to the design rating of 62.5 MWt.

The original design basis has been reviewed thoroughly, the operating characteristics of the reactor and power plant at the higher power have

been predicted, and the conditions in the core at the higher power have been calculated. This proposal presents the reasons for increasing the power level to 62.5 MWt, the procedure for attaining that power, the expected resultant reactor and plant conditions, and the supporting information to justify the planned increase.*

II. REASONS FOR INCREASING POWER

In addition to achieving the original design power and thereby demonstrating the adequacy of the design of the plant, there are reasons for increasing the EBR-II power level that are of direct importance to the LMFBR program. These reasons are:

(1) A 25% increase in power (from 50 to 62.5 MWt) will produce a corresponding increase in flux level. Structural experiments thus will reach a given nvt in only 80% of the time required at present, or, alternatively, will receive 125% as much nvt in a given time.

(2) An increase in power will permit an increased linear heat rating (kW/ft) in experimental fuel elements. Present prototypal LMFBR elements (e.g., 20% PuO_2 -80% UO_2) of the proper diameter, even when fully enriched in ^{235}U , cannot achieve linear heat ratings above 19 kW/ft. The problem is much more severe with carbide elements, whose allowable rating is considerably higher.

(3) An increase in power will permit a more favorable and realistic ratio of the cladding fluence to fuel burnup. For a given linear heat rate, the fuel enrichment may be reduced, and the cladding thus will experience conditions more typical of expected LMFBR-program demonstration reactors.

(4) With a higher power level, some experiments (fuel and structural) might be relocated to positions farther from the core center. If driver-fuel subassemblies then replace the experimental subassemblies in the central region of the core, there will be a potential for enhanced prompt feedback from the driver subassemblies.

* In September 1969, the power level of EBR-II was raised to 62.5 MWt during one run, in accordance with the first stage of the proposal in this report. An evaluation of the reactor and plant data obtained during that run is being published as ANL/EBR-013.

(5) The relocation of experimental subassemblies to outer core locations also will allow more experimental subassemblies in the core without changing the loss of reactivity now caused by such subassemblies.

All the above advantages cannot be obtained simultaneously, at least to the same degree. But the higher power level will permit more freedom in planning experiments. The advantage of the increased flux, for example, may be exploited during one (or several) reactor run(s), and the other advantages may be exploited during one or more other runs.

Organizations using EBR-II as an irradiation facility have been contacted for their views concerning an increase in the EBR-II power level.³ The responses generally have been favorable. Typical comments have been as follows:⁴

GE....This would have the advantage of more closely matching the fuel swelling (burnup rate) with the total integrated flux accumulation rate and therefore clad swelling rates that are more typical of LMFBR's.

ORNL....We feel that increasing the flux and ungrading (sic) the EBR-II performance are extremely important.

GE-NSP....Your proposed increase in the power is welcomed as this would shorten the irradiation time for a given neutron fluence.

WARD....The power increase will be of considerable benefit to future experiments and we look forward to its early accomplishment.

The increased operating power level also presents some problems. Operating conditions are more severe. Fuel performance is apt to be somewhat degraded. There is an increased probability, which cannot be evaluated, of a failure of some component of the overall system. The system, however, was designed to operate at 62.5 MWt, and is believed to be capable of doing so. The advantages of going to 62.5 MWt are believed to outweigh the disadvantages.

III. PROCEDURE FOR ATTAINING 62.5 MWt

The EBR-II currently is fulfilling its role as an experimental-irradiation facility. Irradiation experiments were designed for operation at 50 MWt and allocated specific locations in the core that were based on that power level. The ultimate goal is routine operation at 62.5 MWt. Abruptly increasing the power level, however, is somewhat difficult.

Therefore, the increase to 62.5 MWt will be achieved in two stages. The first stage will demonstrate that the reactor and power plant will effectively operate at 62.5 MWt. The second stage will be continual operation at the higher power.

Before the first stage, all the present experiments that would be adversely affected by the increase in power where they are located will be removed from the core or relocated. All fueled experiments will be removed from the reactor vessel and placed into the storage basket; structural experiments, with two exceptions, will all be outside the core, in row 7. The two exceptions are structural experiments X021 and X057, which will be moved from row 2 to row 5. Dummy structural experiments and half-worth subassemblies will be used in the core region as required by reactivity considerations to produce a core with 88-91 subassemblies. Figure 1 shows a core-loading diagram similar to that expected for the first 62.5-MWt operation.

The core changes will not affect the portion of the reactor plant outside the core, but the change in power will. Therefore, the first stage should demonstrate that the entire EBR-II system is capable of sustained operation at 62.5 MWt. Although unexpected, some unforeseen difficulty or problem with a plant component may limit the power level to some value below 62.5 MWt. The two-stage approach has the advantage that no redesign or modification of the experiments is required before demonstrating that the plant is capable of 62.5-MWt operation. If the maximum attainable power level should be less than that value, the lower value can be set as the new operating limit and the experiments suitably modified.

After this first-stage successful demonstration of 62.5-MWt capability, the reactor will be shut down and the core layout restored to the 50-MWt experimental loading. The reactor and plant then will be operated according to the regular 50-MWt operating schedule for approximately another six months. At the end of the first stage, the experimenters will be informed that in six months the normal power level will be increased to 62.5 MWt and that experiments should be planned accordingly. The experimental subassemblies that must make the transition from 50 to 62.5 MWt should be modified, if necessary, during this period to accommodate either higher flow in a similar core location or a new core location. At the end of the six-month period, the normal operating power level will be raised to 62.5 MWt.

KEY: B - DEPLETED URANIUM
 C - CONTROL ROD
 D - DRIVER FUEL
 P - 1/2 DRIVER FUEL--1/2 STAINLESS STEEL
 S - SAFETY ROD
 SSCR - STAINLESS STEEL CONTROL ROD
 ◼ - EXPERIMENTAL SUBASSEMBLY (X000--stainless steel dummy)

Fig. 1. Typical Core Loading in Initial 62.5-MWt Operation

The detailed procedure for the first stage, the increase to 62.5 MWt with the core containing no fueled experiments, is as follows:

(1) Load the core in a pattern similar to that shown in Fig. 1 so as to achieve criticality with 88-91 subassemblies and with 0-3 blanket subassemblies in row 6. All subassemblies in the core are expected to be capable of 62.5-MWt operation without ill effect.

(2) Attain 50-MWt power level and operate there for about one day. Reduce power and perform rod drops at 27 MWt and at 50 MWt. Read and record necessary reactor and plant conditions as a base.

(3) Increase power to 56-57 MWt. Perform rod-drop tests. Operate at 56 MWt for about two days. Record and compare measured reactor and plant conditions with predicted conditions.

(4) When the Project is satisfied that performance at 56 MWt is satisfactory, increase power to 62.5 MWt. Perform rod-drop tests. Record and compare measured reactor and plant conditions with predicted conditions. Operate at 62.5 MWt for eight days. Bypass turbine while at 62.5 MWt to determine if the heat-rejection system can handle the entire reactor output.

(5) Return to 50 MWt and repeat the tests conducted in (2).

(6) Shut down the reactor. Restore fueled and structural experiments to their 50-MWt positions. Continue 50-MWt operation as scheduled for six months.

(7) Inform all experimenters that EBR-II will commence operation at 62.5 MWt within six months from date of notification.

This two-stage approach to power is in accordance with the answers received from the experimenters to the question³ "What is the minimum notice you would require from the EBR-II Project before EBR-II is operated continuously with a 25% or more increase in flux level?" Nine experimenters answered with a definite time. The answers ranged from two months to six months, with AI, GE, GE-NSP, INC, and ORNL answering five to six months.

IV. CONDITIONS AT 62.5 MWt

The EBR-II facility was designed for operation at 62.5 MWt as an experimental breeder reactor, not as an irradiation facility. The reactor core has changed considerably from that projected originally.^{1,2} Further, the core changes somewhat from run to run, so that one can not talk of "the"

EBR-II core, but rather of "a" core. The plant apart from the core changes little from run to run. Therefore, the reactor and the remainder of the plant are discussed separately below.

A. Reactor Conditions

As discussed in Section III, the increase in power to 62.5 MWt is proposed as a two-stage process. The first core loading (typically shown in Fig. 1) will be used to demonstrate that the entire facility is capable of attaining 62.5 MWt. This typical loading is similar to that previously used in run 30E, when all fueled experiments were removed from the core while searching for a "leaker." Although the actual first 62.5-MWt loading is expected to be very similar to that shown in Fig. 1, there may be definite variations in locations of half-worth, dummy, blanket, and driver-fuel subassemblies. As the core loading changes, the power produced by a particular subassembly may also change; replacing four driver subassemblies by four half-worth subassemblies, for example, results in increased power production (~3%) by the unchanged subassemblies. Therefore, the values of temperature, flux, fission rates, etc. given in this section are subject to slight change. The conclusions, however, are valid for any core configuration expected to be used.

The second stage of the power increase will involve a core loaded with the normal complement of fuels and structural experiments after they have been modified to accommodate the higher power. Figure 2 shows a typical core loading during this second stage, the continual operation at 62.5 MWt. The core shown in Fig. 2 was used as the basis for the calculations of the extended operation at 62.5 MWt.

Physics calculations for this report were carried out using the DOT 5 code with 6 energy groups on the IBM 360/75 computer. The fission source rate as a function of space was calculated using a two-dimensional (x-y) geometry. Each hexagonal subassembly was replaced by an equivalent rectangular region with the same area and center; the physical composition of the region corresponded to that of the specific type (e.g., driver, control rod, experiment) of subassembly in that core location. Four mesh points per subassembly were used. The result was a subassembly-by-subassembly fission source (power) distribution.

KEY: B - DEPLETED URANIUM
 C - CONTROL ROD
 D - DRIVER FUEL
 P - 1/2 DRIVER FUEL-- 1/2 STAINLESS STEEL
 S - SAFETY ROD
 SSCR - STAINLESS STEEL CONTROL ROD
 ◻ - EXPERIMENTAL SUBASSEMBLY

Fig. 2. Typical Core Loading in Extended Operation at 62.5 MWt

The output of the DOT 5 calculations then was processed for use in the thermal analysis. Radial relative source distributions for driver fuel were derived for the two types of cores shown in Figs. 1 and 2. These derived distributions, shown in Fig. 3, then were used as input to the POWDIST code. This code prepares the detailed power distributions for EBR-II driver-fuel subassemblies, which are used as input to HECTIC. (The experimental source distribution obtained for the run-29D core also is shown, for comparison, in Fig. 3.) The detailed fuel and coolant temperatures for the various channels in the reactor then were obtained using a modified HECTIC program.

Table I shows the results of power-distribution calculations for several cores. Two sets of data (Cases 1A and 1B) are shown for the core with no fueled experiments; two (Cases 2A and 2B) for the typical experimental core; and two (Cases 3A and 3B) for the core presented in ANL-5719 (Addendum).² Each value in Table I represents the power produced in a standard EBR-II Mark-IA driver-fuel subassembly placed in the listed position in the given core. (There may not always be a standard driver-fuel subassembly in a given position; i.e., 5N1 and 5N3 normally contain control-rod subassemblies, and any location could contain an experimental subassembly.) Also shown in Table I is the number of effective subassemblies that constitute the given core, where an effective subassembly is defined as one corresponding to a standard driver subassembly; a safety rod, for instance, is 61/91 of an effective subassembly.

The power distributions for the cores of Cases 1A and 2A were originally calculated on the basis of projected core loadings (similar to Figs. 1 and 2) for the stage-1 and stage-2 power increases. These power distributions then were used for calculating the temperature distributions in the fuel and in the coolant channels throughout the two cores, as reported later in this section and shown in Figs. 4-7. Subsequently, the core loadings shown in Figs. 1 and 2 were determined to be more representative of typical stage-1 and stage-2 cores. These new cores, the Case-1B and -2B cores in Table I, each contained several more effective subassemblies than the earlier respective cores, and the power/subassembly thus was reduced slightly. There was insufficient time to repeat the detailed calculations, but examination of the Case-A and Case-B data presented for each stage demonstrate the fact mentioned earlier: namely, that variations in core composition, even

Fig. 3. Calculated Relative Radial Distribution of Fission Source for Two Projected 62.5-MWt Cores and Measured Distribution of Run-29D Core

TABLE I. Power Distributions per Standard Driver-fuel Subassembly for Various Cores at 62.5 MWt

Core Position	First-stage Core (no fueled experimental subassemblies)		Typical Second-stage Core (with fueled experiments)		ANL-5719 (Addendum) Core	
	Case 1A (basis of Figs. 4 & 6 of this report)	Case 1B (Fig. 1 of this report)	Case 2A (basis of Figs. 5 & 7 of this report)	Case 2B (Fig. 2 of this report)	Case 3A (book value)	Case 3B (computer calculated)
1N1	976	960 ^a	994	986	1177	1147
2N1	955	935	974	967	1146	1114
3N1	905	886	924	917	1090	1014
3N2	922	902	940	933	-----	1047
4N1	816	799	832	825	-----	865
4N2,3	853	835	871	864	965	922
5N1	685	671	694	688	-----	689
5N2,4	740	725	753	747	785	761
5N3	759	744	772	767	-----	786
6N1	526	515	524	520	-----	-----
6N2,5	596	583	589	585	-----	-----
6N3,4	631	618	631	627	615	621
Number of Effective Subassemblies	73.8	76.8	72.5	74.1	62.3	62.3

^aThis value corresponds to a fission rate of 1.17×10^{13} fissions/g-sec ²³⁵U at the core center.

Fig. 4. Maximum Temperatures at 62.5 MWt for Rows 1-6 in a 91-subassembly Core without Fueled Experiments

Fig. 5. Axial Temperature Distribution at 62.5 MWt in Hottest Channel in Row 5 of a 91-subassembly Core without Fueled Experiments (no uncertainty factors)

Fig. 6. Maximum Temperatures at 62.5 MWt for Rows 1-6 in a 91-subassembly Core with Fueled Experiments

Fig. 7. Axial Temperature Distribution at 62.5 MWt in Hottest Channel in Row 5 of a 91-subassembly Core with Fueled Experiments
(no uncertainty factors)

for essentially similar cores, produce slight changes in power distribution. The temperatures throughout the reactor for the Case-1B and Case-2B cores will be less than those reported later for the Case-1A and Case-2A cores. The actual core loadings in the first and second stages of the power increase may be somewhat different from those used for the calculations, but the temperatures will be similar to the calculated values.

The last two columns of Table I present data relevant to the core originally designed for 62.5 MWt. Case 3A gives data as found in ANL-5719 (Addendum).² The last column gives corresponding results, based on the same core loading and fission distribution but calculated in the manner now used, which takes advantage of the POWDIST program. These last two columns show the very real and very significant effect that the increased core size has on power distribution and power density. The increase in the number of effective subassemblies inherent in a 91-subassembly core relative to that analyzed in ANL-5719 (Addendum) ensures that the power density must decrease relative to the original configuration. This is further reassurance that small changes in actual core loadings at 62.5 MWt will not have a deleterious effect on the temperature distribution relative to that originally analyzed.²

Figures 4-7 show the temperature distributions for core loadings for the two stages of the 62.5-MWt operation. Figures 4 and 6 show, for the hottest channels in each of the six rows of the reactor, the maximum temperatures of the coolant, the inner surface of the cladding, and the surface of the fuel, the average radial temperature of the fuel, and the maximum temperature at the centerline of the fuel. These values are based on new fuel; the temperatures inside the fuel will tend to increase slightly with irradiation, but the temperatures of the coolant, cladding, and fuel surface will not. Figures 5 and 7 present the axial temperature distribution of the same five quantities in the hottest channel of the reactor. The temperatures shown in Figs. 4 through 7 are based on the power distributions of the Case-1A and Case-2A columns of Table I, and thus are higher than those corresponding to the cores depicted in Figs. 1 and 2.

The temperatures discussed up to this point have not included provision for uncertainty factors. Table II compares the temperatures for Cases 1A and 2A (corresponding to Figs. 4-7) as calculated with and without uncertainty factors. The uncertainty factors applied to the cores in Table II are essentially those used in ANL-5719 (Addendum), with one change.

TABLE II. Maximum Temperature at Various Fuel-element Locations in Several Cores, as
Calculated without and with Uncertainty Factors

Location	Case 1A	Case 2A	ANL-5719 (Addendum) Core	
	First-stage Core (no fueled experiments)	Second-stage Core (with fueled experiments)	Case 3A (book value)	Case 3B (computer calculated)
Fuel Centerline	1163 (1236) ^a	1172 (1246)	1213 (1271)	1198 (1276)
Fuel, Average Radial	1123 (1189)	1130 (1196)	1161 (1213)	1148 (1218)
Fuel Surface	1083 (1143)	1090 (1151)	1108 (1154)	1102 (1165)
Cladding Inner Surface	1067 (1125)	1073 (1132)	1087 (1130)	1084 (1144)
Coolant	1018 (1068)	1023 (1074)	1037 (1075)	1030 (1082)

^aValues in parentheses were calculated with uncertainty factors.

The factor for a 5% transient overload was applied as a multiplier rather than in a square-root-of-the-sum-of-the-squares technique. This resulted in increasing the uncertainty factor from 1.114 to 1.157. Also included in Table II are maximum-temperature values for the ANL-5719 (Addendum) core. Those listed in the book-value column were taken directly from ANL-5719 (Addendum); those listed in the computer-calculated column were arrived at using the same techniques as for this 62.5-MWt analysis.

The values of maximum temperature for the 62.5-MWt proposal cores more closely approach the corresponding values from ANL-5719 (Addendum) than the values for power per subassembly approach the corresponding values in Table I. The larger the core, the lower the power density and the less the coolant flow through each subassembly (since total coolant flow is fixed). Therefore, the flow rate per subassembly in the 91-sub-assembly core is less than that in the ANL-5719 (Addendum) core. The ratio of power to flow rate for a subassembly in each core is about the same. Because of the change in technique for calculating the uncertainty factors, the maximum temperatures with uncertainty factors at one point (i.e., cladding inner surface in core with fueled experiments) is calculated to be 2°F higher than calculated in ANL-5719 (Addendum). This difference is not significant.

The increase in power to 62.5 MWt will affect the location of the experiments within the core and inner blanket. For the first stage, the initial increase in power, the experimental subassemblies are to be treated as shown in Table III. At the end of the first stage, the experimental subassemblies will be returned to their original positions in the core for continued operation at 50 MWt. The subassemblies will be modified, as required, during the six months between the first and second stages of the increase in power. At the beginning of the extended operation at 62.5 MWt, the experimental subassemblies will be relocated as necessary to ensure that their operating conditions are as desired. An indication of the type of subassembly relocation and modification to be required is given by Table IV, in which the present locations of experimental subassemblies are compared with the future locations as shown in Fig. 2 for a typical core loading under extended 62.5-MWt operation. Table IV compares the core positions, the fission rates, and the coolant flow rates for the two power levels, and indicates which subassemblies must be removed and reconstituted because

TABLE III. Disposition of Experimental Subassemblies during Initial 62.5-MWt Operation^a

<u>Subassembly</u>	<u>Normal Position</u>	<u>Remove to Basket</u>	<u>Relocate to Position</u>	<u>No Movement Required</u>
XG04	7B1	X		
X012	4B2	X		
X020	6B5	X		
X021B	2D1		5D2	
X027	4B3	X		
X032	6E5	X		
X033	5E2	X		
X035	7B4		7B2	
X041	7A3			X
X042A	7D5			X
X043	4D2	X		
X044	7A1			X
X050	4C2	X		
X051	3A2	X		
X054	4E1	X		
X055	6A4		7D3	
X056	5C2	X		
X057	2B1		5B2	
X058	6F1	X		
X059	4A1	X		
X060	7C3			X
X061	7A5			X
X062	6F3	X		
X063	7F5		7F1	
X064	4F2	X		
X065	7E5 ^b			X

^aSubassemblies X018B, X034A, X040, and X053 will be undergoing inspection and reconstitution during run 37. If they are available prior to start-up of stage 1, they should be held out of the reactor grid until resumption of 50-MWt operation.

^bIf the interim inspection and reassembly of X065 is completed and if the subassembly is available at the start of stage 1, the subassembly should be located in its normal position.

TABLE IV. Effect of Power Increase on Typical Experimental Loading

Subassembly	Type	50-MWt Reactor Power			62.5-MWt Reactor Power			Reconstitution	
		Pos.	Max Fission Rate $\times 10^{-13}$	Flow, gpm	Pos.	Max Fission Rate $\times 10^{-13}$	Flow, gpm	Zone	Flow
XG04	A	7B1	0.420	16.0	7B1	0.525	16.0	-	-
X012	A	4B2	0.821	52.0	6B3	0.818	52.0	R	-
X018 ^a	B7	4E2		12.0	4E2		12.0	-	-
X020	A	6B5	0.625	42.5	7B5	0.619	24.0	-	R
X021 ^a	B7	2D1		12.0	2D1		12.0	-	-
X027	A	4B3	0.821	30.5	6B4	0.818	W	R	-
X033	A	5E2	0.734	48.5	6E1	0.710	~48.0	-	-
X034 ^a	B7	2F1		12.0	2F1		12.0	-	-
X035 ^a	B7	7B4		12.0	7B2		12.0	-	-
X040	B37	5B2	0.734	68.0	6B1	0.710	~66.0	-	-
X041 ^a	B7	7A3		12.0	7A3		12.0	-	-
X042 ^a	B7	7D5		12.0	7D5		12.0	-	-
X043	F37	4D2	0.821	73.0	6D3	0.818	W	R	-
X044 ^a	B7	7A1		11.5	7A1		11.5	-	-
X051	F37	3A2	0.868	60.5	6A3	0.818	60.5	R	-
X053	B37	4E3	0.821	50.5	6E3	0.818	50.5	R	-
X054	F37	4E1	0.788	84.0	6E2	0.780	~83.0	-	-
X055	A	6A4	0.655	30.3	7F5	0.671	30.0	-	-
X056	F37	5C2	0.734	64.0	6C1	0.710	~61.0	-	-
X057 ^a	B7	2B1		11.5	5B2		~11.0	-	-
X058	F37	7F3	0.537	48.0	7F1	0.525	48.0	-	-
X059	F37	4A1	0.788	88.0	6A2	0.780	~87.0	-	-
X060 ^a	B7	7C3		11.5	7C3		11.5	-	-
X061 ^a	B7	7A5		11.5	7A5		11.5	-	-

Table IV (contd)

Subassembly	Type	50-MWt Reactor Power			62.5-MWt Reactor Power			Reconstitution	
		Pos.	Max Fission Rate $\times 10^{-13}$	Flow, gpm	Pos.	Max Fission Rate $\times 10^{-13}$	Flow, gpm	Zone	Flow
X062	F37	6N3	0.655	W	7D3	0.671	W	-	-
X063 ^a	B7	7N5		~11.5	7C1		~11.5	-	-
X064	A	4N2	0.821	27.0	6C3	0.818	27.0	R	-
X065 ^a	B37	7N4		~20.0	7F4		~20.0	-	-
PNL-6 ^b	H37	6N3	0.655	W	7E3	0.671	W	-	-
X069	H38	4N1	0.788	W	6F2	0.780	W	-	-
PNL-8 ^b	H37	4N1	0.788	W	6D2	0.780	W	-	-
NUMEC-4 ^b	B61A	3N1	0.856	W	6D4	0.818	W	R	-
NUMEC-5 ^b	B61A	4N2	0.821	W	6F3	0.818	W	R	-

^a Structural-material subassemblies.

^b A subassembly number had not been assigned at the time of writing.

^W Flow rate had not been set at time of writing.

^R Reconstitution required.

of moving from the core region (rows 1-5) to the inner blanket (rows 6 and 7) (under "Zone" in table) or because of new flow requirements (under "Flow").

B. Plant Conditions

The initial operation at 62.5 MWt should indicate that operation of the primary system, the secondary sodium system, and the steam system are satisfactory. Subsequent changes in core geometry will not affect the overall system. The flow rate through the primary system is constant at all power levels, and the temperature at the outlet of the reactor vessel is determined only by total reactor power.

As mentioned in Section III, the power will be increased to 62.5 MWt in two steps: 50 to 56 MWt and 56 to 62.5 MWt. The system conditions at 56 and 62.5 MWt will be compared with values that have been predicted on the basis of present performance of the plant. If there should be any significant variation between observed and predicted values, the cause of the variation will be sought.

The plant temperature and flow conditions at 56 and 62.5 MWt have been estimated from present operating conditions at 50 MWt. The estimate is partially based on the detailed heat balance reported in ANL-7577 (pp. 55-58).⁵ The results indicated that the net amounts of heat transferred to the primary sodium, from the primary sodium to the secondary sodium, and from the secondary sodium to the steam system at the three power levels differed slightly from each other, which is to be expected when measuring powers by three independent systems. Table V shows a recent heat balance at 50 MWt and the expected values at 56 and 62.5 MWt. The values listed are those expected to be given on the recording instruments or data logger. The conditions at higher power were obtained from a heat-transfer code that considers the heat balance for each system and the rate equation for each heat-transfer unit. The predicted conditions were based on design correlations that have been adjusted to give the measured performance at 50 MWt. Also shown in Table V are the design conditions given in ANL-5719 (Addendum). The secondary-system flow rate is slightly less than the design value, and the inlet and outlet temperatures for the intermediate heat exchanger (IHX) are higher than design values. These differences are due to the closer than expected approach temperatures achieved in the IHX and evaporators.

TABLE V. Heat Balances at Various Reactor Powers

<u>Parameter^a</u>	<u>Reactor Power, MWt</u>			
	<u>50^b</u>	<u>56</u>	<u>62.5</u>	<u>62.5^c</u>
<u>Primary System</u>				
Flow Rate, lb/hr	3.8×10^6	3.8×10^6	3.8×10^6	3.8×10^6
Reactor Inlet Temp, °F	700	700	700	700
Reactor Outlet Temp, °F	847	864	883	883
Power, MWt	50	56	62.5	62.5
<u>Secondary System</u>				
Flow Rate, lb/hr	2.1×10^6	2.2×10^6	2.4×10^6	2.6×10^6
Steam-generator Outlet Temp, °F	580	580	582	588
Steam-generator Inlet Temp, °F	842	858	875	866
Power, MWt	49.9	55.9	62.4	---
<u>Steam System</u>				
Flow Rate, lb/hr	2.0×10^5	2.2×10^5	2.5×10^5	2.5×10^5
Temp of Feedwater to Steam Drum, °F	549	549	549	550
Superheater Outlet Temp, °F	801	811	820	837
Power, MWt	49.4	55.4	61.9	---

^aThe data, except for flow rates, are those expected to be recorded by the data-logging system. The flow rates are those which would be indicated by the graphic-panel instruments.

^bAs measured on April 10, 1969.

^cDesign values from the Hazards Summary Report Addendum, ANL-5719 (Addendum).

The superheated steam flow is slightly higher than design, and the superheat temperature is less than design conditions. The lower superheat temperature is a result of using modified evaporators as superheaters in place of the design superheaters.

Table VI gives the data for heat-exchanger performance. The temperature of secondary sodium leaving the superheaters is higher than design because of the lower steam superheat temperature associated with using the modified evaporators as superheaters. The predicted pressure losses in the secondary system were obtained from a flow-calibration test at 580°F.

V. REACTOR SAFETY ANALYSIS

The Hazards Summary Report¹ and Addendum² cover the operation of EBR-II at 62.5 MWt. Although the core loadings during both stages of the higher-power operation are different from the ones considered in those references, they are not much different from those which have operated at 50 MWt. It is expected that the operating characteristics of the reactor will be similar to those observed in the past.

A. Reactor Temperatures

Temperatures in the core have been compared with those in Refs. 1 and 2 in Section IV. Temperatures in the blanket are more difficult to predict. Based on a linear extrapolation of the power produced in the blanket when going from 50 to 62.5 MWt and on the information sent to the AEC prior to the increase to 50 MWt in August 1968,⁶ the temperatures in the blanket region are expected to be as shown in Table VII.

B. Reactor Stability

The present methods of stability analysis for the EBR-II system are described in ANL-7476.⁷ The methods used include the standard rod-oscillator and rod-drop techniques. The rod-drop method currently is the basic routine analytical method for kinetic monitoring.

Figure 8 shows the power trace and the feedback reactivity following a rod drop during run 29C. Run 29C used a depleted-uranium blanket with

TABLE VI. Parameters of Plant Components at Various Reactor Powers

Parameter ^a	Reactor Power, MWt			
	50 ^b	56	62.5	62.5 ^c
<u>Intermediate Heat Exchanger</u>				
Primary-sodium Flow Rate, lb/hr	3.8×10^6	3.8×10^6	3.8×10^6	3.8×10^6
Primary-sodium Inlet Temp, °F	847	864	883	883
Primary-sodium Outlet Temp, °F	700	700	700	700
Secondary-sodium Flow Rate, lb/hr	2.1×10^6	2.2×10^6	2.4×10^6	2.6×10^6
Secondary-sodium Inlet Temp, °F	584	584	585	588
Secondary-sodium Outlet Temp, °F	846	862	879	866
Secondary-sodium ΔP , psi	2.2	2.5	2.7	---
<u>Steam Generator - Evaporators</u>				
Secondary-sodium Inlet Temp, °F	780	791	803	794
Secondary-sodium Outlet Temp, °F	580	580	582	588
Secondary-sodium ΔP , psi	2.4	2.7	3.0	---
Feedwater Flow Rate, lb/hr	2.2×10^5	2.5×10^5	2.7×10^5	---
Feedwater Temp at Inlet of Steam Drum, °F	549	549	549	550
Steam Outlet Temp, °F	579	579	579	578
<u>Steam Generator - Superheaters</u>				
Secondary-sodium Inlet Temp, °F	842	858	875	866
Secondary-sodium Outlet Temp, °F	780	791	803	794
Secondary-sodium ΔP , psi	14.3	18.1	19.4	---
Steam Flow Rate, lb/hr	2.0×10^5	2.2×10^5	2.5×10^5	2.5×10^5
Steam Inlet Temp, °F	578	578	578	578
Steam Outlet Temp, °F	801	811	820	837

Table VI (contd)

	Reactor Power, MWt			
	50 ^b	56	62.5	62.5 ^c
<u>Turbine Generator</u>				
Steam Flow Rate, lb/hr	1.3×10^5	1.5×10^5	1.6×10^5	1.9×10^5
Steam Temperature, °F	801	811	820	837
Electric Power, MWe	14.6	16.8	18.7	---

^aThe data, other than for flow rates, are those expected to be recorded by the data-logging system. The flow rates are as those which would be indicated by the graphic-panel instruments.

^bAs measured on April 10, 1969.

^cDesign values from the Hazards Summary Report Addendum, ANL-5719 (Addendum).

TABLE VII. Row-7 and Row-8 Temperatures at 50 and 62.5 MWt, °F

	50 MWt		62.5 MWt	
	Row 7	Row 8	Row 7	Row 8
Average Outlet Sodium	913	893	966	941
Maximum Outlet Sodium	950	910	1012	962
Minimum Outlet Sodium	845	870	881	913
Radial Gradient across Subassembly	55-105	35-75	70-130	45-95

Fig. 8. Run-29C 50-MWt Rod-drop Power and Reactivity Behavior

a 91-subassembly core. The feedback reactivity is calculated by subtracting the rod worth from the reactivity as given by inverse kinetics calculation of a rod drop at any power level. Figure 9 shows a typical drop-rod calibration curve. Once the feedback-reactivity has been plotted, a feedback model, given by the following equation, is fit to the data:

$$FB(S) = \sum_i \frac{A_i \exp(S\alpha_i)}{(1 + T_i S)},$$

where

FB(S) = the feedback reactivity,
 A_i = the feedback coefficient,
 α_i = the transport lag,
 T_i = the feedback time constant, and
 S = the LaPlacian operator.

The reactivity feedback shown in Fig. 8 is everywhere positive for decreasing power. This does not mean that every component is positive (e.g., that all of the components of A_i are necessarily positive), but it does mean that the resultant reactivity feedback is everywhere positive. This "positive" value of reactivity is for a decrease in reactor power caused by a negative reactivity insertion (rod drop). The net feedback for increasing power is therefore negative.

To have confidence that one can predict or extrapolate to higher power from data gathered on an approach to power, one must establish, with reasonable certainty, that extrapolation from one power to another is possible. Figure 10 compares a feedback curve obtained by linear extrapolation with one obtained by actual measurement. It shows that the feedback function is not exactly linear with power. However, the extrapolation is a very good approximation for the prompt feedback terms. The long-delayed terms appear to underestimate the feedback, but are obtained at times when system response may be obscuring the extrapolation.

The demonstrated extrapolation and fairly good agreement of the extrapolated data with the measured data will be used as a basis for kinetic monitoring during the EBR-II power increase to 62.5 MWt. Figures 11, 12, and 13 show the predicted feedback (based on run-35 data) for operation at 50, 56, and 62.5 MWt, respectively. It is apparent the system is expected to be stable for all powers considered. However, rod drops will be performed at 27, 50, 56, and 62.5 MWt. At each of these levels, the feedback

Fig. 9. Typical Drop-rod Calibration Curve

Fig. 10. Comparison of Extrapolated and Measured Feedback for Run 29C

Fig. 11. Predicted Feedback at 50 MWt, Based on Run-35 Data

Fig. 12. Predicted Feedback at 56 MWt, Based on Run-35 Data

Fig. 13. Predicted Feedback at 62.5 MWt, Based on Run-35 Data

will be compared with the most recent model. For the 27-MWt drop, the latest prior run and run-29C data would be used for comparison. For the higher-power rod drops, the experimental data will be compared with predicted values from the previous lower-power rod drops. Any significant deviation from the predicted curve would cause immediate cessation of the proposed power increase until analysis indicates the reason for the deviation. Power would be raised beyond 50 MWt only after sufficient reflection and confidence that the next step increase in power level is prudent on the basis of previous data and their correlation.

C. Power-Reactivity Decrement

Operation at 62.5 MWt will increase the power-reactivity decrement (PRD) because the temperatures of the fuel and coolant will be higher. The slope of the PRD curve from 40 to 50 MWt is nearly constant, and it is reasonable to expect a linear extrapolation of that slope over the 50-62.5 MWt range. On that basis, there should be a 20-25 lh greater reactivity decrement at 62.5 than at 50 MWt. However, the actual PRD may be somewhat different than implied by linear extrapolation because reduced-flow, Mark-II, 70%-enriched, and fueled experimental subassemblies will be removed for stage-one operation.

VI. PLANT PERFORMANCE CAPABILITY

Even though the plant originally was designed to operate at 62.5 MWt, a review of the entire plant has been made to ensure that operation at 62.5 MWt is consistent with current plant conditions. The results of that review are given here.

A. Primary System

The normal mode of operation with the reactor at power is with 100% primary flow regardless of power level. Hence, increasing the power level of the reactor will have no direct effect on any of the system components associated with flow. Operation and performance of the primary coolant

pumps remain unchanged. Operation at higher power, however, will result in two major changes in the primary-system parameters:

(1) Since the reactor is operated with a constant coolant inlet temperature and a constant flow regardless of power level, increasing the power level will increase the outlet coolant temperature of the reactor correspondingly. The average outlet sodium temperature will increase from 847°F at 50 MWt to 883°F at 62.5 MWt.

(2) The decay-heat power level after shutdown will be 25% greater than it was at 50 MWt.

The higher sodium outlet temperature will cause larger temperature gradients to exist in the intermediate heat exchanger and may tend to promote further temperature stratification in the primary bulk sodium. The increased temperature also will cause higher thermal stresses during steady-state operation and more-severe thermal shocks during startups and shutdowns. No problem is expected, however, since the design of the plant was based on 62.5-MWt operation.

1. Reactor Vessel

The reactor vessel and its appurtenances were analyzed for static and thermal stresses by the Franklin Institute in 1960.⁸ The final report concluded that the stresses are not excessive and that the design is satisfactory. The loadings on which the analysis was based are:

- (a) steady-state operation at 62.5 MWt;
- (b) reactor scram from full power subsequent to failure of pumping power; and
- (c) reactor scram from full power while all pumps remain in operation.

The assumed boundary conditions used in the analysis are conservative for the following reasons:

(a) The bulk temperature of the primary sodium at the reactor outlet was assumed to be 900°F. This temperature, during steady-state operation, will not exceed 883°F.

(b) A pressure drop of 20 psi was assumed in the primary sodium between the outlet plenum and the tank. This drop is actually about 12 psi.

(c) The analysis assumed sufficient convection of 700°F bulk sodium between the inner and outer radial neutron-shield cans to effect a sodium temperature of 700°F at the bottom of the outlet nozzle. The actual design includes a shelf installed specifically to prevent this natural convection flow. Therefore, the steady-state as well as the transient ΔT values were assumed to be higher than are now expected.

(d) The stresses termed "Allowable Membrane Stress Intensity" used in the analysis are values taken from ASME Code for Unfired Pressure Vessels, Section VIII, 1959 issue, Table UHA-23 for SA-240, Type 304 high-alloy steel. These stresses are relatively low. Furthermore, the listed values are further depressed because of the desire to keep the deformation to a minimum as recommended by the above cited code. Section III, Nuclear Vessels, as well as Section VIII, Division 2, of the 1968 ASME Code allow higher stress-intensity limits for the same material and advocate a more liberal method (hopper diagram) of comparing these with various categories of calculated stress.

2. Intermediate Heat Exchanger

The Franklin Institute performed a final stress analysis of the intermediate heat exchanger (IHX) after the preliminary analysis and design. Stresses due to static, thermal steady state, and thermal transient loadings were investigated and reported in 1960.⁹

This final analysis is quite complete and contains no major simplifying assumptions. Stresses were calculated for the four different loadings listed below:

(a) secondary-sodium pressure of 150 psig, and a resulting reaction of 1271 lb on the 14-in.-OD secondary inlet pipe;

(b) steady-state thermal loading due to 20.9°F temperature difference between the IHX inner shell and tube metal;

(c) reactor-scrum transient thermal loading produced by 60°F temperature difference between the IHX center shell and tube metal; and

(d) transient thermal loading due to secondary-pump failure resulting in 60°F temperature difference between the IHX shell and tube metal and 200°F rise at the outer edge of the bottom tube sheet.

Maximum operating stress intensity during a postulated secondary-pump failure transient occurs in the lower head of the IHX where the outer shell joins the ring body. The resultant stress intensity is approximately 50,000 psi. However, it is mainly a "peak" stress due to thermal transient and can be treated as an alternating stress range. In accordance with Fig. N-415(b) of the Nuclear Vessels Code, the number of cycles to produce fatigue is in excess of 10,000.

The comments regarding the techniques of thermal-stress analysis and the code limitations mentioned under "Reactor Vessel" (Section VI.A.1) above also apply to the IHX.

Two areas of possible deficiency were recently checked to ensure satisfactory operation of the IHX at a reactor power level of 62.5 MWt. The first, and probably more critical, item is the adequacy of the fillet weld between the U-nozzle (IHX well casing) and the bottom plate of the primary-vessel cover. The concern was prompted by the steady-state and transient temperature gradients across the weld and the shear stress imposed by the expansion of the IHX well casing. Since the resultant stress is a combination of primary, secondary, and peak stresses and is related to operating loads, a fatigue analysis has been performed. The results indicate that the maximum allowable number of thermal cycles is greater than 1100. The Project therefore feels confident that the weld integrity will be maintained even at temperatures that will prevail at 62.5-MWt operation.

The second item of concern is the thermal stress imposed on the IHX upper tube sheet, as well as on the secondary pipe, during a down-transient (scram). Calculations showed that, even with a conservative assumption of boundary conditions, the 3/4-in. shock plate underneath the tube sheet and the shock sleeve around the secondary pipe provide adequate protection.

Tabulated below are the expected and design IHX operating parameters for comparison:

	<u>Expected</u>	<u>Design</u> <u>(ANL-5719)¹</u>
Heat output, MWt	62.5	62.5
Primary-Na temp from reactor, °F	883	900
Primary-Na flow rate, lb/hr	3.8×10^6	3.73×10^6
Secondary-Na flow rate, lb/hr	2.4×10^6	2.50×10^6
Secondary-Na temp @ IHX inlet, °F	585	610
Secondary-Na temp @ IHX outlet, °F	879	880

The original stress calculations were based on a primary-sodium temperature of 900°F. The expected value of 883°F provides a certain amount of conservatism to the thermal-stress analysis. An effort is being made to reduce the number of spurious scrams of the reactor. With fewer scrams, the frequency of the cyclic thermal stresses will be less.

3. Primary-system Piping

A thorough stress analysis preceded the design and fabrication of primary piping between the reactor and the IHX.

The largest stress due to radial thermal gradients where inner and outer pipe are separated by sodium occurs during the up transient. Where the pipes are separated by gas, thermal stresses may reach 18,800 psi during a scram. This value is well below the maximum values allowed in accordance with USAS B31.7, the Nuclear Power Piping code.

4. Removal of Decay Heat

The increased decay heat after shutdown is of concern because of the limited cooling capability of the shutdown coolers. Each shutdown cooler was designed to remove 250 kW of heat from the primary-tank bulk sodium (initial temperature = 700°F). Recent tests showed that only about 200 kW are removed per cooler when the ambient air temperature is in the range from 70 to 80°F.

Decay heat normally is removed after shutdown by the steam system via the natural circulation of the secondary sodium system. If some incident would require dumping the secondary sodium system immediately following shutdown of the reactor from a high power level, the shutdown coolers must be capable of removing sufficient decay heat to prevent overheating of the primary-tank bulk sodium.

The results of an analysis of the temperature of the primary-tank bulk sodium vs time after shutdown were presented in Fig. 81 of ANL-5719.¹ That figure shows that with one shutdown cooler in operation, removing 250 kW of heat, the bulk sodium temperature slowly increases to a maximum of about 725°F in about 11 hr. Thereafter, the decay-heat power level is less than 250 kW, and the parasitic heat loss will cause the temperature to start to decrease.

The results shown in Fig. 81 of ANL-5719 are conservative, because they are based on the assumption that the reactor is shut down from full power of 62.5 MWt following infinite operation at that power level.

These results show that the actual heat-removal capability (400 kW) of the two existing shutdown coolers is adequate to prevent overheating of the primary tank in the event of the postulated incident. It appears that in the unlikely event of only one shutdown cooler being available for removing decay heat, the temperature of the bulk sodium could still be maintained below approximately 740°F, a temperature which is tolerable for short periods of time.

Based on the above, it is concluded that the existing shutdown coolers, which have a heat-removal capability of about 200 kW each, do not impose any restrictions on operation of the reactor at a power level of 62.5 MWt.

B. Secondary Sodium System

The secondary sodium system also has been analyzed to determine the effect of 62.5-MWt operation. Unlike the case in the primary loop, an increase in power will result in an increased flow rate of secondary sodium. Two areas of possible concern due to this increased flow rate were noted: the secondary-sodium electromagnetic (EM) pump and the superheater and steam generators. Further study, however, indicated that no problems should develop in either area.

1. Secondary-sodium EM Pump

The currently installed secondary-sodium pump developed a leak in the pump duct in April 1964 after only about 1300 hr of operation. During those 1300 hr of operation, the pump had been operated over a wide range of conditions. For about the last 125 hr of that time, the pump was operated with virtually no cover-gas pressure in the surge tank and with about 2 psig of gas pressure in the pump casing. For at least 16 hr of that last period, the flow rate was 6000 gpm, and for about 60 hr of that period, the flow rate was in excess of 5500 gpm. Prior to those last 125 hr

of operation, the surge-tank pressure usually was maintained in the range from 6 to 8 psig whenever the flow was above 4000 gpm.

Analysis of the duct failure led to the conclusion that failure was due to electromagnetically induced pressure pulsations in the entrance section of the pump duct. These pulsations caused resonance in several of the panels between spacers, and this resonance, in turn, caused fatigue cracks over three of the spacers. This failure mechanism can occur only when an adverse combination of low suction pressure (pressure of the argon cover gas in the surge tank), high flow, and elevated casing pressure is present. Since all these conditions were present just before (and only before) failure, there is little doubt that they are directly responsible for the failure.

The failed pump was repaired and has operated satisfactorily for over 33,000 hr. Operation of the repaired pump has been in accordance with recommendations made at that time by the pump vendor, General Electric.

Increasing the power level of EBR-II from 50 to 62.5 MWt will require an increase in the flow rate of secondary sodium from about 4800 to 5500 gpm. In order to continue to comply with General Electric's recommendations for pump operation, it will be necessary to increase the pressure of the argon cover gas in the surge tank from 6-10 in. of H_2O (0.22-0.36 psig) to 6-7 psig, while continuing to maintain the present vacuum in the pump casing. Increasing the pressure in the secondary surge tank to 6-7 psig will be adequate to ensure that the pump will not cavitate and that the pressure inside the pump duct will always be greater than the external pressure. As previously pointed out, the secondary sodium system originally was operated with cover-gas pressures to about 8 psig in the surge tank. This was later reduced to 6-10 in. of H_2O when it was found unnecessary to operate at the higher pressure. The design pressure of the system is 150 psig. Hence, the advent of higher-power operation and, therefore, a higher secondary-sodium flow rate, makes it necessary and acceptable to increase the pressure of the secondary sodium system as described.

The plant change to increase the pressure in the secondary sodium system by 6-7 psig will be accomplished before reactor power is raised above 50 MWt.

2. Evaporators and Superheaters

No problems are anticipated from operating either the evaporators or superheaters (modified evaporators) at powers up to 62.5 MWt. However, the possibility of flow-induced tube vibrations is recognized. The steam-generating equipment is arranged such that the secondary sodium flows through two superheaters in parallel and then through the eight evaporators, which are in parallel with each other but in series with the superheaters. The sodium flow rate is thus four times greater through the superheaters than through the evaporators, and the superheaters are more likely to have problems with flow-induced vibrations than the evaporators. Plans are being formulated to monitor and characterize the noise originating from each of the two superheaters. Accelerometers will be attached to the superheaters, and data will be collected and examined as the power is increased in approximately 10-MWt increments. On-the-spot analysis will consist of determining the frequency-amplitude spectra at each power level. A sudden change (increase) in amplitude as power is increased will indicate the possibility of detrimental tube vibrations. If this occurs, power will be restricted to a value less than that where indications were first noted until a plan is formulated to proceed.

In addition to the above vibration analysis, cognizant personnel using a stethoscope will listen for unusual or sudden changes in sound originating from anywhere in the steam-generation equipment.

The stress analysis of the evaporators, superheaters, and the secondary-sodium and steam piping was reviewed. This review did not identify or suggest any problem area.

C. Steam System

1. Main Turbine

The EBR-II power plant was designed for an output of 20 MWe. The main turbine-generator is rated at 20 MWe with turbine throttle conditions of 1250 psig and 840°F steam. Predicted turbine-performance information obtained from General Electric (the vendor) is as follows:

- a. Steam Pressure = 1250 psig
Steam Temperature = 840°F
Exhaust Pressure = 1-1/2 in. Hg abs

<u>Load, kW</u>	<u>Throttle Flow, lb/hr</u>
5,000	45,950
10,000	83,000
15,000	120,300
20,000	160,800

- b. Steam Pressure = 1250 psig
Steam Temperature = 800°F
Exhaust Pressure = 1-1/2 in. Hg abs

<u>Load, kW</u>	<u>Throttle Flow, lb/hr</u>
5,000	47,600
10,000	85,900
15,000	124,500
20,000	166,500

Present operating conditions of the turbine with the reactor operating at 50 MWt are approximately:

Steam Pressure = 1250 psig
Steam Temperature = 800°F
Exhaust Pressure = 1.0 in. Hg abs
Turbine-generator Load = 14,900 kW
Throttle Flow = 130,000 lb/hr
Total Steam Flow = 206,000 lb/hr
Turbine Bypass Flow = none

Predicted cycle conditions at a reactor power level of 62.5 MWt are:

Steam Pressure = 1250 psig
Steam Temperature = 819°F
Total Steam Flow = 254,000 lb/hr

On the basis of these data, it is predicted that a throttle flow of 180,000 lb/hr of steam at these conditions would be required to produce 20,000 kW of electrical power. At 50 MWt, the difference between the total

steam flow and the turbine steam flow was 76,000 lb/hr. All this steam was used to heat feedwater and/or run the feedwater (and condensate) pump.

It is estimated that, at 62.5-MWt operation, 170,000 lb/hr of steam will be available at the turbine throttle, and 84,000 lb/hr of steam will be required to run the feedwater pumps and to heat the feedwater to about 550°F, the normal value. Hence, this indicates that the turbine is capable of accepting all the steam available without exceeding the 20,000-kWe rating and without necessarily bypassing steam directly to the condenser.

The original predicted steam temperature in the EBR-II steam cycle at 62.5 MWt was 837°F. As mentioned above, the steam temperature at the present power level of 50 MWt is approximately 800°F, and this is expected to increase to 819°F at 62.5-MWt operation. With this slightly lower steam temperature, only about 3000 lb/hr of additional steam flow will be required to produce the same electrical power. This is less than a 2% increase in turbine steam flow and is expected to have a negligible effect on the life of the turbine.

It is concluded that no problems associated with the turbine-generator will arise when the power level is increased to 62.5 MWt. It is estimated that the turbine-generator will produce about 19,000 kW of electrical power with no steam being bypassed to the condenser.

2. Condenser

The EBR-II power plant was designed to be operated as an integrated facility for generating electricity. During normal operation, steam is passed through the turbine-generator, where a portion of the steam's energy is converted to electricity. The steam then enters the condenser, where additional energy is removed and rejected to cooling water, which in turn carries it to the cooling tower. At 62.5-MWt reactor power and with the turbine-generator operating at 20 MWe, 42.5 MWt must be dissipated by the condenser.

Provision has been made so that steam can be bypassed around the turbine and routed directly to the condenser. The condenser was designed to handle a continuous heat load of 215,000,000 Btu/hr (62.5 MWt) with 75°F inlet cooling water.

The specifications in the purchase order for the condenser are:

a. The condenser shall be capable of maintaining a pressure of 1-1/2 in. Hg abs at the turbine exhaust flange while condensing 145,000 lb/hr of turbine flow (970 Btu/lb enthalpy) plus 5000 lb/hr of bypass steam (1404 Btu/lb enthalpy).

b. The condenser shall be capable of condensing any combination of bypass and turbine flows ranging from 0 lb/hr of bypass and 150,000 lb/hr of turbine flow to 160,000 lb/hr of bypass and 0 lb/hr of turbine flow. A pressure of 1-1/2 in. Hg abs will not be required for heat loads exceeding those under condition a. Cooling-water inlet temperature at maximum heat load of 215,000,000 Btu/hr bypass will be 75°F.

c. In the event of turbine trip-out, the condenser may be called upon to accept up to 200,000 lb/hr of bypass flow within 1/2 sec; after several seconds, this flow will be reduced to the normal maximum of 160,000 lb/hr. A pressure of 1-1/2 in. Hg abs is not required for heat loads exceeding those under condition a. The condenser must be capable of withstanding these conditions.

Operating data to date indicate that the condenser will meet all these requirements at 62.5 MWt. For example, the plant has been operated on several occasions at a power level of 50 MWt without running the turbine. That is, all the power produced by the reactor was rejected to the condenser by bypassing the turbine. This has been done under a variety of weather conditions during the summer, and a condenser pressure of less than 4 in. Hg abs (normally pressure is about 1.5 in. Hg abs) always has been maintained.

These results show that the requirements of condition a, above, are exceeded and that there will be no difficulty in dissipating 62.5 MWt of power if the turbine is operating.

If it becomes desirable or necessary to operate EBR-II at design power level without the main turbine, all the heat will have to be dissipated by the condenser. The most severe set of conditions would arise following a sudden trip-off of the turbine, such that all the steam is suddenly bypassed to the condenser while power is maintained at 62.5 MWt. The sequence of events would be as follows:

(1) Initially, the steam flow to the turbine would be approximately 170,000 lb/hr, and the bypass to the turbine would be zero. At least 20,000 lb/hr would be extracted from the turbine for the purpose of heating

feedwater in the No. 1 and No. 3 feedwater heaters. Less than 150,000 lb/hr of steam from the turbine would be entering the condenser.

(2) When the turbine is suddenly secured, the entire steam flow (170,000 lb/hr) to the turbine throttle is bypassed to the condenser through the desuperheater. (This steam would have the enthalpy associated with 1250 psig, 819°F conditions.) At that time, the feedwater temperature starts to decrease because the heating that was done by the steam extracted from the turbine is no longer available.

(3) When the feedwater temperature starts to decrease, more steam is automatically extracted from the main steam heater to maintain the temperature of the feedwater going to the evaporators at about 550°F.

(4) Owing to the additional extraction, the steam flow to the condenser will decrease to a value which is determined primarily by the hotwell temperature (and, of course, power level).

The steam flow (X) to the condenser can be calculated approximately by a simple heat balance, as follows:

NOTE: The actual condenser pressure has little effect on the calculation because the enthalpy of the condensate varies little compared with the enthalpy of the steam.

$$(254,000 - X) 1390 + X93 = 254,000 (550),$$

from which

$$X = 165,000 \text{ lb/hr.}$$

A check on the calculation is the heat rejected by the condenser, which is

$$\frac{165,000(1390 - 93)}{3.413 \times 10^6} = 62.5 \text{ MWt.}$$

Hence, the predicted steam conditions to the condenser are almost exactly according to the design specification. It is, therefore, concluded that the installed condenser will perform satisfactorily at 62.5-MWt operation, provided:

(1) The capacity of the condenser-cooling system (cooling tower) is adequate to maintain a condenser pressure of less than 4 in. Hg abs when bypassing steam. Water from the cooling tower at 75°F should be adequate, and higher temperatures possibly could be tolerated if the condition described in item (2), below, is observed.

(2) The condensate pumps do not cavitate.

(3) The noise level associated with operation of the condenser when bypassing steam is not excessive.

The above three items will be monitored during the power increase.

3. Condensate Pumps

The two condensate pumps take their suction from the condenser hotwell. The pumps are identical, but one is motor driven and the other turbine driven. They are rated as follows:

Flow--530 gpm (255,000 lb/hr)

Head--550 ft

Speed--1760 rpm

NPSH--3.5 ft

Each pump was designed to be capable of handling the total condensate flow rate. The maximum condensate flow will occur when bypassing steam and is estimated to be as follows:

Bypass steam flow = 165,000 lb/hr

Blowdown water = 13,000 lb/hr

No. 1 feedwater heater = 23,000 lb/hr

Desuperheater spray = 47,000 lb/hr

Total 248,000 lb/hr

The normal operating level in the hotwell is about 5-1/2 ft higher than the suction of the pumps. Hence, the expected operating conditions for the condensate pumps are within the design specifications. Because the condensate-pump lines are not instrumented for flow, it has not been possible to verify the characteristics of the pump. Plans are being made to instrument the condensate lines and to test the pumps before increasing power. The main purpose of the test will be to determine if pump cavitation occurs at high flow rates and/or with abnormally high hotwell temperature. In any event, the condensate pumps are not expected to be limiting with respect to 62.5-MWt operation. If pump cavitation occurs with only one pump operating, both pumps can be run simultaneously. With both pumps running, the operating conditions for each pump will be less severe than they are at 50-MWt operation.

4. Feedwater Pumps

Each of the two installed feedwater pumps is designed to be capable of carrying the total feedwater flow at 62.5-MWt operation. One of the feedwater pumps is turbine driven, and it is run during normal operation. The prime mover for the second pump is an 800-hp motor. The pumps are not now capable of parallel operation. Both pumps have identical design ratings, which are:

Capacity = 295,000 lb/hr at 364°F

Suction Pressure = 161 psig

Discharge Pressure = 1500 psig

The estimated total feedwater flow at 62.5 MWt is 274,000 lb/hr, so either pump should be able to carry the full feedwater load.

Recent operating data for the turbine-driven feedwater pump at a flow rate of 226,000 lb/hr (50 MWt) indicated that the steam consumption by the turbine was higher than that which was predicted from performance curves supplied by the vendor. The measured steam flow was 22,500 lb/hr, and the predicted value would have been about 19,300 lb/hr. The performance curves indicate that the required feedwater flow at 62.5 MWt should be attainable with a steam flow rate of less than 22,000 lb/hr. Because operation of the turbine-driven feedwater pump has been substandard, the unit is being overhauled in an effort to improve its performance. While the

turbine pump has been out of service, the motor-driven pump has been operated. Performance of that pump has been satisfactory, and no problems are foreseen for its operation at the increased power level.

If the performance of the turbine-driven pump is not improved by the current overhaul, additional action will be taken to correct the situation. In all likelihood, this action will consist of procuring and installing new nozzles. It is not considered imperative to accomplish this action before increasing power. The motor-driven pump is fully capable of carrying the total feedwater flow, and although the present performance of the turbine pump indicates that its capacity may be marginal at 62.5 MWt, it may turn out to be completely acceptable.

D. Cooling Tower

Technical specifications written for the cooling tower at the time it was ordered required that it perform in accordance with the following requirements:

"A. The cooling tower shall cool 24,000 gpm of cooling water from 83°F to 70°F with an ambient wet-bulb temperature of 60°F."

A performance test was conducted in September 1968 to ascertain if the tower was capable of satisfying this requirement. The test results indicated that the pitch of the fan blades was improper and must be adjusted to ensure additional air flow for 62.5-MWt operation.

The test results were then analyzed to determine the effect of adjusting the fan-blade pitch to design values. The results of the analysis indicate that 24,000 gpm of cooling water will be cooled from 83.9°F to 70.9°F at an ambient wet-bulb temperature of 60°F.

A projection of the test results indicates that, with a water-flow rate of 22,519 gpm and ambient wet-bulb temperature of 60°F, the tower will reject 62.5 MWt of heat, thereby cooling the water to a temperature of 74°F. Condenser vacuum should be satisfactory with 74°F cooling water.

Climatographic data for the EBR-II location indicate that wet-bulb temperatures exceed 60.1°F only 2-1/2% of the time. The maximum wet-bulb temperature ever recorded there was 67°F. Extreme wet-bulb temperatures might force a temporary reduction of power from 62.5 MWt, with all heat

being rejected by the tower. With the plant in normal operation, i.e., the turbine-generator at 20 MWe, no reduction of power should be necessary.

These projections are based on the pitch of the fan blades being adjusted so that a sufficient rate of air flow will be attained. It is planned to make the adjustment in the four blades prior to the start of initial 62.5-MWt operation.

E. Electrical System

The EBR-II electrical power system is connected to the National Reactor Testing Station (NRTS) 138-kV electrical power system through two main 12,000/16,000-kVA, 138-13.8-kV power transformers. This arrangement provides duplicate main 13.8-kV feeders and will permit essentially full-capacity operation of the EBR-II facility if one of the main transformers is out of service. When EBR-II increases its power production from 50 to 62.5 MWt (15 to 19 MWe), more electrical power will flow through the main 138-13.8-kV transformers to the 138-kV electrical power system. The electrical power demand for EBR-II at 62.5 MWt will be between 4.5 and 5.0 MWe. With EBR-II generating 19 MWe, 14.5 MWe could be transmitted to the NRTS power system. The 14.5-MWe load with a 0.85 power factor corresponds to a 17,000-kVA load, which is greater than the nameplate rating of 16,000 kVA for one main transformer.

Under the condition that one main transformer must be taken out of service, the rating of the remaining transformer could be exceeded if the entire electric load of 14.5 MWe at 0.85 power factor is transmitted to the NRTS power system. One of two minor adjustments to the plant would need to be taken to prevent an overload. Either the electrical output of the turbine-generator could be reduced approximately 10% by bypassing a corresponding amount of steam directly to the condenser, or the kVA loading could be decreased by adjusting the power factor towards unity. Either adjustment is an acceptable solution and will guarantee essentially full-capacity operation on loss of one of the 138-13.8-kV transformers.

The 2400-V and 480-V distribution systems are supplied through parallel feeders by two 2500-kVA, 13.8-2.4-kV and 2000-kVA 13.8-kV-480-V power transformers, respectively. The load for the 480-V system will be the same at 62.5 MWt as it is at 50 MWt. A slight load increase will be noted on the

2400-V system because of the increased power for the secondary pump. The anticipated reactive loads for 62.5 MWt are summarized below:

	<u>50 MWt</u>	<u>62.5 MWt</u>	<u>Transformer Rated Capacity (total of two)</u>
480-V System	2300 kVA	2300 kVA	4000 kVA
2400-V System	2700 kVA	2800 kVA	5000 kVA

If either of the transformers or its associated primary or secondary breakers should fail, the remaining transformer would be loaded beyond its rating during reactor operation at either 50 MWt or 62.5 MWt. This condition exists for both the 480-V and 2400-V systems. A study is being made to determine possible sources of action on loss of either a 13.8-2.4 kV or a 13.8-kV-480-V power transformer. Two plans under consideration are:

(1) Install capacitors to bring the connected reactive load to a power factor near unity and thus reduce the transformer load, and (2) install forced-cooling fans on the transformers to increase transformer capacity by approximately 30%.

Since transformers have no moving parts, their life is limited only by deterioration of insulation. Excessive temperatures reduce insulation life. If overload periods are short (as when a transformer or associated breakers are out of service) or if ambient conditions are such that cooling-oil temperatures do not exceed rated values, overloads can be tolerated without loss of transformer life.

The other portion of the electrical system--including circuit breakers, switchgear, and emergency systems--is fully capable of carrying the loads for 62.5 MWt. It should be emphasized that loads on only the 13.8-kV and 2400-V electrical distribution systems will increase--the former because of the increased output of the turbine-generator, and the latter because of the slight increase in power for the secondary pump. The emergency loads on the diesel generators will be the same and within the design capacity of the diesel generators.

The temperatures of the oil for the 13.8-13.8-kV, 13.8-2.4-kV, and 13.8-kV-480-V transformers will be monitored during the power increase. Reactive loads for the three distribution systems--13.8-kV, 2400-V, and 480-V--will be measured before and immediately after the power increase.

It is concluded, then, that the electrical distribution systems are satisfactory for 62.5-MWt operation of the plant if special consideration is provided for the loss of one of either the 13.8-2.4-kV transformers or the 13.8-kV-480-V transformers.

VII. FUEL PERFORMANCE CAPABILITY

As a prerequisite to operation of EBR-II at 62.5 MWt, an experimental driver-fuel irradiation program has been undertaken. This program has consisted of irradiating 70%-enriched elements and controlled-flow subassemblies. Both types of experiments simulate the maximum temperature conditions at 62.5-MWt operation and the corresponding fission rates and heat fluxes. The program includes six subassemblies containing 70%-enriched fuel irradiated mostly in 4N1 positions with one subassembly in a 3N1 position. Currently, these six subassemblies have been irradiated in the burnup range of 0.3 to 1.8 at. %. The six controlled-flow subassemblies in the program have been irradiated in row 2 to current burnups of 0.4 to 1.5 at. %. The controlled-flow technique has two advantages over the 70%-enriched technique: it uses standard EBR-II fuel elements; and more of the elements see the maximum temperature conditions because there is only a slight flux gradient across the row-2 subassemblies.

Figure 14 shows the axial distributions of the sodium-coolant and the fuel-centerline temperatures for the hottest channel of the 70%-enriched and the controlled-flow subassemblies at 50 MWt and of a row-5 subassembly at 62.5 MWt. The test conditions at 50 MWt closely simulate the axial and radial fuel-element temperature distributions at 62.5 MWt. Also shown for comparison in the figure are the row-5 conditions at 50 MWt.

Figure 15 gives the fuel swelling that has been found in eight of the twelve subassemblies that have been examined to date. This figure shows that the swelling of 70%-enriched and controlled-flow elements lies between the high-silicon and low-silicon elements operated under 50-MWt conditions to burnups above 1 at. % and lies slightly above the 50-MWt subassemblies at low burnups. We can therefore conclude without any reservations that the driver fuel will perform quite satisfactorily with the reactor operating at 62.5 MWt. The only real question is, "What should the burnup

Fig. 14. Maximum Fuel-centerline and Coolant Temperatures

Fig. 15. Swelling of 70%-enriched EBR-II Fuel and of 52%-enriched EBR-II Fuel from Controlled-flow Subassemblies, as a Function of Burnup

limit be at 62.5 MWt?" This is a question which need not be answered until routine extended 62.5-MWt operation is started. Limited information available now indicates that the 1.5-at. % burnup limit now in force could be kept. We will continue to study the behavior of more subassemblies at and just above this limit during the intervening six months to provide a broader base for this decision.

Blanket subassemblies have not been run at simulated 62.5-MWt conditions, but drawing on information obtained from the 50-MWt blanket surveillance program and from the temperature dependence of uranium swelling, blanket performance at 62.5 MWt can be estimated. The increased temperature of operation (i.e., 970°F maximum center temperature at 50 MWt to about 1040°F maximum center temperature at 62.5 MWt) should have some effect on the blanket swelling characteristics. According to Leggett et al.,¹⁰ uranium having about the same impurity content as the EBR-II blanket material exhibits maximum swelling at about 550°C (1020°F) with significant swelling down to 450°C (840°F) and up to 625°C (1160°F). The EBR-II blanket elements now operate in this temperature range, but at 62.5 MWt a somewhat greater portion will be in this range. It is therefore possible that the blanket will swell somewhat more rapidly. Inspection of blanket elements operated at 50 MWt to burnups of 0.27 at. % maximum have shown a maximum cladding strain of 0.9%; however, the maximum burnup increment of the blanket element nearest the core center in one run at 62.5 MWt would be only about 0.04 at. %, so that the present limit of 0.27 at. % burnup will not be exceeded for any blanket subassembly. We therefore anticipate no blanket problems in running for only one reactor run. An experiment during the 62.5-MWt run is planned to obtain more information on the effect of increased power on blanket performance.

VIII. OPERATING PROCEDURES AT 62.5 MWt

A. Revision of Existing Procedures

Only minor changes in operating procedures will be required for operation at 62.5 MWt. Provision must be made to obtain the additional data that are not normally obtained during 50-MWt operation and to account for

the effect of the increased cover-gas pressure in the secondary-system surge tank. The present interlock checksheets, startup checksheets, data sheets, and list of operating set-points, however, are written for 50-MWt operation and must therefore be revised to reflect the new values for the various systems.

The following summarizes those changes of the Operating Manual, check-sheets, and data sheets that are necessary for 62.5-MWt operation:

(1) Operating Manual Section II B (Normal Procedures) will be revised to provide for additional data for power-reactivity decrement (PRD) at 56 and 62.5 MWt.

(2) Operating Manual Section V B (Secondary Cooling) will be revised to reflect the increased surge-tank pressure of 6-7 psig.

(3) The list of operating set-points (Appendix G of the Manual), interlock checksheets, startup checksheets, and operating log sheets will be revised as necessary to reflect the expected values of the trip set-points for those parameters in the shutdown or alarm circuits.

(4) At present, there are two areas that require special attention for the proposed operation at 62.5 MWt. One area is the capability of the turbine-driven feedwater pump to maintain sufficient flow at this power level, and the second is the capability of the condenser, the condensate pumps, and the cooling tower to handle full bypass flow when the turbine-generator is shut down. Special procedures will be prepared to cover this area, and the appropriate sections of the Operating Manual subsequently will be updated to reflect actual operating experience.

B. System and Component Surveillance

An increase from 50 to 62.5 MWt will result in consequent increases of the pressures, flows, and temperatures in the secondary and steam systems and temperature increases in the primary system. Effects on the plant auxiliary systems should be slight or nonexistent.

A thorough surveillance of each system and individual component will be performed at both 56 and 62.5 MWt. Operation at 56 MWt for 248 hr will allow ample time to assess the performance of the steam, secondary, and reactor systems before increasing the power to 62.5 MWt. The details of the surveillance program are to be delineated in a test procedure prepared before raising reactor power.

During the actual power increase, plant systems and components will be observed closely. If an unusual or unexplainable condition occurs, no further increase in reactor power will be permitted until a complete evaluation has been made of the condition. Once power is stabilized at the desired level, a complete set of normal operating data and special readings, as specified in the test procedure, will be recorded. All data and chart readings will be compared against the expected values, and any deviations will be noted.

Particular attention will be given to the subassembly outlet coolant temperatures and the fission-product monitors during the actual power increase. Special vibration instrumentation will be installed on the superheaters. Measurements taken before the power increase will be compared with data taken during the power increase. The evaporators and the secondary EM pump will be monitored for unusual vibration or noise with listening rods or stethoscopes.

The turbine-generator, cooling tower, condenser, circulating pumps, condensate pumps, feedwater pumps, and piping are expected to be capable of satisfactory performance at 62.5 MWt, as discussed in Section VI. Before any increase in power above 50 MWt is attempted, three new flowmeters will be installed in the condensate system, and a test run will be made to determine the flow rates within the system. Further, the hydrogen pressure in the generator casing will be increased to 15 psig to provide for adequate cooling when the turbine-generator produces an electrical load above 15 MWe.

During and after the actual power increase, the normal and standby pumps for each system will be operated to verify that each pump is adequate for flow corresponding to 62.5 MWt. After 62.5 MWt is attained, special tests will be performed on the condenser and the cooling tower. The tests will determine if these systems and their associated components meet design requirements and will establish any considerations for future operation.

IX. CONCLUSIONS

A thorough analysis of the EBR-II reactor and power plant indicates that the entire system is capable of operation at the design power level of 62.5 MWt. The power density and the temperatures in the reactor at 62.5 MWt, as well as the considerations of reactor stability and safety, are within

the bounds given in the Hazard Summary Reports.^{1,2} The Mark-IA driver fuel has been shown to be capable of operation at 62.5 MWt.

Operation of EBR-II at 62.5 MWt is desirable in keeping with the role of EBR-II as an irradiation and engineering facility in the overall LMFBR program. Experimenters have indicated their interest in having EBR-II operate at 62.5 MWt.

A program has been proposed to increase the power for a short time (≈ 600 MWd) to demonstrate the capability of the plant to operate at 62.5 MWt; to then return to operation at 50 MWt for six months so that experimenters will have sufficient lead time to plan subsequent experiments and modify current ones; and to then increase the normal operating power to 62.5 MWt.

ACKNOWLEDGMENTS

The preparation of this report was coordinated by R. A. Cushman. Contributions were obtained from all disciplines of the EBR-II Project, with major efforts by:

N. Bulut
H. W. Buschman
R. A. Cushman
R. W. Hyndman
R. E. Jarka
F. S. Kirn
J. H. Kittel
J. F. Koenig
S. M. Masters
L. B. Miller
R. Neidner
W. H. Perry
M. B. Trillhaase
C. M. Walter

REFERENCES

1. L. J. Koch et al., Hazard Summary Report, Experimental Breeder Reactor II (EBR-II), ANL-5719 (May 1957).
2. L. J. Koch et al., Addendum to Hazard Summary Report, Experimental Breeder Reactor II (EBR-II), ANL-5719 (Addendum), (June 1962).
3. Letter, J. H. Kittel to EBR-II Experimenters, "Projected Increase in EBR-II Flux Levels," June 2, 1969.
4. Letter, W. E. Baily (GE) to J. H. Kittel, June 19, 1969.
Letter, E. E. Bloom (ORNL) to J. H. Kittel, June 11, 1969.
Letter, J. Moteff (GE-NSP) to J. H. Kittel, June 11, 1969.
Letter, A. Biancheria (WARD) to J. H. Kittel, June 16, 1969.
5. ANL Reactor Development Program Progress Report for April-May 1969, ANL-7577.
6. TWX, R. K. Winkleblack to K. A. Dunbar, "Proposal for Increasing Power of EBR-II to 50 MWt," August 9, 1968.
7. R. W. Hyndman and R. B. Nicholson, The EBR-II Feedback Function, ANL-7476 (July 1968).
8. Final Report on Project A 2218, Franklin Institute, April 1, 1960.
9. EBR-II Primary Heat Exchanger Stress Analysis, Final Report on Project 16G-A 2268, Franklin Institute, July 1960.
10. R. D. Leggett et al., "Basic Swelling Studies," AIME Metallurgical Society Conference on Radiation Effects 37, p. 303, Gordon and Breach Pub., New York (1967).