OPERATION OF EBR-II AT 62.5 MWt:
AN EVALUATION OF PLANT AND REACTOR DATA FROM RUN 38A

Coordinated by

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ABSTRACT

During run 38A (September 14 to 28, 1969) the power level of EBR-II was increased in steps from 50 to 56 to 62.5 MWt in accordance with the proposal¹ for 62.5-MWt operation. All reactor and plant components operated in a satisfactory manner. Operation has been returned to the 50-MWt power level for an interim period to permit modification of existing experimental subassemblies and design of future experiments for 62.5-MWt operation.

I. INTRODUCTION

The proposal of August 1969¹ outlines the plan for increasing the operating power level of EBR-II to 62.5 MWt. The first stage of that plan--the initial operation at 62.5 MWt with a special core loading--was performed during run 38A. Total accumulated operation during run 38A was 776 MWd, of which 517 MWd was accumulated at 62.5 MWt. The required changes in core loading and other preparations for high-power operation (e.g., measurements of shut-down circuit response times, increasing secondary-sodium surge tank pressure, etc.) were made before run 38A, which started on September 14, 1969.

II. CORE LOADING

Extensive fuel handling took place between routine operation at 50 MWt in run 37 and the demonstration of 62.5-MWt operation in run 38A. All fueled experimental subassemblies were removed from the core (rows 1-6). One fueled experimental subassembly (X055) was located at position 7D3 in

the blanket region. The final core loading resembled that illustrated by Fig. 1 of Ref. 1; the exact core loading for run 38A is shown in Fig. 1 of this report. The core loading consisted of 89 subassemblies. Two blanket subassemblies were used on opposite corners of row 6 to fill that row. All structural experimental subassemblies except X021B and X057 also were removed from the core region. Three dummy structural subassemblies (two of which were orificed to give a flowrate of about 3 gpm as compared with the 12-gpm nominal flowrate for structural subassemblies) were placed in row 3. Ten half-worth subassemblies were loaded for reactivity control. (Half-worth subassemblies contain 46 standard driver elements and 45 stainless steel dummy rods.) Six of these half-worth subassemblies accepted the full coolant flow of a driver subassembly, and four were orificed to 2/3 flow. Rows 1 and 2 contained only standard driver-fuel subassemblies, as part of an attempt to gain further information on the relationship between power and flow in EBR-II.

III. PROCEDURE AND EXPERIMENTS

The procedure for attaining 62.5 MWt described in Ref. 1 was followed. Operation at 50 MWt began on September 14, 1969, and continued until September 16, when the power level was increased to 56 MWt. Two days later, on September 18, the power was increased to 62.5 MWt. Reactor operation at the new power levels was within expectations and followed the established patterns of operation at 50 MWt. The power-reactivity decrement (PRD) was measured during the initial approach to 50 MWt and during the ascent from 50 to 62.5 MWt. As illustrated in Fig. 2, no change in the basic nature of the PRD was noted.

Rod-drop tests were performed at 500 kW and 50, 56, and 62.5 MWt. Component performance was monitored at all power levels. Acoustical methods were used to monitor the superheaters. Tests of the steam-bypass system were run at 62.5 MWt, and transient tests of the steam system were performed at 62.5 MWt to simulate loss of tie with the NRTS grid and to demonstrate the effectiveness of a turbine-generator trip at full power. The results of operations and tests at powers up to 62.5 MWt during run 38A are summarized in Sections IV and V.

KEY: B - DEPLETED URANIUM
 BETH - BERYLLIUM THIMBLE
 C - CONTROL ROD
 CF - CONTROLLED-FLOW S/A
 D - DRIVER FUEL
 Ni - NICKEL-CORROSION S/A
 P - 1/2 DRIVER FUEL--1/2 STAINLESS STEEL
 S - SAFETY ROD
 SSCR - STAINLESS STEEL CONTROL ROD
 SST - STAINLESS STEEL THIMBLE
 ○ - EXPERIMENTAL S/A (X000--stainless steel dummy)

Fig. 1. Loading Pattern for EBR-II Run 38A, 9/10/69 to 9/28/69

Fig. 2. Curves of Initial Power-Reactivity Decrements in Runs 37 and 38A

IV. CONDITIONS AT HIGHER POWER

A. Reactor

The power was increased to 62.5 MWt in two increments: first, from 50 to 56 MWt and then, two days later, from 56 to 62.5 MWt. The core loading, shown in Fig. 1, differed from the usual recent irradiation loadings (run 29C through run 37) which also used the radial depleted-uranium blanket. Except for a portion of run 32, three structural experiments were located in row 2 in all recent runs. For run 38A, three subassemblies simulating structural experiments were located on row-3 flats. The four row-3 corner positions that did not contain safety rods were occupied by half-worth subassemblies. In effect, the seven standard driver subassemblies in rows 1 and 2 virtually were surrounded by a ring of stainless steel in row 3. The seven standard driver subassemblies were intentionally located in rows 1 and 2 as part of a special investigation relating to temperature prediction in EBR-II. Because of their lower coolant-flow rates, all reduced-flow subassemblies, which were in rows 2 and 3, were removed before the run-38 startup. These included both the Mark-I and Mark-II fuel subassemblies. Only two structural irradiation experiments were left in the core. These were located in row 5.

The effect of the revised core loading (run 38A vs run 37) on the power coefficient was reflected by an overall decrease in the PRD. For example, the PRD at 50 MWt in run 38A was approximately 10 lh smaller than that measured at 50 MWt during the run-37 startup (see Fig.2). Since the slope of the measured PRD curve for run 38A was essentially independent of reactor power, it is reasonable to conclude that the decrease in the PRD was not per se a consequence of higher-power operation. Instead, the decrease is believed to be the result of the removal of the controlled-flow subassemblies and a possible decrease in negative, mechanically originated feedback effects. The slight, but perceptible, increase in the slope of the PRD curve for power levels above 50 MWt has been tentatively associated with the higher fuel-expansion coefficient that resulted from an increase in the fraction of fuel material in the higher-temperature gamma phase above 50 MWt.

Measured control-rod worths during run 38A were approximately 6-8% smaller than those for previous runs, and a 25% reduction in the worth of the experimental stainless steel drop rod was noted. The reduction in drop-rod worth was the consequence of locating X021B (a structural experiment) next to the drop rod.

The results of physics calculations for the run-38A configuration clarified the differences noted in control-rod worths relative to previous runs. The increased steel content of row 3 caused a change in the radial fission-rate distribution for run 38A as compared with those for previous loadings. Such effects are illustrated in Fig. 3, which is essentially reproduced from Ref. 1. An additional curve has been added; this gives the results of DOT-5 X-Y calculations for the detailed run-38A radial fission distribution.* As the figure shows, the calculated radial fission distribution for run 38A is different from that measured in run 29D. The larger maximum-to-average fission-rate ratio for run 38A is the cause of the measured reduction in control-rod worth relative to previous runs. The reactivity decrease due to burnup of fuel that was measured during run 38A was about 0.02 Ih/MWd greater than the 0.12 Ih/MWd generally realized since run 29D. The central peak in fission rate, coupled with the greater worth of central subassemblies, resulted in a larger reactivity-burnup coefficient.

The temperatures indicated by core-outlet thermocouples were found to lie within the allowable range of values predicted from measurements made at 50 MWt. Over the range from 50 to 62.5 MWt, all thermocouple readings increased linearly with respect to power. The readings of outlet thermocouples in rows 1 and 2 (which contained seven standard driver subassemblies) were used to provide data for relating power, flow, and

* The curve labeled "Figure 1" in Fig. 3 of Ref. 1 was mislabeled. It was based on an earlier version of Fig. 1, the proposed initial core layout. When the core layout actually used as Fig. 1 of Ref. 1 was established, a replacement curve should have been provided in Fig. 3. This was not done. Had this been done, the new curve would have been similar to that added in this report for the actual run-38A loading.

Fig. 3. Relative Radial Distribution of Fission Source

temperature. The outputs of all seven thermocouples were consistent within an envelope of $\pm 11^{\circ}\text{F}$. The temperature data suggest that the error in existing power-flow calculations is approximately 2-4% for the cases analyzed. Calculated subassembly-outlet temperatures have been up to 10°F higher than measured temperatures. When the influence of adjacent subassemblies is considered, the discrepancy decreases to about 7°F for the cases studied. If the predicted subassembly flows were slightly too low or the predicted power density slightly too high, or if some combination of these possibilities were to occur, approximately a 4% change in the ratio of power density to subassembly flow would effect a reasonable agreement between measured and predicted temperatures.

The calculated DOT X-Y fission distribution for run 38A was used in two different programs (POWDIST and COOLTEMP) to determine power generation and temperature distributions in the core. The POWDIST program considers the average radial fission-rate distribution and determines the power produced by driver subassemblies located in the various core positions (e.g., 2N1, 4N2, and 4N3) assuming that only "effective" subassemblies (i.e., those containing the equivalent amount of fuel as a driver subassembly) are used. It is quite possible that no subassembly actually produces its designated power, because variations in fuel loadings, burnup, and flux asymmetry are not considered by POWDIST. COOLTEMP, on the other hand, can utilize the detailed subassembly-by-subassembly relative power-generation distribution provided by DOT and thus calculate the power actually produced by each subassembly, whether driver, control or safety rod, or experiment. The differences between the POWDIST and COOLTEMP results are evident from a scrutiny of data summarized in Tables I and II. Table I compares the power-production predictions for driver subassemblies in various core positions, assuming a fission distribution which depends on radial position only. It gives results for the proposed core loading from Ref. 1, the actual run-38A loading as calculated by POWDIST, and those obtained by averaging the COOLTEMP results for driver subassemblies in the various core positions. Table II illustrates the variation in power production in row 2, which contained only driver subassemblies. The variation in power generation is principally caused by angular asymmetry in the core (Sectors C and D were in the depressed flux positions of the core loading--see Fig. 1). Table II also presents the temperatures predicted by COOLTEMP and compares these with the outlet-thermocouple temperature measurements for rows 1 and 2.

TABLE I. Predicted Power Distribution (kW per Driver-fuel Subassembly)
for Various Cores at 62.5 MWt

Core Position	Case 1B (Fig. 1 of Ref. 1)	Run 38A (by POWDIST)	Run 38A (by COOLTEMP)
1N1	960	1024	1022
2N1	935	997	996
3N1	886	914	---
3N2	902	940	939
4N1	799	820	813
4N2, 3	835	853	849
5N1	671	694	---
5N2, 4	725	751	753
5N3	744	769	---
6N1	515	532	532
6N2, 5	583	604	604
6N3, 4	618	639	635
Number of "Effective" Subassemblies	76.8	73.8	73.8

TABLE II. Results of COOLTEMP Calculations for Run 38A at 62.5 MWt

Core Position	Subassembly Power, kW	Temperature at Top of Fuel, °F	Coolant Temperature at Top of Blanket, °F	Measured Outlet Temperature, °F
1A1	1022.4	882.5	882.2	872
2A1	1010.8	880.4	880.5	875
2B1	981.9	875.3	874.1	869
2C1	972.7	873.6	873.2	875
2D1	997.0	878.0	878.1	873
2E1	1007.1	879.8	879.6	880
2F1	1008.3	880.0	879.5	873

The reactivity worth of the drop rod was established through calibration measurements conducted at 500 kW at the beginning of run 38A. The measured worth of \$0.017 indicated an approximate 25% decrease from the worth of \$0.021 established at the end of run 37. As explained above, the decrease is believed to be the result of locating a structural subassembly next to the drop rod and from an increase in the maximum-to-average radial flux profile. Figure 4 shows the actual rod-worth calibrational data.

The reactivity feedback at 50 MWt during run 38A was predicted by adjusting the run-37 50-MWt feedback data by the ratio of the drop-rod worths, i.e., \$0.017/\$0.021. Figure 5 compares the measured time-dependent feedback for run 38A at 50 MWt with that predicted from measured run-37 feedback. A cursory examination of the two curves indicates that while the magnitude of the prompt feedback remained essentially unchanged, the actual amplitudes of delayed feedback terms were less than predicted. The reduction presumably reflects the effects of the changes in core configuration and composition made for run 38A.

A linear extrapolation of the measured run-38A 50-MWt feedback to 56 MWt was made by multiplying the measured 50-MWt feedback by the power ratio 56/50. As shown in Fig. 6, the measured and extrapolated values were in excellent agreement; the prediction is indistinguishable from the measurement. Similarly, the feedback measured at 56 MWt was extrapolated to 62.5-MWt conditions. As shown in Fig. 7, the values again were in excellent agreement; the prediction is indistinguishable from the measurement.

Using the method described by Hyndman and Nicholson,² attempts were made to separate the gross (measured) feedback for run-38A 50-MWt operation into individual time-dependent feedback terms. The results of these efforts are summarized in Fig. 8, which compares the measured feedback with that calculated from the mathematical model also described in Fig. 8. An excellent agreement may be seen. Then, assuming the same feedback components with the same time constants, but with their amplitudes multiplied by the power ratio 56/50, the feedback at 56 MWt was predicted. The results are illustrated in Fig. 9. Similar results for the predicted feedback at 62.5 MWt are given in Fig. 10. Although the agreement between prediction and measurement in the last two cases is not perfect, it is apparent that the nature of the feedback at 62.5 MWt was essentially unchanged as the result of the power increase. The slight reduction in the amplitude of the predicted feedback at 62.5 MWt is believed to be the result of small unpredictable changes in the worth of the drop rod as a function of power.

Fig. 4. Drop-rod Calibration, Beginning of Run 38A

Fig. 5. Predicted and Measured Feedback at 50 MWt and Full Flow, Run 38A

Fig. 6. Predicted and Measured Feedback at 56 MWt and Full Flow, Run 38A
(prediction based on run-38A measurement at 50 MWt)

Fig. 7. Predicted and Measured Feedback at 62.5 MWt and Full Flow, Run 38A
(prediction based on run-38A measurement at 56 MWt)

Fig. 8. Run-38A Feedback at 50 MWt and Full Flow, as Measured and as Calculated with Feedback Model

Fig. 9. Run-38A Feedback at 56 MWt and Full Flow, as Measured
and as Calculated with Feedback Model

Fig. 10. Run-38A Feedback at 62.5 MWt and Full Flow, as Measured
and as Calculated with Feedback Model

Such changes have been previously noted and are only qualitatively understood.

The feedback model used to fit the rod-drop data at the different power levels shows that the character of the feedback at 62.5 MWt and at 50 MWt in run 38A are essentially the same. On the basis of the run-38A data, it is expected that the feedback at 62.5 MWt for a run-37 configuration can be linearly extrapolated from 50-MWt data. This extrapolation would yield the feedback anticipated for the 62.5-MWt operation of an EBR-II irradiation or test reactor configuration. The difference between the feedback for such a future configuration and the measured configuration at 62.5 MWt for run 38A may be estimated qualitatively from the differences between "predicted" and "measured" data in Fig. 5. Such extrapolation is appropriate provided that the character of the configuration for extended operation at 62.5 MWt does not differ significantly from the character of the configuration for run 38A.

The feedback model is extremely useful for deducing the trends cited above. The same model, however, cannot be used to make absolute conclusions regarding the system feedback because it is, in a sense, an empirical and nonunique fit to the data. Another or several other models with different time constants and different feedback amplitudes for some terms could be used in a similar manner and yield trends similar to those cited above. The model described in Fig. 8 is only one of several with similar physical characteristics that can be used to fit the data.

B. Plant

Tables III and IV compare the measured and predicted plant data for 56- and 62.5-MWt conditions. The largest differences between measured and predicted values were in the temperature of the secondary sodium leaving the intermediate heat exchanger (IHX). The predicted outlet temperature was based on an IHX performance at 95% of the design coefficient; this performance was calculated from measurements at 50-MWt conditions. Before the 50-MWt measurements, the readings of the temperature sensors for the primary and secondary sodium had been compared at 580°F isothermal conditions. Additional corrections were applied to the recording system for the measurements obtained at operating conditions. Measurements

TABLE III. Heat Balances at 56 and 62.5 MWt

<u>Parameter</u>	<u>56 MWt</u>		<u>62.5 MWt</u>	
	<u>Predicted^a</u>	<u>Measured</u>	<u>Predicted^a</u>	<u>Measured</u>
<u>Primary System</u>				
Flow Rate, lb/hr	3,820,000	3,860,000	3,820,000	3,810,000
Reactor Inlet Temp, °F	700	700	700	700
Reactor Outlet Temp, °F	864	864	883	883
Power, MWt	56	56.5	62.5	62.3
<u>Secondary System</u>				
Flow Rate, lb/hr	2,240,000	2,230,000	2,380,000	2,360,000
Steam-generator Outlet Temp, °F	580	582	582	581
Steam-generator Inlet Temp, °F	858	857	875	870
Power, MWt	55.9	54.1	62.4	61.3
<u>Steam System</u>				
Flow Rate, lb/hr	225,000	224,000	250,000	250,000
Temp of Feedwater to Steam Drum, °F	549	559	549	560
Superheater Outlet Temp, °F	811	807	820	819
Power, MWt	55.4	53.6	61.9	60.8

^a From Ref. 1.

TABLE IV. Parameters of Plant Components at 56 and 62.5 MWt

Parameter	56 MWt		62.5 MWt	
	Predicted ^a	Measured	Predicted ^a	Measured
<u>Intermediate Heat Exchanger</u>				
Primary-sodium Flow Rate, lb/hr	3,820,000	3,860,000	3,820,000	3,810,000
Primary-sodium Inlet Temp, °F	864	864	883	883
Primary-sodium Outlet Temp, °F	700	700	700	700
Secondary-sodium Flow Rate, lb/hr	2,240,000	2,230,000	2,380,000	2,360,000
Secondary-sodium Inlet Temp, °F	584	588	585	586
Secondary-sodium Outlet Temp, °F	862	858	879	872
Secondary-sodium ΔP, psi	2.5	2.6	2.7	2.8
<u>Steam Generator - Evaporators</u>				
Secondary-sodium Inlet Temp, °F	791	788	803	800
Secondary-sodium Outlet Temp, °F	580	582	582	581
Secondary-sodium ΔP, psi	2.7	3.6 ^b	3.0	4.1 ^b
Feedwater Flow Rate, lb/hr	245,000	244,000	270,000	265,000
Feedwater Temp at Inlet of Steam Drum, °F	549	559	549	560
Steam Outlet Temp, °F	579	580	579	580
<u>Steam Generator - Superheaters</u>				
Secondary-sodium Inlet Temp, °F	858	857	875	870
Secondary-sodium Outlet Temp, °F	791	788	803	800
Secondary-sodium ΔP, psi	18.1	16.8	19.4	19.5
Steam Flow Rate, lb/hr	225,000	224,000	250,000	250,000
Steam Inlet Temp, °F	578	559	578	580
Steam Outlet Temp, °F	811	807	820	819
<u>Turbine Generator</u>				
Steam Flow Rate, lb/hr	146,000	142,000	162,000	170,000
Steam Temp, °F	811	807	820	819
Electric Power, MWe	16.8	16.5	18.7	18.5

^a From Ref. 1.

^b Inlet pressure; outlet transmitter out of service.

indicated an IHX hot-end approach temperature of 3°F at 50-MWt reactor conditions. These data were used to obtain the 95% performance prediction. Based on the operating conditions at 62.5 MWt, when the secondary-sodium outlet temperature was less than predicted, the exchanger performed at 87% of the design coefficient. An error of only 1°F in the determination of the hot-end ΔT at 50 MWt is sufficient to account for an 8% difference in performance, which could lead to an over-prediction of the secondary-sodium outlet temperature at 62.5 MWt. The resulting over-prediction also led to high estimates for the temperature of the superheated steam and of the secondary-sodium temperature leaving the superheaters.

Small differences in heat balance between the three systems were also noted. These were similar to those obtained previously at 50-MWt conditions. Detailed heat-balance data indicate that miscellaneous heat additions and losses in each sodium loop are essentially equal. Therefore, the differences are principally the results of inaccuracies in instrumentation.

V. PLANT PERFORMANCE

The performance of the plant at 56 and 62.5 MWt was satisfactory. No problem area that would tend to preclude continual 62.5-MWt operation was noted.

A. Primary System

The increase in sodium-outlet temperature at 62.5 MWt relative to 50 MWt caused further temperature stratification in the primary bulk sodium. Figure 11 shows the temperature stratification that exists from zero power to 62.5 MWt with 100% primary flow. Temperatures of the sodium at elevations near the top of the sodium surface (1-1/2 ft below the sodium surface) and near the bottom of the primary tank (21-1/2 ft below the sodium surface) are plotted in the figure. The temperatures near the surface increased with reactor power while the temperatures near the bottom of the primary tank decreased. The maximum temperature difference measured in the bulk sodium at 50 MWt was 27°F compared with 37°F at 62.5 MWt. Near the sodium surface, the average temperature was 717°F at 50 MWt, and 723°F at 62.5 MWt,

Fig. 11. Temperature Stratification in Bulk Sodium
at 100% Primary Flow and Various Reactor Powers

an increase of 6°F. These numbers do not include the readings of one thermocouple (R1-TC-501-T), which is located at the 1-1/2-ft elevation and near the intermediate heat exchanger and reactor outlet piping. Readings from that thermocouple cycled slowly between 728°F and 756°F during reactor operation at 62.5 MWt. The varying temperatures indicate that a certain amount of natural convection flow occurs in the section of the primary tank near that thermocouple.

The magnitude of the effects of temperature stratification was no greater than anticipated. No problem resulted from the temperature gradient that actually existed in the bulk sodium at 62.5 MWt. All other components of the primary system performed as expected.

B. Secondary Sodium System

1. Secondary-sodium EM Pump

As described in Ref. 1, the pressure in the secondary sodium system was increased to 6-7 psi before raising the power level above 50 MWt. This increase was in accordance with recommendations made by the pump vendor, General Electric, and was for the purposes of (a) preventing pump cavitation and (b) ensuring that the pressure inside the pump duct was always greater than the external pressure. Operation with the increased pressure at 62.5 MWt was in accordance with expectations. Operating parameters of the pump at this power level were:

Secondary Flow--5300-5500 gpm at 590°F

Pump Head--34 psi

Power Input--240 kW

Pump Voltage--300 V

The pump is rated for operation at:

Flow--6500 gpm at 700°F

Pump Head--53 psi

Power Input--336 kW

Pump Voltage--445 V

Comparison of the actual operating parameters of the pump with the design rating shows that operating conditions are well within design conditions.

Pump operation at 62.5 MWt was satisfactory in all respects. No changes in sound originating from the pump could be detected under the 62.5-MWt operating conditions.

2. Evaporators and Superheaters

The possible occurrence of flow-induced tube vibrations was cited in Ref. 1. During the power increase (0 to 62.5 MWt), the noise originating from each of the two superheaters was monitored. The superheaters are more susceptible to tube vibration than the evaporators because the flow velocities through the former are larger than through the latter.

Accelerometers were mounted at three different locations on each superheater. Equipment was provided for (a) direct audio listening, (b) recording on tape, and (c) making a simple frequency-spectrum analysis. The spectrum analyzer consisted of a Krohn-Hite variable band-pass filter (2-500 Hz) and a TRMS voltmeter. With this equipment, it was possible to perform an "on-the-spot" analysis to determine if there were any sudden increases in noise at any frequency in the range of 2-500 Hz as power was increased. Taking into account the considerable experimental uncertainty associated with these data, the character of the measured frequency spectrum was relatively invariant as the power was increased from 20 to 62.5 MWt. Figure 12 shows typical curves of noise vs frequency at each power level. At power levels less than 20 MWt, the character of the frequency spectrum was somewhat different from that at higher (>20 MWt) powers. This is attributed to system noise directly associated with low-power operation. Such noise originates from the induction heating, check valves, and cycling flow-control valves.

In addition to the monitoring of the superheaters with electronic equipment, cognizant personnel listened with a stethoscope for unusual noises or changes in sound originating anywhere in the steam-generating equipment. The sound in all piping and components in the boiler wing of the sodium boiler building was monitored. Specific items so monitored include the following: superheaters; all evaporators; secondary-sodium EM pump; surge tank; steam drum; and steam, sodium, and feedwater piping. No unusual noise resulting from operating at 62.5 MWt was heard at any place in any of these components and piping.

Fig. 12. Results of Monitoring of Superheater Noise during Increase in Reactor Power

C. Steam System

As the power was increased to 62.5 MWt, the performance of the steam system was satisfactory and essentially as predicted.

1. Steam-bypass System

Tests of the steam-bypass system were performed to evaluate its performance. They provided information for use in improving the system further. Bypassing steam involves routing high-pressure steam through a pressure-reducing station and then directly (through a desuperheater) to the condenser, without passing through the main turbine. Initial testing of the bypass system revealed that the large steam-bypass valve oscillated whenever the steam flow through the valve was about 140,000 lb/hr (corresponding to 56 MWt). Operation was stable at both lower and higher steam flows. After taking action to minimize the valve oscillation, additional tests of the bypass-steam system were performed.

Two tests (TST 41-69) were conducted with the reactor at 62.5 MWt:

- (1) With the reactor at 62.5 MWt and the turbine on initial pressure regulator, the incoming-line breakers were tripped open.
- (2) With the reactor at 62.5 MWt and the turbine on initial pressure regulator, the turbine was tripped off.

The first test demonstrated satisfactory operation of the turbine auto-transfer system. However, the steam-bypass valve did not respond (open) fast enough to avoid high steam pressure and subsequent lifting of the Pressurmatic relief valve. The steam-bypass valve was opened manually about 15 sec after the test was initiated.

The second test demonstrated satisfactory operation of the condenser and cooling tower under maximum heat load. As in the first test, the steam-bypass valve failed to respond (open) fast enough to avoid high steam pressure. The Pressurmatic relief valve again opened, and it was necessary to terminate the test by manually opening the steam-bypass valve. Further testing was discontinued until the improvements shown to be necessary could be made.

The steam-bypass system is being studied, and corrective action is being taken to eliminate the problems associated with its operation. The vendor of the bypass valve was contacted for his recommendations regarding changes that could be made in the existing valve to eliminate the observed instability. The vendor's conclusions and recommendations are as follows:

Conclusions: "Due to the high mass flow and extreme pressure drop, the characteristics of the 4" double port throttle plug are such that it tends to create instability at various points in its travels. It is felt that when instability is encountered, it is caused from a reversal of stem forces."

Recommendations: "Laboratory tests have shown that a V-pup inner valve is more stable through all points of travel, and we recommend that the current throttle plug inner valve be replaced with a V-pup, percentage type inner valve. We would further suggest that a size 60 Type 474 piston operator complete with Type 3570 positioner incorporating a snubber be installed."

Procurement of the valve parts as recommended by the valve vendor has been initiated, and delivery is expected by February 1, 1970.

Tests performed subsequent to 62.5-MWt operation demonstrated that the response time of the steam-bypass valve could be reduced by replacing the installed pneumatic pressure transmitter with a pressure-to-current (P-to-I) transmitter. Additional undesirable delay times associated with the system have been analyzed, and further improvement is possible by eliminating and/or replacing transducers within the control circuit. These changes have been identified and will be reviewed.

The steam-bypass system will be tested after the above changes have been made to verify that the system performance has been improved and is acceptable.

2. Main Turbine

The main turbine performed as predicted in Ref. 1. At a reactor power level of 62.5 MWt, the steam consumption of the turbine was measured to be 170,000 lb/hr. The electrical power produced was about 18,600 kW.

3. Condenser

The performance of the condenser was satisfactory during operation at 62.5 MWt. With the main turbine on the line, condenser pressure always was well below the design specification of 1-1/2 in. Hg abs. Condenser pressure increased approximately 1 in. of Hg when the turbine was shut down and steam was bypassed to the condenser.

Cooling water to the condenser always was below the design value of 75°F; therefore, performance of the condenser has not been demonstrated at the design conditions. Cooling-water temperature is a function of cooling-tower performance as influenced by local weather conditions. Weather conditions were quite favorable during the period of time EBR-II was operated at 62.5 MWt.

As expected, the noise level associated with operation of the condenser when bypassing steam was very high. In general, the sound pressure level was of the order of 90 to 110 db, depending on location in the power plant. Sound-protection equipment was used by personnel requiring access to the area. It will be necessary either to limit access to the area or to provide sound-protection equipment to all personnel who must enter the area during the rare periods when steam must be bypassed to the condenser at design power.

The range of the level controllers for the hotwell may have to be adjusted. At present, the controllers are set to maintain hotwell level within a 2-in. band. Apparently, bypass-steam operation at high power levels causes enough water turbulence in the hotwell to make the indicated level fluctuate more than the control band, thereby causing unnecessary opening and closing of the level-control valves. This effect has also been noted to a lesser extent when bypassing steam at 50 MWt. Adjustment of the control range is expected to minimize unnecessary operation of the level valve and to optimize control.

4. Condensate Pumps

Both the turbine- and motor-driven condensate pumps performed satisfactorily during 62.5-MWt operation. Each pump was capable of handling the total condensate load without evidence of cavitation.

Just before the 62.5-MWt run, flow tests were run on the condensate system to evaluate the overall performance of the system. Data from this test indicated that the flow rate to the small sprays in the desuperheater was much higher than the specified 18 gpm. Subsequent inspection of the spray nozzles revealed that the nozzle inserts had eroded. After the inserts were replaced, the flow was measured to be 18 gpm.

5. Feedwater Pumps

Each of the two installed feedwater pumps was capable of carrying the total feedwater flow at 62.5-MWt operation. The turbine-driven pump performed a little better than expected. At 62.5 MWt, however, there was little over-power margin left. The pressure in the turbine steam chest was the same as the line pressure, thus indicating that near-maximum steam flow to the turbine was attained. During operation, it was possible to obtain some operating margin by shutting off the pump-recirculation flow. At EBR-II, the turbine-driven pump is used only at power levels greater than about 30 MWt. At 30 MWt, the feedwater flow rate is high enough to make recirculation flow unnecessary.

The performance of the motor-driven pump was not quite as good as expected. At 62.5-MWt operation, the outlet pressure-control valve was wide open. This valve normally throttles the feedwater flow as necessary to maintain a 150-psi differential pressure across the feedwater control (boiler level) valve. With the pump's outlet pressure-control valve wide open, there was no measurable differential pressure across the feedwater control valve. However, there was sufficient head to provide the necessary feedwater flow to maintain boiler level when operating at 62.5 MWt. It was not possible to shut off or to reduce the pump-recirculation flow as was done with the turbine pump, because the motor-driven pump is operated at very low flow rates.

Operation of the installed feedwater pumps at 62.5 MWt was satisfactory. However, it appears desirable to provide some additional flow margin for both pumps. Means of providing the desired margin are being considered. For example, tests are planned to (a) establish that the recirculation is not excessive and (b) determine if the boiler-blowdown rate can be reduced. Consideration also is being given to installing an automatic recirculation-flow-control valve.

D. Cooling Tower

As described in Ref. 1, the pitch of the fan blades was adjusted before the start of 62.5-MWt operation. All the fan blades were adjusted to have a pitch of 12.5° . With this setting, the power consumption by each of the fan motors of the cooling tower was 75 hp, their design rating.

During the period of 62.5-MWt operation, the weather conditions were very favorable; hence, the cooling tower could dissipate 62.5 MWt without difficulty. A projection of cooling-tower operating data indicates that, with a wet-bulb temperature as high as 64°F , the tower will reject 62.5 MWt of heat, thereby providing a cold-water temperature of 75°F . Condenser design is based on a maximum cooling-water temperature of 75°F .

A review of the climatological data for the past year at the EBR-II location showed that wet-bulb temperature was never more than 60°F . Records for previous years indicate that a 60°F wet-bulb temperature was exceeded rarely. The maximum wet-bulb temperature recorded at the location was 67°F .

Based on the above, it is concluded that the cooling tower is extremely unlikely to limit the power level of the reactor plant to less than 62.5 MWt. The following events would have to occur simultaneously before this condition would be approached: (a) wet-bulb temperature in excess of 64°F ; (b) reactor operating at 62.5 MWt; and (c) all heat rejected to cooling tower (main turbine-generator inoperable).

Hence, the present performance of the cooling tower is considered acceptable for sustained operation of EBR-II at a power level of 62.5 MWt.

E. Electrical System

The EBR-II electrical power system is connected to the National Reactor Testing Station (NRTS) 138-kV electrical power through two main 12,000/16,000-kVA, 138-13.8-kV power transformers. In turn, the 2400-V and 480-V distribution systems are supplied through parallel feeders by two 2500-kVA, 13.8-2.4-kV and two 2000-kVA, 13.8-kV-480-V power transformers, respectively.

The kVA loads for the three distribution systems--13.8 kV, 2400 V, and 480 V--were measured at 50, 56, and 62.5 MWt. The temperature of the oil in the six transformers also was recorded at the different reactor power levels. Table V lists the maximum recorded kVA loads and the transformer temperatures measured during reactor run 38A.

No detectable load increase was measured on either the 480-V or 2400-V systems. The temperatures of the oil in the transformers remained well below the maximum allowable. A maximum load of 18,000 kVA was recorded on the 13.8-kV system at 62.5 MWt. This value represents a 25% load increase over the load for a reactor power of 50 MWt.

It is concluded that the electrical distribution systems are satisfactory for continuous 62.5-MWt operation of the plant if special consideration is given to loss of a 13.8-2.4 kV transformer or a 13.8-kV-480-V transformer. The loss of a transformer for either the 480-V or 2400-V systems results in the remaining transformer being loaded beyond its rating. This situation exists during reactor operation at either 50 MWt or 62.5 MWt. A plan of action is being formulated to cover the loss of either a 13.8-2.4-kV or a 13.8-kV-480-V transformer.

VI. OPERATING PROCEDURES AT 62.5 MWt

A. Revision of Operating Procedures

Prior to the power increase to 62.5 MWt in run 38A, special operating procedure No. SOP 103-69 was issued. This SOP contained necessary changes of the affected sections of the Operating Manual as well as revised checksheets and data sheets which reflected the plant parameters expected at higher power levels. Also included were special procedures to be

TABLE V. Performance of Electrical System at 56 and 62.5 MWt

<u>Reactor Power</u>	<u>Load on Distribution Systems, kVA</u>			<u>Transformer-oil Temperatures, °C</u>					
	<u>480 V</u>	<u>2400 V</u>	<u>13.8 kV</u>	<u>138-13.8 kV</u>	<u>13.8-2.4 kV</u>	<u>13.8 kV-480 V</u>			
	<u>No. 1</u>	<u>No. 2</u>	<u>No. 3</u>	<u>No. 4</u>	<u>No. 5</u>	<u>No. 6</u>			
50 MWt	2300	2700	12,500	33	34	54	56	60	72
62.5 MWt	2300	2700	18,000	40	40	55	58	60	73
Transformer Rated Capacity, kVA (total of two)	4000	5000	32,000						

followed if certain plant systems and equipment were unable to function adequately at the higher power levels.

During the power increase and subsequent operation at 62.5 MWt, it was found that the operational changes listed in SOP 103-69 were adequate and that only minor changes of the checksheets and data sheets were required to reflect actual values obtained.

The following summarizes those revisions of the Operation Manual, checksheets, and data sheets to be made and issued before any future 62.5-MWt operation.

(1) A revision of Section IIB (Normal Procedures) of the Operating Manual has been prepared to include provision for additional PRD measurements at 56 and 62.5 MWt. This revision also includes the procedures for setting the manual and automatic flux trips and the core-subassembly outlet-temperature trips.

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

(3) The list of operating setpoints (Appendix G of the Manual), interlock checksheets, startup checksheets, and operating log sheets have been revised to reflect the actual values obtained at 62.5 MWt along with the new trip setpoints for those values in the shutdown or alarm circuits.

(4) Two areas requiring special attention were mentioned in SOP 103-69 and in special procedures to be followed if the performance of systems or equipment in those areas were found to be inadequate at the higher power levels. One area was the capability of the turbine-driven feedwater pump to maintain sufficient flow, and the second was the capability of the condenser, condensate pumps, and cooling tower to handle full bypass flow with the turbine-generator shut down. Tests conducted during run 38A, however, showed that plant performance in these areas was satisfactory and that no revision of existing procedures will be required.

B. System and Component Surveillance

A thorough surveillance of plant systems and components was performed at both 56 and 62.5 MWt. In addition, close surveillance of each system and individual component was maintained during the actual power increases.

The reactor and plant systems operated as expected and satisfactorily during the power increases and with the power stabilized at 56 and 62.5 MWt.

During the power increases, close surveillance was maintained on the reactor ΔT , reactor outlet temperature, and the nuclear channels to ensure that the desired power level was not exceeded. The reactor ΔT reading was used to establish power and was verified by the reading of channel 7, the linear level channel. Recorded values of reactor ΔT and channel 7 were 183°F and $42.5\%^* \times 10^{-4}$ A, respectively, at 62.5 MWt. These values represent an increase of 25% in temperature and neutron flux as compared with the 50-MWt readings of a reactor ΔT of 146.4°F and $34.0\%^* \times 10^{-4}$ A on channel 7.

The subassembly outlet-coolant temperatures (as measured by 25 thermocouples located in the upper plenum to sense sodium temperature at the outlet of various subassemblies) all were linear with respect to reactor power. At the higher power levels, the actual temperature readings were very close to predicted values; none deviated from the allowable tolerances.

In summary, surveillance of the components and systems revealed no unusual or unexplainable condition at power levels above 50 MWt. Operation at 62.5 MWt proved all primary, secondary, and power-plant systems adequate. However, operation of the steam-bypass valves was not such as to provide a basis for routine operations. Testing and possible modification of the controller for these valves are scheduled and should improve their operability.

VII. CONCLUSIONS

Operation of the EBR-II reactor and power plant during run 38A, the first phase of the proposed 62.5-MWt operation, was satisfactory. Only minor difficulties were encountered; these are being resolved. This phase of the power-increase program demonstrated that continual operation at 62.5 MWt is entirely feasible.

* Of full scale.

Preparation for the second phase of 62.5 MWt-operation should proceed. Basic to the start of continual operation at 62.5 MWt is an official notification to all organizations using EBR-II as an experimental irradiation facility that such operation will commence on a given date. Accordingly, it is recommended that such an announcement be made. Experimental subassemblies now in the reactor can then be modified, as necessary, to be ready for continual operation at the higher power level.

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