Experimental
Breeder Reactor-II
(EBR-II)

An Integrated Experimental Fast Reactor Nuclear Power Station

By
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Authorized By
Argonne National Laboratory

This book is dedicated to my wife Rosemarie, whose support and patience permitted me to spend most of my more than 60 years of professional life in the pursuit and application of advanced cutting edge technology. First in the development of internal combustion engines and then in the development of nuclear power including its introduction, application, and hopefully the realization of its real potential. Most recently it has included the writing of this book, which is intended to assist in realizing that potential.

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FORWARD

The Experimental Breeder Reactor-II (EBR-II) and the associated pyrometallurical processing facility were remarkable engineering achievements of the last century. We are fortunate that the manager of the project, Leonard Koch, has written this first-hand account of the development, design, construction, and initial operation of this facility, which has contributed to the foundation of the knowledge for all fast reactors in the world. It captures the total process involved, including the very beginning when the basic concept resided in the minds of brilliant men. They had discovered the enormous amount of energy that could be produced by nuclear fission and had conceived of a concept for the controlled extraction of that energy. They had correctly recognized that it would be necessary to recycle the nuclear fuel many times to use it effectively and had calculated that this could only be achieved with fast neutrons.

Leonard Koch's early career at Argonne National Laboratory was exemplary as well. He was on the original team that designed and constructed Experimental Breeder Reactor-I (EBR-I). He participated when EBR-I generated the world's first useful electricity from nuclear power on December 20, 1951. His name appears on the EBR-I wall with Walter Zinn and his other colleagues.

This book on EBR-II is a detailed presentation of the design and construction of EBR-II, which is augmented with numerous original drawings and photographs, and thus will be of great use to the designers of future fast reactors. Leonard Koch was careful to explain why certain design choices were made while others were rejected. He also offers a section on how he believes future sodium cooled fast reactors should be designed, based on the experience gained with EBR-II.

Of general and historical interest, this book includes an appendix that traces the lineage of EBR-II, including original memos and meetings notes, beginning with Enrico Fermi and Walter Zinn and progressing to the formation of the EBR-II project.

We owe a debt of gratitude to Leonard Koch for interrupting his retirement to give us this excellent account of one of Argonne's greatest achievements. This book will be of great value for generations to come.

HERMANN A. GRUNDER, DIRECTOR
ARGONNE NATIONAL LABORATORY

PREFACE

The discovery of nuclear fission and of the self-sustaining, controlled fission in a nuclear reactor led to the development of the atomic bomb and to the recognition of the tremendous potential of this new highly-concentrated energy source. The magnitude of the energy potentially available from application of Einstein's theory was awesome; (i.e., while 1 pound of coal can produce approximately 3 kilowatt-hours of thermal energy, 1 pound of uranium can produce more than 10 million kilowatt-hours). Virtually all of the power plants that now operate in the world (all of them in the United States) extract less than 1 percent of this potential energy. Therefore, we now produce about 100,000 kilowatt-hours of heat energy from each pound of uranium we use (use, not consume). This, of course, is much more energy than we extract from a pound of coal, but we "waste" about 9,900,000 kilowatt-hours of the energy potential in each pound of uranium. A primary objective of the EBR-II project, which is described in this book, was to develop and demonstrate the technology that can provide the capability to extract much more of that energy.

The discovery of nuclear fission was greeted by intense enthusiasm by the technical community and the general public; perhaps over enthusiasm, which is not uncommon for new discoveries. "Atomic power" would become the universal energy source powering automobiles, airplanes, space travel, and producing "electricity too cheap to meter."

In the early 1950s, it was established that nuclear power should be a potential energy source for electric power generation and for submarine propulsion. The former, because it was demonstrated that large quantities of energy could be produced in production reactors, and it would be necessary only to increase the working temperature to accomplish energy conversion to electricity. For submarines, the unique advantage of a non-air burning requirement would permit virtually unlimited submerged operation of the ship at all power levels.

With general agreement that these applications should be developed, attention was focused on the options available and the capability to produce the desired results. The U.S Atomic Energy Commission initiated development of a variety of potential reactor concepts for electric power generation (and two basic concepts for submarine propulsion). EBR-II was one of the power reactor concepts selected for development. Two light water power reactor concepts were also selected—pressurized water and boiling water.

Light water reactors were given initial emphasis because they were considered "technologically easier" and could be developed more quickly, but EBR-II was based on the technology which would (and should) make nuclear power a very long-term energy source with a virtually unlimited supply of fuel.

Depending on the type of nuclear reactor and the fuel system used, the number of neutrons produced and their utilization will determine the conversion ratio of uranium-238 to plutonium. For example, in a uranium-fueled light water thermal neutron power reactor (virtually the only type of power reactor used in the United States), about one atom of plutonium is produced for each two atoms of uranium-235 fissioned while in a plutonium fueled fast neutron power reactor, about three atoms of plutonium can be produced for each two atoms of plutonium fissioned. The former type of reactor is known as a converter while the latter is termed a breeder (because it produces more plutonium than it consumes).

These basic characteristics of nuclear power reactors result in a utilization of less than 1 percent of the energy content of natural uranium in thermal converter reactors, when taking into account the natural uranium used to produce the "low enrichment" fuel for light water reactors; while the potential utilization of virtually 100 percent of the energy content of natural uranium is achievable in fast breeder reactors. In addition, this same level of uranium utilization can be achieved in fast reactors with "depleted uranium," (i.e., natural uranium from which much of the uranium-235 has been extracted to produce enriched uranium with uranium-235 content from about 1 percent to 90 percent for uses, including military and fuel for light water reactors). Achieving a high level of uranium utilization was basic to the Argonne concept of nuclear power. Although the Laboratory was involved in the development of other nuclear power concepts, the primary research and development efforts were directed to support this basic philosophy. The Argonne concept is based on the premise that nuclear power should be produced by "burning" natural uranium (depleted uranium as long as it is available) and is accomplished by converting uranium-238 to plutonium in fast reactors. In this process, plutonium functions as the catalyst for consuming uranium-238 and the reactor is fueled with uranium-238.

Processing nuclear fuel to properly produce the needed recycle products is a basic requirement for operation of a fast neutron power reactor system. One objective of the EBR-II was to demonstrate the integrated operation of the reactor power system and the fuel cycle system as a closed energy supply system.

Demonstrating the feasibility of a closed energy supply system was believed necessary to resolve many uncertainties: Could fuel recycle be accomplished in such a manner that the total fuel inventory in the system would be of acceptable size? Could the buildup of heavy isotopes (uranium-236, uranium-237, plutonium-240, plutonium-241, other actinides, etc.) be accommodated in a closed fuel cycle system? Since pyrometallurgical processes appeared to have the potential for application in such a system, but had the disadvantage of incomplete decontamination of the fuel, could a realistic closed fuel system accommodate the recycle of highly-radioactive fuel? The EBR-II program attempted to address these questions. In the process, the program goal included advancing the technology of fast neutron power reactors for long-term generation of electricity.

From the beginning, it was recognized that fast reactors would be significantly different from thermal (neutron moderated) reactors. The much smaller neutron cross-sections, both fission and capture, led to an entirely different geometry and design of these reactors. The fast reactor required high fuel density and relatively high fuel enrichment to achieve criticality in a fast neutron environment in which the fission cross-section is small. On the other hand, a broad choice of materials was permissible for reactor structures and coolant, also because of the small neutron capture cross-sections. Fast reactors were different and would continue to be different as they evolved.

Fast reactors are relatively insensitive to fission product buildup and their effect on reactivity of the reactor. This unique characteristic of fast reactors to tolerate fission product buildup made it feasible to incorporate into the EBR-II a fuel recycle process that did not remove all of the fission products. Although these fission products capture neutrons, the impact on the total neutron balance is insignificant.

A unique aspect of a liquid metal coolant is its high boiling temperature at atmospheric pressure. As a result, the cooling systems operate at essentially atmospheric pressure and the loss of coolant accident assumes an entirely different dimension. This characteristic permitted the EBR-II concept to include submerging the reactor and primary coolant system in the primary coolant and to devise the many benefits that resulted. Such a concept would be impossible in a high pressure cooling system such as water.

Although it was not originally recognized as a significant characteristic, the EBR-II concept is responsive to many of the concerns about nuclear fuel cycles which have developed since EBR-II. Weapons proliferation has become a major concern related to commercial nuclear power. The EBR-II fuel is naturally proliferation-resistant at all times in the cycle. New reprocessed fuel is still highly radioactive and the fissionable isotope, whether plutonium or uranium-235, is never cleanly separated from this highly-radioactive material. This results in an unattractive source of weapons material. As operations continue and the fuel continues to be recycled it becomes even less attractive as a weapons material source because of the natural buildup of the higher isotopes of plutonium and other actinides.

The use of on-site fuel recycle as pioneered by EBR-II, eliminates the shipment of irradiated material, thus making it far less accessible to would-be proliferators, and decreases the opportunity for theft or misappropriation. The absence of the need for shipment through public space eliminates the public safety concerns about shipping highly-radioactive or toxic materials.

All of these considerations contributed to the genesis of this book, but by far the most compelling was the desire to improve the utilization of uranium.

I have been driven by the conviction that much more than 1 percent of the energy contained in uranium must be utilized if nuclear power is to achieve its real long-term potential. It was correctly established by Enrico Fermi and others more than 50 years ago that this could be accomplished in fast reactors if the fuel was properly reprocessed and recycled repeatedly to extract that available energy (rather than store it as waste).

EBR-II was an early attempt to establish the technology needed to exploit this science. It was partially successful. I was among those that expected the pursuit of this needed technology would continue. Although this has not occurred, I am still of the conviction that it will. This book contains my effort to provide a basis for its continuance.

LEONARD J. KOCH

ACKNOWLEDGMENTS

Some time ago, John Sackett of Argonne National Laboratory recognized that after 30 years of operations and staff turnover, current program personnel were not well-versed in the history of the EBR-II. While preparing a series of seminars to the Argonne-West staff to respond to this situation, I found that, although much of the historical information existed, it was not readily available or usable. Descriptions of what was done were retrievable but information detailing why was essentially unavailable.

It quickly became obvious to me that the record of EBR-II was incomplete and the story of a great adventure had not been told. In response to John's urging, I volunteered to lead an effort to get this story told before it was lost. Time was passing, and the potential story tellers were becoming unavailable. After some discussion, it was decided that I would write only the early history of the EBR-II project (and others would complete the record later).

Even this more modest undertaking proved to be quite challenging. It might not have succeeded without the support, encouragement, and assistance of Argonne National Laboratory staff, particularly Leon Walters. I am also deeply indebted to the following people:

- Carey Walton and Gail Walters for direct assistance in locating historical material and draft preparation
- Wally Simmons, Ralph Seidensticher, and Ron King for their reviews of the draft copy of the book and their advice and constructive comments
- Members of the staff at Argonne National Laboratory-West for a variety of supporting activities
- Members of the staff at RED, Inc. Communications for editing and page layout support.

LEONARD J. KOCH

CHAPTER 1 — INTRODUCTION AND OVERVIEW

BACKGROUND

This book describes the history of the EBR-II during its development, design, construction, and initial operation. The operating history of the total power plant and fuel cycle, including its shutdown and closure are planned to be recorded in other documents. These activities occurred over a period of more than 40 years. During that period, changes in the technical and social/political environment influenced this history.

Early on, it was recognized technically that fast neutron reactors (unmoderated reactors) could utilize natural uranium and thorium most efficiently and that fast breeder reactors could utilize virtually all of the energy contained in uranium and thorium. Although the technical potential for high performance of the thorium–uranium-233 fuel cycle in fast reactors has been established, it has not been pursued actively in the United States and is not discussed further here.

Informally and unofficially there was general recognition and acceptance in the technical community that thermal reactors, primarily water cooled and gas cooled, were logical choices for near-term electric power generation; but for the long term, more efficient use of uranium would be necessary. It was in this setting that the EBR-II nuclear power system evolved and the overall concept of a fuel cycle and a power cycle for fast power reactors was developed.

This book also describes how the overall concept was developed to ensure that the knowledge, experience, and technology produced by EBR-II will be preserved and will be available to apply when this energy resource is needed.

ARGONNE NATIONAL LABORATORY

It would be incomplete to review the history of the EBR-II without also reviewing the organizational entity within which its development took place. The Argonne staff and management provided academic curiosity, superb scientific and technical capability, and a legacy of hands-on demonstrations of its ideas and concepts. The EBR-II concept evolved and materialized in this environment.

In the 1940s, there was general recognition by Argonne staff and management that atomic energy had tremendous potential, that its technical feasibility should be established, and the technology should be developed and demonstrated. EBR-I verified the theory of breeding, and the feasibility of operation of a liquid metal cooled fast reactor. This experience provided substance to the analyses that predicted the virtually unlimited potential of nuclear power. In late 1948, Enrico Fermi presented a seminar at Argonne in which he estimated the probable reserves of uranium and thorium in the world and converted the energy they contained to electric power. He concluded that these reserves could easily satisfy the United States electric power demand for several hundred years. Similar analyses by others produced similar conclusions.

During this period, the U.S. Atomic Energy Commission and the Joint Committee on Atomic Energy of the Congress were supportive of nuclear power development. The U.S. Atomic Energy Commission sponsored and supported (financially and technically) the development of various power reactor concepts. Argonne was involved in three programs:

- Technical support of pressurized water reactor development for the U.S. Naval Reactor Program

- Primary responsibility for developing and demonstrating the boiling water reactor concept by development, design, construction, and operation of the experimental boiling water reactor

- Primary responsibility for developing and demonstrating the liquid metal fast breeder reactor concept by development, design, construction, and operation of the EBR-II.

Although these programs proceeded essentially in parallel, there was general recognition of relative priorities. The U.S. Naval Reactor Program projected a near-term urgency. The boiling water reactor program had the objective of quickly advancing the basic technology of light water reactors to support early commercialization of nuclear power. While, the liquid metal fast breeder reactor program required long-range development to establish the ultimate capability of nuclear power. This judgement was verified by the relatively few years of operation of the experimental boiling water reactor, while EBR-II operated for more than 30 years (and additional operation could have advanced the technology even further).

EBR-II Organizational Structure

A unique aspect of this Laboratory activity involved the organizational structure which produced the EBR-II facility and its operation. There were multiple organizational structures, they overlapped, they interacted, there were multiple lines of authority, there were even divided responsibilities.

There were multiple organizational structures involved in EBR-II because development (including research and invention), engineering, design, and construction proceeded concurrently. Complicating matters further, oversight by the government was provided by three organizational units of the U.S. Atomic Energy Commission—Headquarters in Washington, DC, the Chicago Office of the U.S. Atomic Energy Commission, and the Idaho Office of the U.S. Atomic Energy Commission. Although the Chicago Office had oversight responsibility for engineering and construction, the Idaho Office administered the construction contracts.

The design and construction of EBR-II was accomplished by a temporary project organization superimposed on the permanent disciplinary organizational structure of the Laboratory where the supporting research and development proceeded concurrently. Many of the people in the project organization were also involved in the supporting research and development work. They participated in the development of the EBR-II technology and applied it to the design of the EBR-II facility.

As shown in Figure 1-1, the project organization was a relatively typical organizational structure reflecting the direct line project activities involved. The permanent organizational structure of the Laboratory (Figure 1-2) was directly and indirectly involved in the research and technical development support of the EBR-II project. This ongoing organization produced the technology which was incorporated into the EBR-II. This process involved continuing coordination and communication prompted by a common interest.

Contrary to established management concepts, this total organizational structure succeeded because the lines of communication were extremely effective. The project was faced with the usual requirements related to schedule and cost control while simultaneously depending on the development of the required supporting technology. The technology and concept were finalized concurrently by the normal compromises that must be made to achieve a conclusion. Some compromises resulted from a decision-making process where more than one option was available and which permitted a preferred choice. Examples of this process include the decision to locate the disassembly cell in the Fuel Cycle Facility rather than in the Reactor Plant. Another example was the use of mechanical pumps in the primary sodium system rather than direct current electromagnetic pumps (provisions were made to permit the substitution of electromagnetic pumps if necessary). On the other hand, some compromises were made because the technology was unavailable. For example, the EBR-II superheater concept could not be constructed because of the inability to make the unique tube-to-sodium tube sheet weld on the smaller diameter tubes.

A large number of people participated in both the EBR-II development program and the EBR-II project. Some on a part-time basis and some on a full-time basis. Some were involved primarily in design, incorporating the technology available or being developed, but most were involved in both because so much of the work required development and concurrent application of technology. This is typical of the development of a first of a kind product involving the application of very complex new technology.

The participants in the EBR-II project, including the development of the technology and the achievement of the EBR-II, are listed in Figure 1-3. It was not feasible to differentiate their participation because so much of it was dual. A separation was made between Fuel Cycle and Power Cycle, but even here there was some duality involving the fuel design and some aspects of fuel handling. Since it was not necessary to separate these activities to develop and accomplish the EBR-II, it is certainly not necessary to do so here. It is, however, essential to recognize the contributions and involvement of all of these participants. Finally, in the collegial

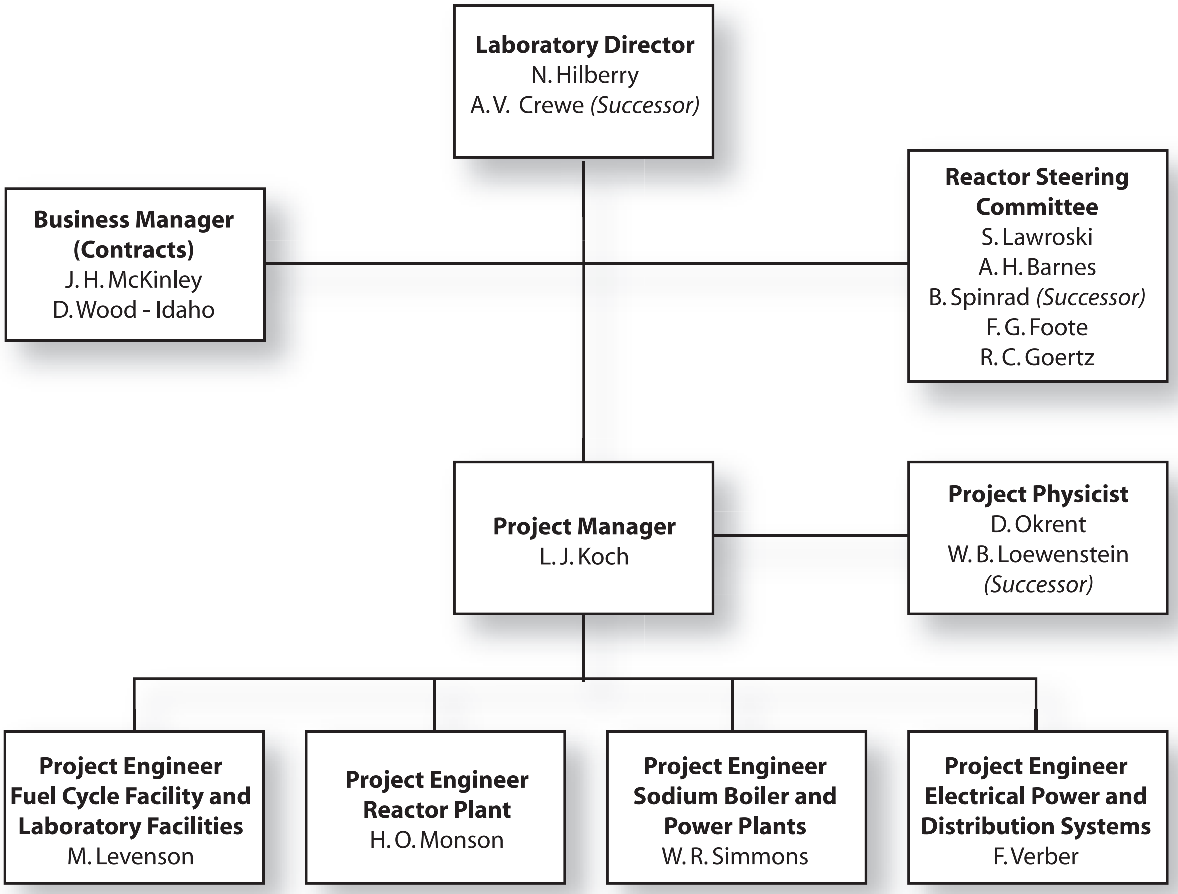


FIGURE 1-1. EBR-II PROJECT ORGANIZATION.

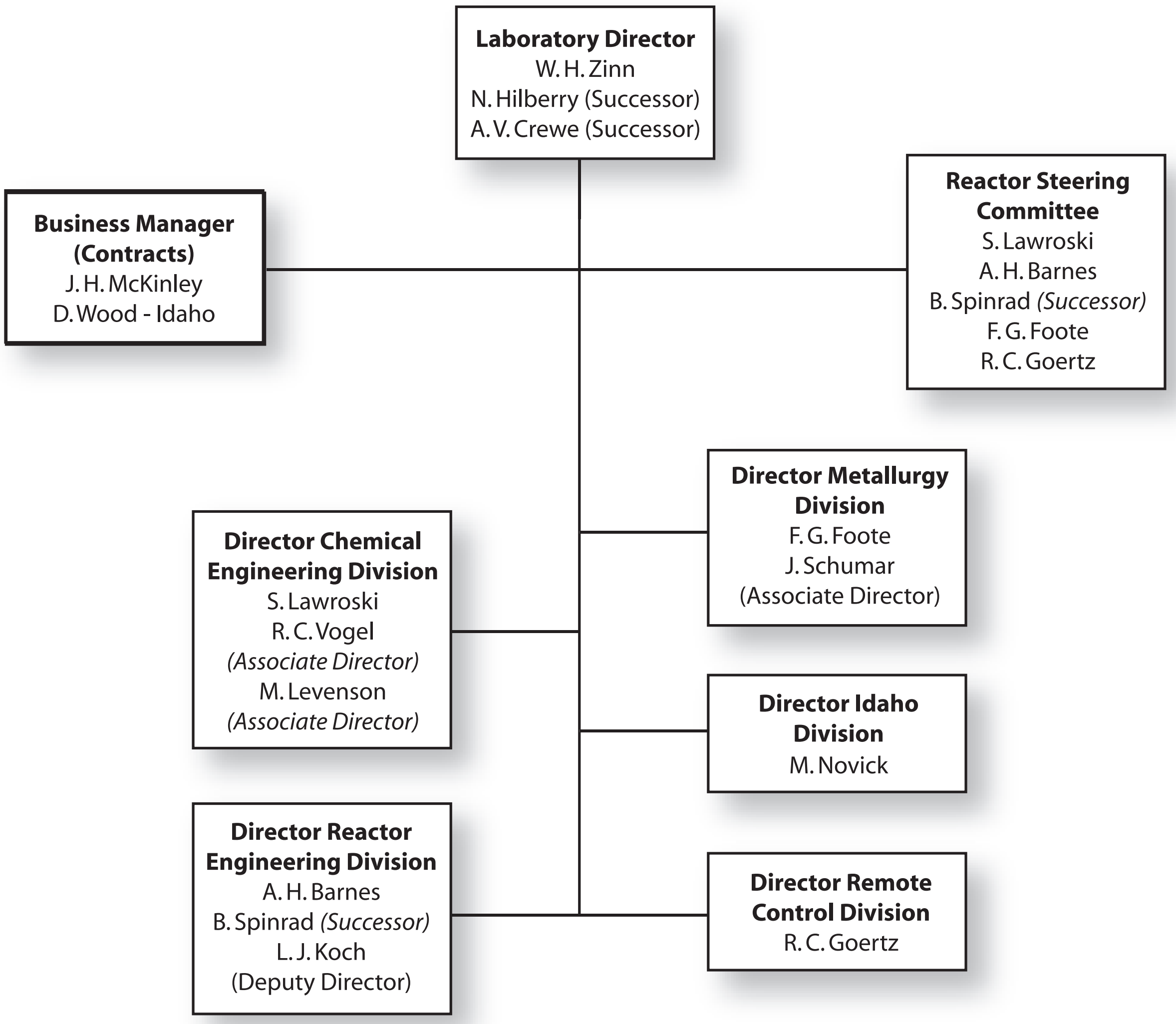


FIGURE 1-2. ARGONNE NATIONAL LABORATORY/EBR-II TECHNICAL SUPPORT ORGANIZATION.

FUEL CYCLE	**POWER CYCLE**
M. Levenson A. B. Shuck J. H. Schraidt J. E. A. Graae K. R. Ferguson	H. O. Monson W. R. Simmons F. Verber E. Hutter H. H. Hooker K. D. Kuczen G. K. Whitham W. B. Loewenstein

J. E. Ayer	N. C. McCown	T. R. Bump	D. Okrent
G. A. Bennett	W. J. McGonnagle	J. P. Burelbach	J. A. Pardini
L. Burris	W. E. Miller	L. Burris, Jr.	C. A. Pesce
T. C. Cameron	J. H. Monaweck	J. D. Carter	R. R. Prina
N. J. Carson	M. V. Nevitt	B. C. Cerutti	R. L. Ramp
A. A. Chilenskas	R. A. Noland	S. H. Fistedis	M. B. Rodin
L. F. Coleman	R. H. Olp	J. D. Geier	A. D. Rossin
I. Dillon	E. J. Petkus	J. R. Ginder	V. Rutkauskas
R. J. Dunworth	R. D. Pierce	H. A. Hummel	H. Sandmeier
T. W. Eckels	G. Pokorny	J. R. Humphreys	C. H. Scheibelhut
H. M. Feder	N. A. Rodger	A. R. Jamrog	R. W. Seidensticker
M. J. Feldman	J. P. Simon	V. C. Jankus	O. S. Seim
N. R. Grant	E. S. Sowa	R. A. Jaross	F. A. Smith
A. P. Grumwald	C. E. Stevenson	W. J. Kann	E. S. Sowa
D. C. Hampson	R. K. Steunenberg	J. Kaeuis	T. R. Spalding
J. C. Hesson	C. Stone	H. Kasinskus	C. Stone
W. F. Holcomb	F. J. Tebo	W. C. Lipinski	T. E. Sullivan
H. F. Jelinek	D. E. Walker	A. Lovoff	F. J. Tebo
J. H. Kittel	J. R. White	D. E. Lutz	D. E. Walker
J. P. LaPlante	F. L. Yaggee	A. E. McArthy	
J. Ludlow	S. T. Zegler	R. A. Noland	

FIGURE 1-3. EBR-II PARTICIPANTS.

environment of Argonne National Laboratory there was strong interest and technical support from people totally uninvolved with EBR-II.

The Laboratory was also responsible for operation of the EBR-II facility. This responsibility was assigned to the Idaho Division which was responsible for all Argonne National Laboratory activities at the National Reactor Testing Station in Idaho, including operation of all Argonne National Laboratory reactor facilities. The Idaho Division was the equivalent of the commercial power plant owner/operator and accepted the transfer of the EBR-II facility from the EBR-II project. This transfer differed from a typical commercial transaction involving a nuclear power facility, because the Laboratory retained responsibility for EBR-II throughout the total life of the project. However, a formal written transfer of technical responsibility for the EBR-II activity from the project organization to the Idaho Division was implemented by the Laboratory Director. At that time the EBR-II project organization was dissolved.

The Idaho Division interacted with the EBR-II project organization in a similar manner as would an owner/operator with the Nuclear Steam Supply System engineer/builder of such a facility. However, since all of the participants involved in the process were employees of the Laboratory, this interaction was relatively informal and extremely effective. The future operators were very interested in, and concerned with design features, particularly those which related to operability and maintainability of the facility. Their comments and recommendations were an important component of the design process.

The Idaho Division had primary responsibility for preparation for operation. This included preparation of operating and maintenance manuals, procedures, and training manuals, and preparing operations and maintenance personnel, including their training and qualification. Needless to say, this complex interaction required effective communication and cooperation. EBR-II was a first-of-a-kind unit and all of these activities required coordination. The project organization was responsible for the design, but utilized input from the operators; while the Idaho Division was responsible for operations, but utilized input from the designers to prepare for operation of this first-of-a-kind facility. Designers described how they expected their equipment to operate and to be operated. The operators identified improvements to design which would enhance operations. This process was accomplished both formally and informally. The formal process included review of, and comments on, specific features and details. The informal process consisted of direct discussion and coordination between the interested parties and was by far the most used process. It was possible and effective because there were no barriers between the participants, they were all employees of the Laboratory and had a single goal. Although the design did not require the formal approval of the operator, the designers had every incentive to produce a design which the operator liked. The primary incentive was to develop a facility in which all parties were comfortable since that would most likely produce the desired product.

The transition from project to operations was smooth and efficient. The coordination and cooperation was most evident and most intense near the time of turnover. Prior to turnover, there were many activities that involved both organizations. Of particular significance was the start-up and testing of components, subsystems, and systems. These were normally performed by teams consisting of members of both organizations with participation and primary responsibility reflecting the personnel requirements for the specific activity being performed. Those activities which involved operations-type activities were led by operations staff and operating personnel with assistance from project staff. These included filling the primary and secondary systems with sodium, installing subassemblies in the reactor, and conducting critical experiments. Those activities that involved the start-up of a component, where the primary purpose was to ascertain that the functional requirements were fulfilled, were led by the responsible design engineer with assistance from operations staff and/or operating personnel. These included pumps, control drives, and fuel handling components.

This capability and opportunity undoubtedly contributed to the very successful operation of EBR-II. It did, however, have one negative impact. Because so much of this work was accomplished relatively informally, it did not produce an extensive formal record. This loss has made it more difficult to reconstruct a detailed record of the history of EBR-II, 30 to 40 years after these significant events occurred.

EBR-II FUEL CYCLE

The EBR-II concept evolved around the fuel cycle. The primary objective of the overall concept was to achieve the fuel utilization, made possible by the breeding process. This approach to the development of a practical breeder reactor system produced the following basic objectives:

- The use of a high density fuel which minimized critical mass and enhanced the breeding characteristics of the reactor.
- The use of a fuel reprocessing system which recycled the fuel efficiently and quickly to minimize the total fuel inventory in the system.
- Demonstrate the feasibility of achieving a total plant operating cycle which required only the addition of uranium-238 to sustain plant operation.

The Metallurgy Division was encouraged in their development of metallic fuel because it provides the highest fuel density of the many possible fuel compositions, and their compatibility with sodium, the preferred reactor coolant was verified by EBR-I. The primary disadvantage of uranium metal fuel is its susceptibility to irradiation damage. However, early work with uranium metal alloys indicated that the irradiation damage resistance might be enhanced by the addition of small amounts of alloying materials.

At the same time, the Chemical Engineering Division was investigating pyrometallurgical processes for removing fission products from irradiated nuclear fuel. These processes had the advantage of being very compact and of recovering the fission products in a very concentrated form. They had the disadvantage of not removing all of the fission products, and therefore the processed fuel was highly radioactive. However, the primary fission product contaminants were noble metals (primarily ruthenium and molybdenum) which had the potential to act as stabilizing alloying agents in the uranium metallic fuel alloy.

These processes for enriched uranium metal fuel alloys were reasonably well-developed and understood. It appeared that they would be feasible for the initial operation of EBR-II. Further, development of pyroprocesses for recycle of plutonium-uranium metal alloys appeared promising, but required additional detailed development of processes and equipment. Of perhaps greatest significance was the fact that it appeared that the same basic facilities could be used, with different process equipment, to apply and demonstrate integrated fuel cycles with both fuel systems (i.e., enriched uranium metallic fuel alloy and plutonium-uranium metallic fuel alloy). Although there were many unresolved problems and uncertainties, there appeared to be a technical fit. The purpose of EBR-II was to make this a reality.

The EBR-II reactor and fuel cycle were developed on the basis of initial use of enriched uranium fuel alloy in the reactor and fuel cycle with the expectation of switching to a plutonium-uranium fuel alloy at a later date. This program was intended eventually to achieve the ultimate objective and demonstrate the integrated operation of power cycle and fuel cycle utilizing a plutonium-uranium-238 fuel cycle. The basic feasibility of manufacturing and assembling fuel assemblies using highly radioactive fuel materials needed to be established. These operations involved complex fabrication, manufacturing, and assembly procedures performed by remote control in heavily shielded facilities that required demonstration.

EBR-II may be unique in the development of nuclear power plants with respect to the influence of the fuel cycle on the overall plant design and operation, but it reflects the need to integrate the fuel cycle into the total operation for nuclear power plants operating on recycled fuel.

EBR-II POWER CYCLE

The required end product of this program was electricity. Therefore, one of the principle objectives of the EBR-II program was to demonstrate the reliable, efficient generation of electric power. Also, to the extent practicable with a small experimental unit, to demonstrate the delivery and availability under conditions comparable to those existing for commercial power generating stations. The EBR-II reactor and Power Plant were designed to achieve these objectives and did so for more than 30 years; even when the reactor was being operated as an experimental irradiation facility, the plant operated primarily as a base load power station.

However, a more fundamental requirement of the primary coolant system, beyond removal of the heat generated in the reactor for power production, was the removal of fission product decay heat after reactor shutdown. This is a requirement unique to nuclear power reactors and is proportional to the power density in the reactor when operating at power. Typically, fast neutron power reactors operate at high power density and shutdown cooling requirements are quite severe. The goal was established to achieve this passively, without the operation of active (powered) systems.

The primary technological emphasis on the EBR-II power cycle was directed to the reactor and primary sodium cooling system. It was recognized that very significant technical advancement from EBR-I and other early work would be imperative if EBR-II was to achieve its objective of advancing the technology to the interesting stage for future commercialization. Therefore, a relatively conservative approach was taken to the technology applied to the balance of plant for EBR-II. There was no effort to develop advanced power cycles or power equipment technology (i.e., the steam conditions selected were quite common for EBR-II size commercial fossil fueled units). On the contrary, the emphasis was placed on reliability and operability. It was thought that the EBR-II fuel cycle should demonstrate the potential for very low fuel cost. Thermal efficiency is less important in systems with low fuel cost; capital cost becomes more significant in establishing the cost of power produced. Also, since capacity factor impacts power generation economics, it was a desirable attribute for EBR-II to demonstrate high capacity factor.

EBR-II Initial Operation Objectives

Power operation of EBR-II began in August 1964 when electric power was first delivered to the National Reactor Testing Station distribution grid. This definitive phase of operation was preceded by a variety of preparatory operations and experiments.

This initial operation of the plant was in accordance with the basic intent and objective of operating a liquid metal cooled fast breeder reactor as an electric power generating plant operating on recycled fuel. Except for a relatively slow and extended start-up process of increasing power levels and examination of fuel (for evidence of irradiation damage), the start-up of EBR-II proceeded well. It operated as an experimental electric power generating station, delivering power to the National Reactor Testing Station 138 kilovolt power loop (about 15,000 kilowatt net power). After the start-up and approach to power phases were completed, the EBR-II plant operated for more than 30 years as a base load generating plant, most of it at its rated power of about 20 megawatt electric.

The initial phase of EBR-II operation lasted almost five years. During that time, the primary emphasis was on verifying and demonstrating the operating characteristics of the power cycle and the fuel cycle. Particular attention was given to the feasibility of the unique EBR-II fuel cycle, especially the processing and fabrication of highly radioactive fuel on site and the power performance of recycled fuel. These operations proceeded extremely well and much experience was gained in the process. The fuel was recycled through the reactor and Fuel Cycle Facility about four times; about 35,000 fuel elements and 400 fuel subassemblies were reprocessed and fabricated on site.

During this time, the U.S. Atomic Energy Commission revised the liquid metal cooled fast breeder reactor fuel development program and assigned essentially total support to the development of uranium-plutonium mixed oxide fuel for liquid metal cooled fast breeder reactors. This resulted in a major change in the EBR-II program. The reactor program was altered to accommodate irradiation experiments on mixed oxide fuels, and the Fuel Cycle Facility was converted to a “Hot Cell” for examination of irradiation experiments. However, because of its excellent performance, the reactor continued to be fueled with the basic enriched uranium/fissium alloy, but it was not recycled. Each fuel loading was fabricated from fresh, enriched uranium, alloyed to simulate recycled fuel, and the spent fuel was placed in storage. This 25-year supply of stored EBR-II spent fuel is included in the Department of Energy stockpile of recoverable material. Its recovery is a part of the EBR-II closure plan.

EBR-II was shut down on September 30, 1994, in accordance with an operational plan developed in response to requirements established by the Department of Energy. This plan incorporated provisions for maintaining the necessary operating systems in either an operational mode or standby mode, as appropriate.

CHAPTER 2 — DEVELOPMENT OF THE EBR-II CONCEPT

The EBR-II concept evolved slowly and along several parallel paths. With limited technology available — EBR-I was the state of the art at the time — developing a working concept required innovation and invention.

The EBR-II concept was strongly influenced by the limited availability of highly enriched fuel at that time. Military applications had priority for using highly enriched uranium and plutonium. It appeared that these materials would have limited availability and would be costly in the future. Consequently, the EBR-II concept was based on the need to achieve a high power density in the reactor and minimize the total fuel inventory by all means available.

EBR-II FUEL ELEMENT

It was evident that an entirely different fuel system was needed. To achieve a much higher power density required a different geometry. To maintain reasonable fuel temperatures, a smaller diameter fuel element would be needed. A smaller fuel element led to a fuel assembly concept that made the fuel package, "the fuel subassembly," the cornerstone of the power cycle.

A tentative target thermal power density of about 1 megawatt per liter of core volume was established for design purposes; the EBR-I operated with a power density of less that 1/6 of a megawatt per liter.

This target power density provided a basis for establishing the physical and thermal parameters that define the reactor core. Although sodium has excellent thermal conductivity, it has relatively low specific heat and requires the movement of a relatively large volume of coolant to remove the heat. These considerations resulted in a configuration that produced high heat flux from the fuel to the coolant, which in turn required a small cross section for the fuel element to maintain acceptable fuel temperature and a large cross section for coolant to provide the necessary flow volume to remove the heat.

Another parameter for a realistic design was coolant temperature rise through the reactor of about 200°F and maximum coolant flow velocity of about 25 feet per second. These reactor conditions coupled with a realistic and conservative steam condition of 850°F and 1,250 pounds per square inch, resulted in reactor operating conditions of about 700°F sodium inlet temperature and 900°F sodium outlet temperature.

Early on, a closely packed hexagonal geometry was selected for the EBR-II reactor configuration. EBR-I had demonstrated improved stability by incorporating cylindrical fuel elements in hexagonal subassemblies as a refinement and modification of the original design.

A series of iterations resulted in a 0.174-inch diameter fuel element and a pin diameter of 0.144 inches for the uranium alloy pin. Since the EBR-II concept was predicated on the use of recycled fuel from highly radioactive uranium alloy that was fabricated remotely, the feasibility of fabricating such small diameter fuel pins had to be established. The usual processes were totally impracticable in the high radiation, remote-controlled environment in which the fabrication would take place. The problem was resolved by the development of a casting process that produced a finished precision metal casting of the proper diameter that could simply be cut to the proper length.

Without this development it was quite likely that the EBR-II fuel recycle concept as conceived at that time would have been unsuccessful. However, subsequent operation of EBR-II demonstrated that the EBR-II fuel element design was overly conservative, and that the desired thermal performance could be achieved with a larger diameter fuel element, approximately 1/4 inch in diameter. This increased diameter would have made all of the fabrication steps much easier, and was being considered for the plutonium-uranium fuel cycle for EBR-II.

EBR-II SUBASSEMBLY

The small size of the individual EBR-II fuel elements required that they be handled and loaded into the reactor as a group or package. That package was the fuel subassembly and consisted of a hexagonal stainless steel tube containing:

- A cluster of 91 fuel elements
- An upper and lower blanket region
- A lower adapter for positioning and support in the reactor and elsewhere in the fuel cycle
- An upper adapter for attachment to various devices utilized to handle, transfer, and transport the subassemblies.

The EBR-II fuel subassembly is shown in Figure 2-1.

The freestanding subassembly concept represented a major departure from EBR-I, which incorporated a series of grids to support the fuel elements. The EBR-II reactor concept consisted of three different types of free standing subassemblies, positioned and supported only by a bottom grid-plenum structure.

The purpose of the subassemblies was to permit assembly of the proper amount of fuel, blanket, and structure in the proper configuration to constitute a nuclear reactor. Individual subassemblies containing properly configured materials were loaded into the reactor in a prescribed sequence.

Initially the reactor was manually loaded with 637 non-nuclear, stainless steel dummy subassemblies. After that, reactor loading was accomplished by removing one subassembly and replacing it with another to reach the desired loading. Similarly, after reactor operation was terminated in 1994, the reactor was defueled on the same replacement basis.

The lower grid plate of the reactor established the configuration of the reactor. Three patterns of different sized holes in the reactor grid matched the lower adapters of the core, inner blanket, and outer blanket subassemblies and produced the configuration shown in Figure 2-2. This figure also shows locations of control and safety rods that were smaller in cross-section by the equivalent of one row of fuel elements. The thimbles, or hexagonal tubes, in which the control and safety rods move vertically, have the same external hexagonal dimensions as the fuel and blanket subassemblies.

Two configuration features were incorporated into the EBR-II concept as demonstrations for potential application in large future liquid metal cooled fast breeder reactors. The first was a provision to prevent installing a subassembly into an incorrect reactor location. The fuel

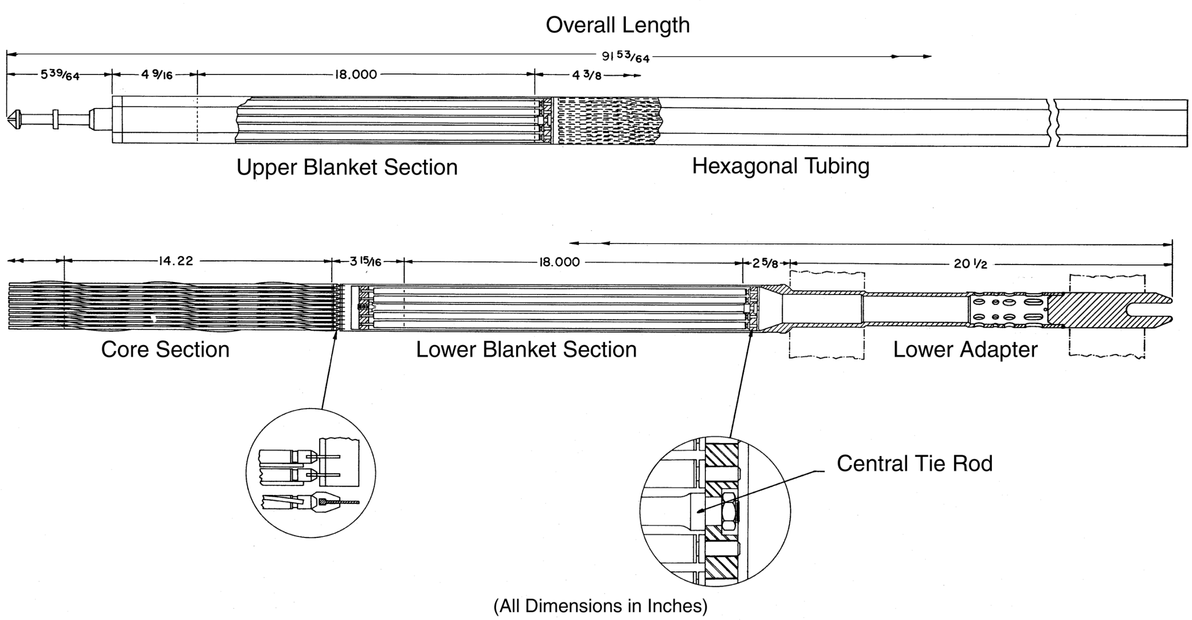
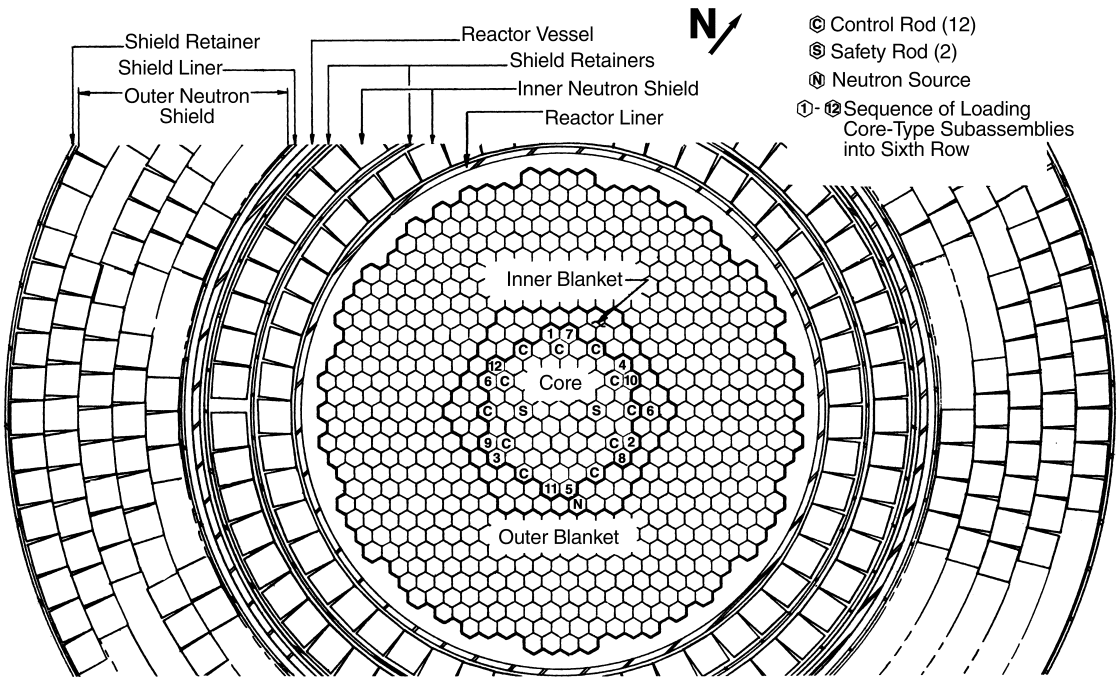


FIGURE 2-1. CORE SUBASSEMBLY (FINAL CONFIGURATION).



Note:
For nominal (67 subassembly) core, Nos. 1 through 6 are only loaded with core type, Nos. 7 through 12 with inner blanket type.

FIGURE 2-2. REACTOR ARRANGEMENT.

subassemblies had a lower adapter with a larger diameter than the other subassemblies and could not be installed in either of the two blanket locations. Similarly, the inner blanket subassemblies could not be installed in the outer blanket. In the opposite direction subassemblies could not be installed closer to the reactor center than their proper zone because the orientation bars, which engaged slots in the bottom of the lower adapters, become wider toward the center of the reactor.

The second configuration feature involved local coolant flow through the subassemblies to control coolant outlet temperature. The subassembly power density decreased as radial distance increased from the center of the reactor. The objective was to demonstrate that controlled coolant flow could be accomplished without placing individual flow control orifices in each subassembly.

The initial approach was to place slots (Figure 2-3) in the lower adapter of each subassembly and to provide steps in the lower grid plate of the inlet plenum to cover a part of the slot, and thus affect flow through slots and into the subassembly as shown in Figure 2-4. This did not work, but the concept was modified to place holes in the adapter rather than slots, as shown in Figure 2-1.

By making each subassembly type identical with identical inlet coolant holes at different elevations and providing steps in the lower grid plate for each row, the amount of coolant flow could be adjusted to correspond to the power generation in that row of subassemblies. This concept was applied to the five rows that constitute the reactor core and the two rows that constitute the inner blanket shown in Figure 2-2. Although this concept was of limited value in a small reactor such as EBR-II, it could be extremely useful in a large reactor and was incorporated and demonstrated for that reason.

Because of the wide variation in power density between the fuel subassemblies and blanket subassemblies, the sodium coolant system was divided into a high pressure system for the core and inner blanket and a low pressure system for the outer blanket.

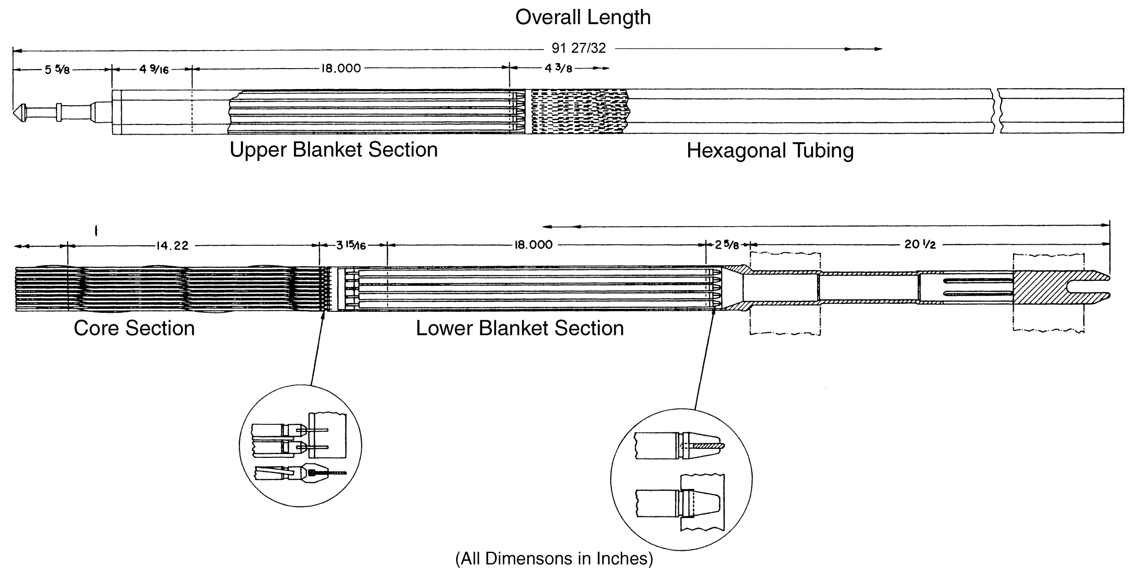


FIGURE 2-3. EBR-II CORE SUBASSEMBLY (EARLY DESIGN).

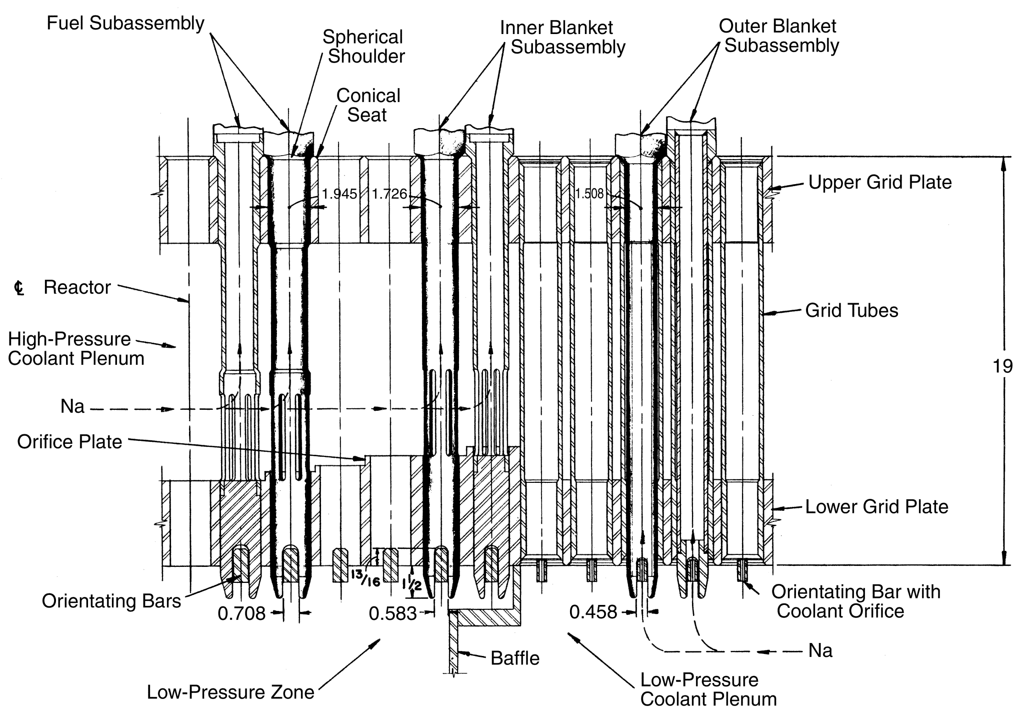


FIGURE 2-4. REACTOR SUPPORT GRID (EARLY DESIGN).

The pressure drop of the coolant flowing through fuel subassemblies was significant enough to lift the subassemblies. This was unacceptable during operation. Mechanical provisions to prevent such lifting could have been incorporated into the design, but there was a strong incentive to avoid incorporating any latches or locks in the assemblies. Stops above the subassemblies to prevent lifting would have had to accommodate thermal expansion of the subassemblies.

A non-mechanical solution was incorporated to permit the subassemblies to expand. The high pressure inlet sodium coolant plenum and the subassemblies were arranged to provide downward hydraulic pressure on the subassembly to offset the upward lifting force of the coolant flow. By introducing the inlet flow into the interior of the lower adapter, pressure was imposed on the closed bottom of the adapter. The hydraulic hold-down force, plus the weight of the subassembly exceeded the lifting force, and no other provisions for hold-down were needed.

EBR-I had demonstrated that mechanical bowing of fuel elements toward the center of the reactor caused by the power gradient across the reactor produced a positive power coefficient. In EBR-I the power density decreased across the diameter of the cylindrical fuel elements depending on their radial position from the center of the reactor.

EBR-II presented a greater potential for a positive bowing power coefficient. In EBR-II the bowing would be produced by a temperature gradient across the fuel subassembly hexagonal tube while the EBR-I temperature gradient was across the much smaller diameter fuel element tube. The EBR-I experience verified that the observed positive power coefficient was produced by a thermal mechanical effect and not a nuclear characteristic. Therefore, an EBR-II concept imperative was the requirement that thermal effects would not produce physical change that would result in a positive power coefficient.

The EBR-II physical and structural configuration was established by positioning and supporting the subassemblies in the grid-plenum structure that was unaffected by the power level of the reactor. This grid-plenum structure was at the temperature of the inlet sodium, which was at the temperature of the bulk primary sodium. Therefore, the thermal and physical effects had to be controlled above the fixed support structure. In response, the reactor was designed to enhance favorable thermal expansion and ensure that bowing would be prevented, or limited to an acceptable level.

The subassemblies of the EBR-II had to be replaced periodically, and therefore had to be movable. Any clearance for movement had to meet the requirements established to prevent subassembly bowing. Appropriate local clearance that satisfied the bowing requirement was achieved by incorporating a button on each of the six sides of the hexagonal subassembly tubes at approximately the vertical center of the reactor (Figure 2-5).

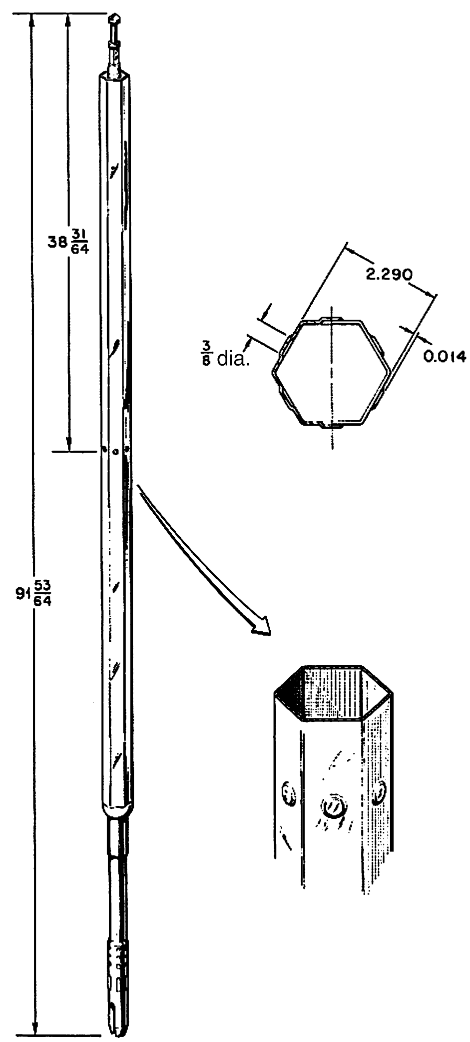


FIGURE 2-5. EBR-II SUBASSEMBLY-SPACER BUTTON DETAILS.

The nominal clearance between the buttons with the subassemblies in place was 0.002 inches. This small clearance produced a very tight structural configuration at the vertical mid-plane of the reactor. The subassemblies attempted to bow toward the center of the reactor, but because they were anchored at the bottom in the grid structure and were confined at the midpoint by the buttons, they tended to bow outward above the buttons.

This solution produced a favorable component of the power coefficient. It should be noted, however, that the unrestrained axial thermal expansion of the upper end of the subassemblies was made possible by the hydraulic holddown concept incorporated into the EBR-II design.

EBR-II Reactor Control

Leakage control was not a feasible option for EBR-II. The use of neutron absorbers was questionable because of nuclear performance uncertainties, but also because of the desire to demonstrate high neutron efficiency. Maximizing breeding ratio was an objective of EBR-II. The movement of fuel appeared to be feasible and was compatible with the basic EBR-II concept of reactor and subassembly. It was recognized that in the EBR-II reactor configuration, a guide would be required for any moveable unit in the reactor, which naturally led to the hexagonal thimble concept located by, and supported in the same manner, as all of the subassemblies. A fueled control rod which would fit in such a guide was made smaller than a fuel subassembly by one fewer rows of fuel elements. The reactor configuration would accommodate 12 such control rod and thimble assemblies.

All subassemblies were freestanding, supported only at the bottom in the reactor grid/plenum and were easily lifted during fuel handling. This arrangement for the control rod thimbles was not acceptable, since the vertical movement of the control rods could lift the thimbles. But it was imperative to incorporate the control rods into the basic EBR-II reactor concept of freestanding hexagonal-shaped containers for the required components comprising the reactor. To prevent movement of the control rods, the thimbles were locked into place by a latching arrangement at the lower end which was effected by rotating the thimble 60 degrees during installation (Figure 2-6). Since rotation of the thimble was prevented by the six adjacent hexagonal subassemblies, a special operational sequence was required to remove and replace a thimble. First, the six adjacent subassemblies were replaced by special dummy subassemblies, with the side adjacent to the thimble scalloped to permit the thimble to be rotated. An attachment had to be made to the upper end of the thimble, which was an open hexagonal tube, that simulated the upper adapter of a subassembly and provided the capability for the fuel handling and transfer machines to remove and replace the thimble. The thimble was replaced, after which the six dummy guide subassemblies were replaced by regular subassemblies. This was a tedious and time-consuming operation, but was required only rarely. Most importantly, it was accomplished without violating the basic requirements of the EBR-II fuel handling concept that there was never more than one vacant lattice position in the reactor at any time and that all components of the reactor consisted of removable, freestanding units. A similar procedure was used for the safety rod thimbles.

Subassembly as a Container of Fuel and its Transfer and Transport

The subassembly was a package in which the fuel elements could perform the function of generating heat while providing the physical capability for that heat to be removed and used productively. This function required the ability to install and remove the fuel subassemblies many times over the operating lifetime of the reactor. Because the EBR-II concept included fuel recycle, extraordinary and unique requirements were imposed on these activities.

In EBR-II, fuel handling consisted of removing and installing subassemblies in the reactor. The concept assumed that fuel handling operations could be required quite frequently, even as often as weekly. This aspect of the fuel handling concept was influenced primarily by uncertainty about the irradiation damage resistance of the fuel. But another influence was consideration of an operating strategy that might be favorable for commercial power generation — refueling the power reactor over a weekend when power demand was lower than during the workweek.

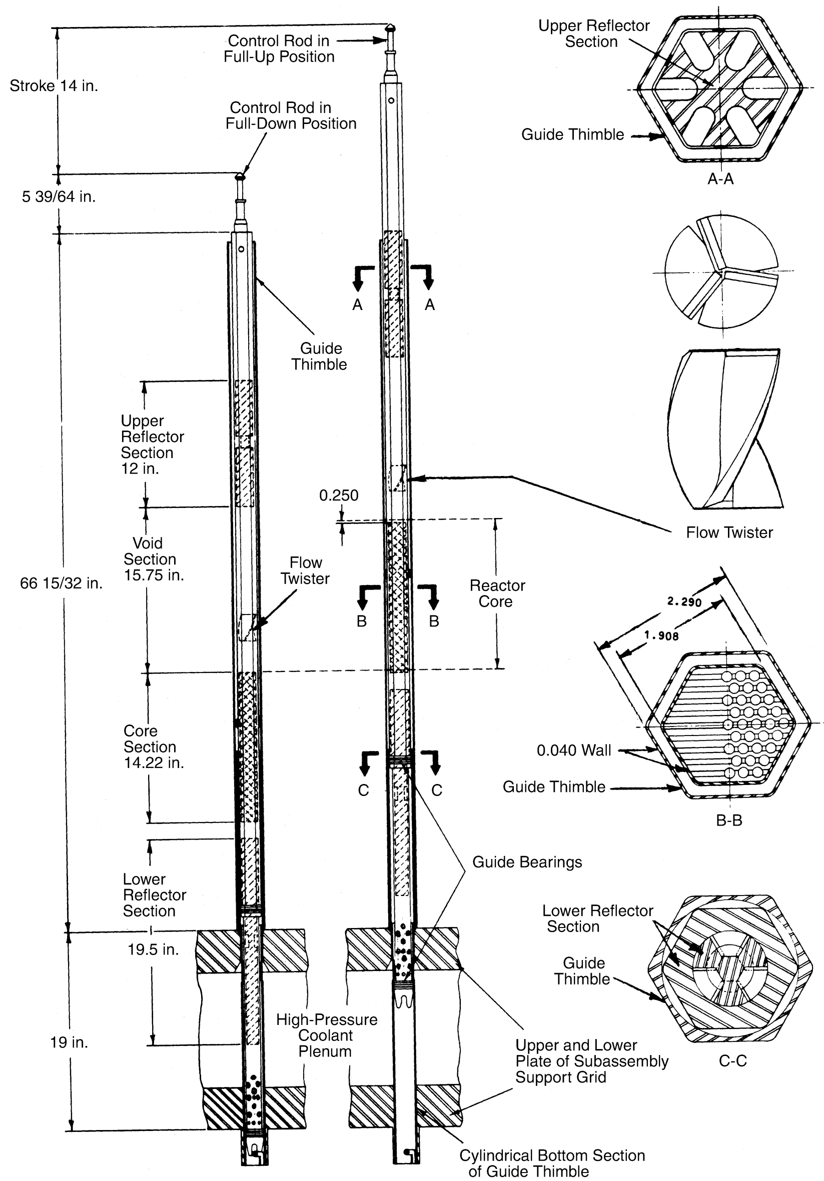


FIGURE 2-6. CONTROL SUBASSEMBLY.

The EBR-II fuel alloy proved to be very durable. After a series of design improvements, 10 percent fuel burnup was achieved routinely, significantly reducing the frequency of refueling. Nevertheless, the capability to perform very rapid fuel handling operations proved to be invaluable in supporting experimental programs.

The EBR-II fuel handling concept incorporated an intermediate storage capability in sodium because the subassembly could not be removed from the liquid sodium environment directly. A storage rack was provided in the primary tank. The operation of the reactor thus was made independent of the transfer and transport of subassemblies to the Fuel Cycle Facility. The time required to replace fuel in the reactor was minimized and passive heat removal was accomplished by natural convection of the sodium in which the subassembly remained submerged. Since the subassemblies continued to be cooled in the storage rack, reliable storage capability was provided indefinitely. This capability was consistent with the EBR-II reprocessing concept of cooling the fuel for as little as 15 days prior to reprocessing, or as long as desired.

The total integrated fuel transfer and transport concept between the Reactor Plant and the Fuel Cycle Facility involved much study and evaluation. Some of the key considerations involved how and when to make the transition from the sodium cooling environment and how to ensure continued reliable cooling of the spent fuel. Fission product decay heat removal occurred easily and reliably in sodium by natural convection, but it was clear that forced convection would be required in an inert gas environment. These cooling requirements would exist for the reprocessed fuel being returned to the reactor at all times that the fuel elements were clustered in the subassembly. Unclustered fuel elements would self cool.

The evolution of the fuel transfer/transport concept focused on the transition from sodium to inert gas coolant medium and the disassembly and assembly process involved in the transition of fuel elements between a tight configuration and a loose configuration. One of the early objectives in the evolution of the concept involved taking the subassembly apart quickly after removal from the sodium coolant in the primary tank.

To achieve this objective, a disassembly cell above the primary tank at the storage basket location was incorporated into the early design concept. In this arrangement the subassemblies were to be transferred from the storage rack directly to the disassembly cell and mechanically disassembled to remove the fuel elements as shown in Figure 2-7. (This concept was not used.)

Although this concept was retained well into the design phase, it was replaced by the final design because of concerns about possible impact on reactor operations and the advantages of physically separating the fuel cycle and power cycle operations.

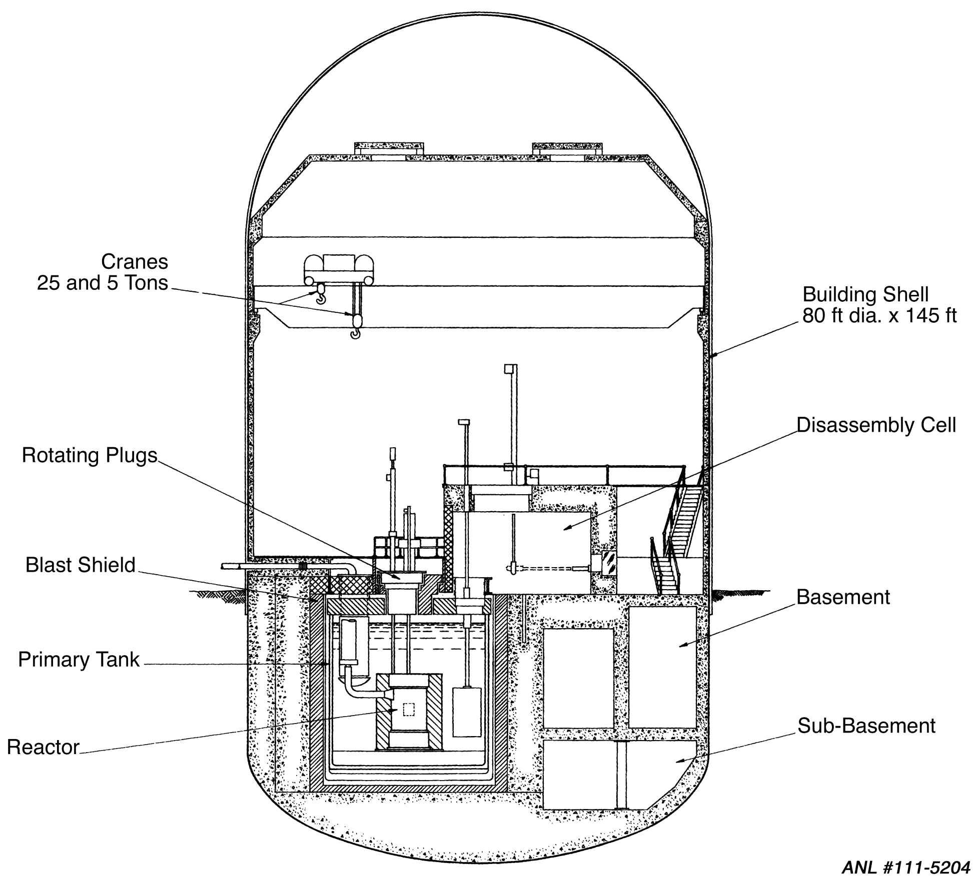


FIGURE 2-7. EBR-II REACTOR PLANT (VERY EARLY DESIGN).

An interesting consequence of this scenario was that the depressed floor area of the disassembly cell was retained and later provided valuable space for experimental systems and components. Figure 2-8 shows the design of the fuel transfer/transport concept after the disassembly cell was deleted.

Elimination of the disassembly cell resulted in the addition of the air cell in the Fuel Cycle Facility and the fuel unloading machine in the Reactor Plant. The cooling environment for the subassembly shifted from liquid sodium to argon gas as the subassembly was lifted into the fuel unloading machine. Forced circulation of argon gas was provided in the fuel unloading machine and in the inter-building coffin during transport and until the residual sodium had been washed from the subassembly components. At that point, heat removal was provided by forced circulation of air until the subassembly was opened and the fuel elements separated from the close packed tight hexagonal geometry. When separated, the fuel elements were cooled sufficiently by natural circulation of air. Because all of the fission products were not removed, decay heat removal was required for the reprocessed fuel. The same equipment and operations were used in the return of the reprocessed fuel to the subassembly storage rack in the primary sodium. Similarly, the same scheme was used subsequently when experimental subassemblies were returned to EBR-II after interim examination. The intermediate storage capability incorporated into the EBR-II concept made an integrated operation of two quite dissimilar operations possible and efficient.

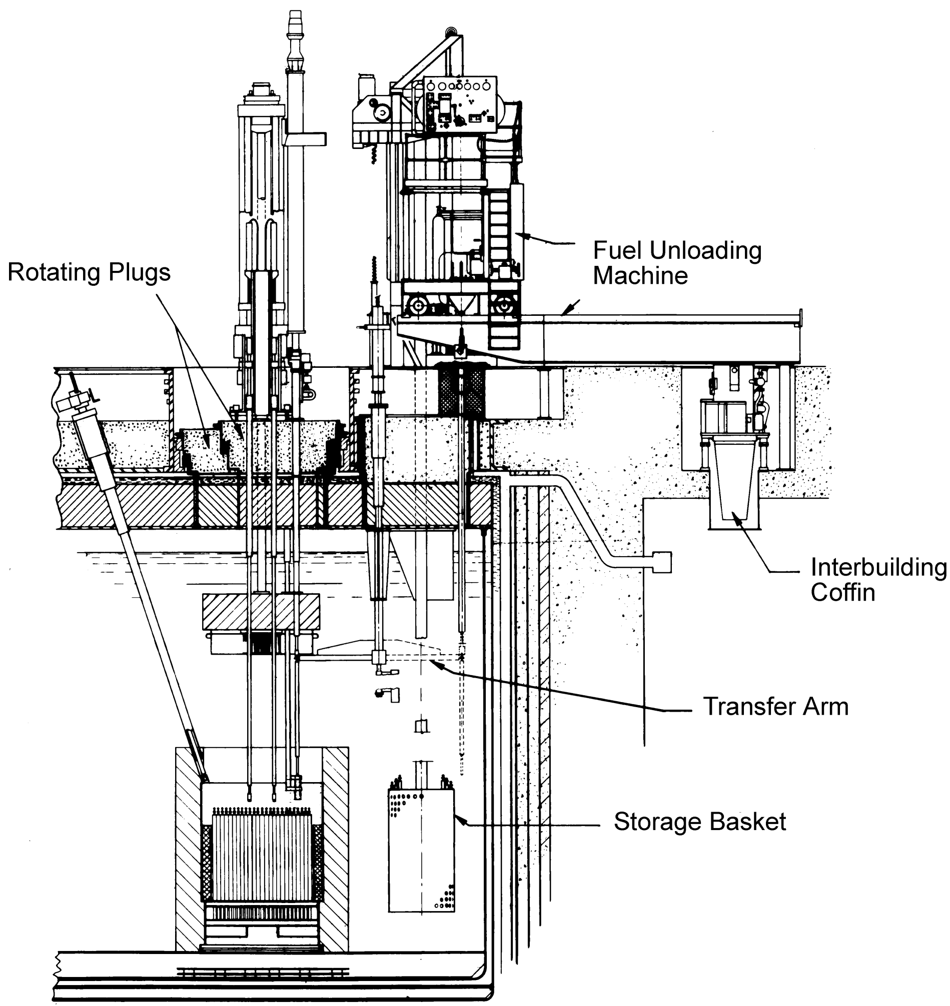


Figure 2-8. FUEL HANDLING SYSTEM WITHOUT DISASSEMBLY CELL.

The details of the design features of the various components comprising the total fuel handling, transfer and transport systems are described later, but the basic concept and its influence on the design can be summarized as follows:

- The same operational requirements, processes, and equipment would be capable of handling all of the subassemblies and related components that were installed in and removed from the reactor.
- Fission product decay heat removal would be ensured at all times during the process.
- At the appropriate times in the process, the subassembly would transfer from a sodium environment to a gas environment, with an attendant change in coolant.
- To reduce the impact of this transition in coolant medium, the EBR-II concept was based on 15 days minimum storage time in sodium coolant before transition to a gas coolant occurred.
- In the reverse scenario, when a subassembly was being delivered to the primary system, adequate preparation had to be made for the subassembly to accommodate immersion in 700°F sodium.

The EBR-II reactor concept introduced some unique requirements during fuel handling. The subassemblies were held in close-packed position by their weight and engagement in the grid. The subassembly involved in the fuel handling operation was lifted from the close-packed cluster of subassemblies. To address concern about the six subassemblies that surrounded the one being removed, a hold-down feature was added to the gripping and lifting sequence involved in removing the subassembly. The hold-down feature was augmented by a spreading feature to move the six surrounding subassemblies away from the one being removed (Figure 2-9).

Swelling of stainless steel and other distortions incurred during long-term residence in the reactor caused interference between some subassemblies. Extra force was required to remove these subassemblies and the hold-down spreading feature permitted the application of such force without jeopardizing the stability and reliability of the reactor configuration. Figure 2-10 is a photo of a removed subassembly that shows the result of interference with other subassemblies.

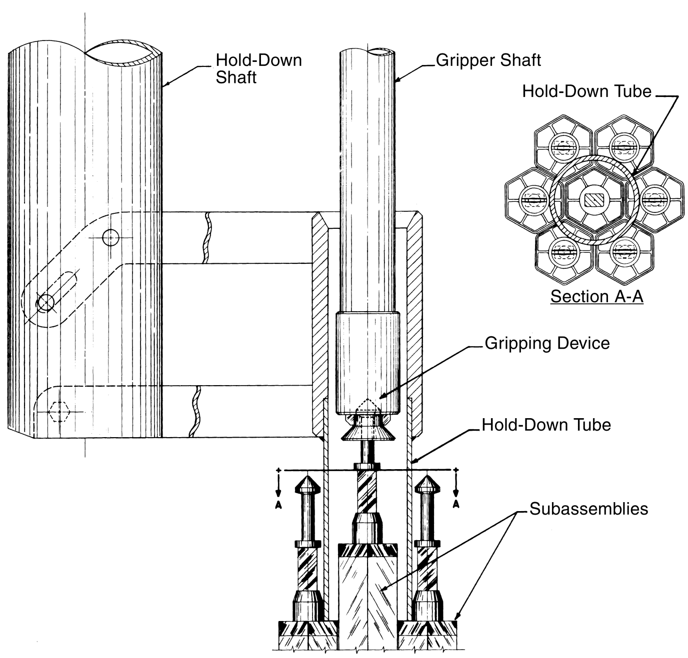


FIGURE 2-9. SUBASSEMBLY HOLD-DOWN AND GRIPPER.

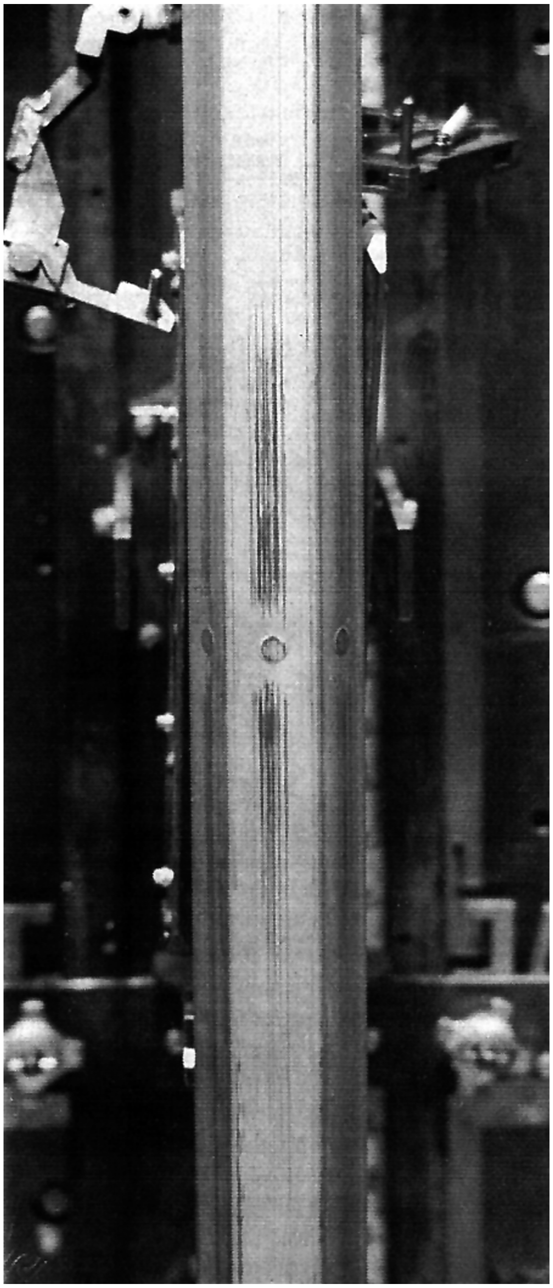


FIGURE 2-10. HIGH-BURNUP SUBASSEMBLY.

THE EBR-II REACTOR AND PRIMARY SYSTEM CONCEPT

In the EBR-II reactor concept, sodium coolant was provided to the reactor grid plenum by two mechanical centrifugal pumps from which it flowed through the subassemblies removing the heat generated by fission of uranium (or plutonium). The heated sodium flowed from the reactor to an intermediate heat exchanger where the heat was transferred to the secondary sodium system. This very simple flow system is shown in Figure 2-11. The primary sodium was radioactive because it flowed through the reactor and was exposed to neutrons. The secondary sodium was not radioactive.

The reactor and primary sodium system were contained in the primary tank and were completely submerged in sodium (see Figure 2-12). Principal benefits included:

- Fuel handling with intermediate storage
- Coolant system containment reliability
- A simple double-walled tank with no openings or penetrations below the sodium level
- Large capacity to absorb heat provided by the large volume of sodium at reactor inlet temperature.

This concept evolved as the additional, specific requirements for reliable heat removal were identified.

As these requirements were evaluated it became apparent that reliable fission product decay heat generation could be more demanding and critical than heat removal during power operation of the reactor. This characteristic resulted from the fact that heat removal during power operation was an on/off situation and could be turned off very easily. On the other hand, fission product decay heat generation continued irrespective of circumstances and could not be turned off. This heat had to be removed reliably at all times, under all conditions, and in all environments. The EBR-II concept required passive heat removal. The submerged primary system concept provided a direct and reliable capability to satisfy this requirement.

Refinements were developed to meet the various conditions that could exist after reactor shutdown. For example, conditions could exist in the secondary sodium and steam systems at shutdown that would affect the heat removal capability through the intermediate heat exchanger from the primary sodium system.

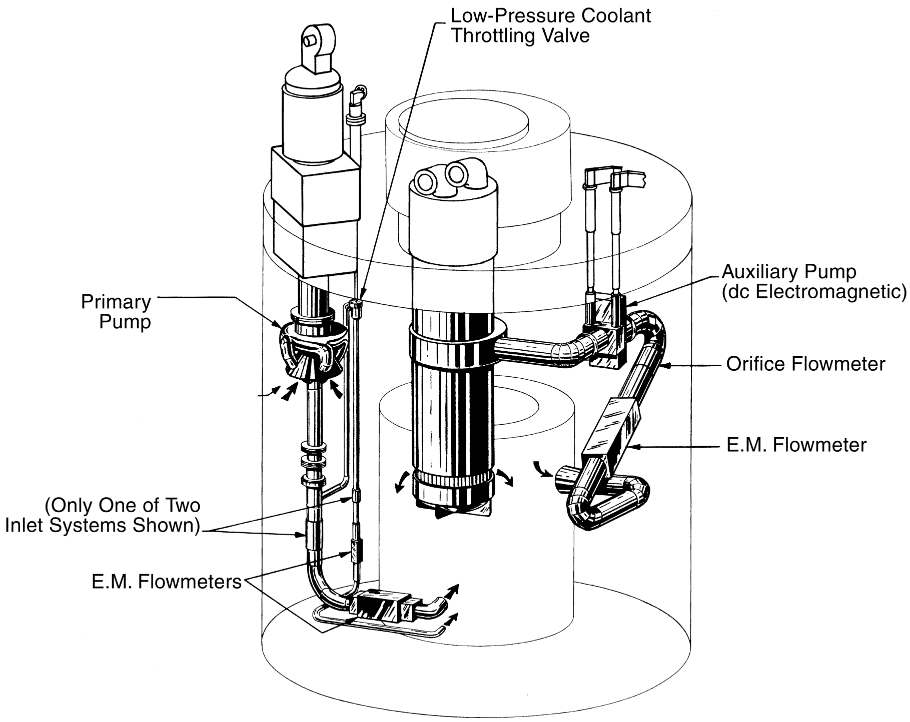


FIGURE 2-11. EBR-II PRIMARY PIPING AND COMPONENT ARRANGEMENT.

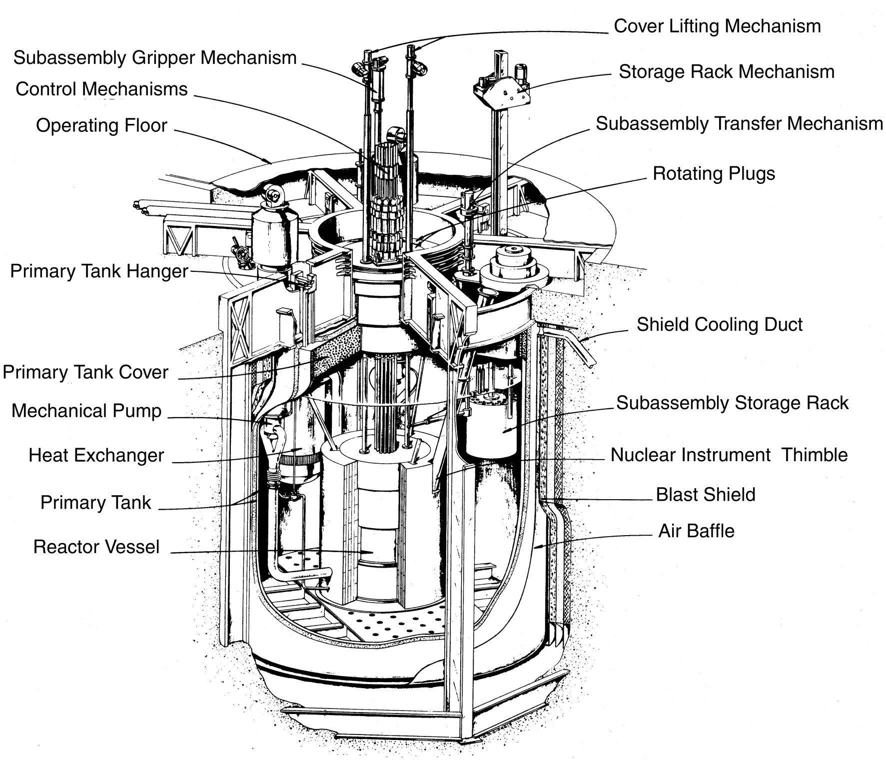


Figure 2-12. PRIMARY SYSTEM.

As a consequence, the absolute basic requirement for the EBR-II concept was to achieve fission product decay heat removal and shutdown cooling totally independent of heat removal by the secondary system. It required passive systems that would remove heat from the fuel and eventually transfer it to the atmosphere, bypassing the secondary system entirely. This was to be achieved without external power using natural convection of the liquid coolants involved and by natural convection of air.

Detailed analyses of a variety of reactor shutdown conditions identified situations which could jeopardize the initiation of natural thermal convection of sodium through the reactor. Under these conditions, the fuel could overheat before natural circulation of the sodium would begin. To avoid such situations, an auxiliary pump was installed in the outlet sodium line (as shown in Figure 2-11) which ensured low flow through the reactor at all times.

It included a direct current power supply to the pump but, in the event of failure of all power supplies, backup battery power would operate the auxiliary pump for at least 30 minutes before the batteries were discharged. This system ensured a reliable transition to natural convection circulation of sodium through the reactor no matter what sodium flow conditions existed at shutdown. This arrangement of the primary system ensured that under the most demanding circumstances the heat generated in the fuel by fission product decay would be removed and transferred to the bulk volume of sodium in the primary tank.

Under normal shutdown conditions, decay heat was transferred to the atmosphere through the secondary sodium system and the steam/feed water system. This transfer happened under controlled conditions that maintained the bulk sodium in the primary tank at the desired temperature.

Under abnormal conditions such as after shutdown where heat was not removed from the primary system through the normal power cycle, the heat was retained in the primary sodium. The 86,000 gallons of sodium provided a huge heat sink but the temperature would slowly rise if heat was not removed.

Two shutdown coolers were provided in the EBR-II primary system to remove this heat. Since this heat removal had to be provided reliably under all conditions, it was a passive system. The system operated by natural thermal convection, removing heat from the primary sodium and transferring it out of the reactor building to the atmosphere. The heat transfer medium was sodium-potassium eutectic alloy that was liquid at room temperature. (This alloy was the primary and secondary coolant for EBR-I for that reason.)

The sodium-potassium eutectic alloy flowed by natural thermal convection through a heat exchanger in the primary tank to an air-cooled heat exchanger in an air stack outside the reactor containment building. Heat was removed by natural convection of air.

To ensure reliable operation of this system, the shutdown coolers operated all the time. Continuous low heat removal was maintained by dampers in the air stack that restricted natural circulation of air through the stack. The damper was held closed by an electrically energized magnet. Upon receipt of a signal, or in the case of a power failure (a fail safe provision), the damper opened, the air flow through the stack increased by natural convection, and the heat removal rate increased (Figure 2-13).

The sodium-potassium eutectic alloy coolant circuit contained no valves and could not be shut off. The heat removal capability was reduced but not stopped by restricting heat removal from the sodium-potassium eutectic alloy to the air heat exchanger.

Figure 2-14 shows the temperature conditions that would result after reactor shutdown without heat removal from the primary sodium through the intermediate heat exchanger, both with one shutdown cooler and two shutdown coolers operating. Fission product decay heat removal would continue indefinitely and was not dependent upon a power supply of any kind.

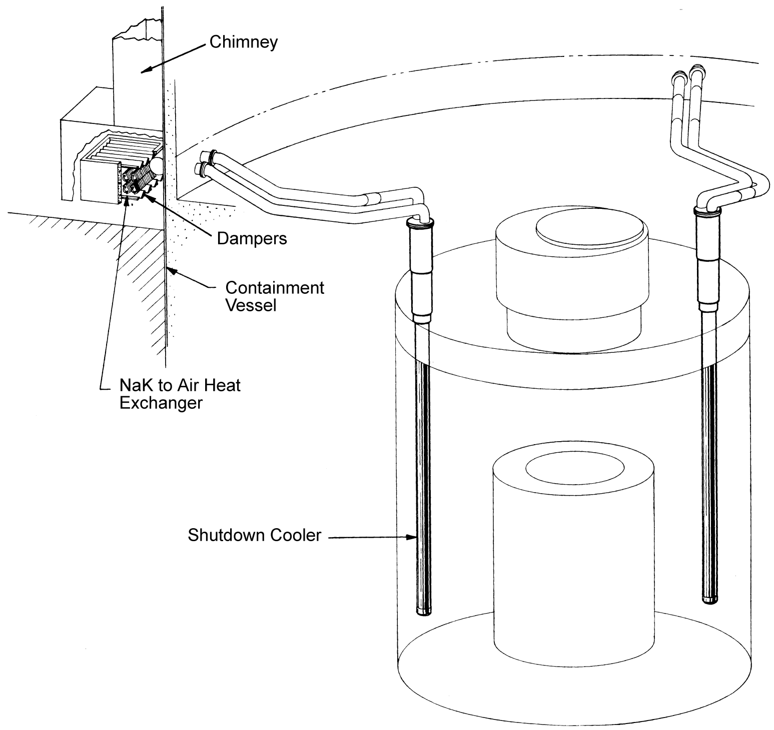


FIGURE 2-13. EBR-II SHUTDOWN COOLING SYSTEM.

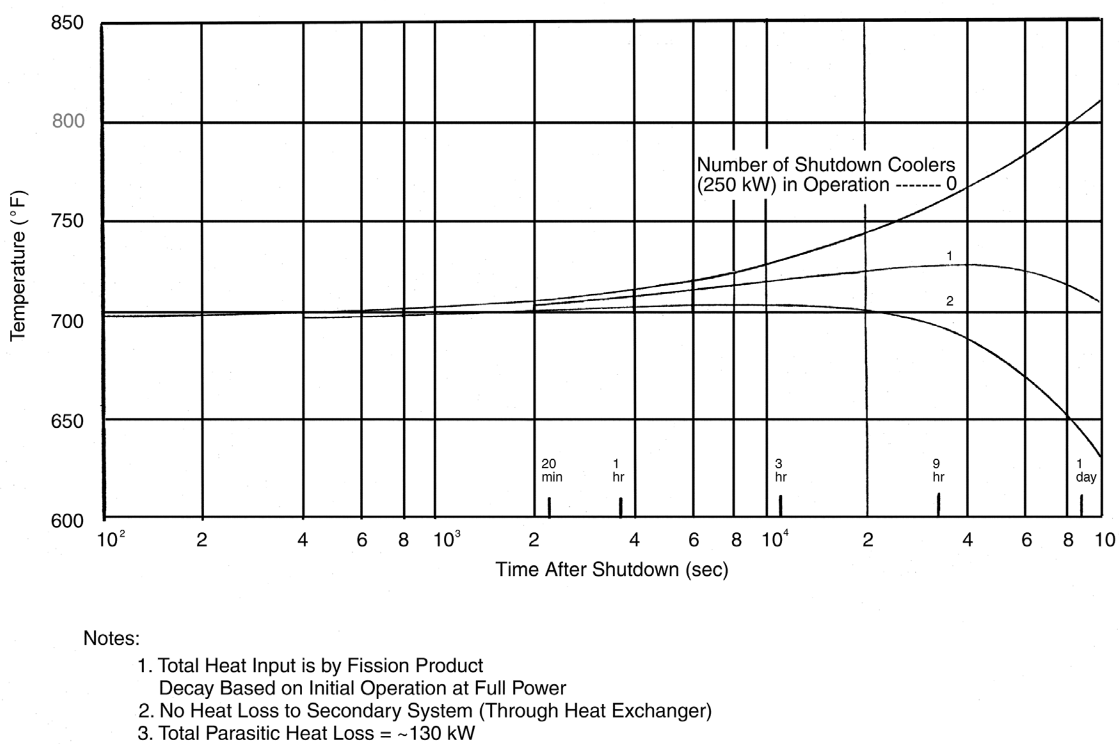


FIGURE 2-14. PRIMARY TANK BULK SODIUM TEMPERATURE VS. TIME AFTER SHUTDOWN.

The very simple and basic EBR-II cooling concept was also very versatile and satisfied a variety of normal as well as off-normal conditions. It was a simple two-step process:

- Remove heat from the fuel and transfer it to the primary sodium
- Remove heat from the primary sodium and transfer it to the atmosphere.

This process not only applied to the sequences described above, but also at all times when the reactor cover was raised, and the reactor outlet piping and intermediate heat exchanger flow system were bypassed. This condition existed during fuel handling and other times when the reactor was open. The basic requirement for this simple process was to keep the fuel, which was the heat source, submerged in the primary sodium, the heat sink.

HEAT REMOVAL, TRANSFER, AND UTILIZATION FOR POWER GENERATION

Although in many respects the EBR-II concept reflected the requirements imposed by reactor shutdown considerations, it also demonstrated the technical feasibility of utilizing a sodium cooled fast reactor as an energy source for generating electricity. Power cycle conditions and requirements were applied to the power system components, while simultaneously ensuring that they would meet the shutdown requirements. Emphasis was placed on reliability of operation and serviceability of components.

A few parameters were set on the basis of judgment and broad objectives. For example, there was a desire to operate with super-heated steam. Steam conditions of 850°F and 1,250 pounds per square inch were selected because they were typical for small plants at that time and the capital cost of associated equipment was favorable. EBR-II was based on the goal that fuel costs for liquid metal cooled fast breeder reactor power plants should be low, and therefore thermal efficiency was not a primary consideration. Capital cost and reliable efficient operation would be more important in evaluating fast reactor power systems.

These considerations led to a 900°F primary sodium outlet temperature. A temperature rise through the reactor of 200°F appeared achievable since the reactor core was only 14 inches high. At a thermal power level of 62.5 megawatt thermal to achieve 20 megawatt electric the other variables such as primary sodium flow rate, secondary flow rate, steam flow rate fell into place.

Although the operating parameters were conventional, many of the components comprising the power system were unique and imposed special requirements.

The intermediate heat exchanger was designed to permit complete removal of the tube bundle. The systems were arranged so that the pressure of the secondary sodium in the tubes of the heat exchanger was higher than the pressure of the primary sodium outside the tubes. This ensured that in the event of a tube leak, non-radioactive secondary sodium would leak into the primary sodium, and not vice versa.

Of necessity, the inlet and outlet secondary sodium lines had to enter the top of the heat exchanger. The inlet cold secondary sodium was directed through a central pipe to a plenum and tube sheet at the bottom of the intermediate heat exchanger. The secondary flow was up through the tubes and the primary flow was down outside the tubes in a conventional counter flow design. In a vertical unit this provided that the heated fluid was flowing up and the cooled fluid was flowing down, the correct arrangement for sustaining natural thermal convection circulation.

The intermediate heat exchanger was not removed during the operating lifetime of EBR-II, but during construction it was verified that it could be removed. The intermediate heat exchanger and the permanent primary sodium piping were installed relatively early in the construction sequence of primary system component installation. The installation of the intermediate heat exchanger tube bundle is shown in Figure 2-15.

The intermediate heat exchanger tube bundle was then removed and stored. The large open nozzle for the intermediate heat exchanger was used as the personnel access to the primary tank during the installation of the balance of the primary system components (Figure 2-16). The final operation to close the primary tank involved the permanent installation of the intermediate heat exchanger.

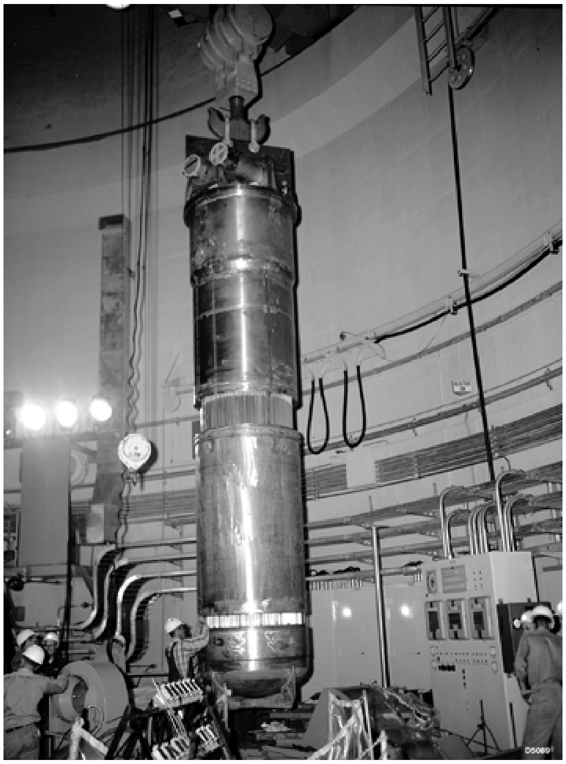


FIGURE 2-15. INSTALLED IHX TUBE BUNDLE (PRIOR TO INSTALLATION).

EVOLUTION AND IMPLEMENTATION OF THE SUBMERGED PRIMARY SYSTEM CONCEPT

The EBR-II submerged primary system concept evolved rather slowly; it was not discovered or invented in a spectacular stroke of genius. The process of identifying and evaluating operating characteristics of liquid metal cooled fast breeder reactors produced a variety of potential concepts.

The needs to achieve high power density for power operation and to accommodate the consequent high fission product decay heat were critical. The considerations involved in the use of sodium on a large scale as a heat transfer fluid were also a major factor in developing the concept. There was very little applicable experience available and the process involved the evaluation of ideas without the benefit of background experience or knowledge. Even those concepts based on more conventional systems required application of undeveloped technology.

Superimposed over all of these considerations was the recognition that this revolutionary reactor concept would require successful demonstration to achieve acceptance. Reliable, predictable operation was a mandatory objective of the project. All the options were evaluated on this basis and, even though a radical concept evolved, the process was conducted very conservatively.

To enhance the achievement of reliable plant operation, reliability, and serviceability of major components were extremely important. Major components and systems were placed into two basic categories: removable and non-removable. The non-removable components were expected to have a lifetime equivalent to the operating life of the plant, or the plant had to be able to operate without them.

As the reactor and primary system concept developed, a non-removable, permanent system evolved consisting of the primary tank, the reactor structure, and the primary sodium piping from the pumps to the reactor and from the reactor to the outer shell of the intermediate heat exchanger. The permanent system did not include the pumps or the intermediate heat exchanger tube bundle; they were removable. The reactor structure consisted of the lower grid-plenum, the cylindrical shell, the lower and radial neutron shield, and the reactor cover. The reactor cover fell into a somewhat different category because it was moveable, but not readily removable. All of the non-removable components were permanently attached to the inside of the primary tank.

The primary tank contained all of the components comprising the reactor and primary sodium system—non-removable as well as removable. Not only did the primary tank contain all of the primary system components, but it also contained the 86,000 gallons of primary sodium. The primary tank was double walled, with inert gas in the annulus between the two tanks. The outer tank was insulated. There were no penetrations or openings in the vertical cylindrical section or the bottom of either tank. All openings and penetrations into the tank were through the top cover. The bulk sodium level in the tank was maintained more than a foot below the underside of the top cover, and therefore there were no penetrations below the sodium level.

The primary tank was hung from the top structure by six hangers equally spaced to permit radial expansion of the tank. The early design of the hangers consisted of a double hinge arrangement shown in Figure 2-17.

FIGURE 2-16. OPEN NOZZLE FOR THE IHX (LOOKING UP).

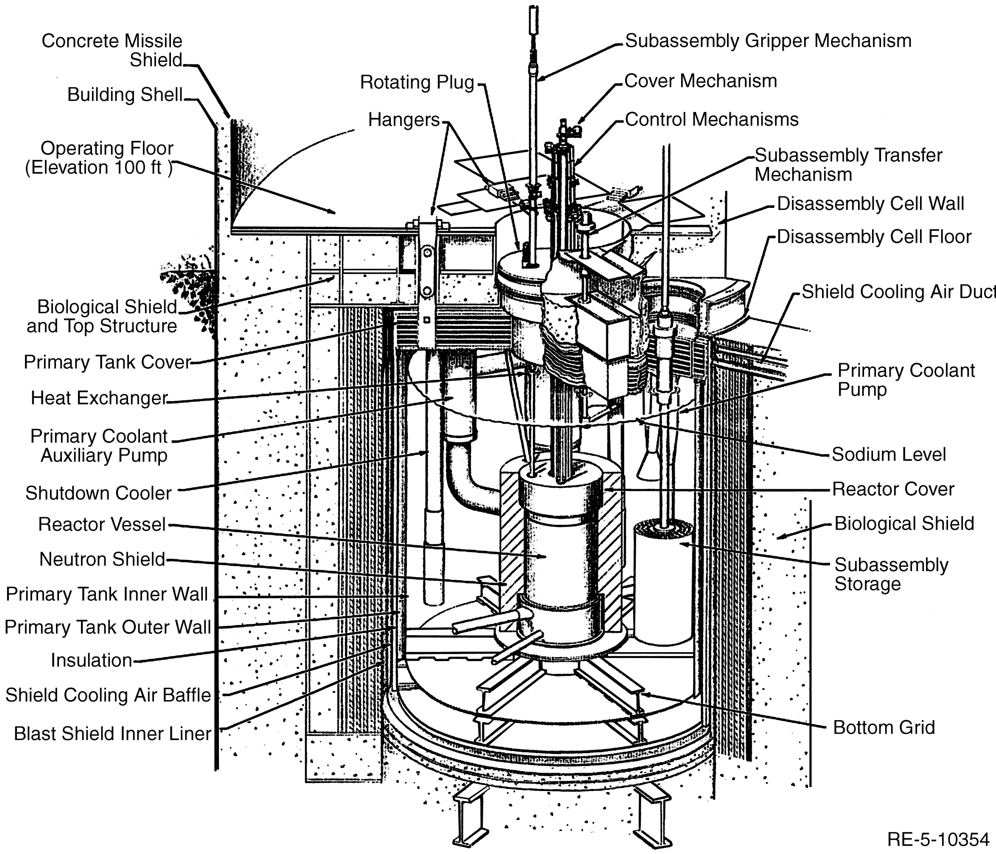


FIGURE 2-17. EBR-II PRIMARY SYSTEM (EARLY DESIGN).

This design was superceded by a roller hanger arrangement shown in Figure 2-12. The roller design permited inspection of the moving parts and measurement of movement during change in temperature of the primary tank. The rollers and support plates were actually removable and replaceable. The inspectability and replaceability of this design represented a significant improvement in reliability, even though no need for repair or replacement arose during the 40 years operating lifetime.

The primary tank and other major components were supported by a symmetrical structure shown in Figure 2-18. It was designed to not only support the total weight suspended from it, but also to survive a high energy release in the reactor and primary system.

The non-removable components, except the intermediate heat exchanger shell, were supported on the bottom of the primary tank. These loads were carried by the cylindrical section of the inner tank wall of the primary tank and the bottom of the inner tank. The details of this lower support structure and a cross-section of the reactor grid/plenum structure are shown in Figure 2-19.

Since the primary tank was hung from the top, it expanded vertically and radially as the bulk sodium temperature increased, and vice versa. This expansion occurred slowly because of the thermal capacity of the bulk sodium that established the tank temperature.

Because of thermal expansion considerations, the most position-sensitive components, such as the reactor and related control and fuel handling components, were positioned at and around the center of the primary tank. Less position-sensitive components were located out from the center, but provisions were made to accommodate movement resulting from thermal expansion.

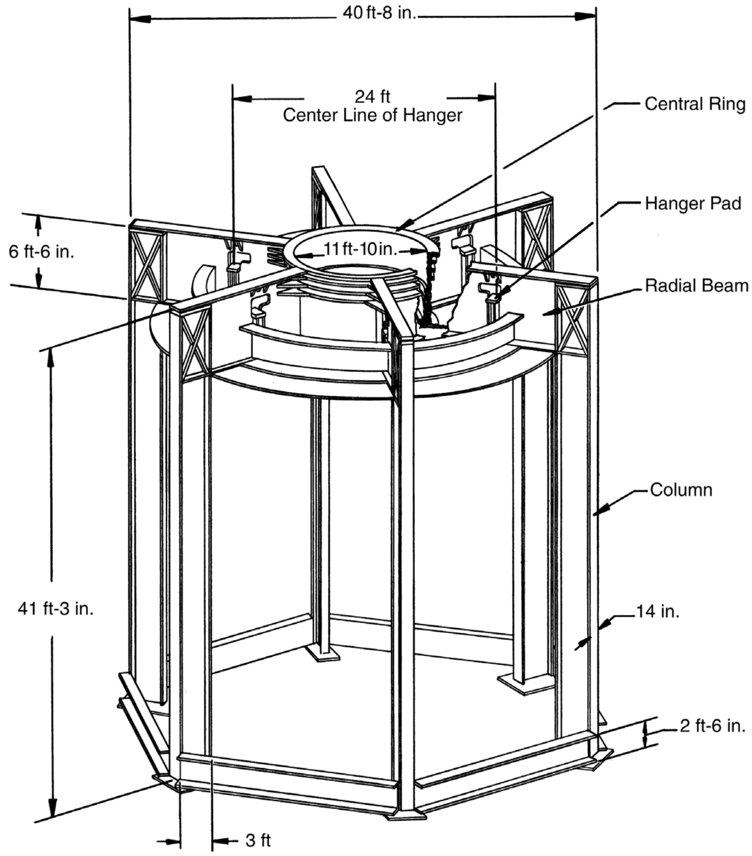


FIGURE 2-18. PRIMARY TANK SUPPORT STRUCTURE.

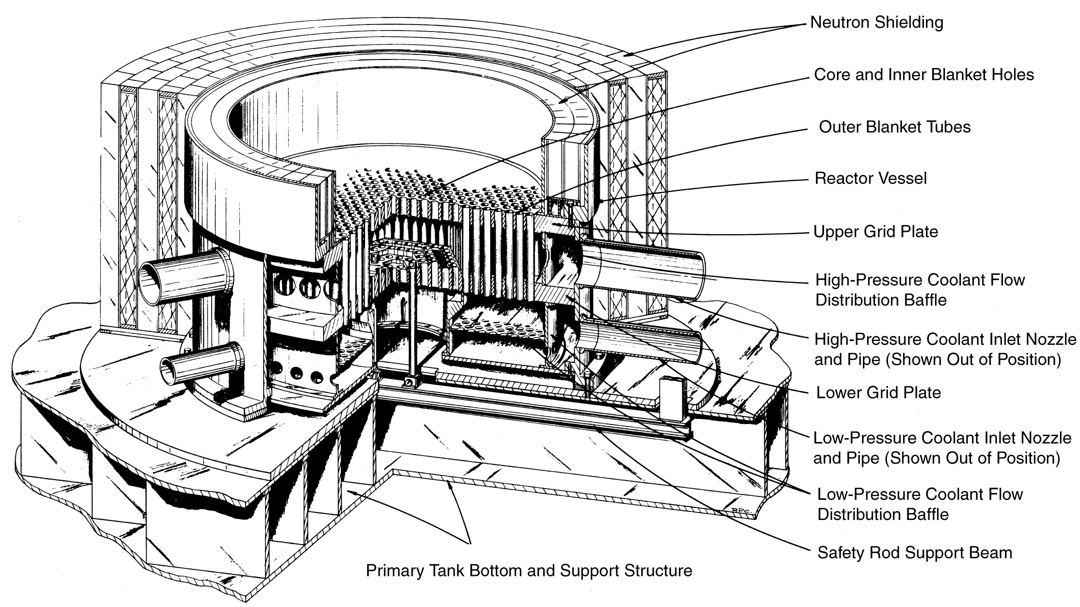


FIGURE 2-19. PICTORIAL OF REACTOR VESSEL GRID ASSEMBLY.

As can be seen in Figure 2-12, the top cover of the primary tank incorporated a large number and variety of penetrations. Most of these housed cylindrical components such as instrument thimbles and were readily removable. Of special interest were the pumps and intermediate heat exchanger, which involved extensive operations for removal. To accommodate pump removal, which in a conventional system involved cutting pipes, a mechanical ball and seat detachable connection was developed as shown in Figure 2-20. Although this was not a leak-tight joint, the leakage was permissible since the sodium leaked into the bulk sodium, which was the intake supply to the pump.

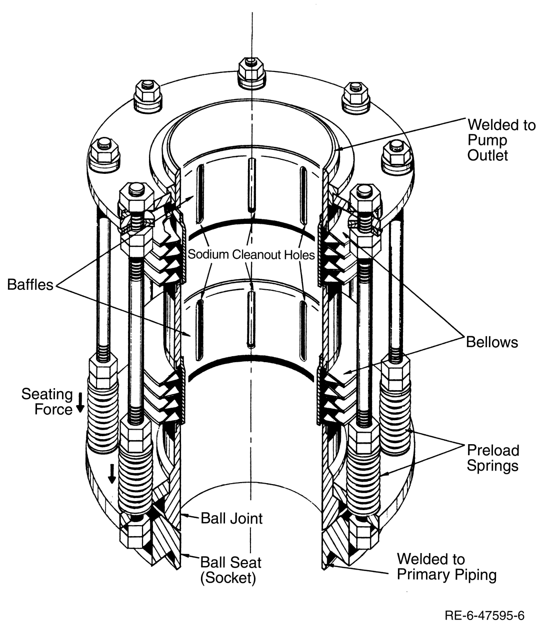


FIGURE 2-20. BALL-JOINT CONNECTOR.

An interesting aspect of the EBR-II concept development involved the primary sodium pumps. At the time, there was very little experience with mechanical pumps and much concern about their reliability. EBR-I employed direct current electromagnetic pumps and early development of alternating current electromagnetic pumps was very promising; the U.S. Naval Reactor Program supported much of this development. The original EBR-II concept incorporated direct current electromagnetic primary sodium pumps. Tests were conducted to verify that operation submerged in sodium was feasible. Their major drawback was the requirement for very high current at extremely low voltage.

During this period, an advance was made in mechanical pumps with the development of the hydrodynamic bearing, which was being used to pump fluids with poor lubricating qualities. This type of bearing proved to be successful with sodium, and mechanical pumps were selected for use in the EBR-II primary sodium system. However, as a precaution, the rectangular shaped penetrations in the top cover were retained so that the mechanical pumps could be replaced with direct current electromagnetic pumps if necessary.

The submerged primary system concept was a radical departure from conventional piped system arrangements and was even more revolutionary because the fluid in which the system was submerged was high-temperature sodium. There was much concern about the feasibility of the concept. The concept evolved because it provided a positive and effective response to two basic requirements:

- Absolute reliability of reactor cooling, particularly for all possible scenarios for fission product decay heat removal
- A realistic process for refueling the reactor in spite of the requirement that these operations be performed in a very difficult and hostile environment.

The absence of extensive coolant piping, the compactness of the system with resultant minimal radiation sources, and system compatibility with a unique fuel cycle were other benefits that contributed to reliability.

The concept became more acceptable, and then preferable as the first two requirements were met and then other benefits were identified. One benefit frequently overlooked was that sodium leaks could be accommodated because they returned the sodium to the system.

Because there was no applicable experience to draw upon as the EBR-II concept was developed, the process really became one of addressing a series of “what ifs.” As scenarios evolved, program teams evaluated virtually every conceivable application of Murphy’s Law. Interestingly, this iterative process served to strengthen the conviction that the system concept could work.

The next chapter focuses on the application of the exhaustive concept planning to the EBR-II systems and components.

CHAPTER 3 — DESCRIPTION OF EBR-II SYSTEMS AND COMPONENTS

This chapter describes the EBR-II. It consisted of four major, integrated, functional systems:

1. **THE PRIMARY SYSTEM** — the reactor and associated equipment, and the primary sodium cooling system. The energy source and the heat removal system.

2. **THE SECONDARY SYSTEM** — the intermediate sodium heat transfer system. The non-radioactive heat delivery system, and the heat source for steam generation.

3. **THE STEAM ELECTRIC SYSTEM** — a conventional superheated, condensing turbine-generator system, which provided the end product — electricity.

4. **The Fuel Recycle System** — the system for decontaminating and manufacturing the nuclear fuel.

The first three systems comprised the power system. The heat produced in the reactor was removed by the primary sodium system and transferred to the secondary sodium system in the intermediate heat exchanger. From the secondary system, the heat was transferred to the steam system in the steam generator to produce superheated steam, which was then delivered to a conventional condensing turbine at 850°F and 1,250 pounds per square inch gauge. A simplified flow diagram of the power system is shown in Figure 3-1. A temperature-enthalpy diagram is included as Figure 3-2.

These systems were housed in four plants and supporting facilities and structures as shown in Figure 3-3. The plants were designated as follows:

The Reactor Plant contained the reactor and primary sodium cooling system and supporting services to these facilities. It consisted of a containment building designed to contain any accidental release of radioactive material within the building. It was interconnected to the Fuel Cycle Facility and the Power Plant.

The Sodium-Boiler Plant contained the entire secondary sodium system, including the steam generator, except for the piping to the Reactor Plant and the intermediate heat exchanger, which was installed in the primary tank. The building had two wings — the sodium wing and the boiler wing. It contained unique features reflecting the incompatibility of sodium and water/steam.

The Sodium-Boiler Plant was somewhat isolated within the system complex. It was linked to the Reactor Plant by 75 feet of sodium lines and to the Power Plant by 200 feet of steam and condensate lines. The building contained only the minimum facilities for operation and was not normally occupied by operating personnel.

The Power Plant contained the turbine generator and associated equipment and the control room for the reactor and power cycle. It was interconnected to the Reactor Plant by means of one air lock to permit personnel access to the Reactor Plant. The building was of conventional construction.

The Fuel Cycle Facility contained two shielded cells for disassembly, processing, and manufacture of fuel elements and subassemblies, and supporting facilities for these operations. It also contained the inert-gas storage facilities, the sodium equipment cleanup cell, and exhaust ventilation system and the stack for the exhaust from the Fuel Cycle Facility and Reactor Plant. It was interconnected to the Reactor Plant.

An additional building, the Laboratory and Service Building, located adjacent to the Fuel Cycle Facility provided supporting analytical facilities for control of the fuel cycle and process operations. It also provided facilities for personnel and supporting services.

PRIMARY SYSTEM

The Primary System (Figure 2-12) was housed in the Reactor Plant and included the following:

- Reactor
- Subassemblies
- Reactor Vessel Assembly
- Primary Cooling System
- Shutdown Cooling System
- Neutron Shield
- Counters, Chambers, and Instrument Thimbles

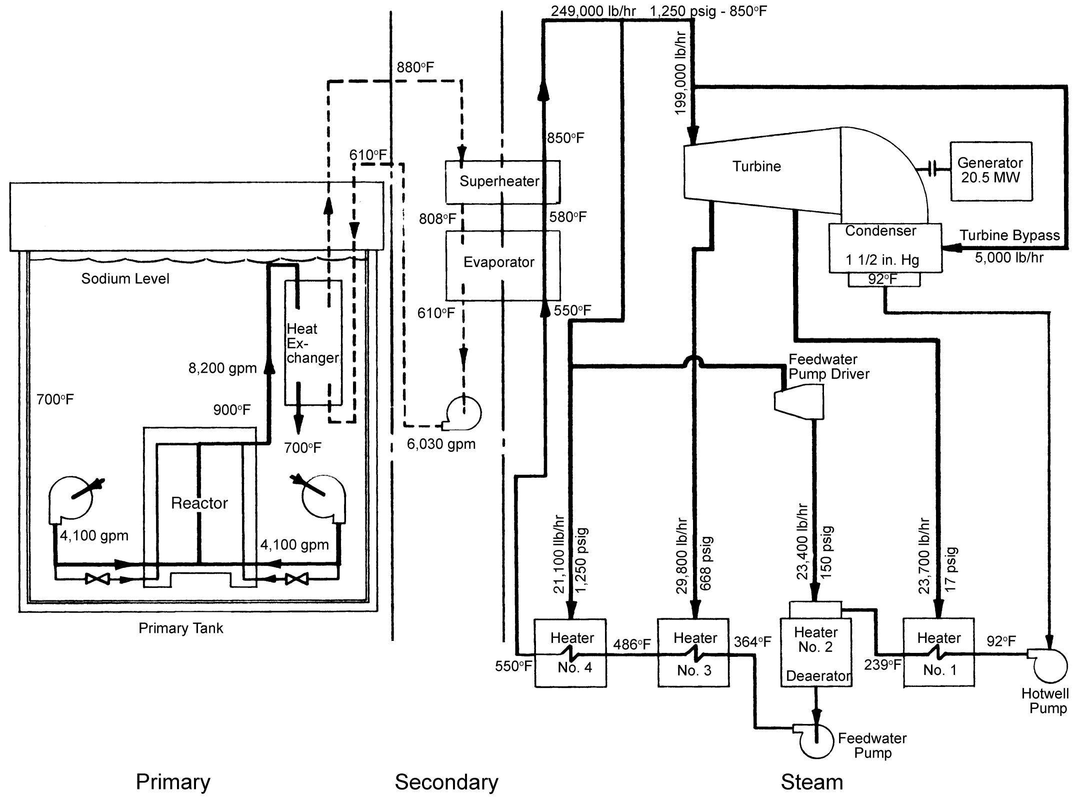


FIGURE 3-1. EBR-II SKELETON FLOW DIAGRAM.

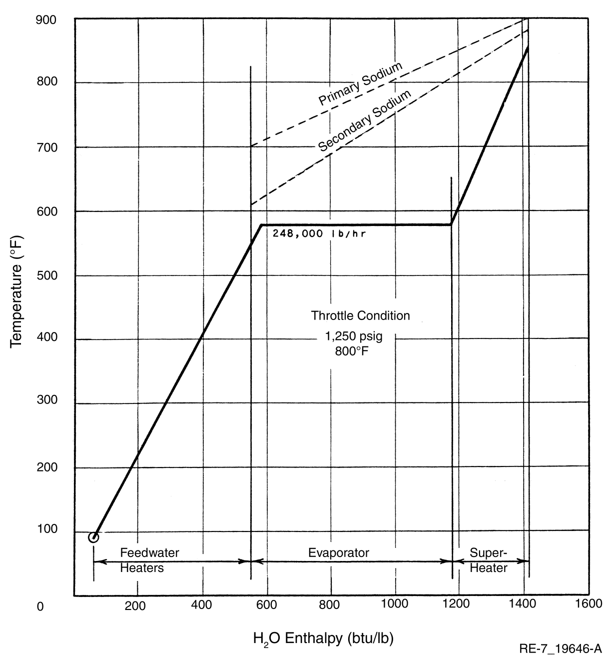


FIGURE 3-2. TEMPERATURE-ENTHALPY DIAGRAM.

- Control and Safety Drive Systems
- Fuel Handling System
- Primary Tank and Biological Shield
- Primary Sodium Purification System
- Inert Gas System.

The reactor, the primary sodium pumps and piping, the heat exchanger, and the fuel handling system were contained in the primary tank submerged in sodium, as shown in Figure 2-12. Coolant was pumped directly from the bulk sodium in the primary tank to the reactor, and after flowing through the reactor, passed through the heat exchanger and back to the bulk sodium. This very simple flow system is shown in Figure 2-11. This submerged concept was employed for the following reasons:

1. The arrangement contributed significantly to the reliability of the primary coolant system. A high degree of integrity could be constructed into the primary tank, since it was of relatively simple design. As an added safety measure, it

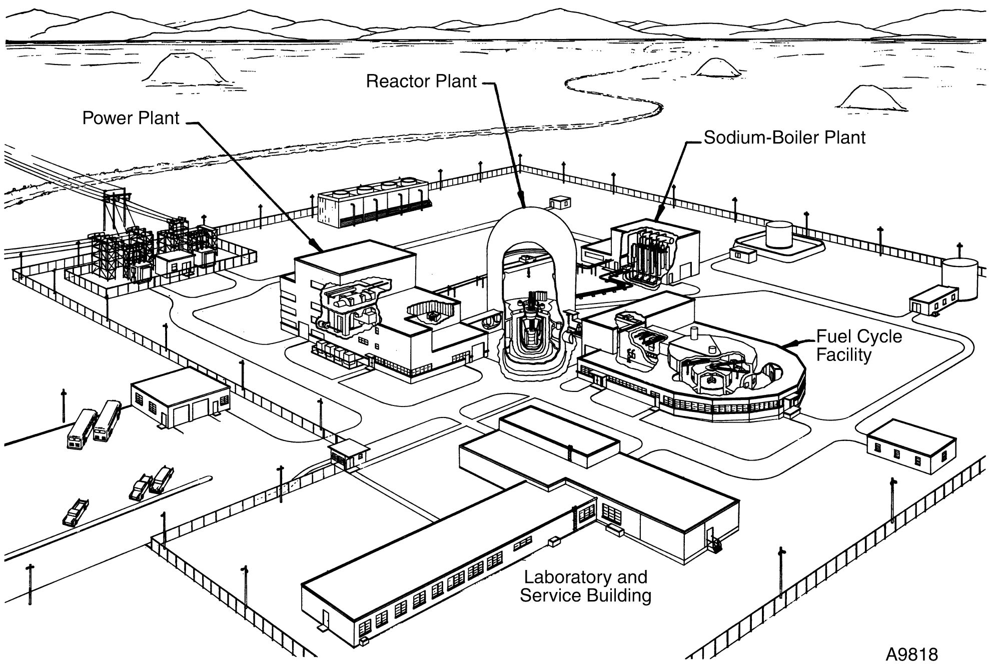


FIGURE 3-3. EBR-II PLANT ARRANGEMENT.

consisted of double-wall construction (a guard tank surrounding the primary tank). Because the entire coolant system was flooded with sodium to a level approximately 10 feet above the top of the reactor loss of coolant for the reactor was virtually impossible. Even if both primary tank walls were to fail, the free volume between the guard tank and the liner of the biological shield was sufficiently small to maintain the sodium level above the top of the reactor.

2. Since the reactor was intended to demonstrate operation suitable for a central station Power Plant, the replacement of fuel was accomplished in a short time. Shortly after reactor shutdown, the heat generation in the fuel was high, and reliable cooling was provided. This was accomplished by handling the fuel subassembly submerged in sodium. The fuel was cooled by natural convection of sodium through the subassembly, and fuel handling could begin immediately after reactor shutdown. The fuel subassemblies were moved to a fuel storage rack within the primary tank where they continued to cool, by natural convection of the sodium, until removed for processing.

3. Leak tightness of the primary coolant system piping was not required. Small amounts of leakage were permissible, since the leakage was internal. A small amount of leakage occurred at the connections between the pumps and the reactor, between the reactor tank and the reactor cover, and around subassembly nozzles.

4. The heat capacity of the very large mass of bulk sodium, approximately 620,000 pounds provided considerable thermal inertia to the primary system. It prevented rapid temperature transients in the primary sodium coolant reactor inlet temperature, and it added reliability to the shutdown cooling system.

5. Maximum integrity was provided with regard to containment of radioactive sodium. The entire radioactive coolant system, with the exception of the single, small, sodium cleanup circulation circuit, was confined within the primary tank.

6. Essentially all of the radioactivity in the Reactor Plant was confined to the primary tank and, therefore, only the primary tank, and the single circuit referred to in No. 5 above, required shielding. Shielded equipment cells and pipe galleries were eliminated.

7. Auxiliary heating of the primary system sodium to prevent freezing was simplified

since the entire system was heated as a unit. The individual components and pipes were in an environment of sodium, and the entire system was at one temperature.

REACTOR

The reactor was divided into three main zones: core, inner blanket, and outer blanket (Figure 2-2). Twelve control rods were located at the outer edge of the core, and two safety rods were located within the core, as shown. Each zone comprised a number of hexagonal subassemblies. The three zones were established by the lower grid-plenum structure, which used three different diameter holes for accepting the three types of subassemblies (including control and safety rods), which comprise the zones as shown. The number of subassemblies comprising the three zones as shown in Figure 2-2 are tabulated below in Table 3-1.

The basic minimum core volume, including control rods and safety rods, consisted of a total of 61 subassembly units (rows 1–5). This represented the minimum core volume for which reactor performance was evaluated and the minimum configuration that was used in the reactor. To provide flexibility of operation and to accommodate variations in core loading, which was practiced throughout the EBR-II operating lifetime, fuel subassemblies identical to those in the core zone were provided and could be installed in the first row of the inner blanket.

TABLE 3-1. Subassembly distribution in reactor.

Core	47[a] (47 to 59)[b]
Safety	2
Control	12
Inner Blanket	66[a] (66 to 54)[b]
Outer Blanket	510
Total	637

a. Minimum volume core.

b. Normal permissible core volume range (to accommodate experimental program).

Analyses were performed for cores incorporating from 1 to 12 of these special inner blanket fuel assemblies. A basic configuration of six of these special subassemblies was considered the nominal reactor loading. The possible arrangements considered where the additional inner blanket type fuel subassemblies were loaded and the sequence and location, from number 1 to 12, were specified. The comparable loading patterns are shown in Figure 2-2. The range of subassembly units in each zone is also shown in Table 3-1. This arrangement provided great flexibility and was extremely useful over the operating lifetime of the reactor.

The core, including the control and safety rods, had an equivalent radius of 9.92 inches (24.17 centimeters) and a height of 14.22 inches (36.12 centimeters); a total core volume of 66.3 liters. The core volume was varied frequently over the operating lifetime of the plant; it was easily increased by placing the special fuel subassemblies in the first row of the inner blanket as described above. The coolant flow control system easily accommodated this arrangement by providing appropriate coolant entry holes in the lower adapter of the subassemblies. Also, using elements with longer fuel sections increased the core height.

The 12 control rods and the 2 safety rods consisted of modified movable fuel subassemblies and were a part of the core zone. The rods, plus their stationary thimbles, comprised the control and safety subassemblies. The external dimensions of the thimbles were identical to the core and blanket subassemblies and the lattice spacing for all units was identical. The reactor could be controlled by moving the control rods in their thimbles in a vertical direction, thus moving fuel into or out of the core.

The safety rods were not a part of the normal reactor operational control system but were maintained in their “full-up” position, or maximum reactive position, at all times during reactor operation. This position was also maintained during fuel handling operation when the control rods were disconnected from their drives and were in their least reactive position.

SUBASSEMBLIES

A single subassembly size was employed throughout the reactor, resulting in a close-packed reactor geometry. The hexagonal subassembly tube was 2.290 inches across external flats with a 0.040-inch wall thickness. The subassemblies were spaced on a triangular pitch of 2.320-inch center distance. The nominal clearance of 0.030 inches between each subassembly permitted removal of the units from the reactor.

Each subassembly was located and supported at the bottom by the combination support grid and inlet coolant plenum (Figure 3-4). The heat generated in the fuel, or blanket material was removed by sodium flowing up through the subassemblies and around the fuel and blanket elements. To accommodate the very large range of flow rates required, two parallel flow systems were employed. A high-pressure system supplied the core and the inner blanket, while a low-pressure system supplied the outer blanket. The two systems had separate inlet plenum chambers as shown in Figure 2-19.

The upper end of each subassembly was identical, and the same handling and transfer devices accommodated all subassemblies. The lower adapters were of different size to differentiate the three types of subassemblies, and were of different configuration to accommodate the two coolant systems (Figure 3-5). Each subassembly contained a number of fuel elements, and/or blanket elements, of size and shape appropriate to the particular type of subassembly.

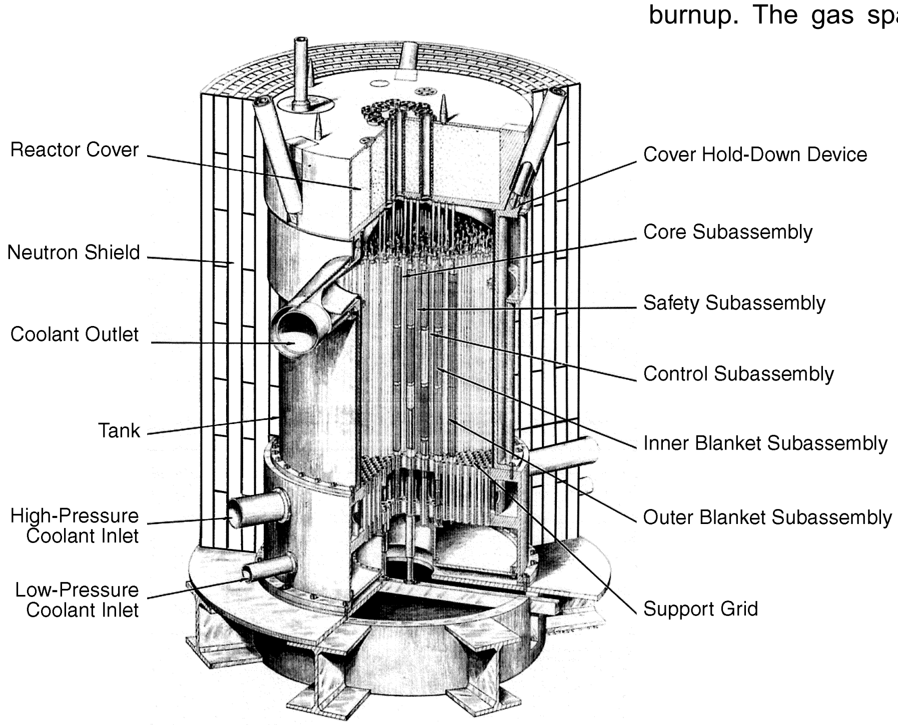


FIGURE 3-4. EBR-II REACTOR.

The core subassembly (Figure 2-1) comprised three active sections: upper blanket, core, and lower blanket. The core section consisted of 91 cylindrical fuel elements spaced on a triangular lattice by a single, helical wound wire on the outside of each element. The elements were supported within the subassembly and fastened at their lower ends to a support grid. The fuel elements (Figure 3-6) were pin type, consisting of a right circular cylinder of fuel alloy (0.144-inch diameter by 14.22 inches long) fitted into a thin-walled, stainless steel tube. The coolant flowed along the outside of the element tube.

The fuel pin was contained in a stainless steel tube (0.009-inch wall thickness by 0.174-inch outside diameter). The resultant annulus between the pin and the inside of the tube (0.006 inch) was filled with static sodium to provide a thermal bond. The sodium bond extended a nominal 0.6 inch above the top of the fuel pin. An inert gas space was provided above the sodium to accommodate expansion of the sodium. The fuel element tube was welded closed at each end. The fuel pin design evolved later to allow for higher fuel burnup. The gas space volume and the sodium bond annulus were increased to accommodate more fission gas and fuel swelling.

The individual fuel elements were contained within the hexagonal subassembly tube. They were fastened to the subassembly at their lower end by hooking to a parallel strip grid, as shown in Figure 2-1. The upper ends of the fuel elements were unrestrained to permit free axial expansion of the fuel element.

The upper and lower blanket sections were identical in construction and each consisted of 19 pin-type elements also spaced on a triangular lattice. The unalloyed depleted uranium pins were 0.3165 inches in diameter and totaled 18 inches long. They were similar in geometry to the fuel elements, being a

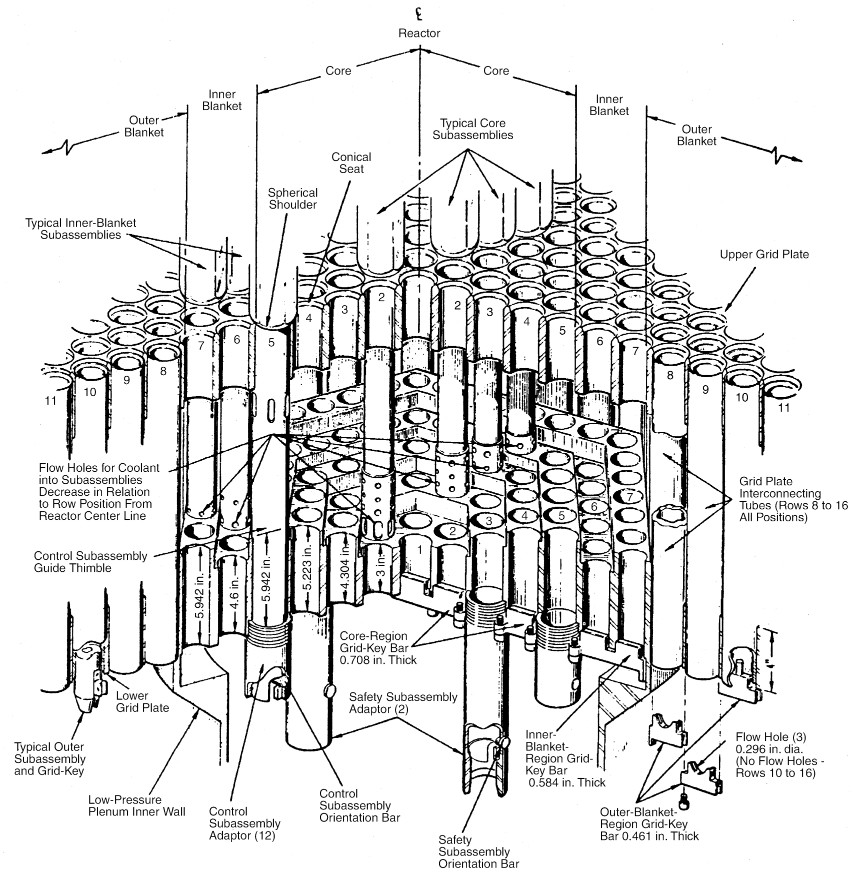


FIGURE 3-5. DETAILS OF GRID-PLENUM ASSEMBLY.

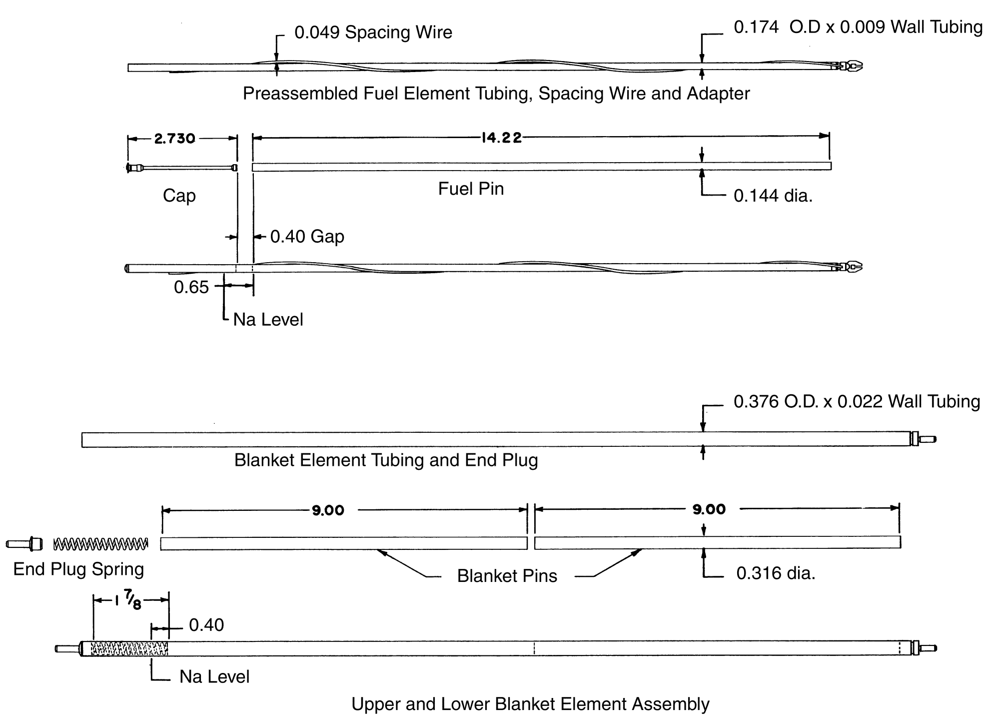


FIGURE 3-6. CORE SUBASSEMBLY ELEMENTS.

loose fit in the blanket element tube that consisted of a 0.370-inch outside diameter by 0.022-inch wall thickness. The 0.008-inch annulus was filled with sodium to provide the necessary thermal bond. Later core assembly designs abandoned the axial blanket sections. The details of the upper and lower blanket elements are shown in Figure 3-6.

The blanket elements were positioned in the subassembly by a grid structure at the lower and upper ends. They were fixed at the lower ends to the grid structure, while a grid also positioned the upper ends, permitting axial expansions but no other movement. Since the blanket elements were positioned at each end, no spacer provisions were made along the length of the blanket elements.

The lower adapter of the fuel subassembly engaged the reactor grid, and contained holes through which the coolant entered the subassembly from the high-pressure inlet coolant plenum chamber as shown in Figure 2-1.

The inner blanket subassembly (Figure 3-7) was made up of 19 cylindrical blanket elements spaced on a triangular pitch and contained in the hexagonal subassembly. The active blanket section consisted of depleted uranium cylinders (0.433-inch diameter) totaling 55 inches in length. They were contained in a stainless steel tube 0.493 inch in outside diameter with a 0.018-inch wall thickness. The resultant 0.012-inch annulus was filled with static sodium to provide a thermal bond. The sodium extended a nominal 2 inches above the top of the uranium, with a 4-inch argon gas expansion region above the sodium. The end closures were welded to provide a sealed unit. Flow distribution strips were included in the outer row of the elements to reduce the sodium flow in the peripheral flow channels to minimize over cooling.

The lower adapter of the inner blanket subassembly was similar to, but smaller in diameter than, the core subassembly. The inner blanket subassemblies also engaged the high-pressure inlet coolant plenum chamber in the reactor grid, as shown in Figure 3-7.

The outer blanket subassembly differed from the inner blanket subassembly in the design of the lower adapter and the design of the flow distribution strips. The lower adapter was arranged to engage the reactor grid in the low-pressure inlet plenum chamber. The two different lower adapters employed in the blanket assemblies are shown in Figure 3-7.

In addition to providing a low-pressure coolant supply to the outer blanket subassemblies, larger flow distribution strips were used in the peripheral flow channels to further reduce the coolant flow to match the lower power density in the outer blanket. The flow distribution strips in both the inner and outer blanket subassemblies are shown in Figure 3-8.

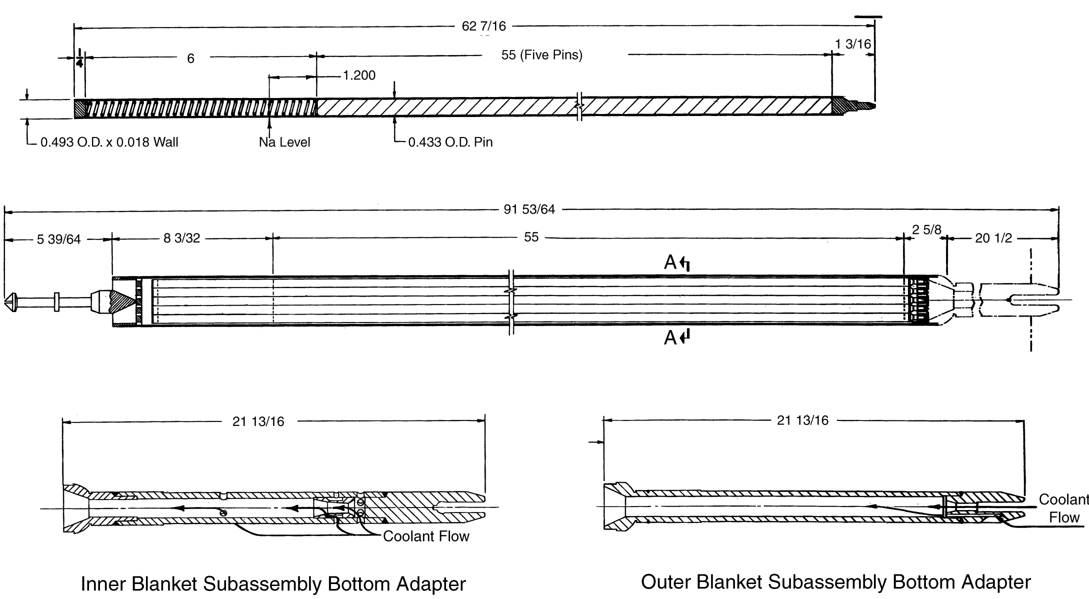


FIGURE 3-7. INNER BLANKET AND OUTER BLANKET SUBASSEMBLIES.

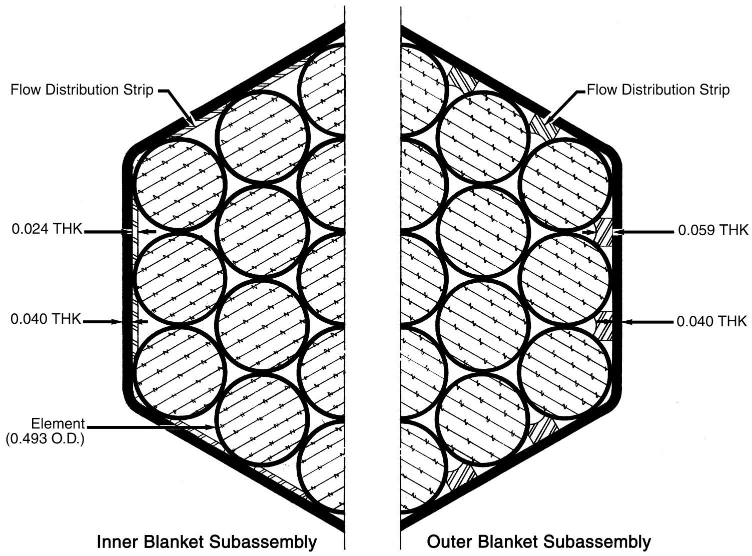


FIGURE 3-8. INNER BLANKET AND OUTER BLANKET SUBASSEMBLIES CROSS SECTIONS.

The control subassembly (Figure 2-6) consisted of a control rod and a guide thimble. The guide thimble occupied a unit lattice identical to those occupied by the various subassemblies.

Twelve identical control rods were employed to provide the operational control of the reactor. In later phases of the program, the number was reduced to as low as eight to permit up to four special test and/or irradiation assemblies to be installed in control rod positions. The control rod consisted of a modified fuel subassembly with a core section comprised of 61 fuel elements identical to the 91 fuel elements contained in the fuel subassembly. The control rod was encased in a hexagonal tube 1.908 inches across flats, which was smaller than the hexagonal guide thimble tube by the equivalent of one row of fuel elements. The control rod did not contain an axial blanket. A void section equivalent in height to the reactor core was provided above the fuel section of the control rod as shown in Figure 2-6.

During operation, this void section was filled with coolant sodium flowing through the control rod. A reflector section of solid steel, except for flow passages for the coolant, was located immediately above the void section. Reactor control was effected by vertical movement of the control rod, adjusting the proportion of fuel or void (sodium) in the core region of the reactor. As discussed earlier, the EBR-II reactor control concept was influenced by the desire to demonstrate high neutron efficiency to demonstrate the potential for maximizing breeding ratio. EBR-II did demonstrate the feasibility of operating the reactor by moving fuel and avoiding the use of parasitic absorber to achieve adequate control.

Subsequent operations of EBR-II demonstrated the use of a combination of absorber and fuel to increase the effectiveness of the control system. The absorber was located above the fuel section (the region of void described above). The upper end of the control rod was equipped with an adapter section identical to the subassemblies and was used for attachment to the control drive unit as well as the fuel gripper unit for unloading. The lower end of the control rod below the fuel section consisted of a cylindrical tube that also contained a steel reflector section. Bearings were provided on this lower section, which provided the guide between the control rod and the guide thimble.

The control rod was cooled in a similar manner to the core subassemblies by sodium delivered from the high-pressure sodium coolant system. Sodium entered through holes in the upper end of the lower adapter of the thimble, and through a second set of holes in the lower end of the control rod. The holes in the thimble section were above the lower bearing of the control rod throughout the control rod travel. The lower end of the thimble was open, and the lower control rod bearing served as a flow restriction to minimize sodium leakage from the bottom of the thimble. The primary system sodium pressure acted across the lower end of the control rod, and therefore exerted a downward force on the control rod. This downward force opposed the lifting force due to the pressure drop of the coolant flowing through the control rod, similar to the arrangement in the fuel subassemblies.

Since the vertical position of the control rods in the reactor varied relative to the stationary reactor core, the heat generation within the control rod was also variable. The coolant flow through the control rod was established to accommodate the maximum heat generation (i.e., with the control rod fully inserted in the reactor). If a constant coolant flow had been employed, the temperature rise in the coolant would have decreased as the control rod was lowered out of the reactor and the heat generated in the control rod decreased. This would have resulted in considerable degradation of the outlet sodium temperature from the control rod and in the mixed coolant temperature from the reactor.

To avoid this condition, an arrangement of the control rod and guide thimble coolant inlet holes provided variable orificing proportional to the position of the control rod in the reactor. This was accomplished by the relative size and locations of the coolant holes in the guide thimble and in the control rod. The coolant flow through the control rod varied with its vertical position in the reactor because the coolant flowed from the high-pressure plenum, through the holes in the thimble, at the top of the plenum, and then through the holes in the lower end of the control rod. The coolant flow path was shortest when the control rod was up and longest when the control rod was down.

This system did not provide precise control of coolant temperature, and the control rods were overcooled, but not to the extent that would have existed in a constant flow system. The flow reduction through the control rod was determined experimentally to be about 35 percent from control rod full up to full down.

A flow twister was installed in the void section immediately above the core section of each control subassembly to reduce the temperature differentials in the control rod hexagonal tube and, therefore, to minimize bowing of the control rod within its thimble. Upon leaving the core section, the hotter coolant flowed along the inside surface of the control rod facing the center of the reactor. The coolant was rotated approximately 180 degrees to the opposite surface by the flow twister. Thus, exposure of the opposite surface to the higher-temperature sodium tended to reverse any bowing of the rod. The flow twister did not introduce any significant pressure drop in the coolant flow. The coolant flow provisions for the control rods are shown in Figure 2-6.

The control rod was removed from the reactor by the fuel handling system in the same manner as the various subassemblies. The same considerations of irradiation damage and fuel recycling that applied to fuel subassemblies also applied to the control rods. The guide thimble was also removable from the reactor in the event of damage. It was locked in the lower reactor grid by a latch that was engaged by rotating the thimble. Rotation of the thimble was normally prevented by the six subassemblies that surround it. To remove or install a thimble it was necessary to first remove the six adjacent subassemblies and replace them with special modified scalloped hexagonal replacements that fill the subassembly space but permitted the thimble to be rotated. This special procedure was used infrequently.

The safety subassembly (Figure 3-9) consisted of a safety rod and a guide thimble. The safety rod and thimble were essentially identical to the control subassembly except for modifications at the lower end. Two safety rods were incorporated in the reactor and located as shown in Figure 2-2. The safety rods were not a part of the normal reactor operation control system. They were fully inserted in the reactor, in their most reactive position, at all times during reactor operation and fuel handling. The purpose of the safety rods was to provide available negative reactivity when the reactor was shut down and the control rods were disconnected from their drives. Their primary purpose was to provide a safety device during reactor fuel loading operations. The two safety rods were attached to the safety rod support beam located below the reactor structure and connected to two vertical drive shafts located outside the fuel transfer system and operable during refueling operation (Figure 2-19 and Figure 3-9).

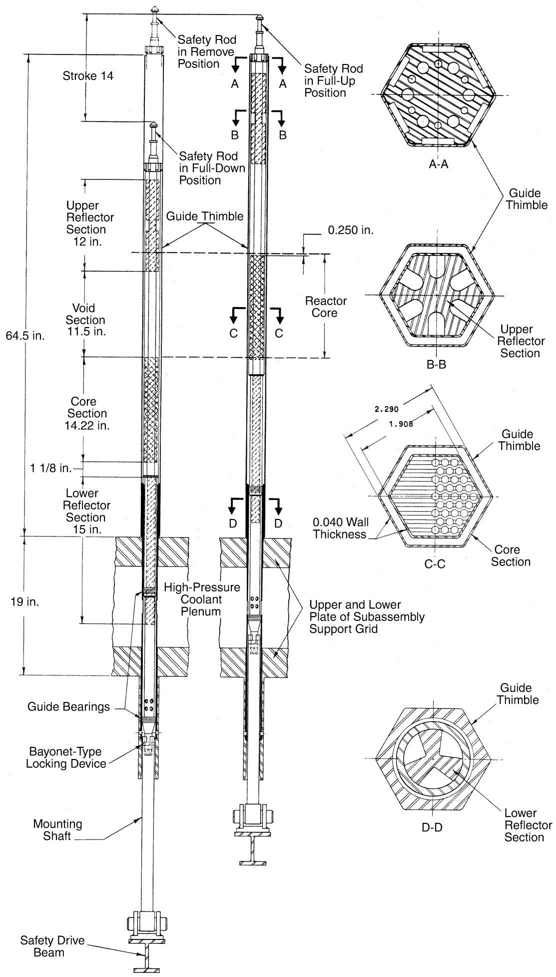


FIGURE 3-9. SAFETY SUBASSEMBLY.

The safety rod guide thimbles were locked to the lower reactor grid structure in a similar manner to that described for the control rod guide thimbles. Each safety rod was engaged to the common drive unit by a rotational locking mechanism. A hexagonal shaped collar on the upper end of the safety rod prevented inadvertent disengagement of the safety rod. This collar normally engaged the inside of the thimble, preventing rotation of the safety rod. To connect or disconnect the safety rod for loading purposes, the safety rod was raised 1 inch above its normal up position by the safety rod drive mechanism to raise the hexagonal collar above the thimble.

The safety rod upper adapter was identical to the control rod and the subassemblies, and was handled in the normal manner by the fuel transfer system. The guide thimble was removable in the same manner as the guide thimble for the control subassemblies.

Cooling of the safety rod was accomplished in the same manner as the control rod, but since it was a one-position device, no provisions were made for variable flow. The safety rods had to be in an up, most reactive, position before the reactor could be made critical or before fuel handling operations could be performed. It should be noted here, however, that this system was never called upon to perform its intended function, which was to shut the reactor down from an unintended reactor critical condition during reactor refueling and with the normal reactor control system inoperative. This feature could be characterized as an ultra-conservative feature that was not necessary and probably could be omitted in future liquid metal cooled fast breeder reactor designs.

Neutron source rods were placed in the outer blanket region. The antimony/beryllium combination provided the neutron source for neutron detector calibration when the reactor was shut down.

REACTOR VESSEL ASSEMBLY

The reactor vessel assembly (Figure 3-4 and Figure 3-10) consisted of the reactor vessel, the grid assembly, and the top cover. It contained the reactor-fuel and blanket subassemblies, and control and safety rods, and provided the proper configuration of these units. The assembly was located and supported at the bottom and on the centerline of the primary tank. It was supported on the structural members that reinforce the bottom of the primary tank inner shell. The vessel assembly was surrounded on all sides by the neutron shield and was submerged beneath approximately 10 feet of sodium.

The vessel assembly consisted of three major units: the grid-plenum assembly, the vessel, and the top cover. To ensure accurate alignment, the vessel was fastened to the grid-plenum assembly by bolts, which were tack-welded to ensure a permanent connection. The vessel cover served as a neutron shield as well as a closure. It was clamped to the vessel flange by means of three

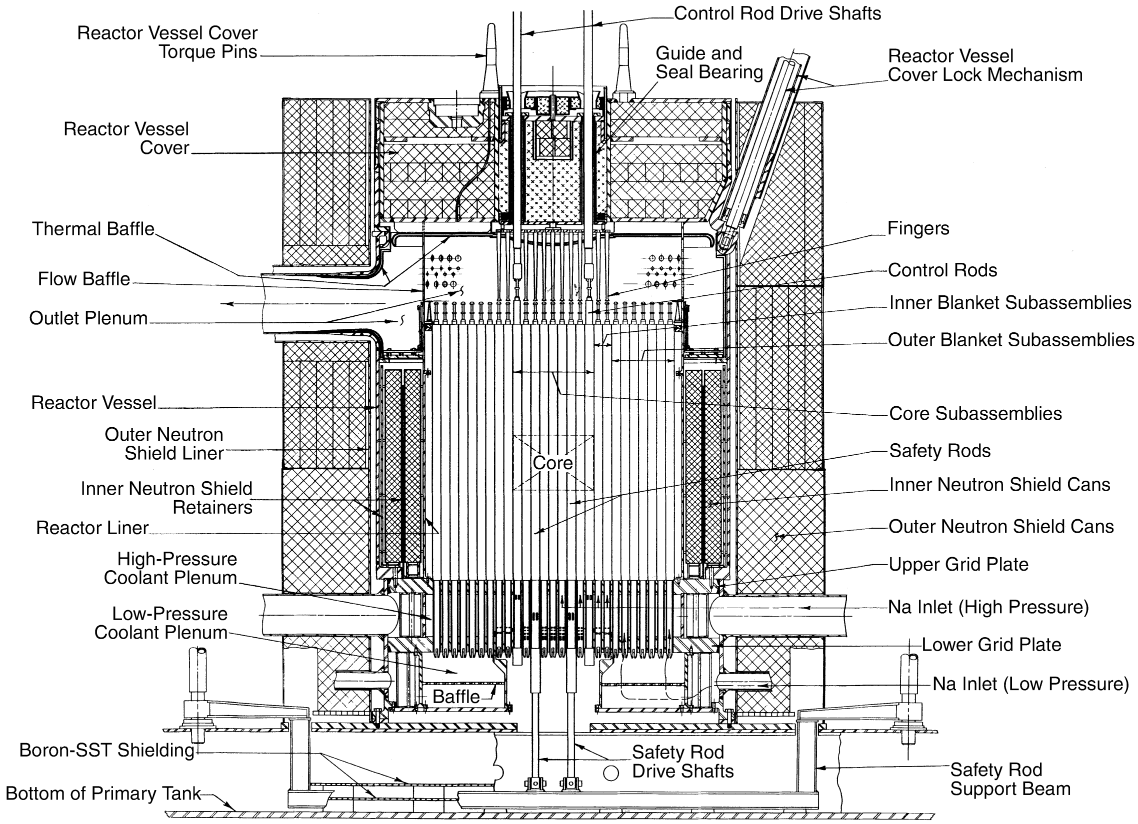


FIGURE 3-10. REACTOR VESSEL ASSEMBLY.

hold-down clamps (Figure 3-10 and Figure 3-11). When the cover was closed, it formed the upper reactor coolant plenum chamber from which the coolant flowed to the heat exchanger. Within the plenum chamber the coolant was at an average temperature of 900°F and a pressure of 18 pounds per square inch gauge. The cover separated this sodium from the ambient bulk sodium in the tank. The sodium seal was formed between the vessel flange and the cover, but some leakage occurred. When it was desired to exchange subassemblies, the hold-down clamps were released and the cover was elevated to allow the fuel handling system to unload the fuel below the raised cover, and transfer the fuel to the storage rack (Figure 2-8).

The reactor vessel was a cylindrical tank with flanged ends. The upper plenum of the vessel, as well as the coolant nozzle, was lined with a thermal baffle (Figure 3-10). The function of this baffle was to reduce the temperature difference across the vessel wall and also the coolant outlet nozzle wall. Below the plenum region the vessel contained a laminated steel thermal shield. The vessel wall was insulated from the bulk sodium in which it was submerged by a steel shell liner that was vented, and therefore contained static sodium. This shell and static sodium combination provided sufficient thermal insulation with acceptable thermal stresses in the vessel wall. The heat loss between the reactor outlet sodium and bulk sodium in the primary tank was relatively small and was not lost from the system.

The grid-plenum assembly (Figure 2-19 and Figure 3-5) was a combination structure that incorporated a grid to support and locate the subassemblies, and incorporated the coolant inlet plenum chambers that supplied coolant to the subassemblies. It consisted of two 4-inch-thick stainless steel plates that contained the locating holes for the lower adapters of the subassemblies. The subassemblies were supported by the upper plate and the lower adapters extended through the lower plate. A spherical shoulder on the

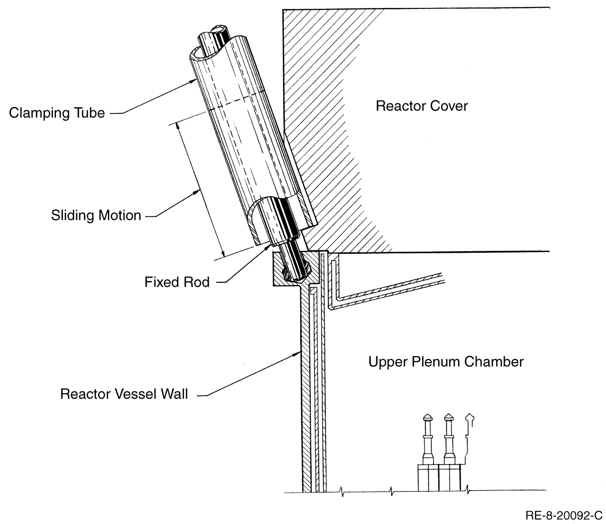


FIGURE 3-11. REACTOR COVER HOLD-DOWN.

subassembly that engaged a conical seat in the upper grid plate supported the subassemblies. This arrangement minimized the leakage flow of coolant along the outside surfaces of the subassemblies.

The high-pressure coolant plenum chamber supply for the core and inner blanket was formed between the two grid plates. The low-pressure coolant plenum chamber supply for the outer blanket consisted of an annular chamber immediately below the lower grid plate. Tubes welded to each plate in the outer blanket zone interconnected the upper and lower grid plates. This prevented short-circuiting of the high-pressure coolant through the outer blanket.

The grid-plenum assembly tube structure also provided the structural system required to support the entire reactor load on the upper grid plate. The high-pressure coolant flowed between these tubes into the core and inner blanket region where it entered the subassemblies. The lower nozzles of the core and inner blanket subassemblies contained holes located precisely between the upper and lower grid plates. The coolant entered the subassembly through these holes and flowed upward through the subassembly. The upper surface of the lower grid plate was stepped to close specific holes; this varied the cross-sectional area of the effective holes in the subassemblies. This arrangement provided orificing of the flow through the subassemblies to match the heat generation rate in each row of subassemblies as described earlier.

The lower end of the subassembly nozzles was closed, forming a hydraulic piston. The sodium in the high-pressure coolant plenum chamber was at a nominal pressure of 61 pounds per square inch, of which 8 pounds per square inch was static head (due to the sodium level in the primary tank). The remainder gave a pressure difference of 53 pounds per square inch acting across the piston. This provided a downward force, or hydraulic hold-down, of 148 pounds on the core subassemblies and 116 pounds on the inner blanket subassemblies.

The low-pressure coolant entered the low-pressure plenum chamber at 22 pounds per square inch, and entered the lower nozzles of the outer blanket subassemblies through openings at the bottom. Because the pressure drop through the outer blanket subassemblies was much smaller and the weight of these units was large, it was unnecessary to provide hydraulic hold-down.

Three different hole diameters for subassemblies were provided in the grid plate. This prevented a fuel subassembly from being inadvertently placed in the wrong position. To prevent the interchange of subassemblies in the other direction, subassembly angular orientation bars were used to provide proper angular orientation of the subassemblies in the reactor. They were fastened to the underside of the lower grid plate and engaged slots in the subassemblies. There were three thicknesses of bars: the core subassemblies engaged the thickest, the inner blanket subassembly slots were thinner and the outer blanket subassembly slots were the thinnest. If an inner blanket subassembly was inadvertently placed in a fuel position, the slot in the inner blanket subassembly tip was too narrow to engage the bar. This prevented engagement of the subassembly at least 2 inches short of its normal position in the grid, which was easily detected by the fuel handling mechanism. The same condition existed if an outer blanket subassembly was placed in an inner blanket position or a fuel position.

This method of loading control was adopted because a core subassembly inserted in either blanket zone introduced both a reactivity problem and a cooling problem, while a blanket subassembly introduced in the wrong zone introduced only a cooling problem. The lower grid was 19 inches deep, while the core was only 14 inches long. Since a core subassembly could not enter the grid in the wrong location because the diameter of the subassembly lower adapter was too large, a loading error would not permit the fuel section of the subassembly to enter the core region of the reactor. In the reverse manner, a subassembly could enter the grid for approximately 17 inches of travel, but the error was detectable.

The reactor cover provided the closure of the top of the reactor vessel and formed the upper surface of the outlet plenum chamber. It also provided the upper portion of the neutron shield. The 12 control rod drive shafts operated through guide, sleeves provided in the cover for these units. The fuel handling gripper mechanism and hold-down shafts also penetrated the cover. A small amount of leakage occurred through these various openings during reactor operation when a sodium pressure differential of approximately 12 pounds per square inch existed across the cover. This leakage flow was employed as a part of the neutron shield cooling system in this region. This too represented a small heat loss because this hot sodium bypassed the IHX but this heat was not lost from the system; it was recycled through the reactor.

The top cover was raised and lowered by two shafts penetrating the small rotating plug. The cover was fastened to the reactor vessel by three clamping mechanisms, and the raising and lowering mechanism was designed to permit free expansion of the two lifting shafts. The drive shafts for the three clamping mechanisms were also permitted to float. This arrangement avoided the large load due to internal pressure being transferred to the cover lifting mechanism, and also avoided problems associated with differential thermal expansion in the system.

The underside of the reactor cover had projections on the same spacing as the core and inner blanket subassemblies. These pins were positioned directly above each subassembly adapter and provided approximately 1/4 inch of clearance between the adapter and the end of the pin. The pins prevented any appreciable lifting of the subassemblies in the event of failure of the hydraulic hold-down system or vertical movement for any reason.

Thermocouple wells were provided adjacent to some of these pins to measure the outlet sodium temperature in various regions of the reactor. The thermocouple leads were introduced through tubes that were brought out through the hollow cover lifting drive shafts. Inside the cover the tubes were routed to the various locations. The tubes were permanently installed in the cover, but the thermocouple junctions and leads could be replaced.

PRIMARY COOLING SYSTEM

The primary system component arrangement is shown in Figure 2-11. The reactor vessel was centrally located at the bottom of the primary tank. The pumps, heat exchanger, and connecting piping were disposed radially around the reactor vessel and elevated above it.

The coolant flow path in the primary cooling system was as follows:

- Two primary coolant pumps took suction from the bulk sodium in the primary tank.
- The flow from each pump separated into two streams before entering the high-pressure and low-pressure reactor inlet plenum nozzles.
- The 12-inch inlet nozzles to each of the high-pressure plenums were approximately diametrically opposite each other.
- Each pump outlet was connected to the corresponding high-pressure inlet plenum nozzle.
- A smaller line connected to each outlet line provided a take off flow through a flow control valve to each low-pressure plenum nozzle.

These valves were set during initial plant operation and remained fixed during most of the operating lifetime of the plant.

Coolant flow in all regions of the reactor was upward through the fuel and blanket subassemblies and into a common upper plenum chamber with a single 14-inch outlet. The heated sodium flowed to the shell side of the intermediate heat exchanger through a permanently installed 14-inch pipe. The pipe had a Z configuration to accommodate thermal expansion. The auxiliary pump was installed in the upper horizontal leg of this outlet pipe line.

The primary coolant flowed downward through the shell side of the heat exchanger and discharged into the bulk sodium in the primary tank (as shown in Figure 2-11). The heat exchanger primary sodium outlet was approximately 7-1/2 feet above the centerline of the reactor. This arrangement assured an inherently reliable natural convection cooling system for shutdown cooling without heat removal by the secondary sodium as discussed earlier.

The sodium line between the upper plenum of the reactor vessel and the heat exchanger shell was permanently attached to these two components. The heat exchanger shell was permanently attached to the underside of the cover of the primary tank; however, the tube bundle, including the upper and lower secondary sodium plenums, secondary sodium inlet and outlet nozzles, and shield plug (as shown in Figure 3-12) could be removed as a unit in a vertical direction.

When the reactor was in operation, coolant was supplied in parallel by the two main primary sodium pumps operating. At 100 percent power operation, each pump supplied approximately 4,700 gallons per minute of coolant at 55 pounds per square inch head; the maximum capacity of each pump was approximately 5,000 gallons per minute at 85 pounds per square inch .

The two primary sodium pumps were vertical-mounted, single-stage, centrifugal-type mechanical pumps (Figure 3-13). These pumps employed a gas-tight motor and sodium hydrostatic bearing. It should be emphasized that the success of this bearing made the use of mechanical centrifugal pumps possible.

Variable-speed motors powered by a motor-generator set providing variable voltage and frequency drove these pumps. They were controllable from about 20 to 100 percent speed with specified rates of acceleration and deceleration. The direct coupled pump drives were special, totally enclosed, gas-tight, 480-volt, alternating current motors. Motor cooling was provided by forced circulation of air through an air-to-argon gas heat exchanger within the sealed motor enclosure as shown in Figure 3-13. Labyrinth-type shaft seals were employed to minimize diffusion of sodium vapor into the motor enclosure.

The inlets to the pumps were open to the primary bulk sodium in which they were submerged. The entire volume of sodium—approximately 86,000 gallons—was at the reactor inlet temperature of approximately 700°F. It was heated approximately 200°F as it passed through the reactor and was cooled approximately 200°F as it passed through the intermediate heat exchanger. From there it returned to the bulk sodium at about 700°F; when the reactor was at power, this was the primary sodium temperature scenario. At full power, about 62.5 megawatt thermal were generated and transferred in this manner. It was achieved by forced circulation of sodium through a very simple heat generation and exchange thermal system. The two primary pumps provided the forced circulation of the primary sodium coolant to achieve this capability under all of the power conditions at which the reactor operated.

This physical arrangement of the primary system components simplified the system enormously. A traditional system of pumps, reactor, and heat exchanger would involve much more piping and support facilities. On the other hand, access to these components submerged in high temperature sodium complicated service and maintenance.

As described earlier, the intermediate heat exchanger internals were completely removable for maintenance or even replacement. Similar capability was required for the pumps. Ball seat type disconnects were used between the pump outlet nozzle and the permanent piping to the reactor inlet plenum. This allowed for removal of these pumps from the primary tank as shown in Figure 3-14. During the operating lifetime of the EBR-II, each pump was removed only two times. Figure 3-15 is a photo of a pump after removal from the caisson used for removal and cooling.

SHUTDOWN COOLING SYSTEM

When the reactor was not operating, fission product decay heat had to be removed. The systems that removed the heat generated at power were perfectly capable of performing the same function when the reactor was shut down. However, these systems were not sufficiently dependable to meet the reliability requirement involved. Shutdown cooling was required at all times and had to be absolutely reliable. This unique requirement of nuclear reactors was particularly important in fast reactors because they operate at very high power density.

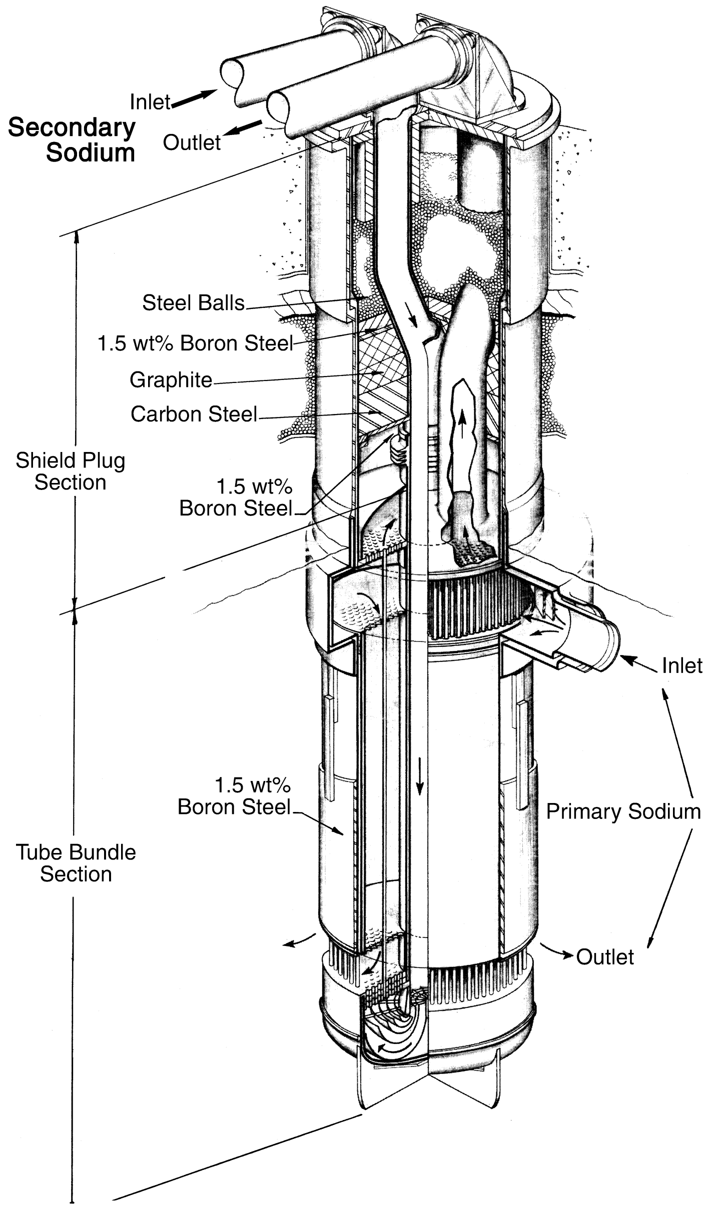


FIGURE 3-12. HEAT EXCHANGER.

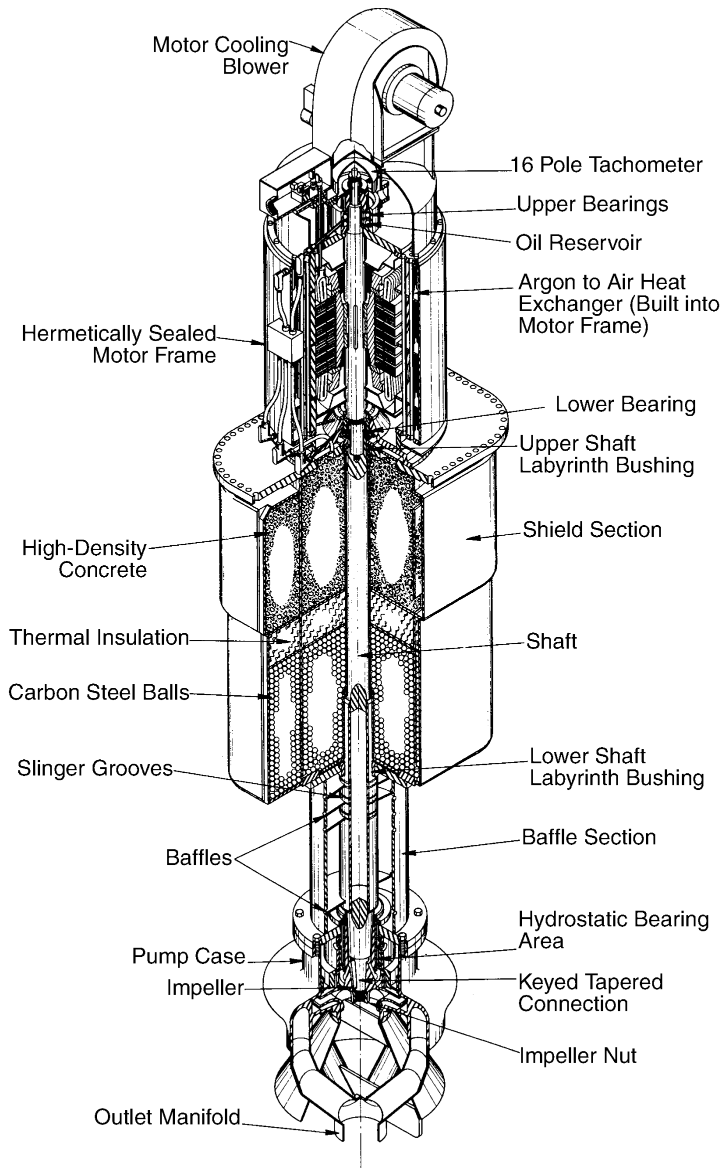


FIGURE 3-13. EBR-II PRIMARY PUMP.

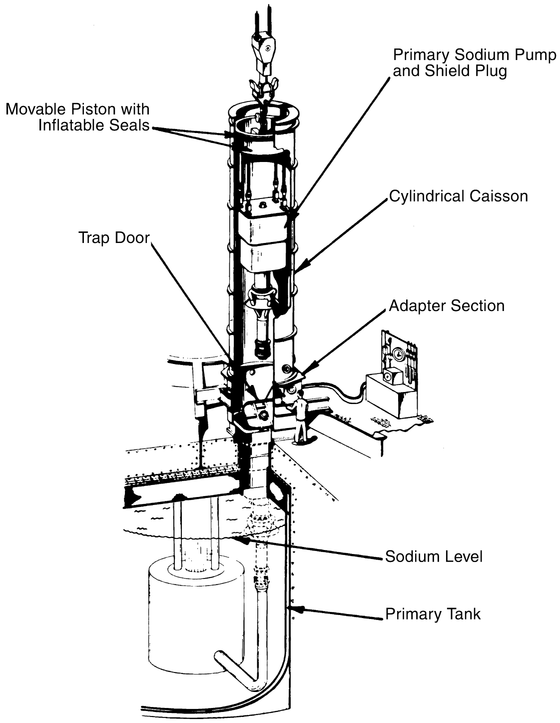


FIGURE 3-14. EBR-II PRIMARY PUMP REMOVAL.

As noted earlier, the EBR-II concept incorporated totally passive systems to remove fission product decay heat from the fuel. The natural thermal convection of sodium through the reactor was enhanced by a unique auxiliary sodium pumping system. This system augmented thermal convection as needed under certain conditions of reactor shutdown which could have inhibited the transition from forced convection coolant flow to natural convection coolant flow.

The auxiliary pump ensured continuity of flow under these conditions and prevented undesirable temperature transients. The auxiliary pump was a direct current electromagnetic pump located in the reactor outlet line, and operated in series with the main pumps. Its design capacity was approximately 500 gallons per minute at 0.15 pounds per square inch and 900°F sodium temperature. The pumping section was incorporated in the 14-inch outlet pipe, with no change in pipe cross section. This was done to maintain the integrity of the piping system at the expense of pumping efficiency, which was not important.

The auxiliary pump electrical power was supplied from metallic rectifier units and storage batteries. The storage batteries, operating in parallel with the rectifier units, assured pump operation in the event of a complete power failure. During normal operation, these batteries floated on the line and remained fully charged at all times. In the event of a sustained power failure, the pump operated until the battery was discharged, which resulted in a gradual decay of the flow rate and an ideal transition to thermal convection.

Removing the fission product decay heat from the reactor fuel after shutdown involved heat removal from the reactor by the primary sodium flowing through the reactor; and heat removal from the primary sodium. After reactor shutdown, coolant flow through the reactor was maintained as follows:

- Operation of the main pumps
- Operation of the auxiliary pump
- Natural convection flow.

Heat removal from the sodium leaving the reactor could be accomplished by two methods:

- The heat could be transferred to the secondary system, then to the steam system, and eventually to the atmosphere.
- The heat could be transferred to the bulk sodium in the primary tank and then transferred to the atmosphere more directly.

If the reactor cover was closed, coolant flow through the reactor by any of the three methods described above followed the normal circuit through the heat exchanger to the bulk sodium. If the secondary system was operating, the heat was transferred in the heat exchanger to the secondary system sodium. The secondary system, in turn, transferred heat to the steam system in the steam generator. The heat left the steam system via the condenser, and was transferred to the atmosphere through the cooling tower.

If the secondary system was inoperative, the heat was transferred to the bulk sodium in the primary tank. The heated sodium leaving the reactor was mixed with the bulk sodium by discharging from either the heat exchanger, or, if the reactor vessel cover was raised, from the top of the reactor. The heat was then removed from the bulk sodium by the shutdown coolers that, in turn, transferred the heat to the atmosphere through a finned-tube air

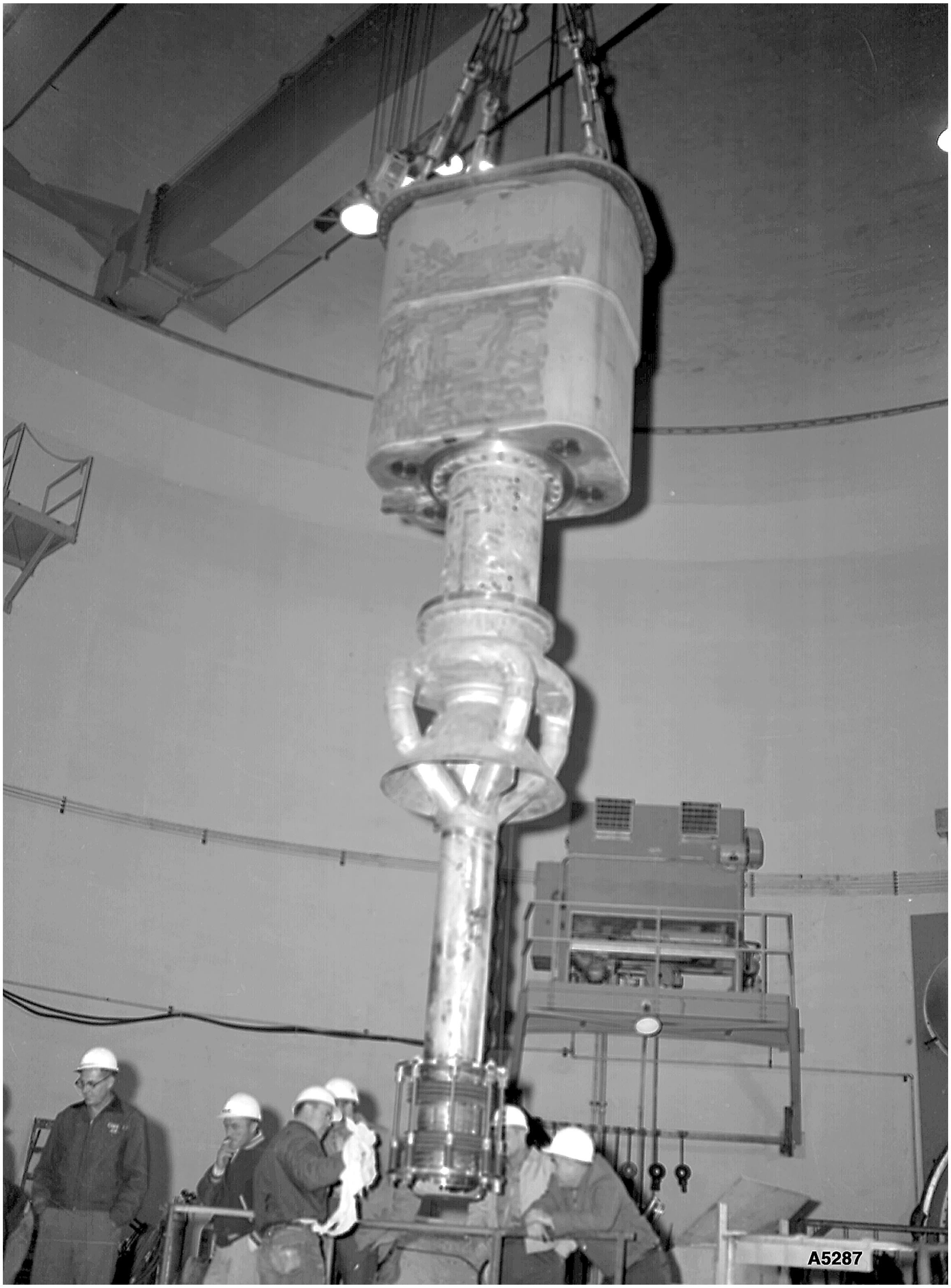


FIGURE 3-15. PRIMARY PUMP AFTER REMOVAL.

heat exchanger. Since the primary system had a very large thermal capacity compared to the amount of fission product decay heat removed from the reactor, the temperature rise of the bulk sodium was slow, and fast response of the shutdown coolers was not necessary. The salient feature of this method of heat removal was the complete independence from any external power source. All fluid flow was due to natural convection.

The shutdown cooler (Figure 3-16) was an immersion-type bayonet heat exchanger. Basically, it consisted of two concentric pipes approximately 26 feet long, the outer pipe being closed at the bottom. An inner concentric pipe produced an annulus between the two concentric pipes. Coolant flowed down through the central pipe and up through the annulus as shown. To enhance thermal convection, the inner pipe was insulated by a void space between two concentric pipes to minimize heat transfer to the downward flowing coolant and thus enhanced heat transfer and thermal convection in the annulus. The coolant was sodium-potassium eutectic that was liquid at room temperature. The bayonet heat exchanger was installed in a thimble with a static sodium bond between to provide effective heat transfer from the bulk sodium surrounding the thimble. This very conservative arrangement incorporating an extra thimble was provided to avoid contamination of the primary sodium with potassium in the event of a coolant leak in the bayonet cooler.

The coolant entered the inner pipe of the bayonet cooler at the top and flowed downward to the bottom of the inner pipe where it reversed direction and entered the annulus. The flow was then upward through the annulus, where heat transfer to the coolant occurred. Leaving the bayonet cooler, flow was upward into a finned-tube air heat exchanger, which was located in a dampered air stack outside the reactor containment building. Here the heat was transferred to the atmosphere by natural convection of air; the cooled sodium-potassium eutectic then flowed downward into the inlet of the bayonet cooler. The balance of the system is shown schematically in Figure 2-13.

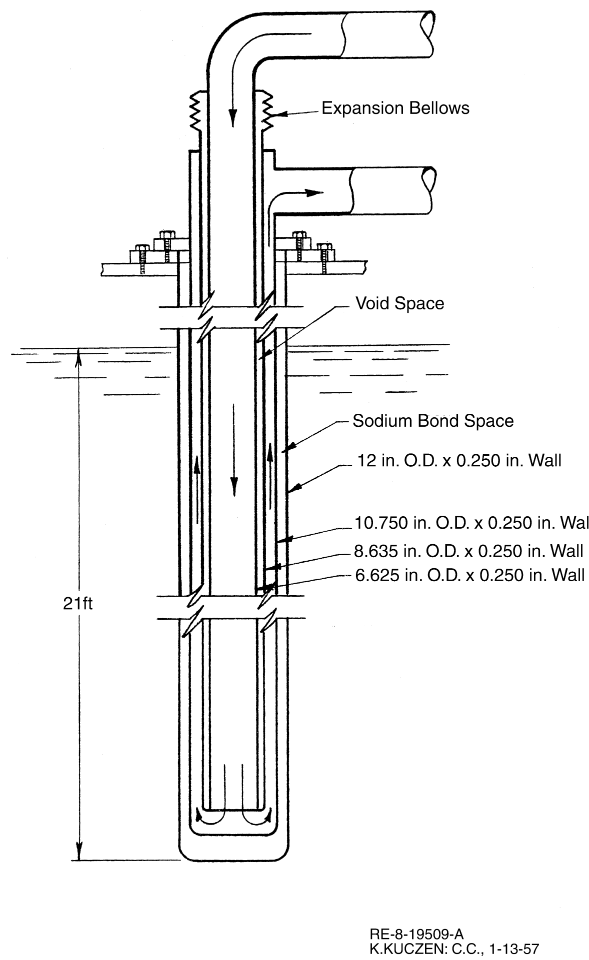


FIGURE 3-16. SHUTDOWN COOLER.

The rate of heat release from the system was controlled by the position of the stack dampers. Normally the dampers were actuated by automatic control, however manual control was also incorporated in the event of failure of the automatic system. During reactor operation, the dampers were held closed by electrically energized magnets, and a minimum flow of sodium-potassium eutectic occurred in the shutdown cooling system. This method of operation prevented the freezing of the coolant in cold weather, provided for positive starting when the dampers were opened, and also reduced thermal shock on the system. When the stack dampers were opened the thermal head on both the coolant and air side was increased. This gave rise to increased flow of both fluids which in turn, resulted in increased heat removal from the bulk sodium.

The sodium-potassium eutectic cooling system, external to the bayonet cooler, was instrumented with thermocouples and an electromagnetic flow-meter. An alarm system was interlocked with these measuring devices to annunciate and indicate abnormal conditions of flow or temperature.

The system was designed for maximum reliability and simplicity. The design of the bayonet coolers provided for minimum internal stresses over large temperature ranges and minimum obstructions in the flow circuit. All welded construction was used and no valves were incorporated in the system.

NEUTRON SHIELD

The neutron shield surrounded the outside of the reactor vessel on all sides and was submerged in the bulk sodium of the primary tank. The shielding material was graphite and graphite impregnated with 3 percent (by weight) of natural boron. To prevent the reaction and contamination of the graphite with sodium, it was canned in stainless steel.

For purposes of description, the shield could be separated into three sections: radial, top, and bottom as shown in Figure 3-17. To facilitate fabrication, handling, and installation, the graphite and the borated graphite were canned in conveniently sized pieces that could be readily stacked and placed in position around the reactor vessel. All cans used for cladding were leak tested, loaded with graphite, and closed by welding. The 1/8-inch clearance space between the can and the graphite was filled with helium. The cans were filled with helium to an absolute pressure of 10 inches mercury, at room temperature, to minimize the internal pressure at operating temperature (12 pounds per square inch absolute at 700°F) and also to provide a heat transfer medium to conduct the heat generated in the graphite to the can wall. The helium generated by the (n,α) reaction with boron, was expected to result in an increase in pressure of approximately 19 pounds per square inch (at operating temperature) during the life of the reactor. This assumed that all of the helium generated in the graphite would be released to the helium atmosphere in the stainless steel can. The cans were designed for a positive internal pressure 50 pounds per square inch greater than the external pressure. They were cooled externally by natural convection flow of sodium.

The radial shield was assembled from graphite blocks fitted in stainless steel cans stacked in three levels to a height of approximately 13 feet. Two rows of canned graphite were positioned inside the reactor vessel periphery and five rows around the periphery of the reactor vessel. Each row was held in place by stainless wire mesh. Clearance was provided between the cans to permit natural convection flow of sodium. Each row was staggered with respect to adjacent rows to minimize neutron streaming. Specially shaped shielding cans were used around the inlet and outlet sodium pipes of the reactor vessel and around the instrument thimbles that terminated in the neutron shield. Retainers and liners provided positive positioning of the shield cans and enhanced natural convection cooling of the shield.

Because of the complex structure of the reactor vessel cover, the cans in the cover were of complex shape. They were stacked to prevent neutron streaming and to permit cooling. The cover contained six layers of cans filled with either

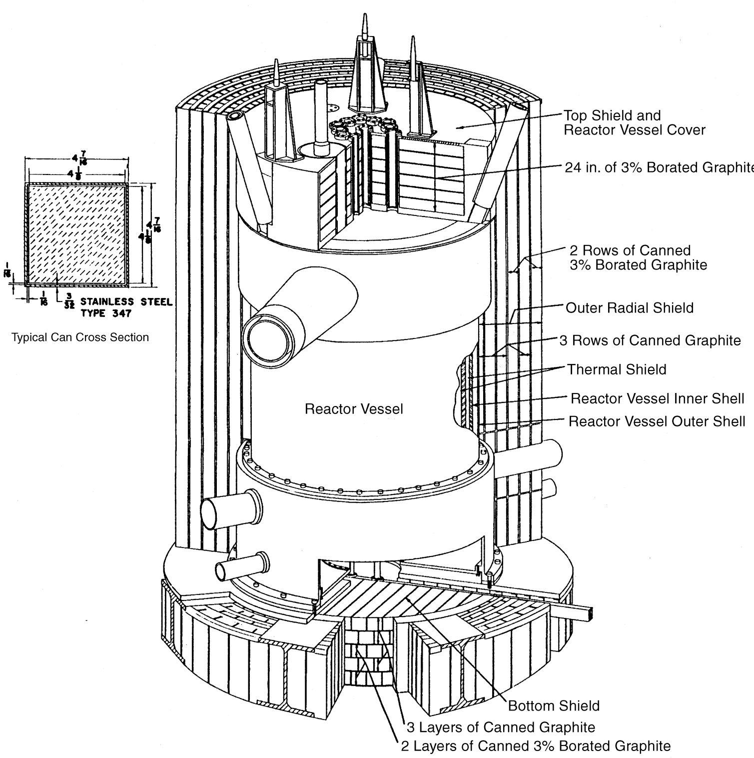


FIGURE 3-17. NEUTRON SHIELD.

3 percent borated graphite or boron carbide. The total thickness of the top shield was 24-3/4 inches.

The bottom shield consisted of borated stainless steel plates located between the vertical webs of the beams on the bottom of the primary tank. The heat generation in the structure below the reactor was not critical. This arrangement provided adequate shielding and a simple structure below the reactor vessel.

COUNTERS, CHAMBERS, AND INSTRUMENT THIMBLES

Three fission counters and eight compensated ion chambers comprised the detectors for the nuclear instrument channels. Since detectors of proven reliability for 700°F operation were not available, conventional detectors were employed in air cooled thimbles. For reliable operation, the temperatures of counters and chambers were maintained below 140°F.

Three uranium-235-enriched fission counters detected thermal neutrons in the startup range of operation. These counters were positioned in "J" thimbles located in the radial neutron shield. Eight compensated ion chambers of the boron-coated type were located adjacent to, or in the radial neutron shield.

The general arrangement of the nuclear instrument thimbles and their associated fission counters and ion chamber is shown in Figure 3-18. Eight thimbles were provided. Four "J" thimbles were imbedded in the radial graphite neutron shield outside of the reactor vessel, and four "O" thimbles were located immediately adjacent to the neutron shield.

Thimble cooling was accomplished by drawing room air through the thimbles. The system did not recirculate, the exhaust air was combined with the biological shield cooling air and discharged through the Fuel Cycle Facility 200-foot stack. Two full capacity blowers were available, one was on standby. For maximum reliability, the standby blower system was provided with automatic switch-over to a 100 kilowatt auxiliary diesel generator power supply. This was in addition to the 400 kilowatt plant auxiliary diesel generator. The reactor was scrammed in the event of thimble cooling failure, but these additional precautionary measures were designed to protect the nuclear detectors and pre-amplifiers from thermal damage. It should be noted that the primary function of these diesel generator power supplies was to protect against economic loss, not reactor damage.

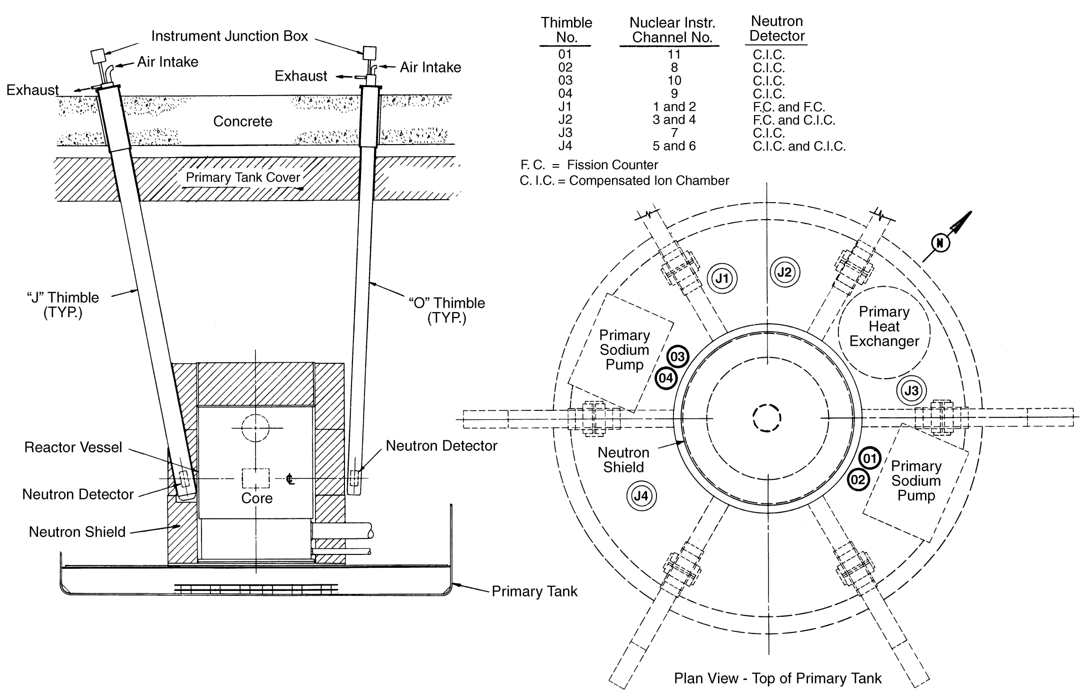


FIGURE 3-18. LOCATION OF NUCLEAR INSTRUMENT THIMBLES.

CONTROL AND SAFETY DRIVE SYSTEMS

Twelve control rods controlled the operation of the reactor. Each rod was independently driven by an electrical-mechanical drive mechanism. The drives were identical and were so arranged that only one drive could be operated at a time, with the exception of scram when all 12 operated simultaneously. Operating control was achieved by a 14 inch vertical motion of the control rods that was provided by a rack and pinion-type drive with constant-speed electric motors, therefore, only one speed of movement was possible. The control rods were disconnected from their drives during fuel handling operations. The disconnect was made with the control rods in their down or least reactive position. The control rods remained in this position during fuel handling operations.

Two safety rods were provided in the reactor in addition to the 12 operational control rods. The safety rods were not a part of the normal operational control system of the reactor. The safety rods were always in the reactor and they were designed to function when the control rods were disconnected from their drives. The primary purpose of the safety rod was to provide available negative reactivity when the reactor was shut down and the control rods were disconnected. They provided a safety factor during reactor loading operations. The safety rod drive mechanisms were separated from and completely independent of the control drive systems. The drives and vertical drive shafts were located outside the reactor and rotating plugs, completely independent of the fuel handling system and reactor components as shown in Figure 3-19. Low level detectors separate from the normal operational control system actuated them.

The control rod drive mechanism performed three major functions: the connection between the drive and the control rod, the slow-speed vertical motion (in both directions) for reactor control, and the high-speed downward motion for reactor scram. These operating functions were combined in a single unit and were appropriately interlocked to ensure proper operation.

The control rod drive mechanism was attached to the control rod by means of a gripper. The gripper attached to the conical top of the control rod adapter (which was also used for fuel handling operations of the control rods). The gripper consisted of two jaws that engaged the control rod adapter and was operated by a cam incorporated in a sliding sleeve; the engagement was very similar to the fuel handling gripper to subassembly engagement during fuel handling operations. Jaw operation was positive; the jaws were opened and closed by the cam, and were locked in position by the cam. The jaws operated through a funnel-shaped guide tube and upon opening, receded beyond the guide tube, providing a smooth interior surface. This eliminated the possibility of the control rod adapter hanging up after the jaws were opened.

The gripper also contained a sensing device that made contact with the top of the control rod adapter. It consisted of a plunger made to move 1/2 inch in a vertical direction by the control rod adapter during engagement and disengagement of the control rod from the gripper. It was spring-loaded and the motion of the sensing plunger was transmitted to a position indicator. If necessary, it could also be used to forcibly eject the adapter from the gripper. A third check was also provided to eliminate the very unlikely possibility of the control rod adapter sticking to the sensing plunger. The relationship between the control rod adapter, the sensing plunger, and the gripper jaws was such that after the control rod was released, and the plunger was in the down position, the jaws would not close if the adapter was still in contact with the sensing plunger. Closing the jaws after the control rod had been released provided a final check that release had actually been accomplished. The arrangement of the units comprising the gripper mechanisms is shown in Figure 3-20.

The gripper device was attached to the main shaft, which extended upward through the biological shield into the operating area above the primary system. The actuating mechanism for the gripper and the sensing mechanism were located above the operating floor and were easily accessible for inspection and maintenance. The necessary motions employed to actuate the gripper and to sense the operations were transmitted by shafts from the gripper to the operating stations. The actuating mechanism shown in Figure 3-21 was constructed in such a way that the control rod could not be released except when it was in the down position of the control stroke. The position of the jaw actuating device and the position of the sensing device were indicated by transducers and were suitably interlocked into the system. The actuating device had to be in its proper position, and the sensing device had to affirm that it was, before subsequent operations could be performed.

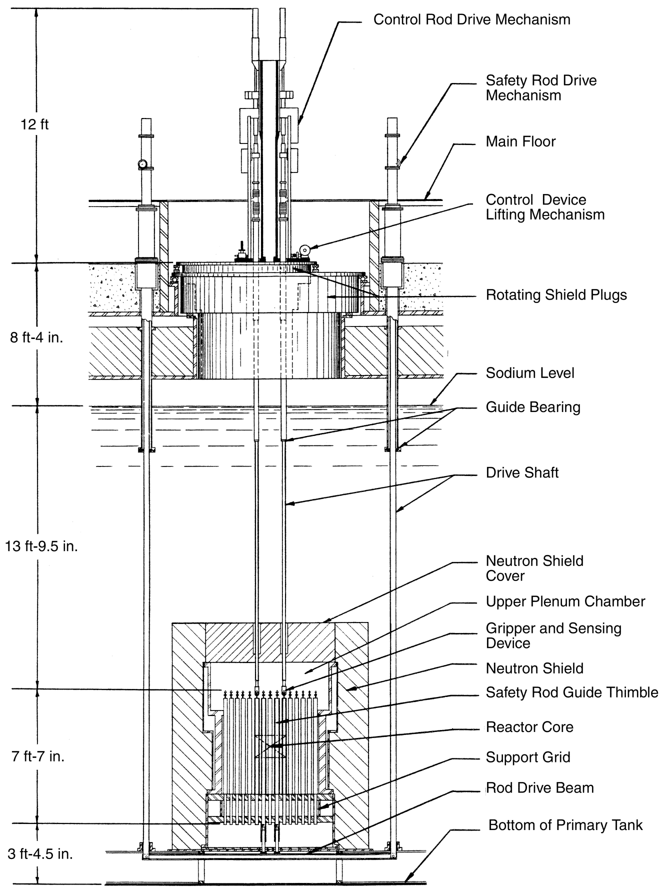


FIGURE 3-19. EBR-II CONTROL AND SAFETY ROD DRIVE SYSTEM.

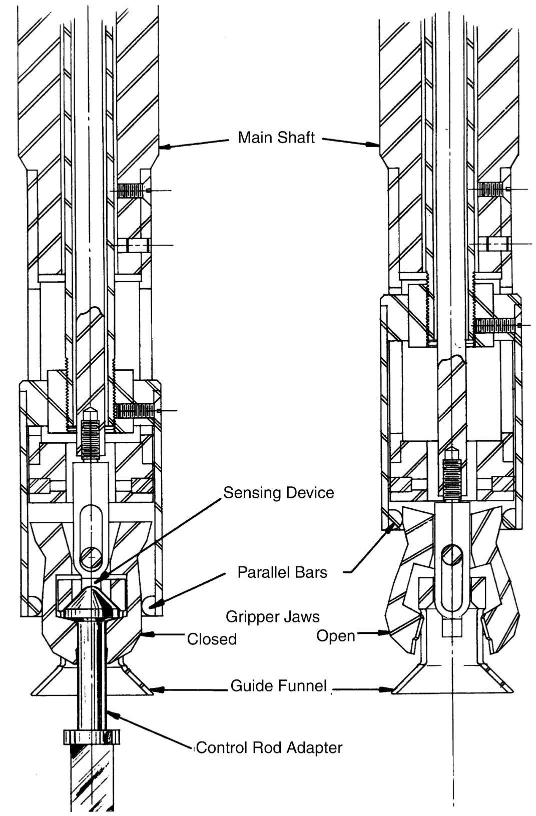


FIGURE 3-20. CONTROL ROD GRIPPER MECHANISM.

The control rod was actuated by a long shaft that extended through the upper biological shield with the control rod attached to its lower end and the drive mechanism at its upper end. The shaft was driven by a rack and pinion at a rate of 5 inches per minute by a constant-speed instantly reversible, polyphase motor. The rack gear teeth were cut on the outside of the tube through which the main drive shaft extended. The drive shaft was connected to the rack tube by a fast-acting magnetic latch. The latch consisted of two rollers that engaged notches in the shaft and were actuated by a magnetic clutch. The magnetic clutch was energized to engage the latch and thereby connect the shaft to the rack tube. The latch arrangement is shown in Figure 3-22.

The main shaft extended upward through the rack tube and was attached to a piston in a pneumatic cylinder. The upper end of the cylinder contained compressed air at a pressure of approximately 50 pounds per square inch gauge. The lower end of the cylinder was open to the atmosphere. The pneumatic pressure was always acting against the piston, tending to drive the shaft, and thus the control rod, down. The latch connecting the shaft to the drive rack prevented motion. Upon a scram signal, the magnetic clutch was de-energized, releasing the shaft from the drive rack and driving the control rod down, out of the reactor core. Scram could occur at any position in the operating stroke of the control rod and was automatically actuated by a power failure, which de-energized the magnetic clutch. This was accomplished in a release time of 0.008 second, including the time elapsed between actuating the scram signal and beginning of shaft motion. To ensure the compressed air supply to the air cylinder, accumulator tanks were provided, which in turn were supplied by an air compressor. Check valves were provided in the connecting lines between the accumulator tanks and the air cylinders, and between the air compressor and the accumulator tanks, to prevent loss of compressed air in the

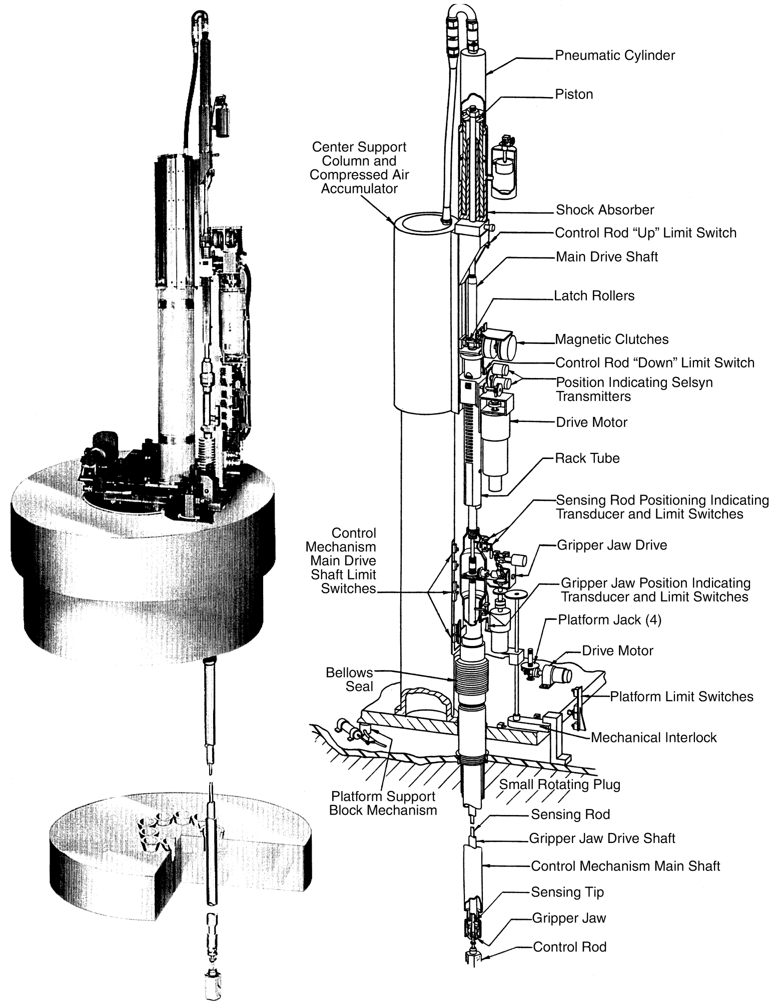


FIGURE 3-21. CONTROL ROD DRIVE MECHANISM.

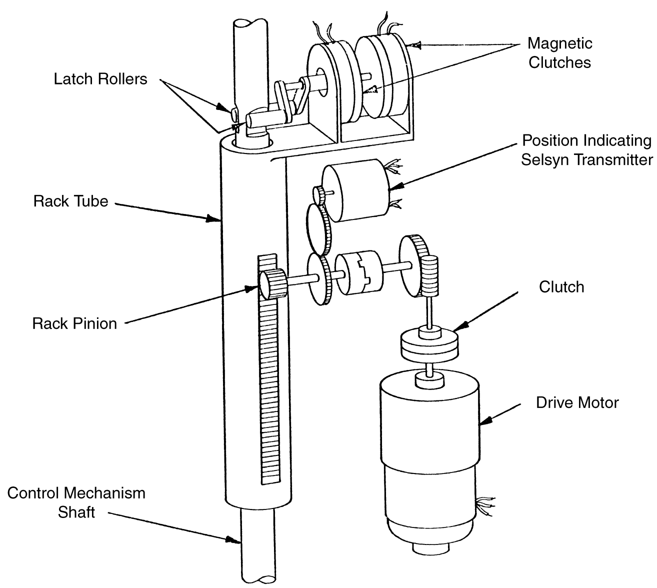


FIGURE 3-22. CONTROL DRIVE AND LATCH MECHANISM.

event of line failure. Pressure actuated switches scrammed the reactor in the event of failure of the air supply. The compressed air available in the cylinder or in the accumulator tanks was sufficient to insure pressure assist during a scram, in addition to the force of gravity. Deceleration of the scram stroke was accomplished by a hydraulic shock absorber connected to the air cylinder. The shock absorber was actuated during the lower 5 inches of travel.

A mechanical stop for upward motion, when the piston reached the top of the air cylinder was built into the system. If the limit switches on the rack driving pinion failed to stop the unit at the upper end of its travel, the drive shaft was stopped, including the control rod, and the rack continued to travel, moving away from the shaft and disengaging the latch. When this occurred, the shaft and the control rod were automatically scrammed by the disengagement of the latch. Over travel of the control rods was prevented and was not dependent upon the operation of the limit switches.

The 12 control drive mechanisms were mounted on a platform that surrounded a central support structure. The platform could be raised 3 inches and lowered 3/4 inches from its normal operation position. The upward movement was required to raise the lower end of the drive mechanisms, after disconnect from the control rods, to clear the subassembly adapters during fuel handling operations. The bottom position of the normal control rod stroke held the control rod 3/4 inches above its bottom seat in the guide sleeve. When released from the gripper, the control rod dropped and was supported by the control and thimble guide sleeve. The downward movement of the platform was required to engage the control rods and grip them when they were in the down position.

The design of the control rods and drive systems was extremely challenging. The space available was very limited due to the close spacing of the rods in the reactor and the demands for very high reliability of operation. A photo of a single drive unit and the cluster of 12 drives is shown in Figure 3-23.

The two safety rods (Figure 3-9 and Figure 3-10) were connected beneath the reactor to a horizontal bar which was connected to two vertical shafts which extended upward outside the biological shield. Each shaft was coupled to a rack tube by a magnetic clutch latch arrangement similar in design to one described above for the control rod drive. The rods were driven by synchronous motor drives and simply raised the system to the cocked position. When the latch was released, the drive mechanism and the safety rods fell 14 inches under the force of gravity. A pneumatic shock absorber decelerated the mechanism during the last 5 inches of movement. All reactor operations, including actuation of the control system or actuation of the fuel handling system, required the safety rods to be in the up position. The safety rods were connected to the horizontal support bar and the entire system acted as a unit with the support bar and both rods being dropped simultaneously.

FUEL HANDLING SYSTEM

EBR-II utilized a series of unique processes to handle reactor fuel (and blanket and other reactor components). These processes were divided into two broad categories:

- Those that were performed with the reactor shut down, designated as unrestricted operations.
- Those that were performed with the reactor in operation, designated as restricted operations.

The restricted operations could be further subdivided into fuel transfer operation and fuel transport operation.

Fuel handling operation involved the movement of subassemblies between the reactor and intermediate storage in the storage rack. All of these operations were performed in the primary tank with the subassemblies submerged in, and cooled by, sodium. These operations were performed with the reactor shutdown, and the control rods disconnected from their drives. Since these operations were performed with the subassemblies submerged in sodium they were not visible. Although the process consisted of a series of relatively simple operations, they were complicated by lack of visibility. They are described here in considerable detail to convey the level of attention that was provided to achieve reliable operation.

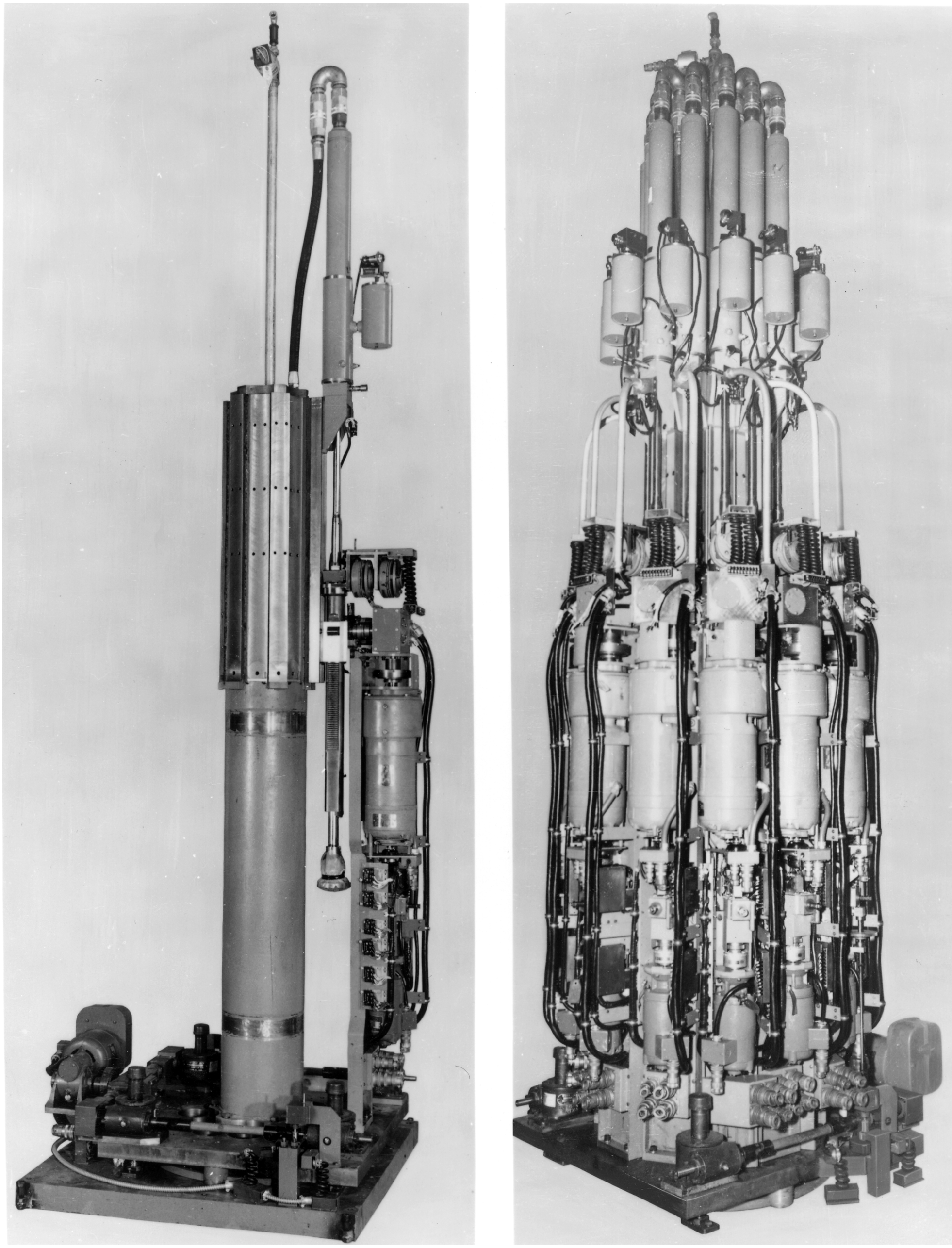


FIGURE 3-23. SINGLE DRIVE UNIT AND CLUSTER OF 12 DRIVES.

Fuel transfer operations involved the transfer of subassemblies between the storage rack and the inter-building coffin. This operation involved the use of the fuel unloading machine to effect this transfer, including the transition from sodium as the coolant to gas as the coolant for spent assemblies and the transition from gas to sodium for new or reprocessed subassemblies. These transitions occurred as the subassemblies were transferred between the sodium environment in the primary tank and the inert gas environment in the fuel unloading machine and vice versa. Fuel transport operations involved the transport of subassemblies between the inter-building coffin transfer station in the Reactor Plant and the air cell in the Fuel Cycle Facility. Included were the transport through the equipment air lock between the buildings, which maintained the containment integrity of the reactor containment building. Both the fuel transfer and fuel transport operations could be performed while the reactor was operating. It was intended that these operations be performed as needed to meet the requirements of the external fuel cycle (i.e., recycle or storage/disposal). The fuel transfer and transport systems will be described later.

The fuel handling system included the gripper and hold-down mechanisms for removing subassemblies from the reactor and installing subassemblies in the reactor; and the transfer arm for transferring subassemblies between the gripper mechanism and storage rack. It also included the rotating plugs and their freeze seals, and certain equipment involved in preparatory operations to permit fuel handling.

The freeze seals for the two rotating plugs provided a combination molten-frozen seal which permitted freezing the upper portion of the seal while retaining the lower region in a molten state. The frozen upper region prevented seal metal loss in the event of a large pressure differential across the seal, while the molten lower region prevented leakage. The entire seal was melted, of course, to permit rotation of the plugs for fuel handling.

After the reactor was shut down, the 12 control rods were released from their individual control rod drive mechanisms and the control rod drive platform was raised 3 inches so that the drives would clear the control rods. The reactor cover hold-down clamps that fastened the cover to the reactor tank were released. The cover elevating columns were raised by two synchronized electric motor-driven lifting mechanisms located on the small rotating plug. In the raised position, the reactor cover engaged pins extending from the cover into the small rotating plug to prevent swinging of the relatively heavy mass (approximately 17 tons) during plug rotation; the reactor cover rotated with the small rotating plug. The cover was raised 9 feet 8 inches to provide clearance below it for removal of subassemblies from the reactor and transfer to the storage rack.

The reactor was now prepared for fuel handling operation. The two rotating plugs were rotated to the proper location to position the gripper over the desired subassembly. Both plugs were supported by roller bearings and rotated by electric motors. In addition to the rotation of the two plugs to the required location, it was also necessary to rotate the gripper unit about its centerline to provide the correct angular orientation of the gripper head.

All operations involved in the fuel handling cycle included provisions for maintaining a known angular orientation of the subassembly. Three locations on the subassembly established its angular orientation:

- The cone-shaped adapter was slotted and engaged a blade in the gripper mechanisms
- The section below the collar was rectangular and engaged a slot at the end of the transfer arm
- The lower nozzles of the subassemblies were slotted and engaged orientation bars in the reactor grid and the storage rack.

Each of these angular orientation controls on the subassemblies was in the same plane. Control of angular orientation and knowledge of angular orientation was maintained at all times during fuel handling.

All of the components and mechanisms involved in these operations were positioned on and in the small rotating plug (Figure 3-24). The small plug, in turn, was located in, and positioned by, the larger plug (Figure 3-25). These various operations were controlled by, and provided feedback for, a variety of circuits requiring a large number of conductors. These were arranged in two multi-conductor cables, one supplying the large plug and one supplying the small plug. To accommodate the rotation of these plugs, these cables were positioned by a festoon cable system, which provided the extension and contraction required as the plugs rotated as shown in Figure 3-26. To minimize the number of conductors involved, the control rod drive conductors were connected only in the reactor operation position of the rotating plugs by multi-conductor electrical plugs that were manually disconnected before fuel handling operations began.

The rotating plugs and gripper head were rotated to the proper position for the particular subassembly to be removed. There was an angular position for each of these three rotating units for each lattice position in the reactor. In preparation for gripping a subassembly, the hold-down mechanism, consisting of a funnel-shaped sleeve, was lowered by an electrically driven screw over the subassembly to be removed. It contacted the six adjacent subassemblies to be removed. This arrangement is shown in Figure 2-9. The hold-down sleeve also acted as a guide for the gripper mechanism.

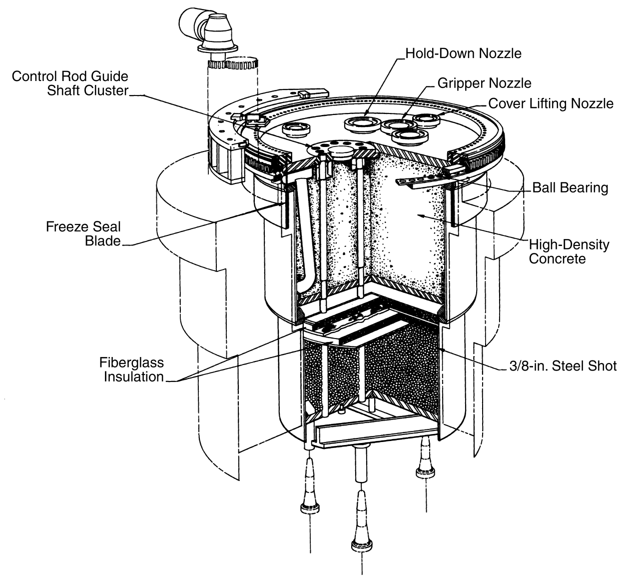


FIGURE 3-24. SMALL ROTATING SHIELD PLUG.

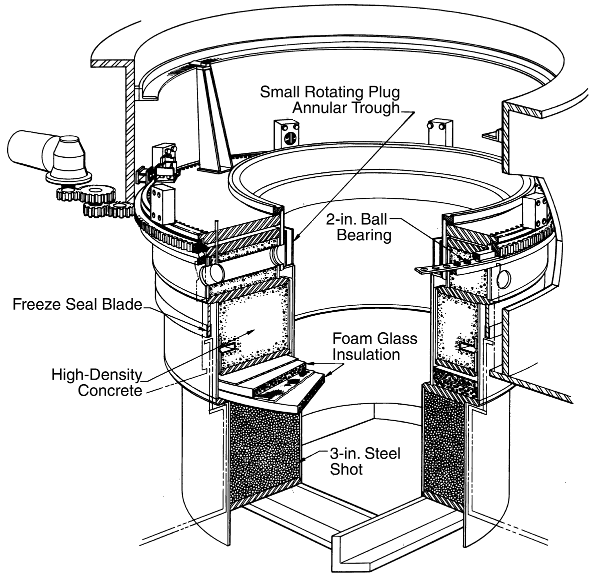


FIGURE 3-25. LARGE ROTATING SHIELD PLUG.

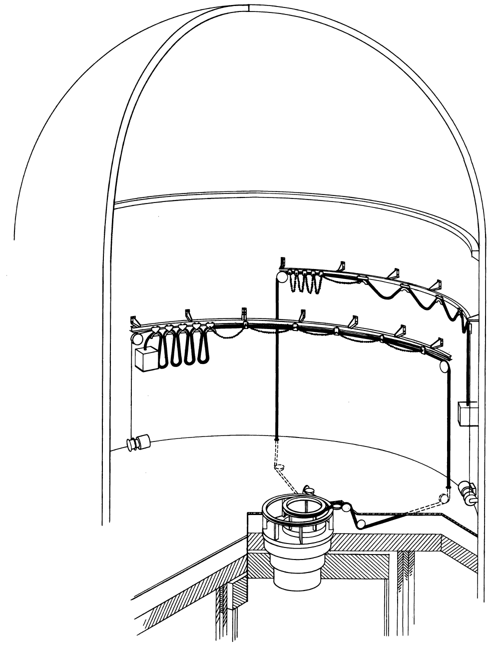


FIGURE 3-26. FESTOON CABLE SYSTEM.

The gripper head was lowered through the hold-down sleeve and contacted the adapter on the subassembly. The gripper device on the lower end of the mechanism gripped the subassembly adapter in the same fashion as the control drive gripper described earlier (Figure 3-20). The orientation blade between the gripper jaws engaged the slot in the conical shaped head. The sensing device also functioned as previously described. The gripping mechanism was moved vertically by an electrically-driven screw drive and the gripper jaws were motor-operated. After the subassembly had been raised out of the reactor, the hold-down tube was raised around the suspended subassembly and provided a lateral support during movement of the two rotating plugs to prevent the subassembly from swinging.

The plugs were rotated to the transfer point, and the gripper head was rotated to the transfer position. The slotted section of the transfer arm engaged the rectangular section of the subassembly adapter to maintain proper angular orientation. The collar of the subassembly adapter fit into the counter-bored recess on the transfer arm holding device when the subassembly was lowered by the gripper mechanism as shown in Figure 3-27. The locking bar on the transfer arm holding device

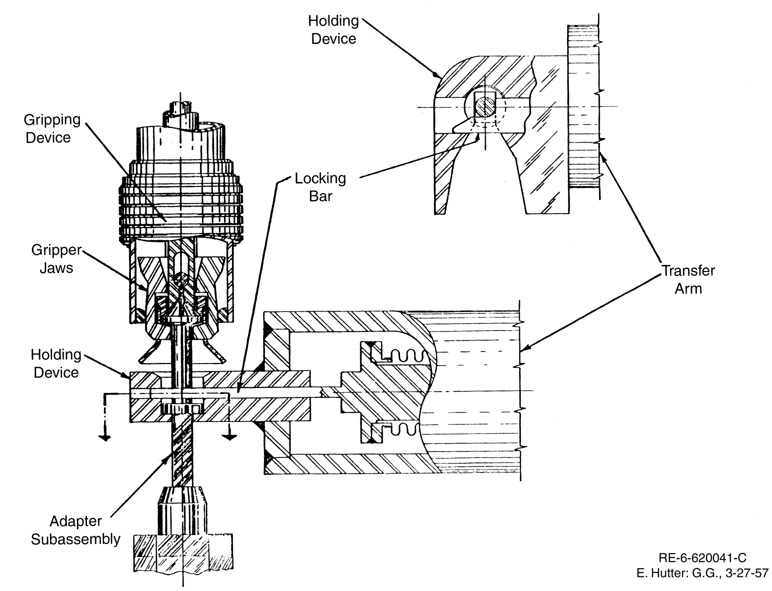


FIGURE 3-27. SUBASSEMBLY TRANSFER.

locked the subassembly positively to the transfer arm. The subassembly was then released from the gripper, the gripper was raised, and the hold-down was lowered below the subassembly.

The transfer arm was rotated through a horizontal arc of about 80 degrees and positioned the subassembly above any one of three concentric rows of storage locations in the storage rack shown in Figure 3-28. The transfer arm was operated manually, and several checkpoints could be felt by the operator. For example, the physical contact between the transfer arm and subassembly at the transfer position was felt by a wiggle test: the transfer arm could not be moved while the subassembly was held by the gripper and hold-down sleeve, and attempting to move it provided a check that the transfer had been made correctly. Similar checks were made between the transfer arm and the storage rack.

The storage rack was a cylindrical vessel providing 75 storage locations for subassemblies in three concentric rows. The storage rack was suspended by a shaft connected to a drive mechanism that provided rotation and vertical movement to the storage rack in the primary tank below the sodium level. An empty storage location was positioned below the subassembly, which was suspended from the transfer arm at the proper angular position. The transfer arm was lowered to provide initial engagement of the subassembly with the storage rack. This was a

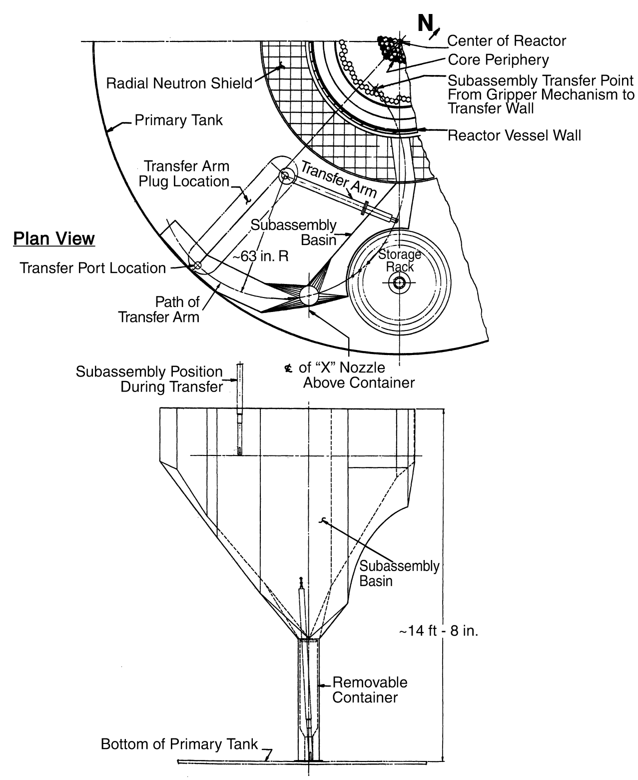


FIGURE 3-28. SUBASSEMBLY BASIN.

manual operation and the operator could feel that the subassembly had entered the storage position. The storage rack was then raised. At the end of the upward movement, the subassembly was lifted from the holding device on the transfer arm.

An additional checkpoint existed here. As long as the subassembly was held jointly by the storage rack and the transfer arm, the transfer arm could not be moved, indicating proper operation of both mechanisms. Following this check, the transfer arm locking bar was released and the transfer arm was rotated to a neutral position while the storage rack was lowered. To remove a subassembly from the storage rack by the transfer arm, the process was reversed.

Although the gripper, transfer arm, and related control circuits were designed to prevent accidental release of a subassembly, provisions were made for recovery. A subassembly catch basin (Figure 3-28), consisting of a funnel-shaped trough, traversed the area under much of the transfer arm path, the arc between the reactor, storage rack, and transfer port. All sides of the funnel sloped toward a depression located directly below an access nozzle (the “X” nozzle) in the primary tank cover. If a subassembly was accidentally released and dropped from the transfer arm at any point along its travel other than over the reactor, the radial neutron shield, or the storage rack, it would drop into the basin and slide to the retrieving position, standing in a near vertical attitude. In this position, the subassembly could be grappled through the access nozzle. Natural convection cooling of this subassembly would occur in a similar fashion as for subassemblies located in the storage rack.

The basin was thoroughly tested after installation by repeated, deliberate dropping of a dummy subassembly from the transfer arm. During the reactor operating lifetime, two subassemblies were dropped out of about 40,000 transfer operations. One subassembly was dropped into the subassembly basin and one on top of the reactor. The first was retrieved as described above; ingenuity and patience retrieved the second. Both experiences will be described as a part of the EBR-II operational experience in other documents.

Primary Tank and Biological Shield

The primary tank, primary tank support structure, biological shield, and shield cooling system comprised an integrated system, designed to meet static load requirements, maintain accuracy of alignment, and contain internal energy release. As shown in Figure 2-12, the tank was surrounded by and supported by the primary structure that included the biological shield.

The primary tank and support structure were separate except at the top. Much of the equipment entering the primary system was large and heavy, requiring adequate support, as did the primary tank itself. The low temperature top structure was designed to support these loads.

The primary structure (Figure 3-29) was also designed to contain the energy release associated with a hypothetical nuclear accident. For design purposes, an energy release equivalent to 300 pounds of TNT at the center of the reactor was assumed. Although the primary tank would be destroyed, the primary structure surrounding the tanks was designed to contain this energy release without failure. It should be noted that these design assumptions were developed in the 1950s and were a very conservative substitution for experience and technology. The operating experience and operational response characteristics of EBR-II would suggest that the EBR-II design was extremely conservative.

The primary tank was constructed with double walls, a tank within a tank to provide maximum reliability of sodium containment. Both the inner and outer tanks were constructed of Type 304 stainless steel. The inner tank had a 26-foot internal diameter. The sidewalls were constructed of 1/2-inch-thick plates, while the tank bottom was constructed of 1-1/2-inch-thick plate. The outer tank sidewalls were constructed of 1/4-inch-thick plates while the tank bottom was constructed of 3/4-inch-thick plate. The 5-inch annulus between the two tanks was filled with an inert gas, which was monitored to detect sodium or air leakage through either tank wall. The outside of the outer tank was insulated to minimize heat loss from the primary system.

The inner tank bottom plate structure was designed to support the reactor tank, the subassemblies, neutron shield, and the entire sodium load. The tank wall transferred this load to

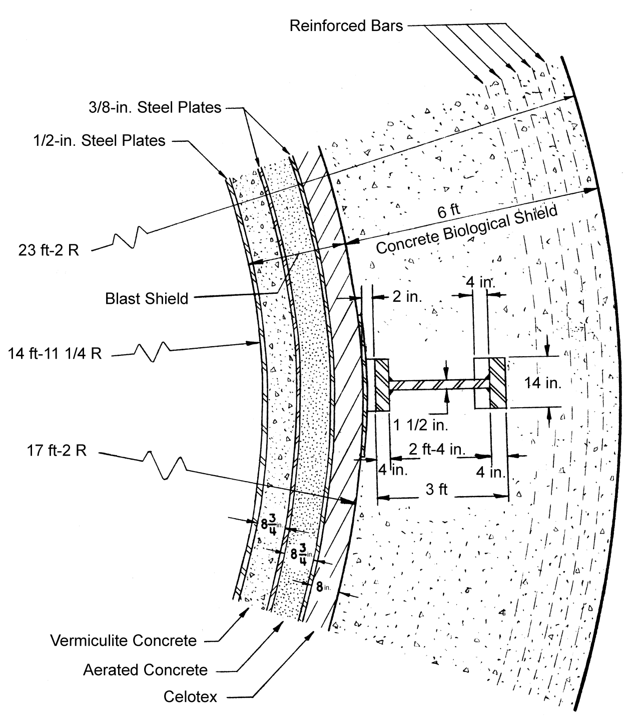


FIGURE 3-29. BLAST SHIELD AND TYPICAL COLUMN DETAIL FOR PRIMARY TANK SUPPORT STRUCTURE.

the top cover where the tank was supported. The outer tank structure was designed to carry only the sodium load in the event of a leak in the inner tank. The bottom of each of the tanks was stiffened with radial beams. The criteria used in the bottom plate structure design were as follows:

1. The inner tank bottom plate structure was designed to support the full load with a maximum deflection of 1/4 inch at a temperature of 750°F. This small deflection was established to minimize misalignment between the reactor and the upper structure of the primary system.

2. The outer tank bottom plate structure was designed to support the uniformly distributed sodium load with an allowable bending stress in the plates and beams of 14,700 pounds per square inch.

The primary tank and its contents, and those components that were connected to the primary tank top cover, were supported by six hangers welded to the top cover beams, which in turn transferred these loads to the top structure beams. Each hanger was supported on a roller to permit radial thermal expansion of the primary tank cover as shown in Figure 2-12.

The primary tank design and the method of support were arranged to provide radial expansion about the vertical centerline of the system. The most critical units, the reactor and the rotating plugs that located the control drives and the fuel handling mechanisms, were located on the physical centerline of the system. Differential vertical expansion was minimized by the use of identical material for all equipment in the system, and maintaining it at the same temperature.

The primary tank support structure (Figure 2-18) consisted of a system of columns and beams that transmitted the loads to the main internal building foundation. In combination with the biological shield, it formed a pressure vessel surrounding the primary tank. The columns were connected to each other by horizontal beams at the bottom, and embedded in the heavily reinforced concrete. These columns were connected at the top to six radial beams which framed into a circular ring (6 inches thick) located on the centerline of the system. With some additional stiffening, this top structure provided the supporting structure for the primary tank and for the major primary system components external to the primary tank. A ring of ordinary concrete (6 feet thick) provided the radial biological shield; the inside diameter was at essentially the same diameter as the inside of the six vertical columns (Figure 3-29).

The radial biological shield and structure was continuous except at an elevation near the top of the primary tank where it was penetrated by several horizontal offset holes, approximately 8 inches in diameter, for the ventilation ducts required for shield cooling. The shield was heated by the heat loss from the primary system, and by energy absorbed in attenuating neutrons and gamma rays. The heat was removed to avoid overheating the steel plates and the concrete.

The shield was cooled by forced circulation of air. It was essentially a recirculation system, however, a fraction of the air was continuously drawn into the system and an equal amount was discharged through the building exhaust system. The shield cooling system operated at a pressure slightly below that of the building atmosphere. This provided in-leakage and also simplified certain areas in the shield that could not be connected to a closed circulation system. The top structure and the shield plugs installed therein were cooled by air drawn from the building atmosphere. The radial shield and the structure below the primary tank were cooled primarily by re-circulated air. Figure 3-30 is a simplified diagram describing the shield cooling system. Air from inside the building was drawn into the primary system through a duct system in the rotating plugs and in the primary top structure, and circulated around the top cover of the primary tank, through ducts in the biological shield into exhaust blowers. It joined air that had circulated through the radial shield and bottom shield air space. The flow then was split into two paths, one to the exhaust stack in the Fuel Cycle Facility, and the other through coolers.

The heat that needed to be removed by the shield cooling system consisted almost entirely of the heat loss from the primary system, the heating in the shield due to neutron and gamma ray attenuation being only a small fraction of the total heat load. The total heat load was approximately 430,000 British thermal unit per hour, of which 415,000 British thermal unit per hour was the heat loss from the primary tank, and approximately 15,000 British thermal unit per hour was due to the neutron and gamma attenuation in the structure and shield.

An air-cooling system of 15,000 cubic feet per minute capacity with a maximum air velocity of approximately 30 feet per second was provided. Reliability of the system was achieved by auxiliary power supplied to the exhaust blowers and coolers. Because of the large heat capacity of the system, brief interruption of the cooling system was not critical.

PRIMARY SODIUM PURIFICATION SYSTEM

A recirculating cold trap system (Figure 3-31) was used for continuous primary sodium purification. This system provided impurity concentrations at or near their greatly reduced solubility limits at temperatures just above the melting point of sodium. Cold trap precipitation was effective in maintaining low concentration of such impurities as sodium hydride, most fission products, and particularly sodium monoxide.

The cold trap consisted of a 500-gallon tank filled with Type 304 stainless steel wire mesh to provide supplementary surface area to enhance sodium crystallization and deposition.

A regenerative heat exchanger was incorporated in the main sodium stream to reduce over-all heat losses in the cold trap system. The cold trap operational temperature of 350°F was maintained by a cold trap coolant loop. Plugging indicators were located on the sodium inlet and outlet sides of the cold trap (plugging/melting temperature increased with impurity concentration of the sodium). They were used to check the efficiency of the cold trapping operations and to provide sampling points for chemical and radiological analysis of the sodium before and after the purification cycle.

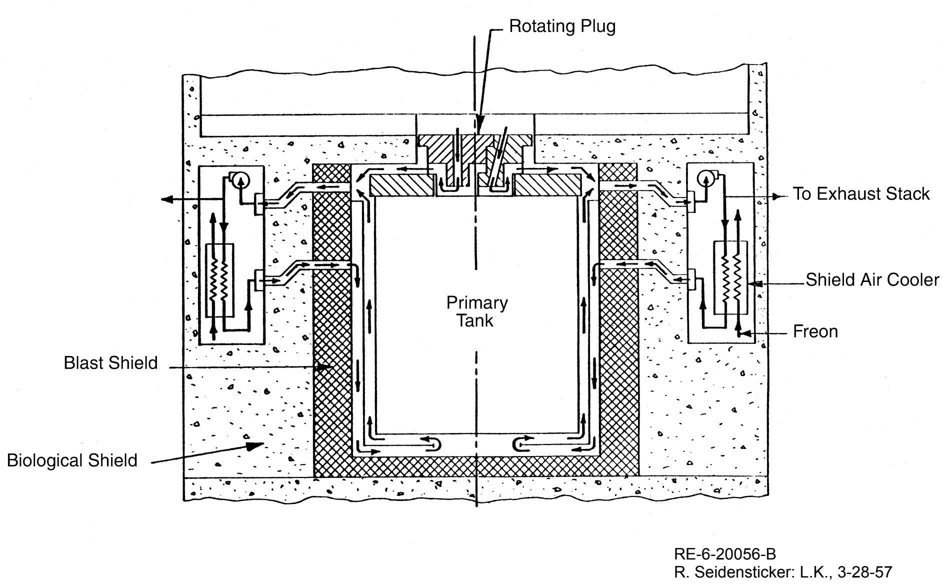


FIGURE 3-30. SHIELD COOLING AIR SYSTEM SCHEMATIC DIAGRAM.

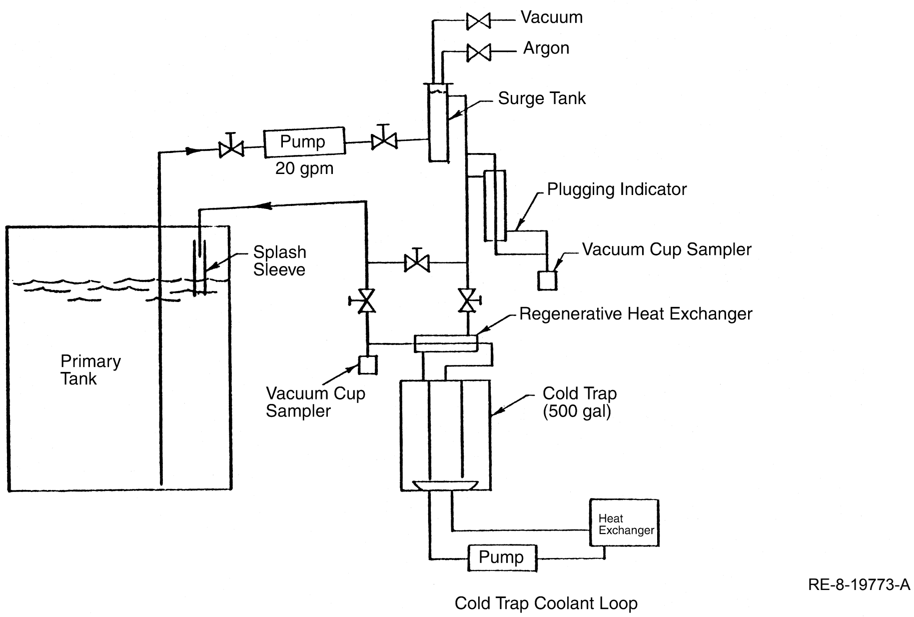


FIGURE 3-31. SODIUM CLEANUP SYSTEM FLOW DIAGRAM.

Parts of the cold trap circuit were below the level of sodium in the primary tank. Since radioactive primary sodium was circulated in the cold trap system, it was essential to eliminate the possibility of an accident or equipment failure resulting in siphoning of primary tank sodium. To avoid this possibility, a surge tank was included in the cold trap inlet line at its highest point of elevation. An argon gas blanket pressure was maintained such that, under static conditions, the sodium level was just below the surge tank discharge opening. With the pump operating, the level rose sufficiently to establish flow. The power supply to the pump was interlocked to a sodium vapor monitor at the cold trap floor level to cut out when a sodium leak was detected, thereby breaking the inlet sodium line at the surge tank. In addition, an argon gas line was provided for positive gas addition to insure breaking the sodium column in an emergency.

INERT GAS SYSTEM

It is necessary to provide an inert gas blanket over sodium. Argon was chosen for this system because of its superiority for pumping, heat transfer, and sealing. To maintain a low level of atmosphere contamination, a gas cleanup system (Figure 3-32) was provided through which the argon could be re-circulated and purified. This system maintained a static argon gas blanket over the primary sodium. The primary tank argon gas blanket pressure is maintained at a positive 1 inch ± ½ inch water pressure differential with respect to building static pressure to prevent excessive loss of blanket gas to the reactor building in the event of a leak. The slight positive pressure also prevents building air from leaking into the blanket gas and contaminating the bulk sodium. The inert gas blanket protects the primary tank sodium from contact with air. Make-up gas was added to the primary circulating gas system, as needed, from the Fuel Cycle Facility argon gas supply system. Excess gas was vented directly through filters to the exhaust stack or to a retention tank for subsequent disposal.

SECONDARY SYSTEM

The secondary system was the non-radioactive sodium heat transfer loop between the radioactive primary system and the steam system (see Figure 3-1). The principal function of this system was to transfer heat from the primary sodium system to the steam system in an efficient manner. The flow rate was 2.5×10^6 pounds per hour (approximately 6,000 gallons per minute). The heat exchanger inlet temperature was 588°F and the outlet temperature was 866°F. The principal components of the secondary system, in flow sequence, were the sodium circulating pump, the heat exchanger, the steam superheater and the steam evaporator.

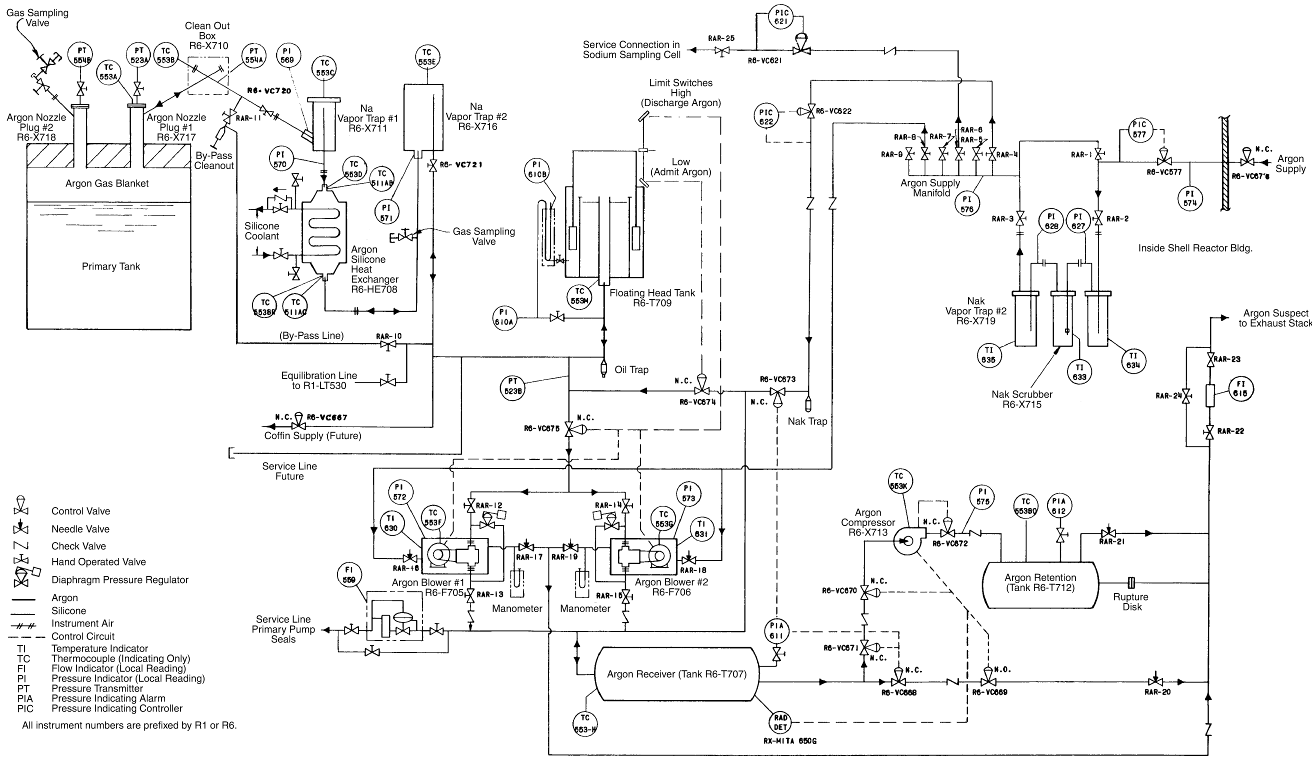


FIGURE 3-32. ARGON BLANKET GAS SYSTEM.

The circulating pump was an alternating current linear induction electromagnetic pump with a capacity of 6,500 gallons per minute at about 53 pounds per square inch. Flow control down to 0 percent of nominal rating, actually to negative reverse flow, was achieved by a generator voltage regulator that used an amplidyne motor-generator set for very accurate voltage control of the main generator output to the pump.

The circulating pump was located in the Sodium-Boiler Plant building which was separated from the Reactor Plant building. This fireproof building also contained the secondary sodium purification system, sodium receiving facilities, and the sodium storage tank. The sodium storage tank was below floor level in this building and the entire secondary system sodium, except that in the heat exchanger, could be drained into this tank.

The surge tank, which was connected into the piping at the circulating pump inlet, maintained a constant head to the pump. The sodium purification system circulated 20 gallons per minute from the storage tank and discharged into the surge tank, ensuring constant level. The overflow returned to the storage tank through an internal overflow pipe in the surge tank. Argon gas at approximately 10 pounds per square inch was provided as an inert gas atmosphere over the sodium in the surge and storage tanks.

The heat exchanger was located within the primary tank in the Reactor Plant. It was suspended from the primary tank cover, and was almost totally submerged in the primary sodium. It was a shell and tube-type exchanger with the secondary sodium on the tube side as shown in Figure 3-12.

The steam generation equipment was located so as to ensure sodium drainage to the storage tank in the sodium plant. The secondary sodium passed through the superheater section and the evaporator section in series (Figure 3-33).

All piping in the secondary system was capable of absorbing thermal expansions due to temperature changes from ambient to 1,000°F. The sodium yard piping was carried on conventional concrete piers fitted with pipe guide or anchor frames as required. The yard piping was heated, insulated, and weatherproofed. Heating was accomplished by 60-cycle induction heating to maintain a temperature above the freezing point of sodium (208°F).

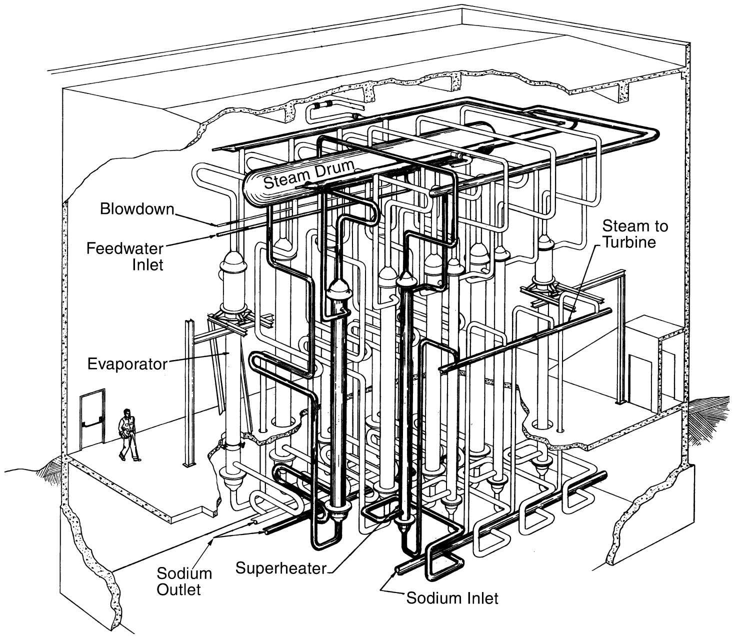


FIGURE 3-33. STEAM GENERATOR.

SODIUM RELIEF SYSTEM

The secondary sodium system included a sodium relief system to accommodate a pressure surge in the event of a sodium-water reaction. This system consisted of two duplex, 10-inch blowout diaphragms. One diaphragm was connected to each 10-inch sodium header that interconnected the superheaters and the evaporators. Each duplex diaphragm, two individual diaphragms in series, was designed to rupture at 100 pounds per square inch. Rupture of the two diaphragms allowed the sodium to flow from the header into a 1,200-gallon pressure relief tank. The tank, in turn, communicated with the atmosphere via two 12-inch lines. Each of these lines was sealed with a rupture diaphragm set for 25 pounds per square inch. The normal sodium pressure in the superheater evaporator headers was about 10 pounds per square inch.

STEAM SYSTEM

The steam system served as a heat sink for power generated in the reactor. Steam was generated at 1,300 pounds per square inch, 850°F from the heat delivered by the secondary sodium system. At 62.5 megawatt thermal reactor output, the steam generator system delivered 248,000 pounds per hour of superheated steam to a conventional 20 megawatt turbine generator system. An induced draft cooling tower provided low-temperature heat rejection.

A steam by-pass system was incorporated around the turbine to permit absorption of all energy produced in the reactor independent of electrical output. The condenser was sized to accept 100 percent of the steam generated.

Steam conditions were selected to provide maximum stability to the heat transfer loops with respect to system temperatures. The saturation temperature of 1,300 pounds per square inch steam (580°F) approximated the minimum temperature of the secondary system. This resulted in a constant high temperature heat sink provided that the steam pressure was maintained constant, which was readily accomplished. The temperature of the secondary sodium seen by the primary sodium coolant system was essentially constant under all conditions of operation.

Achieving reliability of the steam generator unit was a primary objective of EBR-II. High thermal stresses were known to have contributed to failures in other steam generators. In an effort to minimize thermal stresses in the EBR-II steam generator, special feedwater temperature requirements were established. In addition to normal feedwater heating by steam extraction from the turbine, an additional heater supplied with steam directly from the 1,300 pounds per square inch system raised the feed-water temperature further. In this manner, the feed-water was heated to 550°F over the entire load range resulting in a very small temperature difference between the feedwater and the evaporator water (580°F).

The steam generator consisted of a natural circulation evaporating section, a conventional steam drum, and a once-through superheating section. The evaporation section consisted of eight identical shell and tube heat exchangers connected in parallel on the tube side to a horizontal, overhead steam drum with conventional moisture separation internals. Saturated steam flowed from the top of the steam drum downward through vertical shell and tube superheaters, to the turbine generator unit.

The original design of the steam generator system is shown in Figure 3-33 and the original design of the evaporator units and superheater units is shown in Figure 3-34. They were constructed entirely of 2-1/4 percent chromium—1 percent molybdenum material and utilized double-walled tubes. Each duplex tube consisted of two seamless tubes which were individually inspected as single-wall tubes and again inspected as a duplex tube. The units have double-tube sheets at each end; the outer tube was welded to the sodium tube sheet and the inner tube was welded to the steam tube sheet. The space between the two tube sheets communicated directly with the atmosphere. No weld existed in these units with sodium on one side and water and/or steam on the other side. As a result, the only direct path between sodium and water and/or steam was across two seamless tubes that had been individually and jointly non-destructively inspected.

The basic design concept of the evaporators and superheaters was very similar. They were both double tube-double tube sheet designs with sodium and water/steam separated by two barriers. The most significant difference between the two units was the tube diameters and the tube wall thickness. The superheater tubes were smaller in diameter and had a thinner wall. This difference caused a welding problem that impacted the construction of the units.

Difficulty was encountered in fabrication of the superheater units; more specifically, the outer tube-to-sodium tube sheet welds (see inset, Figure 3-35). Many sound and reliable welds were made, but not consistently. This welding problem did not prevail during fabrication of the larger tube evaporators. The smaller tube diameter and thinner wall of the super heater tube could not be made reliably with the welding techniques available at the time. An alternate method of superheating was selected.

Spare parts of evaporator units were available for the fabrication of two additional evaporators. These two units were modified to serve as superheaters. The major difference was the addition of a core tube in each evaporator tube with a 0.812-inch outside diameter that provided increased steam velocity in the 0.125-inch steam annulus (see Figure 3-35).

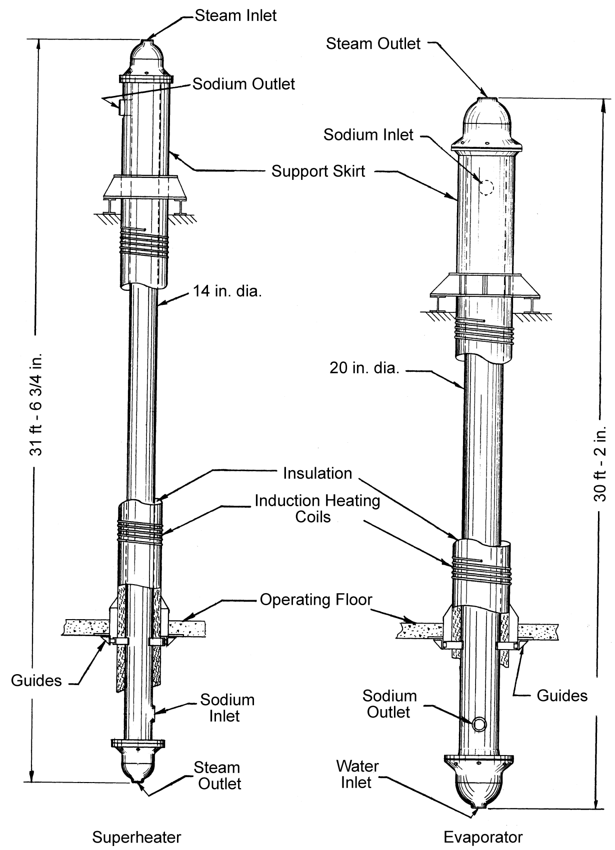


FIGURE 3-34. SUPERHEATER AND EVAPORATOR ASSEMBLIES.

The use of two modified evaporators as substitute superheaters resulted in a reduced steam temperature and a slight increase in the moisture content in the final stage of the turbine. The slight moisture increase did not seriously affect operation of the machine during its 30-year operating lifetime.

At the time this decision was made, it was considered a temporary solution to place the plant in operation, but satisfactory operation of the plant was achieved without any change and the temporary fix was made permanent. As noted earlier, achievement of high thermal efficiency was not an EBR-II primary objective, but reliable operations was.

Owing to the external similarity between the evaporators and superheaters, only minor modifications to the building, supporting structure, and piping were required to effect use of the modified evaporators.

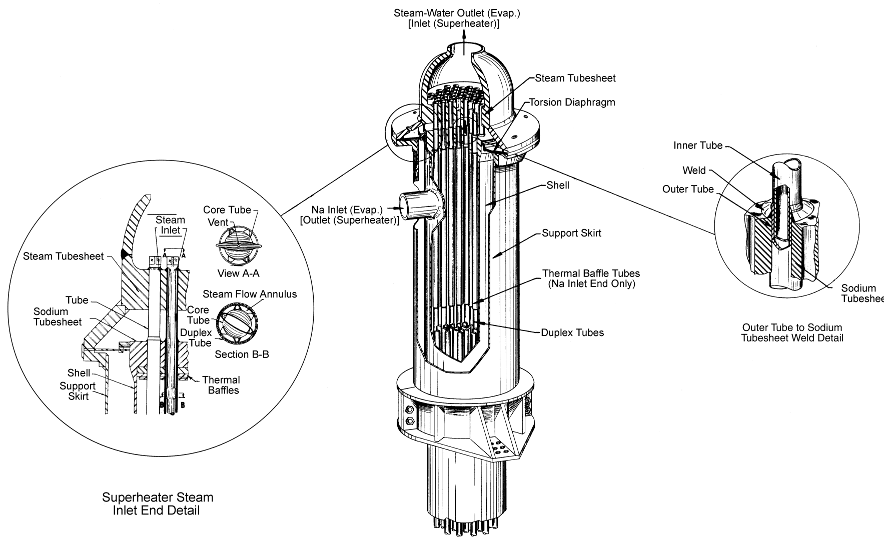


FIGURE 3-35. EVAPORATOR AND "MODIFIED SUPERHEATER" DETAILS.

FUEL TRANSFER AND TRANSPORT SYSTEMS

As described earlier, the fuel handling system delivered subassemblies to the storage rack in the primary tank where they were stored and ready for recycle or other disposition. Subsequent operations involved a two-step process; fuel transfer and fuel transport. These operations could be performed with the reactor in operation. They were independent of reactor operation and were coordinated with the fuel cycle. Although very similar procedures and processes were involved in the transfer and transport of fuel assemblies for either recycle or other disposition, only the operations involved with recycle will be described here. The equipment and components involved in this process are depicted in Figure 3-36. Restricted operation is indicated since the reactor is shown in the operating configuration.

FUEL TRANSFER

The fuel transfer system moved subassemblies between the storage rack in the primary tank and the inter-building coffin. The mechanical components included the storage rack, transfer arm, transfer port, fuel unloading machine and the inter-building coffin (Figure 3-36).

The transfer port provided access to the inside of the primary tank through the transfer arm nozzle. It provided the link between the transfer arm inside the primary tank and the fuel unloading machine, which operated on the main floor above the transfer port. The transfer port was basically a large, manually operated valve, normally closed, which was opened while attached to the fuel unloading machine to permit subassembly transfer. It had provisions for argon gas purging as required for fuel transfer operations.

The fuel unloading machine (Figure 3-37) was an electro-mechanical device that transferred fuel subassemblies from the transfer arm inside the primary tank to the inter-building coffin outside the primary tank. The machine was basically a shielded container-carriage assembly, mounted on a set of tracks on which it traveled between the transfer port and the inter-building coffin. The internal mechanisms included a gripping device, and an argon gas circulating system.

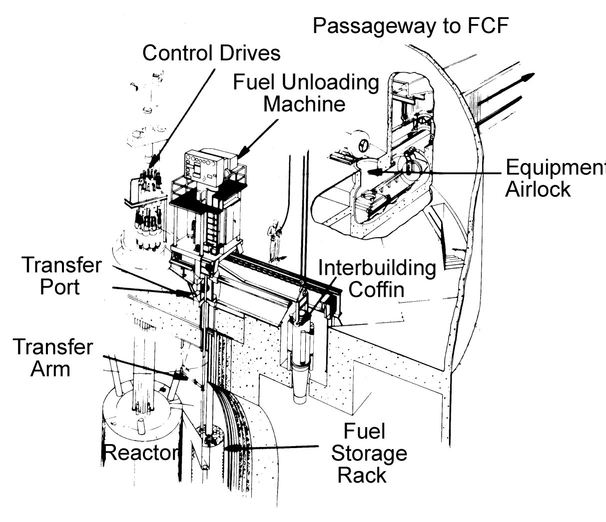


FIGURE 3-36. FUEL HANDLING SYSTEM.

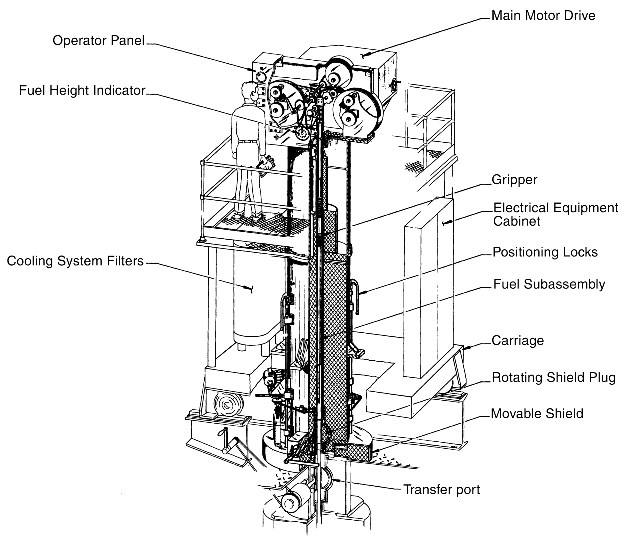


FIGURE 3-37. FUEL UNLOADING MACHINE.

Briefly, the sequence of fuel subassembly transfer was as follows. An irradiated subassembly was removed from the storage rack by the transfer arm and aligned directly under the transfer port. The fuel unloading machine was positioned over and sealed to the transfer port. The transfer port was purged to provide a total argon gas environment, i.e., in the primary tank, the transfer port and the fuel unloading machine. The gripping device was lowered through the transfer port to the level of the transfer arm to engage the subassembly. The transfer arm was disengaged from the subassembly. The subassembly was lifted into the fuel unloading machine, transported to and lowered into the inter-building coffin. The reverse procedure was employed to transfer a recycled subassembly into the storage rack from the inter-building coffin.

The argon gas circulation system on the fuel unloading machine was used to:

- Drain excess sodium from the subassembly as it was removed from the primary tank
- Cool the spent fuel during the transfer to the inter-building coffin
- Preheat a recycled subassembly before insertion into the primary tank sodium.

The transition from sodium cooling to inert gas cooling occurred during the fuel transfer process as the subassembly was raised into the fuel unloading machine. The reverse process occurred when a reprocessed subassembly was preheated and transferred from the fuel unloading machine to the storage basket.

INTER-BUILDING FUEL TRANSPORT SYSTEM

Inter-building fuel transport involved the transportation of a fuel, blanket, or other, subassembly from the Reactor Plant to the Fuel Cycle Facility or the transportation from the Fuel Cycle Facility to the Reactor Plant. The primary vehicle for performing this task was the inter-building coffin. Transporting this 15-ton carrier required cranes, carriages and controlled passage through the equipment air lock between the two buildings.

The inter-building coffin was a portable, sealed, shielded vessel with an integral argon gas (or air) cooling system. The cooling units on the inter-building coffin were battery-powered to ensure continuous operation in the event of transport difficulties or power failure during transit inside and between the buildings. The primary functions of the inter-building coffin were to provide radiation shielding of subassemblies during transport and to provide cooling to remove the heat generated by fission product decay.

A simplified routing of the inter-building coffin is shown in Figure 3-38.

The first step in the fuel cycle after the inter-building coffin arrived in the Fuel Cycle Facility involved removal of the sodium adhering to the subassembly. This was radioactive primary sodium. The predominant isotope was sodium-24, which has a short half-life of 15 hours. The subassembly was blanketed by argon gas that was circulated to cool the spent fuel, and was the environment in which the process began. The sodium was removed from the subassembly in the inter-building coffin in the following steps:

1. Admit oxygen diluted by nitrogen to the circulating gas system to oxidize the sodium adhering to the subassembly.
2. Admit water vapor in nitrogen to allow further reaction with the sodium.
3. Flow water through the coffin and subassembly to remove the oxidized sodium and assure that all sodium was oxidized and removed.

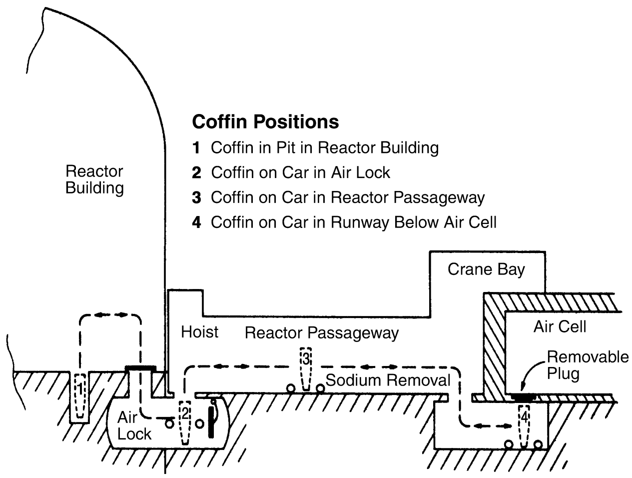


FIGURE 3-38. MOVEMENTS OF SUBASSEMBLY COFFIN BETWEEN REACTOR AND FUEL CYCLE FACILITY.

4. Immediately after the water wash, blow out the water and dry the inter-building coffin and subassembly with a stream of air.

At this point the subassembly was in an air environment and air became the circulated coolant to remove fission product decay heat. The gas cooling system on the inter-building coffin continued to perform this function. Argon gas or air were the cooling medium as appropriate.

The final step in the transport process involved delivery of the subassembly to the air cell for dismantling. The inter-building coffin was lowered onto a cart below and adjacent to the air cell and then moved into position below the air cell, permitting the air cell crane to lift the subassembly into the air cell. During this transfer process, the gaseous medium in the inter-building coffin continued to be air that was circulated through the subassembly to remove fission product decay heat. This cooling process was continued until the subassembly was dismantled and the individual fuel elements were separated from the close packed hexagonal array that existed in the subassembly. When separated, the fuel elements cooled in ambient air and forced circulation cooling was no longer necessary.

FUEL RECYCLE SYSTEM

Fuel recycle was accomplished in the Fuel Cycle Facility which was designed for reprocessing the fuel material discharged from EBR-II by pyrometallurgical methods. The fuel alloy was enriched uranium-5 percent fissium and contains about 46 weight percent uranium-235. The major processes involved are summarized here. A detailed description of the processes and the equipment involved are described in “The EBR-II Fuel Cycle Story,” 1987, by Charles E. Stevenson.

The Fuel Cycle Facility included an argon-atmosphere cell where fuel reprocessing and fabrication could be performed in an inert gas environment, an adjacent air-atmosphere cell where fuel subassemblies could be assembled and disassembled, and an operating area for personnel that surrounded the two cells. Because of the high levels of radioactivity involved, the fuel handling and processing had to be accomplished by remote operation of processing and supporting equipment.

Remote processing was accomplished with the aid of bridge cranes, electromagnetic bridge manipulators, and master-slave manipulators. Transfer ports and air locks were provided for the transfer of materials and equipment into the argon cell and between the two cells. The walls between the cells and the operating areas were heavily shielded, and viewing was provided through thick shielding windows.

EBR-II initial operations included disassembly of fuel subassemblies and their constituent fuel elements, fuel purification, refabrication of fuel elements, and reassembly of the subassemblies for reloading into the reactor. Specific operations included: subassembly disassembly, fuel element decanning, chopping of fuel pins, melt refining, oxidation of skull material retained in melt refining crucibles, injection casting of fuel pins, final pin fabrication, canning of fuel pins, sodium bonding and bond testing, fuel element inspection and testing, and assembly into fuel subassemblies. These operations required a large number of supporting activities which also have been described in Stevenson (1987), including the processes and equipment involved. These include:

- Fuel movement and storage
- Sampling and analysis of fuel and waste
- Preparation, handling, and storage of liquid, solid, and gaseous radioactive wastes
- Disposal of scrap and unrecoverable fuel

- Special controls to avoid criticality and to provide material accountability.

The initial EBR-II fuel cycle consisted of the recycle of enriched uranium containing five weight percent fission products. This alloy was named fissium (Fs) and contained the noble metal fission products that were not removed by the EBR-II pyrometallurgical fuel recycle process. The initial alloy composition was established to approximate the expected equilibrium composition of the alloy and thereby reduce the changes in composition and properties that would occur as the fuel was repeatedly recycled. The EBR-II was the first power reactor system in the United States power demonstration program to operate on a closed fuel cycle utilizing recycled fuel. Many new developments, both in procedures and equipment, were required to perform the various steps in the fuel cycle. Actual plant experience with remote operations and equipment was needed to demonstrate feasibility.

The Fuel Cycle Facility through-put was based on reactor operation at 62.5 megawatt thermal and average burn up of 2 atom percent. This operating mode would produce about 3,130 grams per day of spent fuel. A typical melt refining charge consisted of 10 to 12 kilograms. This was a batch process, but there was adequate storage capability in the primary tank storage rack, and batch size was not critical.

The Fuel Cycle Facility incorporated a unique design developed specifically to perform the processes and operations involved in the EBR-II fuel cycle as summarized in Figure 3-39. This flow diagram depicts all of the direct and supporting activities which had been identified at the time. It defined a total recycle program which attempted to describe a development program for total fast breeder reactor fuel and blanket recycle. Much of the supporting development, such as the blanket material process were not developed, but significant advances have been made in the development of similar fuel recycle of plutonium-uranium metal fuel alloys. Also, it would appear that recycle of these alloys could be performed in an EBR-II-type Fuel Cycle Facility (as originally conceived in the EBR-II plant concept). Development of applicable processes and equipment has continued intended for operation and demonstration in the Fuel Cycle Facility which has been upgraded for this purpose.

The starting point for the EBR-II fuel cycle was the delivery of a spent fuel assembly, which has had the residual sodium removed as a part of the fuel transport, to the Fuel Cycle Facility air cell. As a part of this delivery process, forced convection cooling by air circulation through the subassembly had to be maintained. This circulation was provided in the inter-building coffin and in the air cell dismantler where the first operation was performed. The subassembly hex can was cut at the lower adapter and pulled off exposing the cluster of fuel elements. The fuel elements were removed from the subassembly by rows and placed flat in trays, approximately 30 per tray. In this configuration no forced cooling was required. After appropriate inspection the fuel elements were transferred on the trays through an air lock to the argon cell, they continued to cool naturally in the argon gas environment.

EBR-II fuel processing began by separating the fuel pin from the fuel tube. The fuel tube was sheared at each end at points which also released the spiral spacer wire which was welded to the tube at each end of the wire. The tube was then peeled from the pin, which was accomplished by spiral cutting the tube into a narrow continuous strip. As the clad was removed, the exposed pin was chopped into approximately 1-1/2 inch lengths to provide suitable feed for the next process which involved melt refining.

The chopped pins were fed into a zirconia crucible used for the melt refining operation. Approximately 10 to 12 kilograms of fuel constituted a normal charge. The crucible was heated to 1,400°C and the charge held in the molten state at this temperature for about three hours. During that period, gases and some fission products which volatilize, were removed and some fission products were oxidized. At the end of the heating period, the melt was poured into a graphite crucible and formed a metal ingot. About 90 to 95 percent of the fuel plus noble metals were in the metal ingot, and 5 to 10 percent of the charge remained in the skull and was recovered in a separate process. More detailed descriptions are given in Stevenson (1987). It should be noted, however, that the total recovery of fuel alloy from both processes (i.e., the ingot plus processed skull was 99.8 to 99.9 percent.

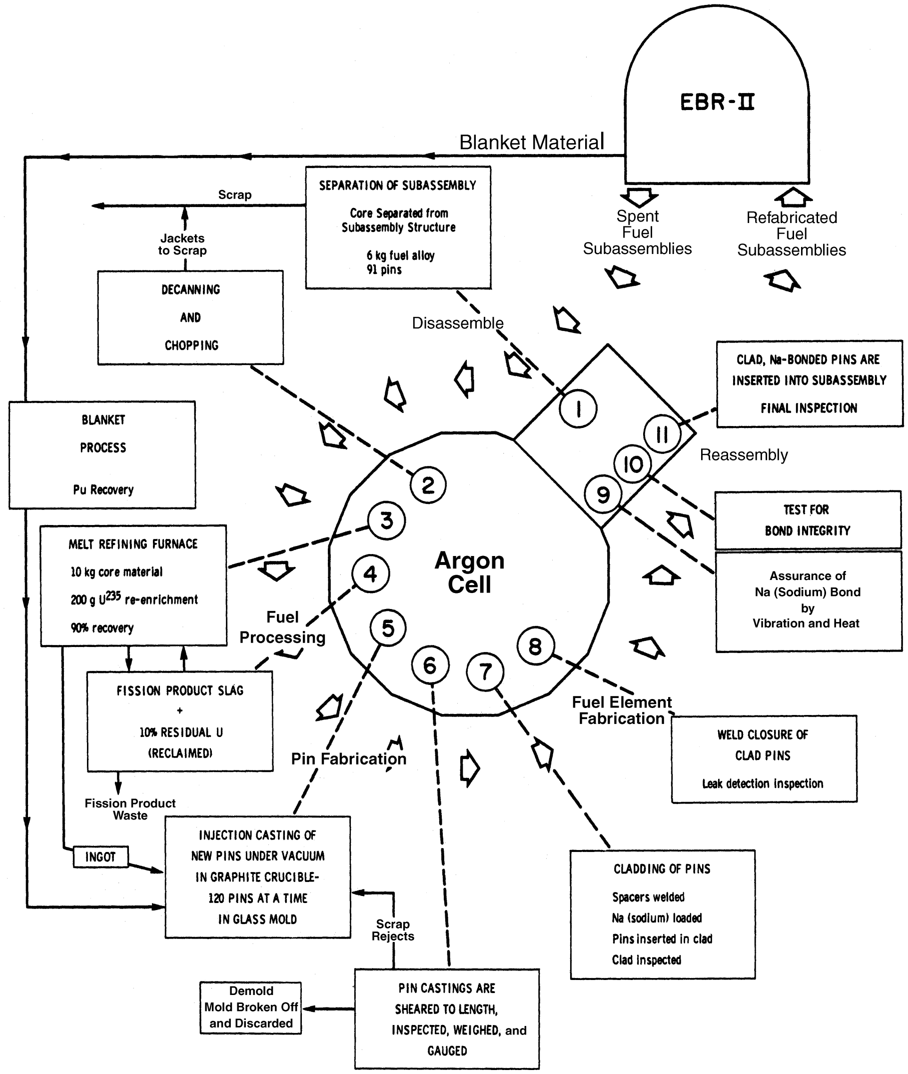


FIGURE 3-39. EBR-II FUEL CYCLE FLOW.

The recovered material was used as feed for the injection casting process. This was a unique machine which produced precision castings in one operation. The casting furnace employed a high-frequency induction heating system and a graphite crucible. The fuel alloy was cast directly into precision Vycor glass molds by a vacuum/gas pressure system. The inside diameter of the molds and the graphite crucible were coated with thoria (ThO_2). The injection casting process involved a series of carefully controlled operations. The furnace assembly, including the crucible containing the fuel alloy and the Vycor glass molds were contained in a gas tight enclosure which was evacuated. The fuel alloy in the crucible, located directly below the cluster of molds, was heated to approximately 1,350°C. The crucible was raised by a pneumatic cylinder immersing the open end of the molds in the molten fuel alloy. The furnace was then pressurized with inert gas and after the alloy solidified in the molds (about two seconds), the crucible was lowered. All of these operations required very careful and accurate control. A normal run involved a fuel alloy charge of 11 to 14 kilograms and approximately 100 Vycor glass molds. The total process was accomplished in about eight hours. The castings were finished pins, except that they required cutting to correct length which was accomplished by shearing. The sheared ends and the heel remaining in the graphite mold were used as feed material in subsequent runs. Approximately 44,000 pins were cast for the initial operating phase of EBR-II.

The finished pins, after detailed measurement, inspection and recording of data were assembled into fuel elements. This step was preceded by the preparation of the fuel element tube assembly which included the tube, the attached lower adapter (hook) and the spiral spacer wire. This assembly was delivered to an argon-atmosphere glove box adjacent to and attached to the argon cell via an air lock penetration. In the glove box, the sodium to provide the bond between the fuel pin and the fuel element tube was installed. This was done by preparing a sodium extrusion slightly smaller in diameter than the inside diameter of the fuel element tube and of the proper length, thus providing the proper volume of sodium.

These fuel tube assemblies were then transferred to the argon cell where the fuel pin was installed, the sodium was melted to permit the fuel pin to settle to the bottom of the tube assembly and the top plug was installed and welded. The top plug also served as a restrainer to prevent the fuel pin from protruding above the sodium bond level. The weld closure was also a very unique concept. It consisted of a flanged plug the same outside diameter as the fuel tube. It also included a small projection in the center for welding as shown in Figure 3-40. The weld was accomplished by a condenser discharge through a tungsten electrode positioned directly above the projection in the center of the plug. The entire top end was fused as shown. The concept and process were developed at Argonne.

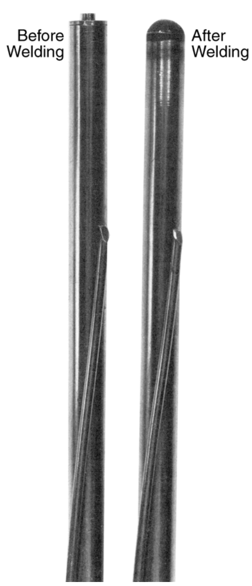


FIGURE 3-40. AN ASSEMBLED FUEL ELEMENT BEFORE AND AFTER WELDING.

The completed sealed fuel elements were then transferred through an air lock to the air cell where a series of processes and tests were preformed to ensure the completeness and accuracy of the sodium bond. This was necessary to ensure the reliability of heat transfer from the fuel to the primary sodium coolant. The completed fuel elements were also leak tested to ensure that the closure was leak tight.

The completed fuel elements were then delivered to the subassembly station for the final assembly operations. Here again the maximum permissible preassembly operations, not involving significant radiation, were employed to produce two major preassemblies; the lower preassembly consisted of the lower adapter, lower blanket section, and the grid to which the fuel elements would be attached; and the upper preassembly consisted of the upper adapter, upper blanket section, and the hexagonal subassembly tube. These two preassemblies are shown in Figure 3-41. The lower preassembly was placed into the assembly machine shown in Figure 3-42. The fuel elements were slid onto the parallel T-strips that constituted the lower fuel element grid. This grid consisted of 11 parallel T-strips to which the fuel elements had to be attached in the proper sequence. Sequencing and proper angular orientation of the fuel element were controlled by the element loading block (Figure 3-43) which permitted the process to begin with a much wider spacing than the grid strips. Each fuel element was first placed into the proper, controlled, position in the loading block and then slid on the guide wire to the grid strip. This action was performed by the use of two master slave manipulators augmented by auxiliary devices to assist specific actions. These devices were incorporated into the fuel element assembly machine (Figure 3-42). The operations were visible through 5-foot-thick shielding windows. Note that provisions were made for cooling the fuel elements during the assembly process.

After all 91 fuel elements were installed and properly positioned and supported, the upper preassembly was lowered over the fuel section down to the lower adapter. At this position the hex tube was spot welded to the lower adapter. The completed subassembly was checked for dimensional correctness and straightness and delivered to the inter-building coffin. It was then transported to the Reactor Plant and transferred to the storage rack in the primary tank. This process was essentially the reverse of the delivery

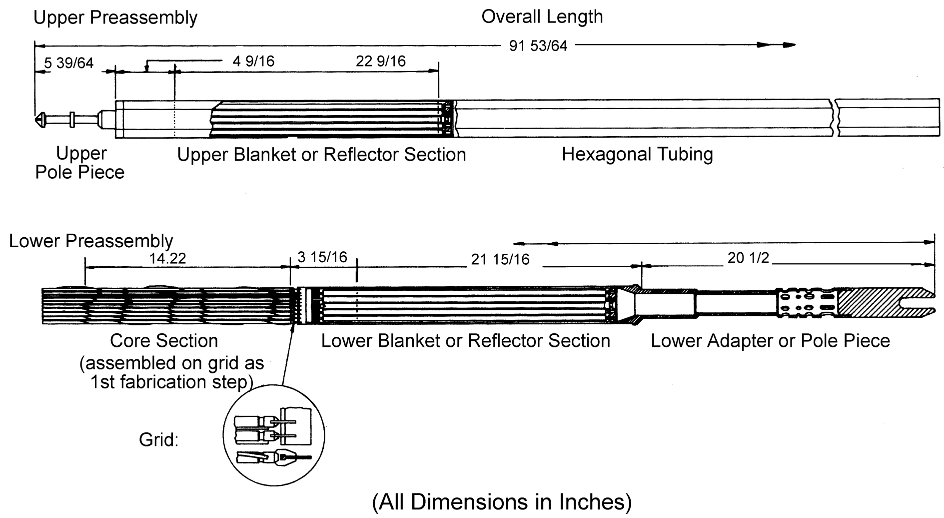


FIGURE 3-41. THE TWO PREASSEMBLED COMPONENTS OF AN EBR-II CORE SUBASSEMBLY.

of an irradiated subassembly to the Fuel Cycle Facility, except that there was no need to wash the subassembly to remove sodium, but there was a need to preheat the subassembly before immersing it into the 700°F primary sodium.

The EBR-II Fuel Recycle System demonstrated that fuel recycle for a power reactor system need not produce a pure, clean product as were required for most military products. It also demonstrated that a relatively small facility, as compared to other purification processes, could accommodate the needs of a power reactor. The Fuel Cycle Facility probably had the capability of processing 5 to 10 times the output of EBR-II. It is quite probable that a facility utilizing this type of process/fabrication cycle could serve more than one reactor. This becomes an exciting possibility when considering the nuclear power park concept.

LABORATORY AND SUPPORT FACILITIES

The Laboratory and Service Building was the facility in which analytical and control support activities were conducted. Six analytical caves (hot cells) were provided to handle small radioactive samples, including irradiated fuel samples and radioactive samples of the fuel elements and process materials. These samples included fuel alloy, cladding, oxidized skull, and scrap. Analytical facilities were also provided to support operation and control of argon and sodium systems, and a variety of waste processes. These facilities and operations are described in Stevenson (1987).

The Laboratory and Service Building also provided personnel support facilities, library, cafeteria, graphic arts, and offices. These proved to be quite inadequate, and during the operating life of the plant required almost constant expansion. This was due, at least in part, to the broadened and enlarged mission of EBR-II.

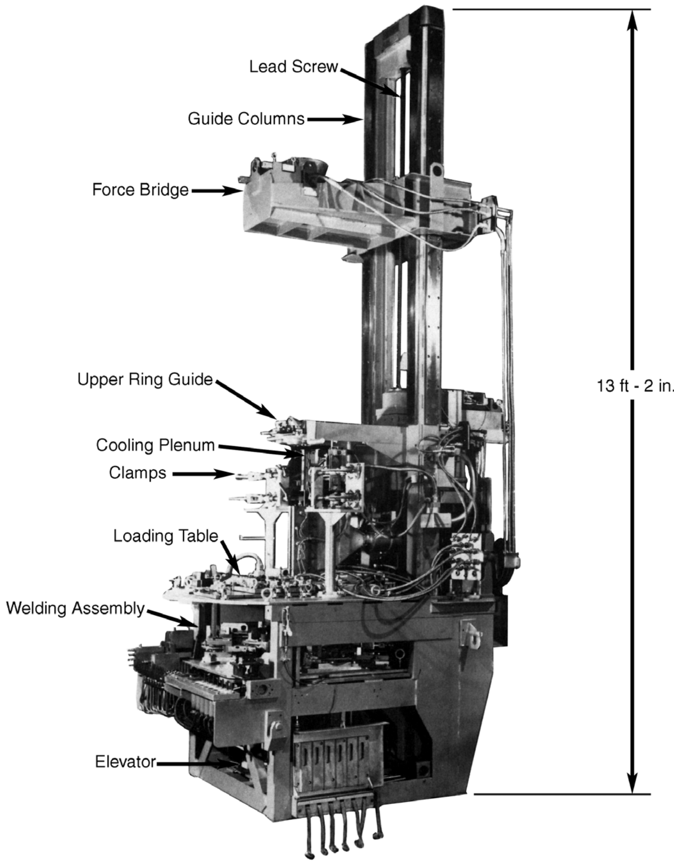


FIGURE 3-42. UNIVERSAL FUEL ELEMENT ASSEMBLY MACHINE.

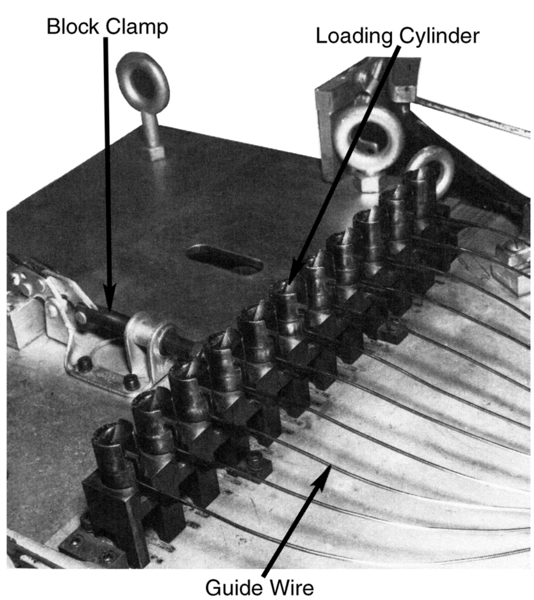


FIGURE 3-43. FUEL ELEMENT LOADING BLOCK.