EXPERIMENTAL GAS COOLED REACTOR

Final Hazards Summary Report. Volume II

August 1, 1963

Oak Ridge Operations Office, AEC
Oak Ridge, Tennessee
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EXPERIMENTAL
GAS COOLED REACTOR

FINAL
HAZARDS SUMMARY REPORT

AUGUST 1, 1963

VOLUME II

PLANT OPERATION

OAK RIDGE OPERATIONS OFFICE
UNITED STATES ATOMIC ENERGY COMMISSION
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Oak Ridge, Tennessee
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INTRODUCTION

The general objective of the safeguards program associated with a nuclear facility is to provide reasonable assurance that the health and safety of both the general public and plant personnel are not endangered. The evolution of facility design, from the time of site selection, is strongly influenced by the hazards analysis of both normal and abnormal operating conditions. The design basis for the EGCR, as well as the accident analysis for credible accidents, is discussed in Volume I of the EGCR Hazards Summary Report. The material presented in Volume II of the Hazards Summary Report describes the safeguards program which deals with all phases of operation of the reactor, including those operations or tests prior to criticality and all subsequent operations including emergencies.

The basic objectives of the safeguards program associated with operation of the reactor are: to establish and execute those programs necessary to prevent conditions which could lead to a hazardous situation; to develop the capability required to minimize the consequences of credible accidents; and to take actions following a hazardous condition to prevent its recurrence.

The first phase of the safeguards program deals with establishing a program to prevent or minimize hazardous conditions. It is assumed that a hazardous condition can conceivably result from improper or inadequate design, fabrication, or inspection; normal failure; operator error; or conditions in the system which were not fully evaluated or understood during the design of the facility. To prevent or minimize such conditions, it is necessary to establish control measures for each of the probable causes of the hazardous condition. To preclude hazardous conditions resulting from improper or inadequate design, fabrication, or inspection, numerous tests and inspections are conducted on the equipment and complete systems. The test program is conducted in several phases to establish the performance of equipment or systems and to assure that the original design intent has been accomplished. Three distinct test periods are discussed in Volume II, precritical shakedown tests (Section 12), initial nuclear tests (Section 13), and tests at power (Section 14). In addition to these initial tests, continuous monitoring of equipment and system performance is required to establish long-term reliability and design adequacy. In certain cases, long-term adequacy of some components is established by periodic inspection (Section 17).

Control over normal equipment failures is established by close observation of performance, by preventive maintenance (Section 19), and by periodic plant tests and inspections (Section 17). The use of duplicate equipment or systems reduces the probability of total failure of a particular functioning unit of the plant.

The control exercised to reduce the probability of operator error is based primarily on effective operator training and administrative control. The duties and responsibilities of plant personnel are clearly defined (Section 10). The training program for all Project personnel has the two-fold purpose of developing and maintaining a well informed and adequately trained staff (Section 11). All operations, including normal and emergency operation and tests, are performed using written instructions contained either in the
Operating Manual, test procedures, or written instructions prepared for special situations. Each procedure is approved prior to execution and any deviation requires additional approval (Sections 12 through 21).

One of the most difficult failures to protect against is that resulting from an unknown cause. This can result from lack of fundamental information, or from conditions not anticipated during the design. To minimize the number and consequences of such failures, reliance is placed on operator training and alertness and on routine plant tests and inspections. In addition, the preventive maintenance program reveals unanticipated problem areas which can be evaluated and corrected to minimize failures in this category.

The second phase of the safeguards program deals with minimizing the consequences of credible incidents which release ionizing radiation. The first step in carrying out this phase of the program is the acceptance and application of established radiation standards (Section 22) and radiation exposure control procedures (Section 23). The section on radiation standards establishes the basic criteria for operations involving a radiation hazard. Section 23 describes measures taken to prevent or minimize radiation exposure of personnel. Although the plant contains many automatic devices intended to restore the system to a safe level, the operator is required to take supplementary actions to assure that the systems remain operable. In addition, during an emergency, the Operator directs those activities which minimize radiation exposure to both plant personnel and the general public (Section 16).

The third phase of the safeguards program deals with those actions taken by the Operator following the development of a hazardous condition to prevent its recurrence. This phase of the program is carried out by investigating failures, reporting the findings to the proper authorities, and recommending changes in plant design and method of operation which result in decreasing the probability of subsequent failures. As a part of this phase of the program, the Operator reviews the causes and consequences of major failures in other reactors and investigates the credibility of such failures occurring in the EGCR.

In summary, information provided in Volume II describes programs, requirements, and procedures established by the Operator to minimize the probability of equipment failure, actions taken during any emergency to minimize radiation exposure dose to the general public and to plant personnel, and actions taken following a hazardous condition to prevent its recurrence.
10. PLANT ORGANIZATION

10.1 General

The Tennessee Valley Authority (TVA) is responsible for managing, operating, and maintaining the EGCR, and for furnishing related technical services including the following:

a. Provide and train personnel for the Project operations
b. Review designs and specifications and recommend improvements related to operating economy and reliability
c. Follow construction and recommend field corrections or changes
d. Collaborate with the AEC and its contractors in designing experimental facilities
e. Design and construct the transmission line
f. Perform other related design and procurement services
g. Construct, alter, or repair facilities as authorized by the AEC
h. Participate in preoperational tests and inspections
i. Manage, operate, and maintain the EGCR facilities for the AEC in accordance with policies, programs, and schedules approved by the AEC
j. Carry out tests and experiments and make continuing studies and analyses of the results and of reactor operating data
k. Integrate the electric power output with the power supply to the Commission's Oak Ridge plants
l. Operate and coordinate the experimental use of future reactor loop facilities
m. Plan and recommend research and development activities related to the EGCR and, upon approval, conduct such activities
n. Make reports to the Commission on program activities
o. Arrange for discussions, consultations, and observations by others at EGCR
p. Plan, schedule, and carry out training programs for employees of other organizations at EGCR
q. Furnish related services as requested by the Commission
r. Furnish standby power from the TVA system.

In order to carry out these varying responsibilities, the EGCR operating organization has been developed so as to require only a minimum amount of technical assistance from other TVA groups or from AEC contractors. However, some assistance is required in order to minimize duplication of certain personnel and facilities. The services that are available from other TVA groups or from AEC contractors in Oak Ridge are discussed in Sections 10.3 and 10.4.

The general organization of the Tennessee Valley Authority is shown in Figure 10.1.1. TVA's responsibilities in the EGCR program are carried out by the EGCR Project organization which is a branch of the Division of Power Production in the Office of Power. Other TVA offices and divisions are assigned responsibilities for various management and technical services as required by the Project. The administrative and personnel staff services units of the Division of Power Production and the Management Services Staff of the Power Manager's Office participate in coordinating the furnishing of management services to the Project by other TVA divisions. The organization of the Office of Power and
ORGANIZATION OF THE TENNESSEE VALLEY AUTHORITY

BOARD OF DIRECTORS

OFFICE OF THE GENERAL MANAGER
Budget Staff
Information Office
Power Financing Staff
Washington Office
Government Relations and Economics Staff

Division of Law
Division of Personnel
Division of Finance
Division of Purchasing
Division of Property and Supply
Division of Health and Safety
Division of Reservoir Properties

Office of Engineering
Divisions:
Water Control Planning
Design
Construction
Navigation and Local Flood Relations

Office of Power
Divisions:
Power Planning and Engg
Power Marketing
Power Production
Power System Operations
Power Construction

Office of Agricultural and Chemical Development
Divisions:
Agricultural Development
Chemical Development
Chemical Operations

Division of Forestry Development

Figure 10.1.1
of the Division of Power Production is shown in Figures 10.1.2 and 10.1.3.

10.2 EGCR Project Organization

The EGCR Project organization chart is shown in Figure 10.2. Two principal groups, the Technical Program Group and the Operations Group, function directly under the supervision of the project manager. Staff assistance is provided by an assistant to the project manager, an Administrative Services Staff, and a Radiological Health Section. The latter section is under the administrative supervision of the TVA Division of Health and Safety. Public safety officers associated with the Project are under the administrative supervision of the TVA Division of Reservoir Properties.

In the following descriptions of the Project organization units, the number of employees in each unit should be regarded as approximate numbers which may be varied to suit the actual work requirements.

Project Manager

The project manager is responsible for the planning, coordination, and direction of all phases of Project operations. He is responsible for the safe operation of the reactor facility and for assuring that all operations meet the requirements of AEC regulations. He approves all operating programs and procedures before they are issued to the operating superintendent.

Assistant to the Project Manager

The assistant to the project manager assists the project manager in planning, coordinating, and directing the Project activities. He is particularly concerned with coordinating the planning and execution of over-all programs which require close timing and cooperation of the technical and operating groups. He assists in liaison activities among the Project, AEC, and other contractors. In the absence of the project manager, he is responsible for management of the Project activities.

10.2.1. Technical Program Group

The Technical Program Group provides the basic technical support for all EGCR operations and associated programs. The technical group is both an advisory and functional group. It is responsible for initiating and carrying out technical support programs, and at the same time, it is responsible for advising the operating superintendent through the project manager in matters pertaining to operation. However, with the exception of the maintenance functions performed by the Controls Engineering Section and the chemical and radioactivity analyses performed by the Chemical Engineering Section, the technical group is not directly responsible for plant operation or maintenance.

Prior to reactor operation, the group is primarily responsible for reviewing plant design and specifications; assisting in the preparation and review of test procedures, the Hazards Summary Report, and the Operating Manual; providing technical assistance during all tests; evaluating tests and planning programs for subsequent operation. After initial reactor operation, the technical group is responsible for a continuing program of performance analysis, prediction of reactor characteristics, planning nuclear experiments, fuel management, analysis and recommendations relating to the control of potential plant hazards, performing studies and recommending changes in design or operation, and for preparing technical reports on the activities associated with the Project.

The Technical Program Group, under the supervision of the technical program superintendent, comprises five sections, each with a functional responsibility.
OFFICE OF POWER
ORGANIZATION CHART

Manager of Power
   /       \
  /         \
Manager of Power
       /       \
  /         \
Assistant Manager

Advisory Staff

Management Services Staff

Financial Planning Staff

Fuels Planning Staff

Power Research Staff

Division of Power Planning and Engineering

Division of Power Construction

Division of Power Production

Division of Power System Operations

Division of Power Marketing

Figure 10.1.2
DIVISION OF POWER PRODUCTION
ORGANIZATION CHART

Division of Power Production

Power Plant Maintenance Branch

Hydroelectric Generation Branch

Steam-Electric Generation Branch

EGCR Project

Hydroelectric Plants

Steam Plants

Combination Plants

Figure 10.1.3
In addition, technical staff groups provide assistance to the Technical Program Group. The present organization has technical staff groups assigned to the particular fields of hazards control, reactor controls, and reactor physics. Engineer trainees or technical personnel from other organizations who are temporarily assigned to the Project are supervised by the technical program superintendent but are temporarily assigned to one of the five sections or to the staff groups. The responsibilities of the various units in the Technical Program Group are described below.

a. Technical Program Superintendent

The technical program superintendent is responsible for initiating, planning, and coordinating the technical support functions of the Project. He represents the project manager in matters relating to the technical program and advises the project manager relating to the safe operation of the plant. He provides assistance and guidance to the technical sections and coordinates the work of these sections with the Operations Group and with the services provided by other TVA Divisions and AEC Contractors.

b. Hazards Control Staff

The Hazards Control Staff is responsible for analyzing and for evaluating conditions relevant to hazards in all phases of operations.

Prior to operation, the group is responsible for participating in and coordinating the Operators' review of plant design, as relating to safety, and for reviewing hazards analyses and the portion of the Hazards Summary Report that describes the design and accident analyses. The group is responsible for coordinating the preparation of Volume II of the Hazards Summary Report and for coordinating the preparation of the document describing the EGCR operating limits. Prior to operation, the group reviews test procedures and the Operating Manual to assure that the procedures in these documents are in accordance with assumptions and performance postulations in the Hazards Summary Report. It reviews the results of the test programs for hazards implications and proposes changes in the Hazards Summary Report where such changes are required. It provides staff assistance to other technical or operations sections in matters relating to hazards control.

After the plant is in operation, the Hazards Control Staff is responsible for reviewing plant operation and assuring that it is in accordance with the established and approved procedures. Results obtained during operation, including routine plant tests and inspections, are reviewed to assure that unsafe conditions do not exist and to predict if unsafe conditions are building up. The group participates in and coordinates the preparation of proposed revisions or supplements to the Hazards Summary Report which become necessary based on operating experience or as a result of proposed changes in the design or operation of the facility.

c. Controls Specialist

The controls specialist is responsible for the analysis and for the evaluation of the controls and safety system during all phases of operation.

Prior to operation, he makes analyses and recommendations relative to plant design, operation, and testing as relating to the adequacy of the plant control system. He assists in setting up an analog simulation of
the EGCR to be used for training purposes, reviews analyses made on the control system, and recommends additional analyses that are required to demonstrate plant safety or to determine necessary control characteristics. He reviews the results of test programs to assure that all plant controls function in accordance with the design intent.

After the plant is in operation, the controls specialist is responsible for reviewing plant operation to assure the proper functioning of all control systems. He advises other technical or operations sections in matters relating to plant control. He reviews recommended changes in plant design or operation to assure that any proposed changes do not alter the proper functioning of the plant control or safety system. He recommends additions or changes in plant controls and safety system which may be required as a result of proposed changes in the basic design or method of operation.

d. Reactor Physics Staff

The Reactor Physics Staff is responsible for the analysis and evaluation of the physics performance of the reactor core, and for providing staff assistance in matters relating to reactor physics.

Prior to operation, the staff is responsible for reviewing the physics analysis and performing additional analyses that are required to verify the performance of the reactor core. The staff participates in the development of detailed procedures for fuel loading and for the open core physics tests. It investigates control rod programs and fuel cycles which may be used during subsequent operation. It provides staff assistance to sections of the Technical Program Group and Operations Group in matters relating to reactor physics. During the test program, it provides assistance to the Operations Group and evaluates the results of the test program.

After the plant is in operation, the staff is responsible for a continuous program of reactor analysis of the operating core. It is responsible for the preparation and initiation of periodic recommendations to the operating superintendent and the Chemical Engineering Section regarding fuel movements and the fuel cycle program. It maintains an operating record of the fuel inventory in each channel, position and assembly in the core, cross-checking with cumulative and permanent records maintained by the Chemical Engineering Section. In cooperation with the Chemical Engineering Section, it recommends fuel movement and control of experimental materials temporarily placed in the core.

The staff prepares control rod programs to insure control of flux profiles in the core and maintains a continuous record of channel power levels. It evaluates experiments and general reactor performance from a physics viewpoint and furnishes technical services to other sections as required.

It evaluates the effect on reactor physics of all proposed changes in the basic design or method of operation and assists in the preparation of supplements or revisions to the Hazards Summary Report or Operating Manual pertaining to proposed changes. The staff participates in the preparation of technical studies for advanced designs or for improving the basic design. It prepares routine reports dealing with reactor physics work relating to operation of the reactor.

e. Experimental Engineering Section

The Experimental Engineering Section is responsible for the technical
management of experiments for which the reactor serves as a research bed.

Prior to operation and until the reactor is utilized as an experimental facility, the Experimental Engineering Section is responsible for participating in the development of advanced gas-cooled reactor fuel elements and for evaluating the first core loading for the EGCR.

When the reactor is utilized as an experimental facility, the section maintains liaison with organizations designing new experiments or experimental facilities for the EGCR. It participates in the development of the experiment and reviews the design and hazards analyses for each experimental installation. It reviews proposed operating procedures to assure consistency with established reactor operating and control procedures. It assists the Operations Group in developing and improving operating methods which would increase the efficiency and usefulness of experiments. It prepares reports following the completion of all experiments.

f. Reactor Engineering Section

The Reactor Engineering Section is primarily responsible for determining, analyzing, evaluating, and reporting on the thermal and mechanical performance of the EGCR.

Prior to reactor operation, this section coordinates the Operators' review of the Operating Manual, and Phases II and III test procedures. The section plans and schedules the integrated Phase III tests, including manpower estimates, critical path scheduling, and coordination with construction. It reviews the inspection, fabrication, and erection of selected components to assure that they are acceptable for operation and that proper records exist. It assists in the preparation of and coordinates Phases IV and V test procedures. It performs drafting services for the Project. Members of the section witness many of the Phase II tests, provide technical assistance for many of the Phase III tests, and evaluate the test results.

After operations begin, the Reactor Engineering Section is primarily responsible for analyzing the thermal and mechanical performance of the plant and for recommending tests to assure satisfactory performance of equipment and systems. When necessary, it prepares procurement drawings and specifications for plant alterations or additions or such other drawings, specifications, and schedules as may be required. It provides engineering design and drafting services necessary to carry out the experimental and operational program. It makes studies and recommends changes for improvement in the design of the plant or in the method of operation. It also initiates, plans, assists in performing and evaluates special tests designed to advance gas-cooled reactor technology.

g. Engineering Analysis Section

The Engineering Analysis Section is responsible for the analytical investigation of mechanical and thermal features of the reactor and its directly associated systems and components. The responsibilities of this section are similar to those in the Reactor Engineering Section except that emphasis is on analyses in the fields of heat transfer, thermodynamics, fluid flow, and stress analysis.

Prior to reactor operation, the section develops calculational
procedures for analyzing thermal conditions in the core, thermal stresses and thermal cycle limitations on components or equipment, and for specifying fuel channel orifice settings. It is responsible for analyzing data from the EGCR research and development program and for applying the results to the operating program. It participates in the test program and reviews the test results.

After operations begin, the Engineering Analysis Section is responsible for analyses of data derived from operations and for prescribing fuel channel orifice settings on the basis of control rod and fuel movements within the core. It performs analyses to predict the effect of any proposed changes in the basic design or method of operation. The section issues routine reports and provides technical assistance to other sections of the Operations Group and Technical Program Group.

h. Chemical Engineering Section

The Chemical Engineering Section is primarily responsible for all phases of work in the general fields of chemistry, radio-chemistry, metallurgy, and materials, including fuels and source materials.

Prior to reactor operation, the section reviews those portions of the design that are related to its assigned functions and recommends design changes which may be required. The section is responsible for planning and initiating programs pertaining to corrosion monitoring and control, fuel examination, and cleanliness control. The section participates in the Phase III tests and provides analytical services associated with these and subsequent tests. The section prepares programs associated with fuel handling and decontamination.

After reactor startup, the Chemical Engineering Section conducts routine chemical and radiochemical tests on plant gas and liquid process streams. It prescribes treatment to the steam generator feedwater and circulating water; provides technical support for operation of the helium purification and recovery system and demineralizer units; and advises the operating superintendent in matters relating to equipment decontamination and waste disposal. It conducts a program of periodic examination of fuel and other materials exposed in the reactor and arranges for examination of fuel elements or other materials as required. The section is responsible for receiving new fuel, shipping spent fuel, and for fuel accountability during all phases of fuel handling. It reviews proposed applications of materials in the plant for compatibility with the service and environment and recommends tests to verify proposed uses of materials. The section is responsible for the procurement of chemical supplies, helium, fuel, and special activation materials such as foils, wires, and tapes. In cooperation with the Radiological Health Section, it recommends procedures for the handling and use of radioactive, toxic, or other potentially hazardous materials.

i. Controls Engineering Section

The Controls Engineering Section is responsible for the operating adequacy and functional reliability of all controls and instruments at the EGCR, including the reactor, its auxiliaries, the steam plant, and any experimental facilities associated with reactor operation.

Prior to reactor operation, the Controls Engineering Section is responsible for the inspection, testing, and calibration of all of the process plant instruments prior to installation by the construction
forces. The section assists in reviewing the adequacy of instrumentation used in the test program and participates in the selection and design of special instrumentation circuits required by the test program. It procures and supervises the installation, calibration, and removal of special test instrumentation. It participates in the test program and reviews the results of the test program relating to instruments and controls. It prepares and reviews maintenance procedures for all process instrumentation and controls and determines what spare components, parts, materials, and supplies are needed for subsequent operation. The section reviews vendor data and test procedures for adequacy and completeness. It prepares a program of routine maintenance for subsequent operation and initiates this program for those systems that are placed into operation.

Following reactor operation, the Controls Engineering Section is responsible for all inspection, testing, calibration, and maintenance of instrumentation and controls and for planning and supervising the installation, replacement and repair of such systems and components. It coordinates the control and instrumentation work with operation, maintenance, and engineering schedules to minimize outage time and to assure that the required data can be obtained. The section analyzes and reports on the long-term performance of components of the control and instrumentation system, proposes tests aimed at improving existing design, and carries out those programs that are approved. It participates in the review or preparation of design changes or changes in the method of operation as relating to instrumentation or controls. It reviews the instrumentation and controls of proposed experimental facilities to assure that such additions are in accordance with established maintenance procedures. It participates with other sections in advanced studies aimed at improving the gas-cooled reactor technology.

10.2.2 Operations Group

The Operations Group is responsible for the operation and maintenance of all plant facilities with the exception of those specific operations and functions performed by the Radiological Health Section, Chemical Engineering Section, and Controls Engineering Section. It carries out all tests performed at the site in accordance with operating authorizations and approved test procedures. It is responsible for industrial safety at the site.

Prior to plant operation, the group is responsible for planning and initiating programs directed toward training a competent operating and maintenance staff and for preparing programs to be utilized during subsequent operation.

The Operations Group, under the direction of the operating superintendent, comprises two major sections, Plant Operations and Plant Maintenance. An assistant to the operating superintendent assists in all phases of operation. The responsibilities of the various units in the Operations Group are described below.

a. Operating Superintendent

The operating superintendent is responsible for the operation and maintenance of the plant, including existing or future experimental facilities. He is responsible for assuring that all operations are in accordance with written authorizations and procedures and within established operating limits. He is responsible for notifying the project manager of any abnormal operating conditions and for taking actions to prevent the development of unsafe conditions. He plans and schedules operations to minimize downtime and to utilize the reactor to its
greatest capability. He collaborates with the technical program superintendent in developing procedures and revisions and in recommending them to the project manager for approval. He assures that adequate operating and maintenance records are kept and submits reports as required. He provides consultation and assistance to other groups on problems relating to operation and maintenance. He recommends training requirements for operating and maintenance personnel and is responsible for implementing and carrying out on-site training.

b. Assistant to the Operating Superintendent

The assistant to the operating superintendent assists his supervisor in reviewing, coordinating, and planning the operational activities of the EGCR. This includes developing detailed operating plans and schedules, participating in program coordination, preparing special operating procedures and instructions, making special studies of operational problems, serving as liaison with the Technical Program Group, and continually studying plant operations to assure that program requirements are being met. He collaborates with the technical and maintenance sections in planning schedules for maintenance and other work to be done during plant outages. He formulates the overall training program for the Operations Group and participates in operator training activities. In the absence of the operating superintendent, he is responsible for the operation of the plant.

c. Plant Operations Section

Prior to reactor operation, the Plant Operations Section is responsible for and participates in programs that are designed to prepare the operating crews and the plant for operation. The section participates in reviewing plant design, Operating Manual, Hazards Summary Report, and related documents, and recommends supplements or revisions as required. The section participates in the preparation of material for the Hazards Summary Report, operating limits, test procedures, and, in general, reviews all pre-operational programs. The Plant Operations Section reviews the Operating Manual and recommends changes to incorporate valve numbers, individual system and equipment checklists for startup and shutdown, diagrams, and other modifications to facilitate operation. Prior to operation, all members of the Plant Operations Section become familiar with the location and operation of equipment in the plant. It is responsible for coordinating and scheduling the training program for all operations personnel. It provides the nucleus of the site firefighting organization and escorts official visitors at the site. It provides all operating personnel for the testing program conducted by the Operator.

The Plant Operations Section is responsible for all plant operations. There are five shift crews, each consisting of a plant operations supervisor, a shift engineer, three unit operators, and two assistant unit operators. There will also be one additional unit operator and two more assistant unit operators on the day shift for fuel handling operations. The plant operations supervisor is responsible for all operations on his shift and coordinates the work of the Plant Operations Section with the Plant Maintenance Section. He plans and schedules operations in accordance with instructions issued by the operating superintendent. He supervises the operation through the shift engineer, who is directly responsible for the detailed operation of the plant. He provides special direction to the shift engineer for emergency conditions not specifically covered by written instructions.
The shift engineer sees that the plant is operated in accordance with established procedures and objectives and that the operation is carried out to insure the safety of the plant and personnel. During a normal shift, one unit operator is stationed at the turbine control board and one is stationed at the reactor control board. The remaining unit operator is not assigned to a specific location but works throughout the plant as directed by the shift engineer. One assistant unit operator in the basement area operates the water treatment equipment, takes samples, and, in general, monitors equipment operation. The remaining assistant unit operator works at various locations as assigned.

Special operations, such as routine testing, decontamination, and waste disposal, are performed by those operators who are not assigned to a specific post.

d. Plant Maintenance Section

The Plant Maintenance Section is responsible for planning, coordinating, and carrying out all plant maintenance functions with the exception of those performed by the Controls Engineering Section.

Prior to reactor operation, the section is responsible for preparing maintenance procedures and schedules, and for procuring materials and tools needed for operation. The section participates in the preparation or review of selected portions of the Operating Manual, Hazards Summary Report, and test procedures. It participates in preparing routine plant tests and inspections, particularly from the standpoint of personnel and material requirements. It reviews maintenance procedures prepared by the designer or vendors, refining such procedures for application at the EGCR. The section supervises the planning, construction, and operation of mock-ups required to develop special maintenance techniques. It plans and initiates training and educational programs for maintenance personnel. It establishes and maintains an accessible file of design and vendor information, parts data, preventive maintenance records, and records of equipment performance. During the test program, the section observes equipment performance and assists in performing tests relating to maintenance.

Following reactor operation, the Plant Maintenance Section is responsible for maintaining the plant. Under normal conditions, mechanical and electrical maintenance will be accomplished on a day schedule five days per week, but special maintenance work may be scheduled on other shifts. The section performs technical studies relating to the maintenance of equipment and recommends design changes for improving safety, economy, or ease of maintenance. It reviews all proposed changes in the design or method of operation of plant facilities from the standpoint of maintenance. The section maintains the radiological cleanliness of the plant as required by radiological health standards and procedures.

10.2.3 Radiological Health Section

The Radiological Health Section is responsible for radiation protection activities at the Project. This section is under the general administrative supervision of the chief of the radiological health staff, located offsite in the TVA Division of Health and Safety. This section provides both advisory and monitoring services in radiation protection at the plant as required by the project manager.

Prior to reactor operation, the section is responsible for establishing radiation standards and procedures for controlling radiation exposure to plant
personnel, visitors, and the general public in the vicinity of the site. It reviews the adequacy of the design and proposed method of operation from the standpoint of radiation protection. It participates in the preparation and review of certain sections of the Operating Manual, Hazards Summary Report, test procedures, and related documents. It prepares a Radiological Health Manual and assists in the preparation of the Emergency Plan. The section is responsible for the orientation and indoctrination of all Project employees, assignees, and visitors in the principles of radiation protection and regulation. It establishes an accurate record of background activity at and adjacent to the site. It establishes records for personnel radiation exposure, instrument calibration and performance, equipment and area decontamination, and waste disposal controls. It provides specialized training to health physics technicians as such technicians join the Project. It participates in the test program relating to plant radiation monitoring systems. During the final phases of testing, it provides health physics coverage for those tests where radiation is involved. It maintains records of exposure and processes film badges during and subsequent to the startup program.

After the reactor is in operation, the Radiological Health Section is responsible for advising the project manager and for assisting the Operations Group in matters relating to radiation protection. The section functions as a service group providing continuous health physics coverage for all operations where radiation hazards exist, including waste disposal, fuel handling, maintenance, and decontamination operations. It maintains records of personnel exposure and assures that personnel exposure is within the limits established in the Radiological Health Manual. The section maintains records of all radioactive wastes discharged during normal and abnormal operations and advises the operating superintendent through the project manager on acceptable waste disposal procedures. The section assists other technical sections in resolving radiological health problems. It reviews all proposed changes in reactor design or method of operation to assure that such proposed changes are in accordance with established radiation standards. It provides radiological health coverage for all existing or future experimental facilities.

10.2.4 Administrative Services Staff and Public Safety Officers

The Administrative Services Staff, under the supervision of the administrative officer, performs management service functions and clerical services for the Project. These functions include consolidation and preparation of budget estimates; maintenance of operation accounts and cost control; maintenance of stores for plant equipment, materials, and supplies; provision of general clerical services such as maintenance of files, preparation of records and reports on payroll, personnel records, travel requests, and security clearances.

Public safety officers associated with the Project are responsible for maintaining the security guard at the plant. The public safety officers are administratively supervised by the Division of Reservoir Properties but report directly to the project manager at the site. The guards are responsible for maintaining security in the vicinity of the site (checking the fences) and for cooperating with the operating superintendent in maintaining plant security within the fenced area. The guards also provide escort services and cooperate with the Plant Operations Section in taking actions to protect personnel or property in the event of fire, sabotage, or other emergencies. At the time of an emergency, the public safety officers control access to the area and cooperate with the Plant Operations Section in executing the Emergency Plan.

10.3 Services Available from Other Organizations

The EGCR Project organization is designed to require a minimum amount of assistance from other organizations. However, some services can be provided
most economically by utilizing the personnel or facilities of other TVA groups or AEC contractors. The organization described in Section 10.2 provides a full-time staff for all normal operating, maintenance, administrative, technical, and service functions, and depends upon other organizations for only part-time services which cannot be provided economically at the EGCR. Other organizations, upon request from the project manager, assist during the execution of the test program, and provide required consultation, design, construction, repair, and general services during operations.

10.3.1 Services Available from Other TVA Divisions

The basic responsibility for operating the EGCR rests with the EGCR project manager. Services from other TVA groups are obtained at the request of the project manager when such services are required to supplement existing facilities or staff capabilities at the site. The organizations principally involved are as follows:

a. Division of Power Production

This division provides general supervision and management services for the EGCR Project. The Power Plant Maintenance Branch can provide technical services of a central office staff on electrical and mechanical maintenance problems and major shop services from its central service shops. The Steam Electric Generation Branch can provide technical advice and services in addition to the emergency services of the major steam plant operating and maintenance organizations. Most of the plant operators and maintenance men and their supervisors for the EGCR have been recruited from the ranks of the steam plant organizations.

b. Division of Power System Operations

This division provides the services of its Electrical Laboratory and Test Branch, including the central laboratory and technical staff. In addition, field test engineers are provided for chemical and laboratory tests, and for solution of special technical problems. Employees from this division are utilized as required during various phases of the test program.

c. Division of Power Planning and Engineering

This division furnishes advice and assistance regarding the engineering and design of the electrical transmission lines, substations, and communication facilities. The division designed the EGCR transmission line.

d. Division of Power Construction

This division provides assistance and advice regarding the construction of transmission lines and substations. This division constructed the EGCR transmission line.

e. Division of Design

The Division of Design, of the Office of Engineering, is a major engineering organization which has the responsibility for planning and designing all TVA power plants and other major facilities. The services of this organization are available for advice and consultation on all kinds of engineering problems and for carrying out the design of plant changes or additions which may be required at the EGCR. The Division
of Design designed the substation for the EGCR.

f. Division of Construction

The services of the Division of Construction, of the Office of Engineering, can be made available for any plant construction that may be required at the site.

g. Division of Chemical Development

This division can furnish technical advice and services on chemical and chemical engineering problems and can furnish chemical engineering development services in certain fields.

h. Division of Health and Safety

As noted previously, the Division of Health and Safety provides supervision for the Radiological Health Section at the EGCR. The division assists in the preparation and review of radiological protection standards, provides employee health services for the EGCR Project, and furnishes special services in fields of industrial hygiene, environmental hygiene, pollution control, and other areas.

i. Division of Reservoir Properties

This division provides public safety officers to carry out plant guard service and related duties at all TVA plants. It provides these services as required at the EGCR.

j. Division of Purchasing

All supplies, material, or equipment not obtained through AEC contractors or transfers are purchased by the Operator through the Division of Purchasing.

k. Other Management Services

The Divisions of Law, Personnel, Finance, and Property and Supply furnish the required services within their assigned fields of activity as required by the EGCR Project.

10.3.2 Services Available from AEC Contractors and Consultants

In addition to the services obtained from other TVA divisions, services of AEC contractors in Oak Ridge are utilized as required. In general, these services are utilized for fulfilling requirements for specialized equipment, facilities, or technical competence which are uniquely associated with the design, operation, or maintenance of reactor facilities. Facilities owned by the Commission at Oak Ridge and operated by the Union Carbide Nuclear Corporation, or other contractors, are utilized to supplement the facilities provided at the site.

Assistance from ORNL is of two types, utilization of facilities and utilization of technical personnel. During the various phases of reactor testing and continuing through initial full-power operation, technical specialists are utilized to supplement the TVA organization at the site. The amount of assistance during this period is estimated to be between one and four man-years. Some of the areas where assistance is to be provided during this period are described below:
a. Reactor Physics

Assistance is provided in planning the startup program, reviewing the physics program and physics test procedures prepared by TVA for the Phases IV and V tests, consulting during tests, and in analyzing or interpreting the results of the test program.

b. Reactor Control and Instrumentation

Assistance and consultation are provided for reviewing the design, selection, installation, calibration, and test procedures for the startup instrumentation. In addition, assistance is provided for planning tests, analyzing, and interpreting test results.

c. Reactor Engineering

Services and consultation of engineers and specialists are provided for reviewing Phases IV and V test procedures and for consulting on special problems in the areas of core thermal performance and analysis, blower performance and testing, transient heat transfer, statistical analysis of test data, and stress analysis. In addition, the services of a project engineer during Phases III, IV, and V testing are required to assist with test planning, execution, analysis, and interpretation of results.

d. Other

Prior to operation of the facility and subsequent to operation, consultation with ORNL specialists on technical problems is utilized by the various units of the Technical Program Group and the Radiological Health Section. In addition to the services of personnel, numerous facilities associated with ORNL are utilized. These include, but are not limited to, the following:

1. Hot laboratory facilities and other special laboratory services not provided at the site are available at ORNL
2. Waste disposal facilities at ORNL
3. Emergency services under the UCNC area emergency plan
4. UCNC stores for certain spare parts and supplies
5. Shops or special equipment located at ORNL, Y-12 and ORGDP (operated by UCNC)
6. Computer services associated with the Central Data Processing Center at the ORGDP and at ORNL.

10.4 Plant Safeguards Review

A system of written procedures and approvals is employed to control potential hazards from physical changes in the plant or changes in operating procedures. The operating superintendent is responsible for carrying out all plant operations in accordance with this system, and for assuring that all safety requirements are met.

Procedures covering all plant startups, operations, maintenance work, tests, experiments, equipment changes, and other activities which might adversely affect safety are put into effect only after being authorized in writing by the project manager or his delegated representative. It is the project manager's responsibility to assure that proper hazards investigations are made and that the required reviews and approvals are completed before the authorizations are issued.
The technical program superintendent, the operating superintendent, and the radiological health section supervisor are responsible for initiating and reviewing proposed authorizations for changes in plant design or operating procedures. The Hazards Control Staff is responsible to the technical program superintendent for coordinating and supervising the preparation of hazards analyses and reports pertaining to such changes. Each of the supervisors of the several technical units is responsible for preparing information for that portion of the analysis which is within his field of technical responsibility. The technical program superintendent, operating superintendent, and radiological health section supervisor review the completed report and the proposed authorization and make their recommendations to the project manager. The project manager and staff members may obtain the technical assistance and advice of the Oak Ridge National Laboratory in preparing and reviewing hazards analyses when the project manager determines that such assistance is required.

On the basis of the hazards analysis report and recommendations by the principal staff members of the three main groups, the project manager determines whether further review and approval is required. Before a change authorization is issued, the following factors are considered:

a. If it is clearly established that the proposed authorization does not result in hazards greater than or different from those analyzed in the Final Hazards Summary Report and does not fall into a category designated in the operating limits document or other AEC instructions as requiring approval by AEC, the authorization may be issued without further approval. Copies of the authorization and the report are furnished to AEC (ORO).

b. If the proposed authorization falls in a category requiring AEC approval, it is submitted to AEC (ORO) together with the hazards report and TVA's recommendation.

In addition to planned changes in the plant and operating procedures discussed above, there is a further area requiring hazards control. This is the area of accidental or gradual changes in characteristics or condition of plant and equipment, or development of unanticipated situations. Each principal staff member, section supervisor, and plant operator has an inherent responsibility to be continually alert for such changes and for reporting them upon detection. The periodic inspection of plant equipment and the continuing review and analysis of data from plant logs, instruments, and special tests provide regular sources of information on plant conditions. Subsequent hazards analysis reports and recommendations arising from such changes in plant conditions are handled in accordance with the procedure for planned changes.
11. TRAINING PROGRAM

11.1 General

The basic objective of the training program established for the EGCR Project is to develop and maintain a well-qualified organization capable of performing those functions delegated to the operator by the AEC. In order to fulfill this basic objective, the training program is divided into two phases, a program to develop or train currently assigned Project personnel to perform those functions described in Section 10.2, and a program to maintain an effective organization by providing continuous training as required by either change in the organization, plant design, or method of operation. The first phase of the training program is discussed in detail in the sections that follow. The second phase of the training program is not discussed in detail since it is a program that must be developed and carried out after operation has begun.

In order to establish a training program which satisfies the requirement of developing a staff competent to carry out all phases of plant operation, it is necessary to compare the background and experience of Project personnel with the various group functions described in Section 10.2. A brief review of the background and experience of those personnel assigned to the EGCR Project indicates that the individual nuclear and conventional power plant experience varies widely. Most of those individuals with experience in operating TVA power stations had little or no nuclear experience prior to initiating the training program. Conversely, those individuals with considerable experience in the nuclear field had little or no experience with the operation of a power station. As a result of these two extremes in background and experience, it has been necessary to develop a training program with sufficient diversification to meet the needs of both of these groups as well as to supplement the background of those persons recruited from other organizations.

The training of EGCR personnel is described for each of the major groups of the plant organization: the Operations Group, the Technical Program Group, and the Radiological Health Section.

The training of Project personnel is accomplished by several methods including academic training, assignment to other organizations, and on-the-job training. A general description of each of these methods is included in Sections 11.2 through 11.4. This description indicates that the program is directed primarily toward the Operations Group rather than the Technical Program Group or Radiological Health Section. Employees in the Operations Group, with but few exceptions, had little nuclear experience when joining the Project, and, therefore, both academic training and assignments to other organizations were necessary to qualify these staff members for certification and for carrying out their assignments. The Technical Program Group supervisors had an average of about 5-1/2 years of experience including, in most cases, both academic training (ORSORT) as well as on-the-job training at some AEC facility. Assignment of personnel to the AEC facilities has been such as to acquire a general knowledge of the problems associated with reactor design and development as well as ability in the specific field the supervisor is now assigned. Background and training is tabulated in Table 11.1.
11.2 Academic Training

Academic training includes both formal and informal classroom instruction. The following courses of instruction are provided as a part of the training program. A brief description of each course, except the ORSORT courses, is included to indicate the scope of training. Participation in one or more of the academic training courses is indicated in Table 11.1.

The academic courses in which members of the three groups have participated in the same academic training, are described below.

a. Oak Ridge School of Reactor Technology (ORSORT)

All ORSORT courses are presented by the Education Division of the Oak Ridge National Laboratory.

1. Health Physics (30 lecture hours, 25 laboratory hours)
2. Mathematics (60 lecture hours)
3. Pre-Physics (30 lecture hours, 18 laboratory hours)
4. Reactor Instrumentation and Controls (22 lecture hours, 40 laboratory hours)
5. Experimental Reactor Physics (25 lecture hours, 75 laboratory hours)
6. Reactor Analysis (90 lecture hours)
7. Reactor Chemical Technology (36 lecture hours)
8. Reactor Engineering Science (75 lecture hours)
9. Reactor Materials (60 lecture hours)
10. Reactor Shielding (30 lecture hours)
11. Economics of Nuclear Power (20 lecture hours)
12. Geology (21 lecture hours)
13. Meteorology (21 lecture hours)
14. Hazards Evaluation (100 lecture hours)

b. TVA Management Training-EGCR I (20 hours)

This course is given by TVA staff members. The course consists of a series of lectures on the fundamentals of management.

c. Elements of Health Physics-EGCR II (20 hours)

This course consists of eight 2-hour lectures, a 2-hour laboratory session, and a final examination. This course is presented by the radiological health supervisor to all personnel in the Technical Program Group, and to the supervisors of the Operations Group. Subjects covered in the course were:

- History and Chronology of Radiation Discovery
- Philosophy of Radiation Safety at the EGCR
- Review of Atomic and Nuclear Physics
- Theory of Radioactive Decay
- Units and Equivalents of Radiation
- Radiation and Contamination Control
- Personnel Monitoring
- Personnel, Area, and Equipment Decontamination

11.2.1 Operations Group

a. ORSORT

The operating superintendent and three plant operations supervisors attended ORSORT for varying periods of time. Refer to Table 11.1 for the time and courses taken.
b. **EGCR Basic Reactor Technology—EGCR III**

This course is designed to present a fundamental understanding of general nuclear considerations of practical value to the shift engineers, unit and assistant unit operators. Instruction is provided by EGCR staff personnel. The subjects comprising the course are:

1. **Applied Mathematics** (78 lecture hours)—Algebra, trigonometry, vectors, graphs, geometry, periodic functions, differential and integral calculus, probability, and statistics.

2. **Slide Rule** (35 lecture hours)—Powers of ten, common logarithms, natural logarithms, logarithms to any base, antilogarithms, roots, slide rule construction, theory and operation, and application to trigonometry.

3. **Basic Nuclear Physics** (46 lecture hours)—Review of basic physics, electromagnetic radiation, energy and units, atomic and nuclear structure, binding energy, radioactivity, interaction of radiation with matter, nuclear reactions, cross-section, resonance, and mean-free path.

4. **Nuclear Power Engineering** (45 lecture hours)—Review of nuclear physics; reactor theory; radiation shielding; power reactor design, control, and operation; and nuclear instrumentation and controls.

5. **Health Physics** (15 lecture hours)—History of health physics; review of serious nuclear incidents; radiation safety and protective measures; radiation type, origin, and interaction with matter; radiation units; definitions and equivalents; permissible radiation limits; radiation control measures and procedures; and decontamination. Included also is a demonstration of portable survey meters, air monitors, hand and foot counters, and student participation in smearing and air sampling techniques.

c. **EGCR Process and Control Systems—EGCR IV** (690 hours)

The purpose of this course is to familiarize the operator with the individual plant process and control systems. It presents a description of component and system construction, location, and application. A comprehensive review is made of all system design criteria and operations. Included is a detailed study of reactor instrumentation, control, and protective systems with major emphasis placed upon system kinetics and response. Instruction is provided by various members of the Operations Group, and is supplemented by instruction from members of the other Project groups where appropriate. Volume I of the EGCR Operating Manual is used as the principal reference.

The study plan involves the following major areas:

- Nuclear Plant (320 lecture hours)
- Steam Plant (110 lecture hours)
- Electric Power System (170 lecture hours)
- Station Auxiliary Systems (90 lecture hours)

d. **EGCR Plant Operations—EGCR V** (150 lecture hours)

This course is designed to present a detailed study of integrated plant operations with special emphasis placed upon emergency operations and equipment limitations. The principal reference is Volume II of the
Operating Manual utilizing the following study format:

System review and operating assumptions
Outline of normal operating procedures
Pre-operational checkouts, interlocks, and precautions
Startup procedures
Normal operating procedures
Abnormal operating procedures
Emergency operating procedures
Scheduled shutdown procedures
Precautions during shutdown

During the last part of this training phase, a detailed study is made of the EGCR Final Hazards Summary Report, Volumes I and II.

Instruction is provided by various members of the Operating Group, and members of the Technical Program Group provide instruction in their various fields of specialization.

e. EGCR Supplementary Training for Operating Supervisors and Shift Engineers-EGCR VI (90 lecture hours)

This course of instruction was given during March and April of 1961, by various EGCR staff specialists as a familiarization course for key operating personnel. Course curriculum consisted of: control rod and drive systems; reactor coolant system; feedwater, steam, and condensate cycles; water treatment; compressed air systems; fire protection and service water systems; circulating water system; helium purification and recovery system; electrical distribution system; fuel handling and waste disposal systems; reactor and reactor shielding; burst slug detection system; containment hazards; charge and service machines; nuclear instrumentation and flux scanning; and radiation monitoring.

f. EGCR Controls Engineering Training-EGCR VII (77 lecture hours)

These lectures were part of the training program established for instrument mechanics (Section 11.2.2e). During March, April, and May of 1962, two plant operations supervisors, all shift engineers, and six unit operators attended selected lectures involving the following process and control systems: control rods, drives and instrumentation, nuclear instrumentation, reactor safety system, burst slug detection, pneumatic temperature monitoring, core viewing, radiation monitoring, plant alarm and annunciator systems, fuel handling, instrumented fuel assemblies, charge and service machines, failure-free and emergency power systems, and heating and ventilating systems.

g. ORNL Reactor Operations Lectures-ORNL I (55 lecture hours)

This course was given by ORNL staff members to all five shift engineers while on assignment at ORNL for on-the-job training at the Oak Ridge Graphite Reactor (ORGR), the Oak Ridge Research Reactor (ORR), and the Low Intensity Test Reactor (LITR). The topics covered were: heat transfer, shielding, general reactor kinetics, practical reactor operations, and the philosophy of reactor construction, location and control.

h. ORNL Radiation and Safety-ORNL II (15 lecture hours)

These lectures were given by ORNL staff members to the EGCR shift engineers during their 7-months on-the-job training assignment. Topics covered include biological effects of radiation, shielding,
instrumentation, and waste disposal.

i. PRTR Assistant Engineer Training Program-PRTR I (48 lecture hours)

This course was attended by the two shift engineers and four unit operators while on assignment at the Plutonium Recycle Test Reactor (PRTR) for on-the-job training. The course, consisting of 16 lecture sessions presented by PRTR operations staff members during the period September 5 to December 26, 1961, was designed to familiarize PRTR assistant engineers and EGCR operators with PRTR operations.

j. PRTR Operators Training Program Lectures-PRTR II (9 lecture hours)

These lectures were attended by the two shift engineers and four unit operators during their on-the-job assignment at PRTR by PRTR shift supervisors. The purpose of the lectures was to acquaint the PRTR and EGCR operators with PRTR plant design changes and operating procedures.

k. GETR Fundamentals of Radiation Protection Methods-GETR I (20 lecture hours)

This course, designed to provide the basic fundamentals of radiation protection methods, was completed by the two unit operators assigned for on-the-job training at the General Electric Test Reactor (GETR). This instruction, given by GETR shift health physics technicians, includes the following topics: review of atomic theory, radiation units, administrative and physical control of radiation and contamination, and exposure records.

l. GETR Operator Training Program Lectures-GETR II (20 lecture hours)

These lectures, presented by shift supervisors and technicians, were attended by the two unit operators assigned to GETR for on-the-job training. The lectures were designed to familiarize GETR and EGCR operators with GETR operations.

11.2.2 Technical Program Group

a. ORSORT

Six members of the Technical Program Group completed ORSORT as a part of the training program. Two additional engineers in the Controls Engineering Section attended ORSORT on a part-time basis, completing one or more courses.

b. Oak Ridge Institute of Nuclear Studies-ORINS

One chemist, assigned to the Chemical Engineering Section, was given a 4-week isotopes course and a 2-week course in activation analysis. The other chemist in the Chemical Engineering Section is to receive the 6-week isotopes course given at ORINS.

c. Chemical Analyst Training

Three chemical analysts, assigned to the Chemical Engineering Section, are to receive approximately four months of training in the fundamentals of radioactivity and its measurement. This training is directed by the EGCR chemical laboratory supervisor. Approximately 25 percent of the time is devoted to lectures, and the remaining time is used for conducting experiments involving the handling and measuring of radioactivity using EGCR radioactivity measurement systems. The program
also includes about two months of training in analytical techniques peculiar to EGCR systems (gas chromatography, chemical treatment of water systems, etc.). The chemical analysts also receive the health physics course described in Section 11.2.c.

d. **Instrumentation Course** (80 hours)

One engineer in the Controls Engineering Section received formal training at the Minneapolis-Honeywell school. This training consisted of lectures, demonstrations, and laboratories on the fundamentals of instrumentation.

e. **EGCR Controls Engineering Training-EGCR VIII**

This program provides instrument mechanics with special training in the principles and theories of nuclear power plant instrumentation with special emphasis on EGCR requirements. The program is directed by the Controls Engineering Section. The training is provided to the instrument mechanics in two groups. The first group consists of the instrument foreman and three instrument mechanics who are presently on the job. The second group will receive the training after joining the Project. All instrument mechanics and instrument foremen are required to complete the training program. The training for both groups is described below, and is essentially the same except as noted. Members of the Operations Group attended selected lectures as indicated in Section 11.2.1.f.

1. **EGCR Process Systems** (150 hours)

   This course presents a description of the construction, location, operation, and application of all systems and components.

2. **EGCR Special Systems** (100 hours)

   This course covers special or auxiliary systems not studied during the process systems courses. Some of these systems are the burst slug detection system, flux scanning system, pneumatic temperature monitoring system, etc. The design, operation, and application of the systems are discussed.

3. **Factory Maintenance Schools** (40 hours)

   The participants are given training by factory representatives either at the EGCR site or at the school of a particular company. This training is primarily on components and systems that are new to the industry or peculiar to the EGCR application.

4. **EGCR Components** (100 hours)

   This course consists of lectures and demonstrations on plant components with particular emphasis on those components that are unfamiliar to the participants.

5. **Electronics Training** (150 hours)

   Electronics training is designed to make the mechanic proficient in maintaining and in troubleshooting electronic equipment used in a nuclear power plant. The training includes circuit analysis and troubleshooting, application of electronics to nuclear instrumentation, and transistor theory and application.

11-6
6. Test and Maintenance Equipment (40 hours)

Instruction is given in the use of test and maintenance equipment. The participant demonstrates proficiency by conducting experiments under the direction of the instructor.

7. Special Shop Projects (165 hours)

This phase of the training consists of shop work in support of the test program.

8. Basic Reactor Technology

This course includes the following:

- Slide rule (40 hours)
- Mathematics (80 hours)
- Basic nuclear physics (50 hours)
- Nuclear power engineering (50 hours—second group only)
- Health physics (10 hours)
- Field trips, movies, study periods, and examinations (170 hours)

The material covered in each of the above subjects is essentially the same as described in Section 11.2.1.b.

11.2.3 Radiological Health Section

The training program for the health physics technicians comprises eight months of combined academic study and systems familiarization. Formal classroom instruction in both theory and application of radiation protection is included. Coincident with classroom lectures is a generalized acquaintance of all plant systems and components with particular emphasis on potential radiation and contamination hazards. During the training program, procedures, forms, and records are developed which will incorporate certain phases of the radiological health program.

11.3 Assignment to Other Organizations

Assignment of personnel to other organizations has been made in those cases where it is necessary to provide nuclear experience or to supplement previous experience prior to assignment to the EGCR Project. Although most assignments of personnel are to organizations with operating reactors, some assignments are to obtain other special training required for a given job. Assignments to other organizations are described below.

11.3.1 Operations Group

a. Berkeley Nuclear Power Station

The Berkeley Nuclear Power Station, located near Bristol, England, is a natural uranium, heterogeneous, graphite moderated, carbon dioxide cooled reactor owned and operated by the Central Electricity Generating Board. There are two reactors in the plant, each rated at 557 MwT. Eight steam generators for each reactor deliver steam at two pressures to two turbo-alternators, each rated at 85 Mw. Fueling operations are carried out with the reactor on power and at full pressure.

Assignment of personnel from the Operations Group to the Berkeley Station was scheduled during several periods to obtain experience in all phases of reactor operations. The first group, consisting of the
operating superintendent and three plant operations supervisors and
designated below as Berkeley I, performed various jobs during the period
between May and October 1961. The second group, consisting of two plant
operations supervisors (Berkeley II) was on assignment to the Berkeley
reactor during the time period between October 1961 and April 1962.

1. Berkeley I—During the period between May and October 1961, the first
Berkeley reactor was undergoing pre-operational tests and achieved
criticality on August 29, 1961. During this period, the four
assignees participated in the following activities:

   (a) **Pre-operational Equipment Tests** (May 2 to August 12)

   Assisted the plant operating crews on work involving fuel
   charging equipment tests, carbon dioxide system commissioning,
   operation of reactor coolant system on air using the main blower
   losses as heat source, control rod and safety system tests,
   burst cartridge detector system operational tests, and spent
   fuel handling equipment tests.

   (b) **Precritical Fuel Loading Operations** (August 12 to August 29)

   Acted as supervisors for the small pile fuel loading operations.

   (c) **Reactor Parameter Measurements** (August 29 to September 16)

   Assisted operating forces in the performance of the following
tests: variation of $k_{eff}$ with pile radius, air pressure
coefficient of reactivity, calibration of low power instrumenta-
tion, initial criticality predictions, effect of zirconium
brackets on flux and reactivity, control rod calibrations,
radial flux distribution, effect on reactivity and flux of
charging a channel by the charge machine.

   (d) **Fuel Reactor to Full Size** (September 16 to October 6)

   Assisted operating forces in preparing to fuel and fueling
reactor to full size.

   (e) **Gas-Cooled Reactor Simulator Course** (80 hours)

   Received simulator instruction identical to that given to all
Central Electricity Generating Board operating supervisors.
The simulator represented a graphite-moderated, gas-cooled,
natural uranium reactor with six main blowers and six heat
exchangers supplying two turbo-alternators working on a dual
pressure cycle. Normal plant startups and shutdowns were
simulated, using both positive and negative temperature
coefficients of reactivity. Operating drills were held on
abnormal conditions such as feedwater, main blower, and boiler
failures at various loads.

2. Berkeley II—During the period October 2, 1961, to April 1962, two
plant operations supervisors were assigned to the Berkeley reactors.
The two assignees participated in the following activities:

   (a) **Loading to Full Size** (October 6 to 21)

   Assisted in neutron flux measurements and reactivity deter-
minations at the various loading stages.
(b) **Channel Air Flow Tests on Loaded Reactor** (October 25 to November 22)

Assisted in positioning anemometers in top plenum and obtaining channel flow readings to verify proper gag installation.

(c) **Full Pile Critical Tests** (December 9 to March 15)

Assisted in calibrating control rods for various rod configurations, reactor temperatures and pressures; calibrating low power equipment; and measurements of the combined temperature coefficient. Supervised foil insertions for flux shape determinations.

(d) **Proving Run of the Carbon Dioxide Circuits** (March 16 to April 3)

Assisted operations personnel in initial charging of reactor and loops with CO₂; combined blower run; moisture, leak, and temperature checks; burst slug detection gear mechanical proving run; and functional tests on fuel handling equipment.

(e) **Gas-Cooled Reactor Simulator Course** (80 hours)

Received instruction identical to that obtained by the Berkeley I assignees.

(f) **Supplementary Trips**

Visited a number of nuclear installations in the United Kingdom. Included was a weeks attachment at the Bradwell Nuclear Power Station during the final proving run of Reactor No. 1, visits to the Dragon Project and Zenith critical facility at Winfrith Heath, and visits to the Chapel-cross and Advanced Gas-Cooled Reactor projects.

3. **Berkeley III**—During the period April 25 through June 21, 1963, the assistant to the operating superintendent was assigned to the Berkeley reactors. This period was selected to take advantage of plant operations which were optimum for the experience desired. One reactor was shut down for major maintenance, refueling, and control rod replacement while the other was in full operation. He participated in all phases of reactor operation and observed all major maintenance which was in progress during his attachment.

b. **Calder Hall**

The Berkeley II assignees also completed a six-weeks attachment to the Calder Hall reactors during April and May 1962. During this period, both were assigned to the shift managers in charge of operating all four reactors. Operating experience obtained included detection of a fuel element fast burst; shutdown of the reactor to discharge and recharge the failed fuel channel and restoration of the reactor to full power; and shutdown, refueling, and maintenance on Reactor No. 1.

The assignees also participated in performing analyses to determine the amount of blower cooling required on reactor shutdown and to maintain the reactor temperature during refueling operations.
c. Oak Ridge National Laboratory (ORNL)

Two groups of operations personnel were assigned to the ORNL Operations Division where experience was provided by on-the-job training at the ORGR, ORR, and LITR.

1. **ORNL I**

All shift engineers were assigned to ORNL Operations for the period November 1960 through May 1961. Their on-the-job experience included prestartup checks, startup, normal operation, and shutdowns of the three reactors.

During this training, each of the assignees completed the ORNL Operations Division requirements for probationary operators as included in ORNL-CF-59-110, Reactor Operator Training Manual.

2. **ORNL II**

One plant operations supervisor was assigned for three months, and one unit operator for six months, to ORNL Operations in the period February to August 1962. The on-the-job training that each received was similar to the training received by the shift engineers. Each assignee satisfactorily completed the checklists in the ORNL Reactor Operators Training Manual. These checklists are used at ORNL for operator training prior to operator certification.

d. **Plutonium Recycle Test Reactor (PRTR)**

The PRTR is a heavy water, moderated and cooled, pressure tube reactor operated by General Electric for the AEC. The reactor is rated at 74 MwT, and achieved criticality in November 1960.

Two shift engineers and four unit operators were assigned to this reactor for on-the-job training. The shift engineers were assigned for separate periods of three months and the unit operators for periods of six months. The training received by these assignees consisted of formal instruction by PRTR staff members and practical experience while attached to operating shifts. Prior to completion of the training assignment, the assignees satisfactorily passed a written examination prepared by the PRTR Operations Section after which they received further training, primarily with reactor controls. Subsequently, each man satisfactorily completed an operational checklist prepared by the PRTR Operations Section.

Two members of the Plant Maintenance Section were assigned to PRTR for a period of three weeks to study and participate in maintenance operations.

e. **General Electric Test Reactor (GETR)**

The GETR is a pressurized water reactor operated by General Electric. It has a power level of 30 MwT and is operated as an experimental facility to test materials.

Two unit operators were assigned to the GETR for a period of six months. The training received by these assignees consisted of formal instruction by GETR staff members and practical experience while attached to operating shifts. Both assignees were permitted to operate the reactor to perform startups, shutdowns, recovery from scrams, and steady-state operations throughout their assignment. Each satisfactorily passed a
written qualifications examination prepared by the GETR Operations Section.

f. **Kingston Steam Plant**

The Kingston Steam Plant is a modern conventional steam plant owned and operated by the TVA. This plant consists of nine units with a total rated capacity of 1440 electrical MW. The objective in assigning staff members to the conventional plant is to obtain experience with the organization, operation, and maintenance of a conventional plant. Two operations supervisors, who had little previous conventional plant experience, were assigned to this plant for a month each in November and December 1962. One member of the Plant Maintenance Section was also assigned to the Kingston plant for two weeks.

g. **Consolidated Edison Thorium Reactor (Indian Point)**

The Consolidated Edison Thorium Reactor is a pressurized water, thorium converter reactor owned and operated by the Consolidated Edison of New York. The reactor is rated at 585 MWt, and achieved criticality in August 1962.

Three shift engineers are scheduled for a period of four months of on-the-job shift-work experience at this facility during the period June through September 1963.

h. **EGCR Operators Simulator Training Program**

Arrangements have been made to utilize the ORNL analogue computer to simulate the behavior of the EGCR and control systems. An Electronics Associates, Inc., analogue computer model 231-R is used to simulate the reactor and a model TR-10 computer manufactured by the same company is used to represent the elements of the reactor control system.

Each of the operators to be certified before startup receives this training to become familiar with the characteristics and kinetic response of a reactor of the EGCR type. Each operator is scheduled to receive a minimum of 40 hours of instruction on the simulator early in 1964. The plant operations supervisors assist in the trial runs of the simulation and prepare an operating manual to be used in the simulation.

Typical simulations include plant heatup using reactor coolant blowers, approach to criticality, effects of temperature coefficients, steady-state and transient power level operation, effects of xenon, effects of changes in blower speed, blower control response to power and flow mismatch, controlled shutdown, and reactor scram.

11.3.2 **Technical Program Group**

a. **Berkeley Nuclear Power Station**

Employees from the Technical Program Group have been assigned to the Berkeley Station during three time periods between October 1962 and February 1963. Some assignments were during the time periods when members of the Operations Group were also assigned to the Berkeley Station and, in these cases, a portion of the training received is similar to that previously described and is so noted.
1. Berkeley II

During the period between October 1961 and March 1962, the assistant to the project manager and the reactor engineering section supervisor were assigned to the Berkeley Station. (Although the assistant to the project manager is not a member of the Technical Program Group, his training is described here since it is similar to that of the reactor engineering section supervisor.) The purpose in assigning members of the Technical Program Group (or of the Manager's Office) to an operating reactor, is to obtain not only operating experience, but also to obtain information relative to the planning and development of technical information relating to pre-operational tests and operations. Therefore, both of the assignees devoted a large portion of their time in obtaining, reviewing, and discussing the precommissioning test program. Training during the period also included the full-pile critical tests, the gas-cooled reactor simulator course, and supplementary trips as described previously for the plant operations supervisors.

2. Berkeley IV

During the period between February and June 1962, one member of the Reactor Physics Staff was assigned to the Berkeley Station. During the period, the assignee functioned as a member of the station physics staff, performing analyses in support of the test program and participating in experiments as described below. Particular attention was devoted to obtaining and discussing information that could be directly applied to the EGCR physics test program. The assignee participated in various tests including the following:

(a) Control rod calibration and pressure coefficient of reactivity measurement on both reactors.

(b) Insertion and removal of foils to obtain flux plots. Served as shift physicist during the irradiation period, assuring that proper care was taken in preparing, handling, and counting foils.

(c) Making subcritical flux plots.

(d) Calibration of low-power equipment and associated flux plots.

(e) Flux fine structure test.

(f) Burst cartridge detection gear tests on Reactor I.

(g) Initial power phase tests on Reactor I.

The assignee participated in performing numerous analyses supporting the test program, including the following types of calculations:

(a) Air pressure required in the reactor vessel during rod calibration tests.

(b) Analysis of systematic errors between predicted and experimental flux plots.

(c) Correcting experimental control rod calibration data for temperature effects.
(d) Analyses of pressure coefficients of reactivity, including corrections due to temperature, rod position, etc.

(e) Analysis of results of safety rod worth experiments.

3. Berkeley V

During the period between September 1962 and February 1963, the superintendent of the Technical Program Group was assigned to the Berkeley plant. The purpose of the assignment was to review the completed commissioning program for the Berkeley I reactor, to observe and to participate in some of the commissioning experiments for the Berkeley II reactor, to study the civil station commissioning procedures, and to confer with UKAEA and CEGB staff on reasons for and results obtained from the experiments. A further objective was to obtain a document collection which would comprehensively describe a typical civil station commissioning program, with particular emphasis on papers of principle which supports the experimental work. The assignee was primarily concerned with the commissioning programs to see that the elements of such programs were given due consideration in the EGCR planning.

As a part of this assignment, the assignee visited and conferred with staffs at several other British installations including the Advanced Gas-Cooled Reactor.

b. Oak Ridge National Laboratory-ORNL III

Assignment of TVA technical personnel to ORNL for specialized experience in the nuclear field was initiated by TVA in 1953. During the period between 1953 and 1960, as many as 12 engineers were assigned to various groups at ORNL on a loan basis. This experience is indicated in Table 11.1 as nuclear experience.

Assignment of technical personnel to ORNL as a part of the EGCR training program is indicated below. Some of the assignments are primarily for training purposes, others represent cooperative efforts in carrying out the EGCR program.

1. The instrument foreman of the Controls Engineering Section was assigned to ORNL for a period of approximately two months. During the first month, the assignee observed operation and maintenance procedures at the ORR, ORGR, and LITR, with special emphasis on becoming familiar with nuclear instrumentation, observing reactor startup procedures, and studying instrument maintenance and the preventive maintenance program. The last month of the assignment consisted of attending the ORSORT instrumentation and controls lecture series and in assisting in the preparation of the ORSORT instrumentation and controls experiments.

2. An analytical chemist in the Chemical Engineering Section is to be assigned to a radiochemical laboratory for a period of four months for on-the-job training in the general field of radiochemistry.

3. The controls specialist was assigned to the Instrumentation and Controls Division for three months during the review of the EGCR instrumentation and controls. He also received the same training as previously described for the shift engineers, (ORNL I).
4. The supervisor, Experimental Engineering Section, is on full-time assignment to the Metallurgy Division to participate in the fuel development program in support of the gas-cooled reactor program. He was previously assigned to the Reactor Division and participated in the preliminary design and hazards evaluation of the EGCR experimental loops.

c. Kingston Steam Plant

Three engineers in the Controls Engineering Section were assigned to the Kingston Steam Plant for periods of two to four weeks. The purpose of these assignments was to study the organization and functioning of the instrument maintenance group at the Kingston plant, and to become more familiar with instrumentation and controls problems associated with conventional plants.

d. Dresden Nuclear Power Station

Two members of the Controls Engineering Section were assigned to the Dresden Station for two weeks to study and participate in the instrument maintenance program.

11.3.3 Radiological Health Section

All of the shift technicians are to be assigned to the ORNL Applied Health Physics Section for four weeks of field training. This training is to include participation in personnel monitoring, area monitoring, reactor monitoring, and sample counting activities.

11.4 On-The-Job Training

All training provided to TVA personnel at the EGCR site, with the exception of the academic training described in Section 11.2, is designated as on-the-job training. This training is aimed at developing an understanding of the basis for the design and operation of the EGCR. The following types of work comprise the on-the-job training:

a. Instruction of staff members.


c. Familiarization with design basis, operation, maintenance, testing, and location at the site of equipment and systems.

d. Participation in pre-operational testing, initial startup of plant systems, analysis of equipment malfunctions, correction of faults, and other engineering work in getting the plant ready for initial startup.

e. Participation in design review.

f. Planning and scheduling pre-operational and operational programs.
11.5 **Staff Training Prior to Operation**

The experience and background and training of supervisory personnel assigned to the EGCR Project is summarized in Table 11.1. In addition, general information is provided about shift engineers, unit and assistant unit operators. In general, only that experience relating to the design, operation, maintenance or testing of nuclear facilities, conventional power plants or equivalent type of experience is included in the tabulation.

11.6 **Training Program Following Initial Operation**

The basic training program described in previous sections is designed to develop a well-qualified operating organization capable of performing all the functions delegated to the operator by the AEC. This program is necessarily supplemented at the EGCR by a continuing training program which has the dual purpose of maintaining a well-qualified operating organization and by providing training to employees of other organizations as requested by AEC.

11.6.1 **EGCR Staff Training**

The training program following initial operation is directed at accomplishing the following:

a. **Changes in Staff**—All new employees are provided with adequate training commensurate with their position and background. This training is provided in a manner similar to the pre-operational training previously described.

b. **Changes in Design or Method of Operation**—Personnel are provided with adequate instruction regarding changes in reactor design or method of operation. Instruction is provided by senior staff members.

c. **Dissemination of Information**—Periodic lectures and discussions are provided to disseminate information pertaining to:

1. Plant performance
2. Fire prevention and protection
3. Industrial safety
4. Plant security
5. Radiological health and safety

11.6.2 **Employees of Other Organizations**

Training of employees from other organizations is provided when requested by AEC. This training is carried out in the same manner as for new staff members as discussed in Section 11.6.1. The amount of training varies, depending on the objectives and the amount of responsibility the employee is given while at the EGCR. In no case is an employee from another organization placed in a responsible job without proper training and supervision.
## Table 11.1
Background and Training of EGCR Staff

Manager's Office

<table>
<thead>
<tr>
<th>Job</th>
<th>Education</th>
<th>Nuclear Experience</th>
<th>Non-Nuclear Experience</th>
<th>Academic</th>
<th>Assignment to</th>
<th>On-The-Job (Months)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Project</td>
<td>B. S. in E. E. Rensselaer Polytechnic Institute, 1925</td>
<td>8 yr in charge of TVA nuclear power study, including 2 yr in special assignment to AEC Division of Reactor Development</td>
<td>2½ yr electrical test, GE; 7 yr electrical design and construction, Penn. R. R.; 18 yr electrical eng and admin, TVA</td>
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<td>Manager</td>
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<td>(W. R. Cooper)</td>
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<tr>
<td>Assistant to</td>
<td>B. S. in Marine Engr U. S. Merchant Marine Academy, 1946; B. S. in Ch. E., Alabama Polytechnic Institute, 1948</td>
<td>1½ yr Brookhaven National Lab., 4 yr ORNL, design and dev. of reactors; 1 yr AEC-ORO, design coordination EGCR</td>
<td>2½ yr TVA steam plants engineering EGCR II-20 hr laboratory</td>
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<td>Berkeley II 5 months 45</td>
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<td>the Project</td>
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<td>Manager</td>
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<td>(H. L. Falkenberry)</td>
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*Months of on-the-job training are based on completing all pre-operational training by October 1, 1964.*
### TABLE 11.1 (continued)

**Background and Training of EGCR Staff**

**Radiological Health Section**

<table>
<thead>
<tr>
<th>Job</th>
<th>Education</th>
<th>Nuclear Experience</th>
<th>Non-Nuclear Experience</th>
<th>Academic Organizations (Months)</th>
<th>Assignment to Other Job</th>
<th>On-The-Job Experience</th>
</tr>
</thead>
<tbody>
<tr>
<td>Radiological Health</td>
<td>B. S. in Chem., Austin Peay</td>
<td>1½ yr Radio-</td>
<td>3 yr operational</td>
<td>ORSORT 121 hr</td>
<td>None</td>
<td>41</td>
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<tr>
<td>Operating Superintendent (J. R. Calhoun)</td>
<td>B. S. in E. E. Tennessee Polytechnic Institute, 1949</td>
<td>None</td>
<td>11 yr TVA steam plant operation and maintenance. Assistant Superintendent of a 1500-Mw station</td>
<td>ORSORT 400 hr Berkeley I 35</td>
<td>Courses 1-3(100%) 6 months, 4(99%) EGCR Simulator 5-11 (1/3-1/2)</td>
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<tr>
<td>Job</td>
<td>Education</td>
<td>Nuclear Experience</td>
<td>Non-Nuclear Experience</td>
<td>Academic</td>
<td>Assignment to Other Organizations</td>
<td>On-The-Job (Months)</td>
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<tr>
<td>Assistant to the Operating</td>
<td>B. S. in Eng, U. S. Naval Academy, 1953.</td>
<td>2½ yr operation</td>
<td>5 yr operations</td>
<td>EGCR I-20 hr</td>
<td>Berkeley III</td>
<td>34</td>
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<td>Superintendent</td>
<td>(H. J. Green) Advanced Nuclear Power School, 1958</td>
<td>and maintenance</td>
<td>and maintenance</td>
<td>II-20 hr</td>
<td>Kingston Steam</td>
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<td></td>
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<td>with SSG and USS Skipjack</td>
<td>on submarines and ships</td>
<td>IV-690 hr</td>
<td>Plant - 1 month, EGCR Simulator</td>
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<td>V-150 hr</td>
<td>Training</td>
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<td>VI-90 hr</td>
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<td>ORSORT 400 hr</td>
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<td>Courses 1-3(100%)</td>
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<td>-4(99%) ORNL II</td>
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<td>5-11 (1/3-1/2)</td>
<td>3 months, EGCR Simulator</td>
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<td>II-20 hr</td>
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<td>IV-690 hr</td>
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<td>V-150 hr</td>
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<td>VI-90 hr</td>
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<td>ORNL I-25 hr</td>
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<td>ORNL II-7 hr</td>
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<td>Plant Operations Supervisor</td>
<td>B. S. in E. E., University of Tennessee, 1959</td>
<td>11 yr electrical</td>
<td>12 yr electrical</td>
<td>ORSORT 500 hr</td>
<td>Berkeley II</td>
<td>31</td>
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<td>(J. M. Ballentine)</td>
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<td>maintenance at</td>
<td>engineering, TVA</td>
<td>Courses 1-6(100%)</td>
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<td>TVA hydro and</td>
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<td>steam plants</td>
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<td>1⅔ months, EGCR Simulator</td>
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<td>II-20 hr</td>
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<td>VI-90 hr</td>
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### TABLE 11.1 (continued)

**Background and Training of EGCR Staff**

**Operations Group (continued)**

<table>
<thead>
<tr>
<th>Job</th>
<th>Education</th>
<th>Nuclear Experience</th>
<th>Non-Nuclear Experience</th>
<th>Academic Experience</th>
<th>Assignment to Other Organizations (Months)</th>
<th>On-The-Job Experience</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plant Operations Supervisor (R. G. Metke)</td>
<td>B. S. in M. E., None</td>
<td>12 yr TVA steam plant operation</td>
<td>ORSORT 500 hr</td>
<td>Berkeley II</td>
<td>31</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Tri-State College, 1948</td>
<td></td>
<td>Courses 1-6 (100%)</td>
<td>TVA steam ORSORT</td>
<td>1 yr, 1/3 months</td>
<td>6 months, Calder Hall</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>7-11 (1/3)</td>
<td></td>
<td>EGCR I-20 hr</td>
<td>1 yr, 1/3 months, EGCR Simulator Training</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>EGCR II-20 hr</td>
<td></td>
<td>IV-690 hr</td>
<td>EGCR Simulator Training</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>V-150 hr</td>
<td></td>
<td>VI-90 hr</td>
<td></td>
</tr>
<tr>
<td>Plant Operations Supervisor (F.A. Szczepanski)</td>
<td>B. S., U. S.</td>
<td>3½ yr operations</td>
<td>1½ yr Third Assistant</td>
<td>EGCR I-20 hr</td>
<td>Berkeley I</td>
<td>27</td>
</tr>
<tr>
<td></td>
<td>Merchant U. S. Naval</td>
<td></td>
<td>Engineer, SS Constitution</td>
<td>EGCR II-20 hr</td>
<td></td>
<td>6 months, Kingston Steam</td>
</tr>
<tr>
<td></td>
<td>Marine Academy, Power Training</td>
<td>1956; Nuclear Prototype</td>
<td></td>
<td>IV-690 hr</td>
<td>Plant</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Program - SLC Training</td>
<td></td>
<td></td>
<td>V-150 hr</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Prototype, Chief Reactor Opera. Training Course, 1960</td>
<td></td>
<td></td>
<td>VI-90 hr</td>
<td>EGCR Simulator Training</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>VII-77 hr</td>
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</table>
### Table 11.1 (continued)

#### Background and Training of EGCR Staff

**Operations Group (continued)**

<table>
<thead>
<tr>
<th>Job</th>
<th>Education</th>
<th>Nuclear Experience</th>
<th>Non-Nuclear Experience</th>
<th>Academic</th>
<th>Assignment to Other Organizations</th>
<th>On-The-Job Training (Months)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plant Operations Supervisor (C. C. Wheeler)</td>
<td>B. S. in Ch.E., 7½ yr general</td>
<td>reactor experience, including design, operation</td>
<td>None</td>
<td>EGCR I-20 hr II-20 hr IV-690 hr V-150 hr VI-90 hr VII-77 hr</td>
<td>Berkeley I 6 months, Kingston Steam Plant 1 month, EGCR Simulator Training</td>
<td></td>
</tr>
<tr>
<td>Shift Engineers (5) Graduate of TVA Student Generating Plant Operators' Training Program</td>
<td>High school</td>
<td>None</td>
<td>9 yr (avg) operating TVA steam plants with average of about 5 yr experience as asst shift engineer or shift engineer</td>
<td>ORNL I-55 hr II-15 hr PRTR I-48 hr (2 shift eng), (2 shift eng)</td>
<td>ORNL I 7 months (5 shift eng), PRTR 3 months (2 shift eng), PRTR 3 months (3 shift eng), EGCR Simulator Training (5 shift eng)</td>
<td></td>
</tr>
<tr>
<td>Job</td>
<td>Education</td>
<td>Nuclear Experience</td>
<td>Non-Nuclear Experience</td>
<td>Academic</td>
<td>Assignment to Other Organizations</td>
<td>On-The-Job Training (Months)</td>
</tr>
<tr>
<td>---------------------</td>
<td>--------------------</td>
<td>--------------------</td>
<td>------------------------</td>
<td>----------------</td>
<td>----------------------------------</td>
<td>-----------------------------</td>
</tr>
<tr>
<td>Unit Operators (15)</td>
<td>High school graduate or better.</td>
<td>None</td>
<td>9 yr (avg) operating experience TVA steam plants as unit or assistant unit operator</td>
<td>EGCR III-184 hr IV-690 hr V-150 hr VII-77 hr PRTR I-48 hr PRTR II- 9 hr GETR I-20 hr GETR II-20 hr ORNL I-25 hr ORNL II- 7 hr</td>
<td>ORNL II 6 months (1 operator), PRTR 6 months (4 operators), GETR 6 months (2 operators), EGCR Simulator Training (7 operators), None</td>
<td>26 (6)</td>
</tr>
<tr>
<td>Assistant Unit Operators (11)</td>
<td>High school graduate or better.</td>
<td>None</td>
<td>None</td>
<td>EGCR III-184 hr IV-690 hr V-150 hr</td>
<td>None</td>
<td>7</td>
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</tbody>
</table>

**Table 11.1 (continued)**

**Background and Training of EGCR Staff**

**Operations Group (continued)**
<table>
<thead>
<tr>
<th>Job</th>
<th>Education</th>
<th>Nuclear Experience</th>
<th>Non-Nuclear Experience</th>
<th>Academic</th>
<th>Assignment to Other Organizations (Months)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plant Maintenance Supervisor (B. D. Draper)</td>
<td>B. S. in M. E., University of Tennessee, 1948</td>
<td>ORNL - 3½ yr maintenance and development of Homogeneous Reactor; AEC-ORO, 1 year design coordination, EGCR</td>
<td>Mechanical design TVA steam plants 7½ years</td>
<td>EGCR I-20 hr II-20 hr</td>
<td>PRTR 3 weeks 44</td>
</tr>
<tr>
<td>Technical Program Superintendent (J. C. Ebersole)</td>
<td>B. S. in E. E., Mississippi State College, 1938; ORSORT, 1956</td>
<td>4 yr ORNL design and review of reactors and loops</td>
<td>Electrical design TVA steam and hydro stations, 13½ years</td>
<td>EGCR I-20 hr II-20 hr</td>
<td>Berkeley V 5 months 45</td>
</tr>
<tr>
<td>Chief Reactor Physicist (D. Smith)</td>
<td>Sheffield University 1953 Physics Degree</td>
<td>10 years - UKAEA reactor physics, kinetic analysis for gas-cooled reactors; physics commissioning experiments.</td>
<td>None</td>
<td>EGCR I-20 hr II-20 hr</td>
<td>None 12</td>
</tr>
</tbody>
</table>
TABLE 11.1 (continued)

Background and Training of EGCR Staff

Technical Program Group (continued)

<table>
<thead>
<tr>
<th>Job</th>
<th>Education</th>
<th>Nuclear Experience</th>
<th>Non-Nuclear Experience</th>
<th>Academic</th>
<th>Assignment to Other Organizations</th>
<th>On-The-Job (Months)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Hazards Control Engineer (H. N. Culver)</td>
<td>B. S. in C. E., Union College, 1950; B. S. in E. E. Union College, 1954; ORSORT, 1957</td>
<td>5 years ORNL design review and hazards evaluation of reactors and loops</td>
<td>Civil design TVA steam plants, 4 years</td>
<td>EGCR I-20 hr II-20 hr</td>
<td>None</td>
<td>47</td>
</tr>
<tr>
<td>Controls Specialist (D. C. Keeton)</td>
<td>B. S. in E. E., Alabama Polytechnic Institute, 1952</td>
<td>None</td>
<td>8 years testing electrical and mechanical equipment, TVA hydro and steam plants</td>
<td>ORSORT - 9 mo</td>
<td>ORNL I-7 mo</td>
<td>28</td>
</tr>
<tr>
<td>Reactor Engineering Supervisor (D. R. Patterson)</td>
<td>B. S. in M. E., University of Tennessee, 1948</td>
<td>4 yr Argonne Natl Lab, 1 yr Brookhaven Natl Lab, design and design review of reactors and loops; 1½ yr AEC-ORO design coordination, EGCR</td>
<td>Mechanical design TVA steam plant and hydro station, 5 years</td>
<td>EGCR I-20 hr II-20 hr</td>
<td>Berkeley II 4 months</td>
<td>41</td>
</tr>
</tbody>
</table>
### TABLE 11.1 (continued)

**Background and Training of EGCR Staff**

**Technical Program Group (continued)**

<table>
<thead>
<tr>
<th>Job</th>
<th>Education</th>
<th>Nuclear Experience</th>
<th>Non-Nuclear Experience</th>
<th>Academic</th>
<th>Assignment to Other Organizations</th>
<th>On-The-Job Job (Months)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Engineering Analysis</td>
<td>B. S. in M. E., 1½ yr USAEC Army Reactors Branch; M. S. in M. E., University of Tennessee, 1959; ORSORT, 1956</td>
<td>1½ yr USAEC Maritime Reactors Branch; 1½ yr USAEC Maritime Reactors Branch; 1½ yr USAEC Maritime Reactors Branch; 1½ yr USAEC Maritime Reactors Branch; 1½ yr USAEC Maritime Reactors Branch; 1½ yr USAEC Maritime Reactors Branch; 1½ yr USAEC Maritime Reactors Branch; 1½ yr USAEC Maritime Reactors Branch; 1½ yr USAEC Maritime Reactors Branch; 1½ yr USAEC Maritime Reactors Branch; 1½ yr USAEC</td>
<td>Mechanical design; TVA steam and hydro stations, reactor design; reactor design; reactor design; reactor design; reactor design; reactor design; reactor design; reactor design; reactor design; reactor design</td>
<td>ORNL; design review of reactors and loops</td>
<td>EGCR I-20 hr</td>
<td>47</td>
</tr>
<tr>
<td>Supervisor (E. G. Beasley)</td>
<td>Engineering University of Kentucky, 1951; M. S. in E. E., loops</td>
<td>6 yr ORNL design and development</td>
<td>15 months</td>
<td>electrical engineer, TVA</td>
<td>ORNL III</td>
<td>None</td>
</tr>
<tr>
<td>Experimental Engineering</td>
<td>University of Kentucky, 1952; ORSORT, 1954</td>
<td>6 yr ORNL design and development</td>
<td>15 months</td>
<td>electrical engineer, TVA</td>
<td>ORNL III</td>
<td>None</td>
</tr>
<tr>
<td>Job</td>
<td>Education</td>
<td>Nuclear Experience</td>
<td>Non-Nuclear Experience</td>
<td>Academic</td>
<td>Assignment to Other Organizations</td>
<td>On-The-Job Job Education Experience Experience Academic</td>
</tr>
<tr>
<td>----------------------</td>
<td>------------------------------------------</td>
<td>--------------------</td>
<td>------------------------</td>
<td>----------</td>
<td>-----------------------------------</td>
<td>--------------------------------------------------------</td>
</tr>
<tr>
<td>Controls Engineering Supervisor (R. M. Pierce)</td>
<td>B. S. in M. E., University of Tennessee, 1950</td>
<td>5 yr ORNL, instrument maintenance at the HRT, design and review of instrumentation and controls for reactors and loops, design review and analysis of EGCR instrumentation and control</td>
<td>Mechanical design EGCR I-20 hr TVA steam and hydro stations, 5 years</td>
<td>Dresden 2 weeks</td>
<td>42</td>
<td></td>
</tr>
<tr>
<td>Chemical Engineering Supervisor (W. L. Albrecht)</td>
<td>B. S. in Ch. E., University of Wisconsin, 1939; LLB, Jackson College of Law, 1949</td>
<td>4½ yr ORNL design and studies pertaining to reactor chemical problems</td>
<td>TVA Division of Chemical Eng, 3 yr: chemical plant design, 9½ yr Patent</td>
<td>None</td>
<td>50</td>
<td>Liaison Officer</td>
</tr>
</tbody>
</table>
### Background and Training of EGCR Staff

#### Technical Program Group (continued)

<table>
<thead>
<tr>
<th>Job</th>
<th>Education</th>
<th>Nuclear Experience</th>
<th>Non-Nuclear Experience</th>
<th>Academic</th>
<th>Assignment to Other Organizations</th>
<th>On-The-Job (Months)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Technical Staff</td>
<td>College graduate or better</td>
<td>Approximately 50 years - design, analysis, operation of reactors and nuclear facilities</td>
<td>Approximately 50 years - design or testing associated conventional power plants</td>
<td>See Section 11.2.2</td>
<td>See Section 11.3.2</td>
<td>40 (average)</td>
</tr>
<tr>
<td>(Other than Supervisors)</td>
<td>16 engineers, 1 chemist, 2 physicists</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
12. **PRECRITICAL SHAKEDOWN TESTS**

12.1 **General Objectives**

The precritical shakedown tests demonstrate proper functioning of all equipment, including instrumentation, alarms, and control systems and establish the preliminary operating characteristics of the plant. These tests assure that the plant is operable in order that tests at startup power, intermediate power (1-17 MW), and at power (17-85 MW) can be initiated.

The plant testing program is divided into five phases. Precritical shakedown tests (Phases I, II, and III) are described in this section. Initial nuclear tests and tests at power (Phases IV and V) are described in Sections 13 and 14.

12.2 **Phase I Tests**

   a. **Scope**

   Phase I tests (and inspections) include calibration of instruments, wire checking, continuity checks, dimensional checks, alignment checks, construction and erection inspection, and necessary installation checks and component tests to ensure conformance to drawings and specifications.

   b. **Organization**

   These tests are the responsibility of the construction contractor, H. K. Ferguson, with the designers, KE, A-C, and UCNC providing necessary field inspection. All necessary pre-installation calibrations of instruments and controls are performed by TVA.

   c. **Procedures**

   There are few written procedures for Phase I tests. Important checks and tests which must be made are covered in detail in the plant construction specifications.

   The responsibility for the review and/or approval of all quality control procedures, vendor tests, material certifications and other related documents that establish that the important checks and tests have been accomplished during construction has been assigned during the construction phase of the project to the respective designer of the feature, namely, KE, A-C, or UCNC. As these tests, material analyses and certifications and other related documents have been reviewed and/or approved for compliance with the design and specifications, record copies of all the important checks and tests are transferred for record purposes to the operator, TVA.

12.3 **Phase II Tests**

   a. **Scope**
These comprise pressure tests of all pressure-containing components and systems to verify structural adequacy and leak-tightness. Included in this phase is the establishment of clean conditions for various systems.

b. Organization

Test procedures for all systems are prepared by the architect-engineer, reviewed by TVA, ORNL, HKF, and AEC, and approved by TVA, ORNL, and AEC. The procedures give detailed instructions for testing the piping and equipment. Tests are conducted by HKF and will be witnessed by the architect-engineer representative at the site. TVA will also witness all of these tests. The AEC will make frequent or daily visits to the site to review progress and to observe firsthand the performance of this testing. Upon completion of the test, a report is written and approved by the witnessing organizations.

The construction contractor, HKF, and the KE field inspection organization develop details for performing Phase II pressure tests on the controls and instrument lines. These procedures are reviewed and approved by TVA, ORNL and AEC. The testing methods are reviewed and tests witnessed by Kaiser Engineers. TVA also witnesses these tests.

12.3.1 Leak and Pressure Tests

a. General

The test classification applicable to each of the piping systems is listed in Table 12.3.1. In-line equipment with a pressure rating below that permitting application of the test pressure for the piping in the system is isolated during the preliminary test. The entire system, however, is tested at the working pressure for the system during the final test (See Table 12.3.1). A complete written record of all lines tested is maintained. Each test is witnessed and certified by cognizant inspectors representing TVA, KE, and HKF. The test data is further reviewed and concurred in by AEC. In certain selected cases, ORNL is used as a consultant to AEC in the review of the test data.

The major items of equipment in the reactor coolant system, such as the reactor vessel, steam generators, reactor coolant blowers, attemperators, vessel cooling compressors and the like, are subjected to pressure and in-leakage tests in the manufacturers' shops before shipment to the site. Such units are, however, subjected to pressure tests after installation in accordance with the test procedures described herein.

b. Test Procedures

Hydrostatic tests of piping systems are conducted at ambient temperatures and, for certain critical piping, the test pressures are compensated for temperature accordingly. However, in no case does the test pressure exceed the adjusted pressure-temperature rating for 100 °F as given in the American Standard for Steel Pipe Flanges and Flanged Fittings, ASA B-16.5. The requirements for the several classes of tests are listed in Table 12.3.1.
<table>
<thead>
<tr>
<th>Piping System</th>
<th>Cleaning Class&lt;sup&gt;b&lt;/sup&gt; Prior To Erection</th>
<th>After Erection</th>
<th>Leak and Pressure&lt;sup&gt;a&lt;/sup&gt; Test Class</th>
</tr>
</thead>
<tbody>
<tr>
<td>High Pressure Superheated Steam</td>
<td>II</td>
<td>IV</td>
<td>IV</td>
</tr>
<tr>
<td>Extraction Steam</td>
<td>I</td>
<td>III</td>
<td>IV</td>
</tr>
<tr>
<td>Desuperheated Steam</td>
<td>I</td>
<td>III</td>
<td>IV</td>
</tr>
<tr>
<td>Feedwater</td>
<td>II</td>
<td>IV</td>
<td>IV</td>
</tr>
<tr>
<td>Condensate</td>
<td>I</td>
<td>III</td>
<td>IV</td>
</tr>
<tr>
<td>Air Piping at Condenser</td>
<td>II</td>
<td>IV</td>
<td>IV</td>
</tr>
<tr>
<td>Chemical Feed</td>
<td>I</td>
<td>III</td>
<td>IV</td>
</tr>
<tr>
<td>Drains and Vents:</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Superheated Steam</td>
<td></td>
<td>IV</td>
<td></td>
</tr>
<tr>
<td>Steam Generator</td>
<td></td>
<td>IV</td>
<td></td>
</tr>
<tr>
<td>Extraction Steam</td>
<td></td>
<td>IV</td>
<td></td>
</tr>
<tr>
<td>Desuperheated Steam</td>
<td></td>
<td>IV</td>
<td></td>
</tr>
<tr>
<td>Relief Valves</td>
<td></td>
<td>IV</td>
<td></td>
</tr>
<tr>
<td>Feedwater</td>
<td></td>
<td>IV</td>
<td></td>
</tr>
<tr>
<td>Condensate</td>
<td></td>
<td>IV</td>
<td></td>
</tr>
<tr>
<td>Feedwater Heater</td>
<td></td>
<td>IV</td>
<td></td>
</tr>
<tr>
<td>Air Ejector</td>
<td></td>
<td>IV</td>
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</tr>
<tr>
<td>Chemical Feed</td>
<td></td>
<td>IV</td>
<td></td>
</tr>
<tr>
<td>Building Heating</td>
<td></td>
<td>I</td>
<td></td>
</tr>
<tr>
<td>Building Drains (cold)</td>
<td></td>
<td>I</td>
<td></td>
</tr>
<tr>
<td>Circulating Water</td>
<td></td>
<td>I</td>
<td></td>
</tr>
<tr>
<td>Experimenters' Cooling Water</td>
<td></td>
<td>IV</td>
<td></td>
</tr>
<tr>
<td>Service Water</td>
<td></td>
<td>I</td>
<td></td>
</tr>
<tr>
<td>Building Heating Boiler</td>
<td>I</td>
<td>IV</td>
<td>I</td>
</tr>
<tr>
<td>Building Heating Hot Water</td>
<td>I</td>
<td>III</td>
<td>I</td>
</tr>
<tr>
<td>Blower Seal Water</td>
<td>II</td>
<td>IV</td>
<td>IV</td>
</tr>
<tr>
<td>Chilled Water</td>
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<td>V</td>
<td>I</td>
</tr>
<tr>
<td>Circulating Water</td>
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<td>III</td>
<td>I</td>
</tr>
<tr>
<td>Demineralized Water</td>
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<td>IV</td>
<td>IV</td>
</tr>
<tr>
<td>Domestic Water</td>
<td>I</td>
<td>V</td>
<td>I</td>
</tr>
<tr>
<td>Experimenters' Cooling Water</td>
<td>II</td>
<td>IV</td>
<td>IV</td>
</tr>
<tr>
<td>Fire Protection Water</td>
<td>I</td>
<td>III</td>
<td>I</td>
</tr>
<tr>
<td>Service Water</td>
<td>I</td>
<td>III</td>
<td>I</td>
</tr>
<tr>
<td>Hot Waste</td>
<td>I</td>
<td>IV</td>
<td>III</td>
</tr>
<tr>
<td>Warm Waste</td>
<td>I</td>
<td>IV</td>
<td>III</td>
</tr>
<tr>
<td>Helium</td>
<td>II</td>
<td>IV</td>
<td>V</td>
</tr>
<tr>
<td>Hydrogen</td>
<td>II</td>
<td>IV</td>
<td>II</td>
</tr>
<tr>
<td>Nitrogen</td>
<td>II</td>
<td>IV</td>
<td>II</td>
</tr>
<tr>
<td>Oxygen</td>
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<td>V</td>
</tr>
<tr>
<td>Instrument Air</td>
<td>II</td>
<td>IV</td>
<td>II</td>
</tr>
<tr>
<td>Plant Air</td>
<td>II</td>
<td>IV</td>
<td>II</td>
</tr>
<tr>
<td>Diesel-Generator Starting Air</td>
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<td>IV</td>
<td>II</td>
</tr>
<tr>
<td>BSD and PMT</td>
<td>II</td>
<td>IV</td>
<td>V</td>
</tr>
<tr>
<td>Caustic and Acid Waste</td>
<td>I</td>
<td>III</td>
<td>I</td>
</tr>
<tr>
<td>Chlorine Solution</td>
<td>I</td>
<td>III</td>
<td>I</td>
</tr>
<tr>
<td>Sulphuric Acid</td>
<td>II</td>
<td>IV</td>
<td>I</td>
</tr>
<tr>
<td>Fuel Oil, Including Vents and Drains</td>
<td>I</td>
<td>IV</td>
<td>I</td>
</tr>
<tr>
<td>Diesel Engine Exhaust</td>
<td>I</td>
<td>IV</td>
<td>I</td>
</tr>
<tr>
<td>Lube Oil, Including Vents and Drains</td>
<td>I</td>
<td>VI</td>
<td>IV</td>
</tr>
<tr>
<td>Gasoline</td>
<td>I</td>
<td>III</td>
<td>I</td>
</tr>
</tbody>
</table>

<sup>a</sup> Leak and Pressure Tests are defined below:
Class I - A preliminary test at 1-1/2 times the design pressure (125 psig minimum) is applied, using water as the testing medium. The final test is conducted at working pressure with the normal system fluid as the testing medium.

Class II - A preliminary test at 1-1/4 times the design pressure (100 psig minimum) is applied, using oil-free dry air as the testing medium. After purging operations have been completed, a final test is conducted at working pressure with the normal system fluid as the testing medium.

Class III - A static water test is applied to the drainage system by closing all openings, except the highest one, and filling the system with water to the point of overflow.

Class IV - A preliminary test at 1-1/2 times the design pressure is applied, using demineralized water as the testing medium. The final test is conducted at working pressure, with normal system fluid as the testing medium.

Class V - All piping systems, valves, fittings, and other components are shop-tested hydrostatically in accordance with ASA B-31.1-1955, paragraph 121 or ASME Boiler and Pressure Vessel Code, Section I. After erection is completed, the piping systems or parts thereof are purged, and are pneumatically tested in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, paragraph UG-100, using oil-free air with a dew point less than minus 40 F.

Acceptance testing for leak-tightness is by evacuation of the system to 50 microns Hg abs, or less, and measuring in-leakage from the atmosphere. The system is held evacuated for a minimum of 4 hours, and the pressure continuously recorded. The maximum total pressure rise acceptable during the 4-hour period, after temperature correction, is 12 microns Hg.

b Cleaning Classes are defined below:

I Wire brushed, blown with compressed air, capped until erection.

II Sandblast or shotblast (carbon steel and 304 stainless steel only), disassembled, chemically cleaned, rinsed, dried, assembled, sealed. (Carbon steel prime painted.)

III Water flushed, dried with compressed air.

IV Helium system, protected by temporary filters, flushed with helium. Other systems flushed with preliminary test medium.

V Domestic water flush, final disinfection with chlorine, followed by water flush, and sample until sterile.

VI Hot alkaline detergent wash, circulation of descaling solution, neutralizing final rinse (bearings and sensitive components are isolated) pipes are hammered during descaling and detergent wash. Final rinse of circulating hot rust inhibiting flushing oil for minimum of 48 hours with no scale trapped in strainers during a 2-hour hammering period.

NOTE: Classes III, IV, V, and VI include strainer and trap inspection and cleaning as required.

12.3.2 Cleaning Procedures

a. General
A cleanliness control program is established prior to and maintained during erection of the equipment and all related systems within the reactor building. Detailed cleaning procedures are also established and maintained for equipment during manufacture, shipment, storage, and installation, and for all piping systems prior to and after erection.

b. **Equipment**

The surfaces of all equipment intended to be in contact with helium in the reactor coolant systems—including all shop-fabricated components such as compressors, blowers, steam generators, attemperators, valves, and tanks—are fabricated and cleaned to meet the cleanliness requirements of Specification MIL-C-19874, Cleaning Requirements for Nuclear Primary Coolant Equipment Including Piping Systems. An approved cleaning procedure is established for each item of equipment. Preparation for shipment requires that the equipment be thoroughly dried, purged with dry nitrogen, and packaged in a sealed container and, where possible, pressurized with dry nitrogen to 5 to 10 psig before shipment. Vessels that can be pressurized are filled with dry nitrogen at 5 to 10 psig. The equipment is stored in the pressurized condition until it is installed at the plant.

c. **Piping Systems**

Piping material and systems are cleaned in accordance with the individual classifications listed in Table 12.3.1.

d. **Instrument Components**

Equipment and material which will be in contact with helium gas is fabricated, cleaned, inspected, handled, stored, and packed by methods that eliminate the possibility of contamination of the helium. Following cleaning, all surfaces are dried thoroughly and are enclosed in a sealed, impermeable plastic covering, along with adequate desiccant, to maintain the dryness of the enclosed air.

Other equipment and material not requiring the above degree of cleanliness are fabricated, cleaned, handled, inspected, stored, and packed by methods which produce commercial grades of cleanliness.

12.3.3 **Containment Test**

The containment test is performed in the same manner as the leak-rate test described in Section 5.3.5, Volume I, EGCR Final Hazards Summary Report, except that all penetrations, valves, and relief devices are in place and are operable. Initial pressurization is to approximately 3 psig for soap bubble testing of all joints, valves, and seating surfaces that have been added since the initial test. After all leaks at 3 psig have been repaired or reduced to a minimum, the pressure is raised in increments to the leak-rate test pressure of 9 psig. Soap bubble testing is made after each increment of pressure increase.

Prior to pressurizing the containment building, the air lock doors are operated through a minimum of five complete cycles to determine that the doors, interlocks, signal lights, energizing devices, and valves operate correctly, and that bearings have been lubricated.

During the pressurizing of the containment shell, the seals of all doors are checked for leaks with soap solution at each pressure level.

12.4 **Phase III Tests**
a. **Scope**

These comprise all operational tests, that are practicable, for components and systems prior to fuel loading and provide maximum assurance that the plant is ready for fuel loading. Heat can be added to the reactor coolant system by operation of the reactor coolant blowers, permitting operational tests at temperatures of 450 to 500 F. This source of heat and a portable steam boiler available for use during construction tests will permit hot tests on the steam plant prior to raising steam with nuclear heat.

b. **Organization**

Most of the Phase III test procedures are prepared by the architect-engineers with a small portion prepared by TVA or UCNC. Procedures are reviewed by HKF, TVA, ORNL, and AEC, and approved by TVA, ORNL, and AEC. TVA has been assigned the responsibility for direction of all Phase III testing.

The TVA has designated engineers to act as the person responsible for each of the various systems and components of the plant. These persons perform the various pretest tasks associated with these systems, including review of the Operating Manual and test procedures, and providing technical support during the tests. In all cases, however, the operating superintendent retains responsibility for operation of all equipment and for safety of personnel. The assistant to the operating superintendent schedules and coordinates the individual tests between the Technical Program and Operations Group. Over-all planning and coordination of Phase III testing is under direction of the technical program superintendent. Changes or alterations to the test procedures which do not alter the test objectives can be approved by the designated TVA test engineer subject to later approval by the other organizations concerned and AEC. Changes which alter test objectives require prior approval by others, including AEC. Employees of HKF, KE, A-C, ORNL, and AEC are utilized in the test program where special knowledge and consultation are required. Engineers from KE and A-C are assigned to the Project at the site during the period of Phase III tests. In addition, engineers from ORNL and AEC will participate in certain selected Phase III testing. These persons, together with the ECCR operating superintendent and technical program superintendent, comprise a group for reviewing the results of all tests, recommending test acceptance, recommending additional tests, and approving deviations from the approved procedures.

12.4.1 Reactor Coolant System

a. **General**

The major part of the tests on the reactor coolant system are performed with the system pressurized with helium. The blowers are operated and the resulting heat of compression is used to raise system temperature to approximately 500 F. The design pressure rise across the blowers is obtained by throttling the isolation valves in the reactor coolant outlet piping since the reactor coolant system pressure drop, in the absence of fuel, will be lower than the design value. Excess thermal energy input to the helium gas by the blower operation is removed in the steam generators to control reactor coolant system temperature. System pressure drops, temperatures, pressures, and flows are obtained under test conditions along with other checks of component operation to determine that the system is operating satisfactorily.
Whenever practicable, off-normal operating conditions are simulated and the system response and blower unit performance during these transients are observed and analyzed carefully.

Particulate matter such as graphite dust, rust, weld spatter, and lint may remain in the coolant system despite intensive efforts to maintain clean conditions. In addition, the phosphate coating on the interior of the reactor coolant system will decompose during the circulation runs. One of the decomposition products is a finely divided, phosphate-containing powder. Temporary filters are installed in the reactor coolant system piping to prevent damage to system components.

b. **Leakage**

System leak-tightness is verified, as necessary, before proceeding with the tests.

c. **Relief Valves**

The main relief valves for this system are tested as a part of the gas vent system.

d. **Reactor Coolant Blowers**

The main drive motors and the reactor coolant blowers receive extensive shop tests, witnessed by a qualified inspector representing the AEC, before shipment to the job site. Shop tests of the motors are designed to demonstrate conformance with basic equipment specifications and with the applicable AIEE codes. The shop tests of each of the reactor coolant blowers include single-loop tests in helium at design temperature and 35 psig pressure, over the design speed range, to establish the complete operating characteristics of the blowers and to verify performance of the shaft running seals. The variable speed couplings are operated for eight hours at the factory to demonstrate compliance with manufacturing standards. At least one of the speed increasers is tested at the EGCR site under the supervision of the blower manufacturer to demonstrate operability, gear tooth backlash, and horsepower losses.

Precritical testing of the completely assembled blower units is performed to determine the following:

1. The actual operating characteristics of the units as installed, over the full design speed range from 20 percent to 110 percent of nominal rated flow, when circulating helium at a range of pressure rises approximating those existing under normal operating conditions. These blower characteristics will be obtained under the pressure, temperature, and flow at test conditions when operating singly or in parallel.

2. The operation of the blower units during automatic rampdown in various ranges from 110 percent to 20 percent rating.

3. The blower coastdown time and the operation of the blower units during coastdown in various ranges from full speed to zero speed.

4. The accuracy and speed response of the blower speed and flow control systems, including blower differential speed controls.

5. The adequacy and operating characteristics of both the static and running seals.
6. The proper functioning of all systems auxiliary to the blowers, such as the lubricating oil and blower seal water systems.

7. The verification of the starting procedure.

e. **Piping and Expansion Joints**

Position, alignment, and movement of expansion joints and piping are established and verified as much as practicable during these tests.

f. **Isolation Valves**

Tests are conducted on the isolation valves to determine the closure time and to verify operation of interlocks controlling valve operation. In addition, the capability of the isolation valves to hold pressure across the valves is determined. Valve stem and seat leakage is also determined during the tests.

12.4.2 **Steam Plant**

A portable boiler is used to supply a minimum of 30,000 lb/hr of steam at a pressure of about 400 psig and a temperature of approximately 700 F. Steam from this boiler is used initially to blow out the entire steam system piping, and then is used functionally (within the temperature and pressure limitations noted above) to test the automatic and manual operation of the pressure and temperature controls of heat exchangers and other equipment.

Sufficient tests are included to provide satisfactory assurance that the shut-down or decay heat dump system operates in accordance with design requirements. Tests of controls and controls systems associated with the steam plant are described in Section 12.4.8.

Testing and verification of the operation of the steam generators, main steam system, and turbogenerator under design conditions are accomplished during the Phase IVc and Phase V testing described in Sections 13 and 14.

12.4.3 **Electrical System**

a. **Electrical Systems Excluding Emergency Power**

To facilitate the testing of both the electrical equipment and the other plant systems, electrical Phase III tests are conducted in parts. Each section of the bus circuitry, line, motor tests, and protective relaying is tested as the necessary construction is completed.

The electrical systems for Phase III are generally tested in the following categories:

1. **X-10, EGCR Transmission Line and Auxiliary Equipment**

Tests in this category include:

(a) Operational tests to verify proper operation of the transfer trip circuitry.

(b) Simulated faults on the open line between X-10 and the EGCR (including the EGCR main transformer) to verify that the pilot-wire differential relay functions properly.

(c) Tests to verify proper operation of overcurrent relays on the
transformer at the EGCR which will trip the X-10 line isolating transformer circuit breaker S-1340.

(d) Tests to verify proper operation of the transformer ground overcurrent relay protection.

(e) Calibration and functional tests to assure that indicating and recording instruments are operating properly.

2. **Main 13.8 kv Bus**

Phase III tests consist of operational tests to assure that the protective devices of the equipment connected to the 13.8 kv bus through circuit breakers are functioning properly, and that the circuit breakers for each item of equipment function correctly. The air circuit breaker S-1340 is tested at this time for proper functioning, including the opening and closing times.

Protective equipment includes bus differential protective relays which, for a fault on the 13.8 kv station bus, open the X-10, EGCR line air circuit breaker S-1340, the main generator breaker S-1360, and all the air circuit breakers which connect feeder circuits to the 13.8 kv station bus. These latter include the experimenters' transformers, the circuits to the 13.8 and 2.4 kv buses, and the circuits of the 3100 hp reactor coolant blower motors. This equipment is tested by simulating faults that normally would cause the differential protecting equipment to function. In addition, the undervoltage and phase sequence relay on the 13.8 kv bus is tested.

The protective equipment of the main generator, consisting of differential relaying, current balance relaying, time overcurrent relaying, loss-of-generator excitation relaying, reverse power relaying, ground overcurrent relaying, and the tripping of the turbine stop valve on the action of breaker S-1360 is tested under Phase III by means of a phantom or simulated fault test. Simulated fault tests consist of determining actual parameter values that cause relay operation, determining that the relay does operate properly, and determining that the contacts of the relay, when operated, trip the circuit breaker associated with the protection of the equipment concerned. Protective equipment for the blowers and auxiliary transformers is tested in a similar manner.

In the Phase III tests of the 13.8 kv bus, the metering of the main transformer and main generator, including thermal converters, ammeters, watthour meters, voltmeters, frequency meters, and protective relays, is tested for correct functional operation. The equipment will be calibrated by introducing phantom loads, currents, and voltages to the instruments and comparing the meter readings with calibration standards.

3. **2.4 kv Bus**

The tests on the 2.4 kv EGCR buses consist of functional operation of the breakers on the 2.4 kv side of the 13.8-2.4 kv transformer, the bus tie breakers, and the related protective relaying. The breakers are tested for proper operation and timing, and to assure that the individual breakers are operated by protective equipment in the manner according to design for simulated faults. In the case of breakers S-2403 and S-2412, which are associated with 13.8-2.4 kv transformers, the simulated fault will consist of bus differential
operation and transformer differential operation. In the case of the bus feeder breakers, the faults simulated are bus differential operation and feeder circuit overcurrent operation. Bus differential and overcurrent faults, which normally open bus tie breaker S-2407, are simulated. Undervoltage conditions which normally cause the loss-of-power scram of the reactor and which are detected by undervoltage relays are simulated. Feeder breakers on the 2.4 kv bus to the feedwater pumps, circulating water pumps, fire protection water pump, and the main generator exciter are tested functionally.

4. Normal Load Centers

Circuit breakers associated with the normal load center of bus A and bus B are tested functionally in a manner similar to that described for the 2.4 kv buses. Inductive-type relaying, or protective devices associated with bus A and bus B of the normal load center, are operated manually to determine that the associated circuit breakers function properly. Such relay equipment including undervoltage, overcurrent, and time delay relaying are tested to determine their characteristics before the Phase III tests are performed. The relay contacts are manually operated to show that, when these relays function, it causes the desired action of transferring loads from one load center to another, or simply opens circuit breakers to remove faults from the equipment.

The electrical equipment connected to motor control centers 4, 5, 6, and 7 as feeders will have current characteristics, voltage characteristics, and load data recorded as part of the test of the Phase III system to which this equipment is connected.

5. Station Main Generator

Phase III tests for this part of the electrical system are described under plant control system, Section 12.4.8.c.

b. Emergency Power Systems

The purpose of the Phase III tests for the emergency power systems is to verify the operation of the automatic transfer circuits, the automatic start circuits, and the time-delay load application and load transfer circuits with the power circuits energized and loaded. The test verifies the mechanical operation of both diesel engine-driven generators, as well as the proper functioning of the auxiliary systems and alarms. The tests establish the ability of each diesel engine-driven generator to pick up the connected loads within specified times without exceeding the established voltage and speed limits. All tests are conducted with 90 percent of the connected load running to duplicate transient conditions which would exist under actual operating conditions.

1. Diesel Engine-driven Generator Tests

The test is conducted for the following diesel engine-driven generator conditions: stand-by, starting, running, and shutdown. Stand-by tests include operation of the stand-by heating systems and alarms. Starting tests include automatic and manual starting and a starting air capacity test. Running tests include normal operation under engine speed governor and voltage regulator control, operator throttle control, and tests of the stand-by jacket water, lube oil, and fuel oil pumps. Shutdown tests include normal shutdown after power is restored and emergency shutdown.
2. **Motor Control Center Tests**

The following tests are made on the emergency motor control centers 1, 2, and 3:

(a) Loss of voltage on bus A
(b) Loss of voltage on bus B
(c) Simultaneous loss of voltage on buses A and B
(d) Restoration of power to bus A only
(e) Restoration of power to bus B only
(f) Restoration of power to buses A and B simultaneously.

3. **Emergency Load Center Tests**

(a) Loss of voltage on buses A and B--both diesel engine-driven generator units start.
(b) Loss of voltage on buses A and B--diesel engine-driven generator 1 fails to start.
(c) Loss of voltage on buses A and B--diesel engine-driven generator 2 fails to start.
(d) Loss of normal power while diesel engine-driven generator 1 is on "Test."
(e) Loss of normal power while diesel engine-driven generator 2 is on "Test."
(f) Loss of normal power while diesel engine-driven generator 1 is supplying power to bus A.
(g) Loss of normal power while diesel engine-driven generator 2 is supplying power to bus B.

### c. Failure-Free Power System

Phase III tests for the failure-free power system verify that the battery can carry the design load for an 8-hr period and can maintain an output voltage above specified limits. The tests also verify that each of the failure-free motor-generator sets will carry the load of the two instrument supply panels MG-1 and MG-2 and will maintain an output voltage and frequency within prescribed limits.

The ability of the switchgear control power supply and emergency lighting battery to operate under design conditions is verified.

1. **Switchgear Control Power Supply**

The switchgear control battery, N-81, Phase III tests will verify that the battery can supply the design load for eight hours and can maintain an output voltage within prescribed limits.

The following tests will be made to verify that the d-c control power to the following equipment will automatically transfer to the alternate source panel XI:

(a) 13.8 kv switchgear d-c transfer
(b) 2.4 kv switchgear d-c transfer
(c) 480 v emergency load center d-c transfer, buses A and B
(d) Diesel-generator local panels 1 and 2 d-c transfer
(e) Turbine-generator field control panel N-8 d-c transfer
(f) Duplex switchboard d-c transfer
(g) Main control room panel K-1-4 d-c transfer
(h) Main control room panel K-1-5 d-c transfer
(i) Motor control center 1 d-c transfer
Motor control center 2 d-c transfer
Motor control center 3 d-c transfer.

2. **Emergency Lighting Battery**

The Phase III tests on the emergency lighting battery N-82 verify that the battery can supply the design load for an 8-hr period and can maintain an output voltage within prescribed limits.

12.4.4 **Charge Machine**

The Phase III tests include alignment and attachment checks for all reactor charge nozzles, rehearsal area nozzle, spent fuel nozzle, and shield plug loading nozzle. However, to reduce wear on the machine internal mechanisms, the functional tests of fueling and adjusting of the bottom dummy orifices are made with the machine attached to only one reactor nozzle.

The reactor coolant system is charged with helium at 315 psia and the charge machine coolant temperature is maintained at 350 F. The machine and auxiliary systems are tested for performance and ability to meet design requirements.

The charge machine tests include the following:

- **Position tests** to verify the alignment of the bridge and carriage with the nozzles and the proper operation of the transit and cross-hair alignment system. Attachment to the nozzles to check operation of the position lights and interlocks prohibiting movement.
- The capability of the vacuum system to lower pressure in charge machine vessel to 1.5 mm Hg. Pressurizing the charge machine with helium from either the high-pressure storage cylinders or the transfer compressors.
- Leak test attachment joint, service line connections, and charge machine vessel flanges.
- Test all controls, instruments, and equipment required for startup, including portions of those systems associated with the charge machine.
- Test transfer of controls from charge machine room to charge machine control room, including "strong box" installation. Check warning lights and door locks.
- Test shield plug unloading operation including controls, interlocks, and alarms.
- Test entire fuel loading operation including controls, interlocks, and alarms associated with placement and orientation of transfer tube and latch assembly, orientation of magazine, and capability of fuel ram to position six fuel assemblies and bottom dummy in the core. Test load and position-indicating devices for the fuel ram. Verify that proper helium flows can be maintained during fuel loading operation.
- Test orifice adjusting capability of fuel ram. Establish the position of the ram plunger in relation to the indicated instrument readings.
- Test unloading operation including controls, interlocks, and alarms associated with removing six fuel assemblies and bottom dummy from the core and storage in the charge machine, orientation of fuel ram, orientation of magazine, and removal and storage of transfer tube. Check effect of helium flow on fuel ram weight-sensing readout system.
j. Test shield plug loading operation including controls, interlocks, and alarms.

k. Repeat steps f through j using alternate control, indicating, and alarm circuits. Test transfer tube boom under different fuel channels.

l. Leak-test the bottom shield plug gaskets.

m. Test capacity and operation of auxiliary blower system on charge machine vessel.

n. Test ability of shaft seals on blister drives to seal against 315 psia internal pressure of charge machine. Vent drive blister to atmospheric pressure and check seal leakage.

o. Discharge helium to low-pressure storage system. Using transfer compressor, pump charge machine down to 20 psia. Reduce charge machine helium pressure to one atmosphere by venting to the stack. Check ability of systems to perform these functions.


q. Test spent fuel discharge procedure.

r. Test new fuel loading in rehearsal area.

12.4.5 Service Machine

The Phase III tests include alignment and attachment checks for all reactor top nozzles, rehearsal area nozzles, spent fuel nozzle, and storage hole nozzles. To reduce wear on the machine internal mechanisms, tests of servicing operations are made with the machine attached to only one reactor control rod nozzle.

The reactor coolant system is charged with helium at 315 psia and the service machine coolant temperature is maintained at 125 F. The machine and auxiliary systems are tested for performance and ability to meet design requirements.

The service machine tests include the following:

a. Position tests to verify the alignment of the bridge and carriage with nozzles and the proper operation of the vertical alignment mechanism. Checks track shear lock and interlocks prohibiting movement of service machine bridge and carriage. Attachment to nozzles to check interlocks, controls, and indicating instrumentation.

b. The capability of the vacuum system to lower pressure in the service machine to 1.5 mm Hg. Pressurizing the service machine with helium from either the high pressure storage cylinders or the transfer compressors.

c. Leak test attachment joint, service line connections, and service machine vessel flanges.

d. Test all controls, instruments, and equipment required for startup, including portions of those systems associated with the service machine.

e. Check door locks, radiation warning lights, and warning alarms. Transfer control to service machine control room.

f. Check turret indicating instrumentation for closure tool position. Lock
turret and open valve. Lower closure tool. Unlock nozzle closure. Retract closure tool. Check controls, interlocks, position indications, and alarms associated with these operations.

g. Close valve and unlock turret. Rotate turret to shield plug removal tool position. Lock turret and open valve. Lower tool and grapple shield plug and control rod assembly, including shroud, and raise into service machine. Check controls, interlocks, position indicators, and alarms associated with these operations.

h. Check for correct helium flow through machine.

i. Close valve and unlock turret. Rotate turret to fuel chute position. Lock turret and open valve. Lower fuel chute to full-down position. Index basket. Check controls, interlocks, position indicators, and alarms associated with these operations.


k. Check operation and handling of temperature calibration instrument.

l. Repeat step j in reverse, and unload fuel assemblies from the core and store in the spent fuel cask in the service machine.

m. Test ability of fuel hoist to grapple bottom dummy and store in the service machine.

n. Raise fuel chute to the full-up position. Close valve and unlock turret. Rotate turret to the plug installation position. Lock turret and open valve. Check controls, interlocks, position indicators, and alarms associated with these operations.

o. Check backflow effect of helium from the reactor vessel top plenum into the service machine under simulated failed vessel cooling compressor operation. Check operation of emergency helium coolant supply on loss of vessel cooling supply.

p. Lower plug installation tool and replacement shield plug, control rod assembly, and shroud to the full-down position. Release grapple. Raise plug installation tool to the full-up position. Check controls, interlocks, position indicators, and alarms associated with these operations.

q. Close valve and unlock turret. Rotate turret to nozzle closure position. Lock turret and open valve. Check controls, interlocks, position indicators, and alarms associated with these operations.

r. Lower nozzle closure tool. Lock nozzle closure. Retract closure tool. Close valve and unlock turret. Check controls, interlocks, position indicators, and alarms associated with these operations.

s. Repeat steps f through r using alternate control, indicating, and alarm circuits.

t. Depressurize expansion joint. Check tightness of nozzle closure.
Depressurize service machine. Check capability of helium transfer system. Disconnect service machine from reactor nozzle and install blind flange closure. Check tightness of blind flange closure joint.

u. Deposit shield plug, control rod assembly, and shroud in a storage hole. Select some phases of these operations and use the manual drives associated with the drive blisters. Check controls, interlocks, position indicators, and alarms associated with all these operations.

v. At spent fuel shaft, transfer six fuel assemblies from the service machine spent fuel cask to the spent fuel transfer mechanism. Utilize experimental tool turret position for this operation. Check controls, interlocks, position indicators, and alarms associated with these operations.

w. Test new fuel loading at the rehearsal area. Transfer and store a top dummy assembly and six fuel assemblies in the service machine. Check controls, interlocks, position indicators, and alarms associated with these operations.

x. At the rehearsal area, test the ability of the fuel grapple to remove a simulated stuck fuel assembly.

y. Remove fuel grapple and install neutron source grapple at the rehearsal area. Test ability of neutron source grapple to latch to an unirradiated neutron source.

z. Install special control rod shroud grapple on the experimental tool. Position and attach service machine to a reactor control rod nozzle. Remove shield plug and control rod assembly from the reactor, leaving the shroud in place. Test ability of experimental tool grapple to retrieve shroud from a control rod nozzle and store shroud in the service machine. Check all controls, interlocks, position indicators, and alarms associated with these operations.

12.4.6 Auxiliary Systems

Each of the auxiliary systems is functionally tested during Phase III. These tests include setting of relief valves, additional calibration and check of instruments and controls not completed during Phase I, and operating the system as a unit to the extent possible. Specific tests designed to demonstrate system and plant characteristics are described in the following paragraphs;

a. Vessel Cooling System

The vessel cooling compressors are shop-tested, individually, in a helium loop at design pressure, temperature, and flow. A complete compressor map is obtained. The performances of the lubricating oil system, static seals, and running seals are evaluated.

Precritical testing of this system includes the following:

1. Determination of the operating characteristics for each compressor after it is installed in its operating location.

2. Establishment of initial settings of the manual globe valves which control flow to the nozzles.

3. Automatic startup of the stand-by compressor.
4. Loss of normal power to the operating compressor to check the automatic transfer to emergency power and the proper closure of the reactor coolant inlet isolation valves.

5. Tests to confirm the functioning of excess flow valves and check valves for simulated ruptures at various locations.

6. Tests to confirm the functioning of controls and alarms for normal and abnormal operating conditions.

b. **Blower Seal Water System**

In addition to normal operating tests and checks, these tests demonstrate that one blower seal water pump can supply the system with seal water, and that the high-pressure head tanks can supply the system seal water for a minimum of 30 minutes with the blower seal water pumps not operating. Also, stand-by filters and dryers are replaced with the system in operation. Control system operation during a simulated pipe rupture or loss of pumps is demonstrated.

c. **Helium Purification System**

The blower seal leakoff compressors are tested while circulating helium in a closed loop to and from the vessel cooling system to verify manual and automatic operation of the compressors. Functional operation of the coolers and the performance of the dryer is verified. The operating pressure drop across the filters is established.

The purification system is tested as a unit during a period in which helium is circulated in the reactor coolant system.

During operation of the reactor coolant system, outgassing of the moderator graphite occurs, releasing impurities such as H₂, CO, CH₄, CO₂, H₂O, and N₂. When the combined concentrations of the oxidizable impurities H₂, CO, and CH₄ reach approximately 1000 ppm by weight, the catalytic converter is tested. If the combined concentration is substantially less than 1000 ppm, hydrogen from a cylinder is added to the coolant. Performance of the catalytic converter in removing the oxidizable impurities is determined. Of particular interest is the oxygen content of the converter effluent.

The performance of the CO₂ adsorber in removing CO₂ from the coolant is determined. Regeneration of the adsorber is demonstrated.

The operation and regeneration of the helium dryers are demonstrated.

d. **Gas Recovery System**

The operability of flow elements and flow alarms is verified by flowing helium through the system. The flow elements are calibrated by the manufacturer. Functional operation of the low-pressure leakoff recovery compressors is demonstrated.

e. **Gas Vent System**

The precritical testing of the gas vent system is conducted primarily to verify proper operation of the reactor coolant system relief valves, and the three-way diversion valves, and secondly, it determines pressure drop data (clean) through the filtering systems.
Proper operation of the relief valves is demonstrated by pressurizing each valve individually with helium from a high-pressure tank until the valve lifts. The helium source is then removed and the valve allowed to reseat.

Proper operation of the three-way diversion valves is demonstrated by manipulation of the control switch and observing valve actions. The valve actions upon a containment isolation signal due to low coolant pressure and high-stack radioactivity are verified in other tests.

Pressure drop data (clean) is established for the main relief valve filters and the controlled vent filters by passing 100 lb/hr of plant air through each bank of filters and measuring the pressure drop. During testing of the evacuation system, the pressure drop (clean) is measured across the atmospheric filters by operating the vacuum pump at the rated capacity of 300 cfm.

f. Helium Transfer and Storage Systems

The helium receiving and storage system is tested by receiving helium from a transfer trailer and witnessing the functional operation of all instruments and controls. The delivery of emergency helium to the service machine from storage is demonstrated.

The helium transfer system is tested by performing the design functions of the system while transferring helium. The capability of the transfer compressors is demonstrated. Valve operation resulting from signals from radiation monitoring equipment is demonstrated.

g. Helium Evacuation System

Tests are performed to establish the capability of the evacuation system: to evacuate the charge machine to the design pressure level in the required time, and to determine the actual pressure-time curve for this system during evacuation; to evacuate the reactor coolant system and the auxiliary helium-containing system, and to establish the actual pressure-time curve for the system during this operation; and to evacuate the service machine, and to establish the actual pressure-time curve for the system during this operation.

h. Containment Shell Spray Cooling System

Phase III tests on this system consist of operating the system to check the proper functioning of valves, instrumentation, and flow distribution to ring headers and spray nozzles. Automatic startup of the system is verified by causing isolation of the containment shell.

i. Biological Shield Cooling System

The biological shield cooling system is tested to verify the system's ability to provide a flow of cooling air over the biological shield surface, and to circulate the gaseous contents of the reactor building during containment isolation. The tests verify the automatic operation of the fans, fan isolation valves, containment isolation valves, filter differential pressures, and system control circuitry.

j. Hot and Warm Radioactive Liquid Waste

Functional operation of the controls and equipment of the hot and warm radioactive liquid waste system is demonstrated by the transfer of water...
as required. A test of the system used to transfer waste from tank F-15 to a hot waste transfer truck is performed by making a water transfer.

The ability of sampling equipment to provide a sample of the contents of a waste tank is verified by actual operation of that equipment for each waste tank. The filling rate of the sample bottle at each sample station is also established. It is also demonstrated that the heating and ventilating system exhaust fans are capable of maintaining the specified negative gage pressure levels in the waste tanks.

k. Decontamination Fluid Supply System

Each reagent tank is filled with water, and during filling, the mixer is checked and the capability of the heating coil established.

Each of the reagent pumps is tested while pumping water to the reactor building charge and service machine rooms. Pumping rates to each location are established and pressure indicators checked. Proper functioning of the tank level switches in stopping the pumps is verified.

Operation of airbreak suction tank F-50 is tested in conjunction with pump J-60. Performance of the level control system is verified.

The capability of the system to deliver hot rinse water to the charge and service machine rooms is verified.

l. Heating and Ventilating System

Along with the normal functions of a heating and ventilating system, this system is tested to verify its ability to perform those functions unique to a nuclear plant. Design flows, temperature distributions, and proper control systems operation are verified with the entire system, including hot water generator C-7 operating under various load conditions. That portion of the system within the reactor building is tested with the containment shell open and also with the containment shell isolated.

Automatic isolation of the control room and heat removal during isolation is tested.

m. Air Systems

The principal purpose for testing the air systems is to demonstrate that the compressors can maintain minimum header pressure while supplying the combined design loads of instrument and plant air. The tests also demonstrate that the plant and instrument air receivers capacity is capable of supplying 100 cfm of instrument quality air for a period of 30 minutes at a pressure no lower than 60 psig with the compressors isolated. It is demonstrated that sufficient air flow and pressure are provided to remaining instrument air headers when one instrument air header rupture is simulated.

n. Service Water System

The precritical testing of the service water system is conducted to verify the proper operation of the fire protection water backup supply and system response to a loss of electric power or instrument air. The test also establishes system pressures and flows for each of these conditions.
o. **Demineralized Water System**

After the demineralized water plant has been placed in operation by the manufacturer's engineer, the ability of the plant to supply water of the required quality at the specified flows is verified. Distribution of demineralized water from the storage tank to the spent fuel storage basin, shutdown feedwater pumps, shutdown make-up pumps, diesel-engine jacket water system, and the main condenser is demonstrated.

p. **Fire Protection System**

The precritical testing of the fire protection system is conducted to demonstrate the proper operation and capacities, where applicable, of the fire protection system pumps and supervisory controls, fire protection water storage tank instrumentation, pressure relief valve, hydrant delivery, heat exchanger (C-40), and CO₂ system activating controls.

q. **Alarm Systems**

The systems tests are performed on the fire alarm and evacuation alarm systems separately. Component tests are performed by the manufacturer. Complete operational tests are performed after final installation.

The fire alarm system actuating devices are operated to assure proper alarm initiation. These tests show that the coding, coding repeats, data print recorder, control cabinet annunciator, control room annunciator, local CO₂ alarms, gongs, horn, and the master alarm coding to the X-10 fire alarm system operate properly. The system's internal monitoring circuits, battery, battery charger, and gong control voltage loss supervision are checked for proper operations. Supervisory alarm actuating devices are operated to ensure proper alarm initiation.

The evacuation alarm system is actuated by the radiation monitoring evacuation alarm selector switch and containment shell isolation circuits to ensure actuation of evacuation codes.

r. **Fuel Handling System**

Operations performed in handling new and spent fuel outside the reactor are tested using simulated EGCR fuel assemblies. In conjunction with testing of the charge and service machines, loading of fuel into these machines is demonstrated.

Transfer of fuel from the charge and service machines to the spent fuel transfer mechanism is verified. Discharge of fuel from the transfer mechanism to the spent fuel storage basin is tested. Proper functioning of interlocks and position indicators is verified. The capability of maintaining a negative gage pressure inside the spent fuel transfer shaft is verified.

Handling of fuel within the storage basin is tested with the basin empty and flooded. Proper functioning of the long-handled tools is verified. Simulated fuel assemblies are dismantled using the dismantling device, and the force required to accomplish dismantling is measured.

The bypass demineralizer pump, basin cleaning pump, and monitoring pump are tested, and flows measured by pumping to waste sump tank F-14 and determining build-up rate. The pressure drop across filters is established.
The spent fuel monitor is tested by adding approximately ten microcuries of radioactive iodine to water being circulated through the monitor.

s. Reactor Vessel Components and Internals

A visual inspection is performed of the accessible reactor vessel internals and nozzles for any apparent damage or interferences which may occur during the prior Phase III tests. These previous Phase III tests include heatup and cooldown of the reactor coolant system.

t. Hydrocarbon, Gas, and Moisture Analyzers

The hydrocarbon analyzer sequentially analyzes helium discharged from each of five diaphragm compressors. The analysis is made to detect the presence of oil resulting from a ruptured diaphragm. In Phase III tests cylinders of helium and methane, together with flow meters for each, are used to supply various concentrations of hydrocarbon to the analyzer to verify the instrument calibration and to verify proper functioning of alarms. During tests of the reactor coolant system (Section 12.4.1) and the helium transfer system (Section 12.4.6.f), the analyzer is placed in operation, and performance of the sequential sampling system verified.

The gas analyzer is a gas chromatograph which sequentially analyzes helium in the reactor coolant system, the catalytic converter discharge, and the CO$_2$ adsorber discharge. Concentrations of H$_2$, CO, CO$_2$, CH$_4$, O$_2$ + N$_2$, and H$_2$O are measured and recorded. The performance of the analyzer is verified using a calibration gas which contains known concentrations of each of the constituents.

Three moisture analyzers measure moisture content in the helium leaving each of the steam generators and the gas purification and recovery system. Each of these instruments is tested using a cylinder of nitrogen containing a known amount of water vapor.

u. Burst Slug Detection System

The precritical tests performed on the BSD system include vendor pre-delivery tests on all components, vendor tests on the detection and readout instrumentation subsystem, recalibration of all process instruments at the site prior to installation, and Phase III testing after installation.

The Phase III tests verify that:

1. All gas handling components such as selector valves, flow control valves, valve manifold assemblies, temperature switches, compressors, and associated alarms function properly.

2. All electrically operated circuits are wired according to the schematic diagrams and that all indicators and controls are labeled correctly.

3. The detection and readout instrumentation such as monitors, programmers, high-count decoder, scalers, printers, recorders, and associated alarms function properly.

4. The gas handling components and the detection and readout instrumentation are properly coordinated and function properly as an integrated system.
5. The BSD system samples, detects, and locates a source of fission products within a reactor fuel channel.

v. Pneumatic Temperature Monitoring System

The tests consist of four functional checks on the system. These are: to confirm the correct operation of the sampling system, the sample cooling system, and the programmer; to confirm that the line being sampled is correctly identified on the printout; to make an approximate check on the temperature at each fuel channel exit; and to check the operation of the alarm trips. The entire test is conducted with the reactor at approximately 500°F and at a helium pressure of approximately 300 psia.

w. Flux Scanning System

The precritical tests on the flux scanning system include the vendor tests and the Phase III tests after installation. A portion of the vendor tests determines the operability of all electronic components without failure during a burn-in period. The balance of the vendor tests center on a mock-up of the worst tubing configuration. The Phase III tests after installation verify the following:

1. That all circuits are wired according to the schematic diagrams and that all panel indicators and controls are labeled correctly.

2. That all mechanical components such as ion chamber and flux wire connectors, transfer devices, storage drums, and drive mechanisms function properly.

3. That all electrical components such as amplifiers, discriminators, counting circuits, pointers, meters, and recorders operate correctly with simulated signals.

4. That all automatic control circuits operated by the position transmitters, timing devices, counting devices, or logic elements operate properly. An example of this type of device is the adjustable limit switches operated by the position transmitters.

5. That the flux scanner system functions properly in all modes of operation and that the tape readout device indicates the proper guide tube code, time, position, and count information using a gamma source to simulate induced radiation in the flux wire.

x. Temperature Monitoring System

All instruments, components, and auxiliary devices are vendor-mounted, wired, and tested as an assembly before shipment for installation in the main control room at the EGCR. After installation, but prior to Phase III tests, the complete system is checked for proper identification and labeling of all circuits. All thermocouples are checked for continuity, resistance to ground, and total loop resistance.

The system is then subjected to a cold and hot test. The cold test is performed while the reactor coolant system is at ambient temperature, whereas the hot test is performed with the reactor coolant system operating between 450 and 500°F.

Scanning provisions are verified prior to the Phase III tests and also during the cold and hot tests.
y. **Helium Leak Detection System**

Exclusive of the system tubing and external wiring, the helium leak detection equipment is furnished as a packaged unit. The vendor is responsible for the procedures required for shop and field testing, the performance of shop testing, and the supervision of field testing. After installation the following specific items are checked and tested:

1. The completeness of the tubing runs from the sampling points to the helium leak detection cabinet
2. The completeness and proper functioning of the wiring from the helium leak detection cabinet to the annunciators, indicators, and switches located in the main control room
3. The proper operation and capacity of the sampling pumps
4. The proper sequencing of the commutating valves and indication of the valve position, both locally and remotely
5. The calibration of the integrated system
6. The drift of the integrated system
7. The accuracy of the integrated system by using a known helium-air mixture and injecting a quantity at accessible sampling points.

z. **Radiation Monitoring System**

Components of the system which are integral parts of the reactor safety system are tested simultaneously with other equipment to assure sequential operation. The stack monitors are tested with calibrated sources to establish automatic containment isolation set points. A standardized radiotracer is injected into the stack to verify flows, detector sensitivity, and geometry.

Detectors in the liquid waste monitoring system, coolant loop monitoring system, and gas vent system are checked with calibrated sources.

Continuous air monitors, remote area monitors, hand and foot counters, and local rate meters are checked for sensitivity and calibrated with standard sources to verify conformity to specifications and proper operation. The main control room multipoint recorder for the ORNL perimeter air monitoring system is synchronized with the master transmitter located at ORNL.

Meteorological instrumentation is shop calibrated and tested before installation and functional response is verified during and after installation.

12.4.7 **Nuclear Instrumentation**

The nuclear instrumentation system is subdivided into detectors, detector positioning mechanisms, pulse and current signal amplifiers and meters, and the readout meters and recorders. The system is given a complete functional test in the vendor shop in accordance with approved procedures.

During Phase III tests on the system neutron and gamma sources are applied to the detectors, and the flux-level readout of each channel is checked for proper operation. Appropriate test signals are injected at various points in the
channels to verify proper performance. Circuit leakage currents are measured with an electrometer, and signal pulse shapes are observed on an oscilloscope for adequate rise and decay times.

12.4.8 Plant Control System

a. Control Rod Drive System

The precritical tests on the control rod drive system are performed in five stages as follows:

1. Vendor tests on a prototype mechanism are performed before the operational drives are fabricated. The vendor tests on the prototype mechanism are conducted at ambient temperature and pressure and include:

   (a) Component tests on the drive motor, drive motor brake, cable, coolant baffle, shock absorber, and scram control mechanism.

   (b) Assembly tests for directional response; normal operation relative to rod speed, rod overtravel, and drift in hold position; scram operating performance; cycling tests to check limit switches, slack cable switches, and position transmitter and indication; and manual drive tests to determine operability and torque requirements.

2. The environmental tests of the prototype mechanism are conducted at ORNL and consist of cycling tests, scram tests, and normal operating tests. These tests are conducted in helium at 150 °F and 305 psia, in helium at 80 °F and 14.7 psia, and in air at ambient pressure and temperature.

   Following acceptance of the prototype tests, the production drive mechanisms are given the same vendor tests listed in item 1, but are not given the environmental tests at ORNL.

3. The endurance tests on one of the production drive mechanisms are conducted at ORNL for six months followed by further testing as required. The tests include scram or free-fall, momentary power failure, graphite dust contamination, cyclic (straight and bowed channel), slack cable, rod manual drive, shroud-rod manual lock, flow and pressure drop, and loss of coolant flow.

4. The tests on each drive mechanism in the rehearsal shaft are conducted using a jumper cable to the appropriate control rod nozzle, and using the readout instrumentation and controls in the main control room. The checks which are made include directional response, setting of upper and lower limit switches, operation of scram velocity mechanism, and accuracy check of rod position versus readout in main control room. In addition, readout instrumentation and control functions are checked out at the rehearsal shaft panel.

5. Tests on the mechanism and control instrumentation are made after the mechanism is inserted in the reactor. The final precritical tests on the control rod system are made following the insertion of all rods and drive mechanism in their respective nozzles. The following tests are made:

   (a) Verify directional response and limit switch settings for each rod, including rod and shroud lock indications.
(b) Verify operation of the scram velocity mechanism and manual scram switch by using the individual clutch release key switch to scram each rod from the full-up position.

(c) Verify that with each rod switch in the gang position, the gang switch controls the insertion and removal of the rod.

(d) Verify that the rod limiting mechanism limits the number of rods selected for gang or automatic in the four modes of operation; startup, intermediate, power manual, and power automatic. Selection of more rods than allowed gives a rod withdrawal prohibit and an alarm.

(e) Verify that the inlet gas temperature controller cannot operate the rod drive motors in any position of the mode selector switch except the power-automatic mode.

(f) Verify that all rod withdrawal prohibit interlocks associated with the control and safety system are operable.

(g) Verify operation of auxiliary switches associated with the system, such as master power "on" switch.

(h) Verify rod position indication.

b. Reactor Coolant Blower Control System

The blower control system, which consists of direct coolant flow controls, speed synchronizing between the two blowers, and coolant flow controls actuated by the reactor outlet temperature are tested for proper component operation at the vendor plant and again after installation at the site. Blowers are operated by manual control during tests described in Section 12.4.1. Phase III tests on the blower control systems are conducted in two ways. The coolant flow controls and the speed synchronizer circuit are tested by having them in service while operating the blowers. The controls actuated by reactor temperature are tested by using simulated input signals to replace the signals normally derived by the temperature sensors. From such tests, information concerning the system stability is determined, and the proper gain settings calculated for comparison with design values.

The flow control is tested first on each reactor coolant loop separately, and then with both loops operating simultaneously. The coolant flow controls, which are normally actuated by reactor temperature, are disconnected for these tests.

The speed synchronizing controls are tested with both blowers operating, by manually causing the blower speeds to separate, and then allowing the automatic circuits to restore synchronism. These tests are done for several blower speeds corresponding to flows of between 20 and 100% of full flow.

Tests of blower controls, using only a signal which simulates control by reactor outlet temperature, are run on each reactor coolant loop separately. The speed synchronizer and the direct flow controls are set at fixed values for this test. After the temperature control for each loop is tested, the speed synchronizing circuit is placed in service, and the control of parallel operation of the two blowers is tested using simulated reactor temperatures. From these tests, control stability and proper gain settings are calculated and compared to design values.

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At the conclusion of the above tests, an over-all blower control test, using both loops and proper gain settings, is run using the flow control systems, the synchronizing circuits, and a simulated signal to the flow control systems that are actuated by reactor outlet temperature.

Proper operation of the emergency manual control system is verified by adjusting the speed of the blowers over the operating range.

c. **Turbine Generator Control System**

The turbine generator control system is checked partially under Phase III tests. A portable boiler is used to furnish steam to the turbine.

The full-speed test consists of upsetting the governor (leaving the set point or speed adjustment fixed) in both an increased and a decreased condition and observing the recovery. After this test is made, movement of the linkages on the governor is measured to determine speed droop.

With the successful completion of the above test, the unit is increased slowly in speed until the overspeed trip functions.

With the unit operating at full speed on speed control, the speed adjustment is increased slowly until the over-frequency relay is tripped. The unit then is reduced below normal speed until the under-frequency relay is tripped.

The unit is transferred from speed control to initial pressure control. The normal controlling pressure input to the initial pressure regulator of 1250 psig is furnished by test equipment. When the unit is running at normal speed, the pressure set point is decreased slightly so that the unit speed is increased to the point where the overspeed relay trips the unit to speed control.

The above procedure is repeated, but the speed of the unit is reduced to a point where it trips from pressure control to speed control.

The operation of the voltage regulator is tested with the unit at full speed, and the applied unit field controlled automatically. With the voltage set on the unit, the speed of the unit is increased from normal, noting that the voltage regulator controls voltage within the specified limits. The same test is run with the unit speed decreased, noting that the voltage of the unit remains within the specified limits.

Further tests are made after the reactor, steam generator, and turbo-generator are in service.

d. **Steam Generator Drum Level Control System**

Vendor tests are performed to assure circuit continuity and proper functioning of components. Phase III tests are then performed to determine the over-all operability of the drum level control system.

Phase III testing is done by introducing variable simulated signals into each of the three elements of the controls. The data collected from the response of drum level control and the signals to the feedwater control are used to calculate and set the necessary gains for proper operation of these systems.

e. **Steam Dump Control System**
Phase III testing of the steam dump system is simulated by use of auxiliary pressure applied to the controls through test equipment. This pressure, applied to the steam pressure sensor of the equipment under test, is varied from 1250 to 1290 psig. The dump system otherwise is in normal operating condition. By increasing the pressure from 1250 psig, the operating characteristics of the dump valve are determined and the proper values set according to specifications.

The controls of the desuperheater and the condenser low-vacuum prohibit device associated with the dump controls are tested for proper operation at the time of the dump control tests.

12.4.9 Safety System

a. Integrated Safety System Performance

The safety system panel is assembled and tested in the vendor plant in accordance with approved procedures. Following installation, the components of the safety system are carefully checked and calibrated.

In Phase III testing, scram and alarm circuits are actuated and set properly. Separate testing need not be made for alarms except for high start-up rate and high count rate in the start-up range, and high start-up rate in the intermediate range. These alarms are tested using a simulated signal.

As simulated signals are increased, actuation of many rod withdraw prohibits (RWP) occurs before a reactor scram so that in testing the scram systems, RWPs for high flux minus flow, high reactor outlet temperature, low coolant pressure, flux greater than 1 MW in the start-up range, and count rate less than five counts per second in the start-up range are tested at the same time. The remaining RWPs, which include prohibits for flow less than 75% in the start-up and intermediate ranges, rod upper limit switch actuated and number of rods selected for each range of operation, and flux less than 17 MW in the power range, are tested to determine that the RWP deactivates the rod controls in the withdrawal direction.

During Phase III tests of the reactor scram system, fluxes and temperatures are simulated. Flux and temperature simulated voltages are connected directly into the reactor scram circuitry and are varied until the reactor scrams. Such simulated signals are used to check scrams on high neutron flux in the start-up range, and flux minus flow and reactor coolant outlet temperature in the power range.

High coolant pressure and low coolant pressure scram actions are checked by creating the pressures of 325 psig and 225 psig, respectively, and noting that the reactor scram action occurs.

Nozzle annuli flow and nozzle interior flow reactor scram actions are checked by reducing these flows from the vessel cooling compressor by valve actions and noting at what actual flow the scram action occurs.

Nozzle interior coolant temperature reactor scram action is simulated by increasing the temperature of the sensor above 150 F. This is done separately and with the sensor removed from its normal location.

Scram action, due to loss of normal plant power, can be simulated by de-energizing the 2.4 kv electrical buses. This is done, noting that proper action occurs in sequence.
The reactor safety system scram action due to the loss of either reactor coolant blower is simulated by inserting a test block (with the current circuits properly shorted) into the current circuits of the relays, causing a reactor scram action of the blower. The scram from both reactor coolant blower circuits is checked in this manner. Additional tests are made by de-energizing the breaker of the single blower and noting the scram action.

Scram action due to the steam generator low drum level is tested by initiating a low drum level signal in each of the steam generators.

b. Reactor Scram and Emergency Blower Action

During Phase III tests, the blower is tripped and coolant flow and the time to coast down to zero speed is recorded. This information is compared to computer studies to predict reactor and steam generator temperatures and rates of temperature change during power operation.

Another Phase III test verifies emergency blower action due to a simulated reactor scram. The blower is automatically ramped down to a condition of approximately 23% speed.

Reactor scram and emergency blower actions are performed initially with a single loop and, after successfully completing the tests on each loop, the test of both loops is run.

c. Containment Test

The containment system is isolated automatically on a high radiation level signal from the stack monitoring system, a low pressure signal from the reactor coolant system, or if switch S-1 is manually placed in the closed position. The Phase III testing confirms these actions. In addition, the time required to close all of the automatic containment isolation valves is determined.

d. Steam Generator Isolation

Automatic isolation of a steam generator is initiated from a low-low drum level signal or from a flood level signal (in the lower plenum of the steam generator). The isolation action is effected by floats on these devices, and in each case proper operation is determined by manually operating the float or by filling the steam generator to the proper level.

e. Emergency Cooling System

Precritical testing of the emergency cooling system is conducted to verify remote manual operation of equipment and valves and proper functioning of individual components. The compressors are operated at design conditions to verify flows and performance characteristics.

The ability of the purge gas system to deliver the design flows of nitrogen to the emergency cooling loop is demonstrated.

Cell containment and the cell air lock are tested in a manner similar to that for the containment shell and air locks.

Precritical testing of the fission product removal system consists of verification of remote manual valve operation, flow tests and filter efficiency tests.
12.4.10 Process Instrumentation

Process instrumentation is defined here as those components associated with a particular process and not tested as separate instrument and control systems. These components are tested during the precritical testing of the appropriate process system.

In general, the process instruments are shop tested as individual components at the vendor's plant prior to shipment, and results recorded and reported to the contractor. All of the "line" installed instruments, except flow elements, are calibrated to check the vendor tests prior to installation. Following installation, the individual components and integrated systems are given a series of operating and functional tests to verify that they meet system requirements for range, accuracy, and response. Each instrument system and its individual components are then given a final check to determine the functional, alarm, and scram settings. The scram set points are coordinated with the safety system tests.

12.5 Test Evaluation and Further Action

Employees of HKF, KE, A-C, ORNL, and AEC are utilized during the precritical testing where special knowledge and consultation are required. These persons, together with the EGCR operating superintendent and technical program superintendent, comprise a group for reviewing the tests, evaluating system performance, and recommending additional tests.

After the precritical testing is completed, this group evaluates the over-all plant performance and reviews plant operating parameters and test procedures before continuing with core loading and further testing. This evaluation provides maximum assurance that the plant is ready for fuel loading and testing with nuclear power.
13. INITIAL NUCLEAR TESTS

13.1 Introduction

The portion of the test program relating to reactor physics is designed to fulfill a number of objectives. In general, the experiments are intended to confirm the design calculations, to establish information necessary for safe operation of the reactor, and to provide information of a general nature useful in the design of future reactors of this type.

Of particular importance in confirming design calculations are critical experiments establishing information on core reactivity and power distribution. Of importance to safe operation of the reactor are experiments to determine worth of control rods in various configurations, the minimum number of withdrawn rods for criticality, and the accuracy and stability of neutron instrumentation.

The program is designed to produce information which verifies a basic proposed operating mode for the reactor and, in addition, produces information which provides the basis for reliable methods of core analysis. When operating as an experimental facility, the reactor undergoes frequent changes of core configuration and each presents its own problems of operation. In particular, the testing of high performance fuel elements presents problems of power distribution and reactivity control which require detailed analysis for each case. Although a few general principles of fuel and control rod management may broadly define a pattern of operation, it is important to select acceptable configurations on the basis of calculations as a routine part of operations. For this reason, the program places considerable emphasis on experiments which are required for core analysis in addition to providing the necessary information to operate in a basic mode.

The initial nuclear tests discussed in this section comprise the Phase IVa tests. Nuclear tests at low power and full power are described in Section 14.

Phase IVa testing program begins with the installation of the in-core neutron detectors and ends with the conclusion of all cold rod calibration tests. The purpose of Phase IVa tests is to accomplish the successful loading of the core, calibrate control rods and nuclear instruments, determine the neutron distribution within the core for various control rod configurations, verify physics calculations, and perform other miscellaneous tests so that subsequent tests may be performed.

Prior to initiating the Phase IVa tests, it is intended that Phase III tests relating to the reactor are completed. However, in the event of delays or unforeseen developments during the Phase III tests, the Phase IVa tests are done prior to or concurrently with certain Phase III tests since comparatively few auxiliary systems are needed for this work. In addition, the control and safety system circuitry are changed from the basic design to accommodate the Phase IVa tests. Such changes are noted in Section 13.3.

During the Phase IVa test program, personnel access to the upper plenum of the reactor pressure vessel is to be available if required. Should such access be required, adequate ventilation is provided, extreme care is exercised to
maintain the cleanliness of the reactor, and radiation surveys are made prior to and during entry. The individual supply circuits to the a-c rod drive motors and scram clutches are opened to prevent inadvertent criticality.

13.2 Organization

A reactor physics task force, composed of representatives from ORNL, TVA, and AEC, has responsibility for preparation of the physics tests to be performed during Phase IVa tests. The program includes estimated times to prepare the tests, listing the papers of principles to be prepared supporting each test, and listing the detailed test procedures for stepwise execution of the work. The program also shows the time estimates for actually performing the tests. The program is reviewed by ORNL, TVA, and AEC, and finally approved by TVA. As the individual tests are written by the task force in the form of papers of principle and detailed procedures, they are reviewed and approved in like manner.

Organization responsibilities for Phase IVa tests are as follows:

a. The Reactor Physics Staff, under the direction of the technical program superintendent, has responsibility for planning and coordinating the performance of physics tests.

b. The assistant to the operating superintendent coordinates the individual tests between the Technical Program and Operating Groups.

c. The Operating Group is responsible for execution of the tests.

d. Throughout the program the operation of the reactor as well as all other plant equipment is the direct responsibility of the plant operations supervisor, under the general direction of the operating superintendent.

e. The shift operating crew is augmented by the addition of at least one duty physicist to provide continuous technical assistance to the plant operations supervisor throughout all tests. The duty physicist is permitted to initiate changes in the test procedures which do not alter the objective of the experiment or alter the hazards aspects of the work. Such changes are noted and considered in the final test evaluation. More significant changes require prior review and approval in accordance with the procedure described in Section 10.4.

f. TVA provides all required operators, craft personnel, data collectors, and technical personnel for performing the tests.

Analyses of test data are performed by the Reactor Physics Staff with assistance from ORNL as required and mutually agreed upon.

13.3 Preloading Checks of Special Instrumentation

The permanently installed neutron detectors, outside the pressure vessel, and normal neutron sources in the reactor are designed and located to give appropriate counting rates when bringing the fully-loaded core from shutdown conditions to 100% power. For the partially-loaded core or the completely-unloaded core, and for initial criticality with the fully-loaded core, special detectors and sources are installed inside the pressure vessel to obtain the desired counting rate of at least two counts per second.

Seven detectors are installed temporarily in the core during Phase IVa tests; three uncompensated ion chambers, two fission chambers, one BF\textsubscript{3} counter, and one compensated ion chamber. Four of the seven detectors are used as safety channel detectors as indicated in Table 13.3. The three uncompensated ion
chambers are located in unused neutron source holes N1, N2, and N4 and remain fixed throughout the Phase IVa tests. Information from these detectors is used during sequential core loading stages to reference readings to a common base. The remaining detectors are placed in empty fuel channels. Prior to each core loading stage, the detectors are relocated, if required, to empty fuel channels immediately outside the expected boundary of the loading stage. In addition, the three permanently installed safety channels are used as a backup. These channels are designated in Table 13.3 as channels 8 through 10. During the Phase IVa tests these channels are capable of detecting a flux level corresponding to approximately 2.5 kw in the core by increasing the gain of the flux amplifiers and by readjusting the scram trip points.

An americium-beryllium neutron source with a strength of $10^7$ neutrons/sec is to be located in source hole N3 in the core. A source of this size provides sufficient flexibility in positioning the neutron counters, but is small enough to minimize problems of fuel element activation.

In addition to special sources and detectors, Phase IVa testing requires certain temporary changes in the reactor control and safety system instrumentation. Changes in the rod withdraw prohibits and scram trip levels for the Phase IVa tests are shown in Table 13.3. A thorough checkout of all changes and temporary circuits is made prior to loading fuel and after each step of the testing that requires subsequent changes. A complete record is made of all changes and restoration to the original design is required and verified before proceeding to power operation in Phase IVc of the program.

**TABLE 13.3**

*Nuclear Instrumentation During Phase IVa Tests*

<table>
<thead>
<tr>
<th>Channel Number</th>
<th>Perma-</th>
<th>Name</th>
<th>Approximate Flux at Chamber</th>
<th>Safety Action</th>
<th>Set Point</th>
</tr>
</thead>
<tbody>
<tr>
<td>Test</td>
<td>nent</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td></td>
<td>BF$_3$ counting channel</td>
<td>0.50 nv</td>
<td>None</td>
<td>None</td>
</tr>
<tr>
<td>2</td>
<td>1</td>
<td>Fission counting channel</td>
<td>1.4 nv</td>
<td>High dpm $^{\text{b}}$ (period) RWP</td>
<td>$1 \text{ dpm} (T = 26 \text{ sec})$</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>Fission counting channel</td>
<td>1.4 nv</td>
<td>High count rate RWP</td>
<td>$10^5 \text{ counts/sec}$</td>
</tr>
<tr>
<td>4</td>
<td>3</td>
<td>Log-N channel (uncompensated ion chamber)</td>
<td>$5 \times 10^4$ nv</td>
<td>Level RWP</td>
<td>$3 \times 10^7$ nv</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Level Scram</td>
<td>$10^8$ nv</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>High dpm $^{\text{c}}$ (period) Scram</td>
<td>$2 \text{ dpm} (T = 13 \text{ sec})$</td>
</tr>
</tbody>
</table>
### Table 13.3 (continued)

<table>
<thead>
<tr>
<th>Channel Number</th>
<th>Permanently</th>
<th>Name</th>
<th>Approximate Minimum Flux at Chamber</th>
<th>Safety Action</th>
<th>Set Point</th>
</tr>
</thead>
<tbody>
<tr>
<td>5</td>
<td>4</td>
<td>Log-N channel (uncompensated ion chamber)</td>
<td>$5 \times 10^4$ nv</td>
<td>Level RWP</td>
<td>$3 \times 10^7$ nv</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Level Scram</td>
<td>$10^8$ nv</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>High dpm (period)</td>
<td>2 dpm $(T = 13$ sec)</td>
</tr>
<tr>
<td>6</td>
<td>--</td>
<td>Auxiliary log-N channel (compensated ion chamber)</td>
<td>300 nv</td>
<td>None</td>
<td>None</td>
</tr>
<tr>
<td>7</td>
<td>--</td>
<td>Linear ion Chamber channel (uncompensated ion chamber)</td>
<td>$5 \times 10^4$ nv</td>
<td>None</td>
<td>None</td>
</tr>
<tr>
<td>8-10c</td>
<td>5-7</td>
<td>Flux-Minus-Flow (zero flow) (2 out of 3 coincidence circuit)</td>
<td>$3 \times 10^5$ nv (\sim 2.5$ kw)</td>
<td>Scram</td>
<td>$3 \times 10^6$ nv (\sim 25$ kw)</td>
</tr>
</tbody>
</table>

a Table 7.1.4.1, Volume 1.

b dpm is decades per minute and corresponds to inverse period.

c Channels 8, 9, and 10 are safety channels installed outside the reactor vessel in instrument tubes. The sensitivity of the flux amplifiers is increased, for the test program, to function at this level.

### 13.4 The Initial Critical Experiment

Preliminary estimates indicate that the core is critical with a loading of approximately 40 channels. The resulting core configuration determined on this basis is illustrated in Figure 13.4 showing the loaded channels, the positions of the seven detectors, and the neutron source.

During the loading process up to the minimum critical core, control rods NGOTM are always fully inserted before fuel is added. In addition, no fuel is added or control rods withdrawn unless a count rate of at least two counts per second is indicated by at least two of the three low level neutron detectors (test channels 1 through 3, Table 13.3). Fuel is loaded with the charge machine at the rate of about two channels per eight-hour shift, and the pattern of fuel in the core is built up as symmetrically as practical about the central control rod N. The service machine also is available for fuel loading as required. After a predetermined addition of fuel, the inserted rods are withdrawn together and count rates obtained with the BF$_3$ counter and the two fission chambers.
FIG. 13.4

LEGEND:
+ Installed Fuel
× Neutron Detector Location for loading Stage 1.
× Neutron Detector Location for loading Stage 2.
○ Installed Control Rod
□ Control Rod Nozzle
□ Installed Source
□ Source Channel containing Ion-Chamber
© Flux Scanning Channel
- Access through Nozzle not located directly above Detector location.
- Detector location change from Loading Stage 1.

APPROXIMATE CORE BOUNDARY AT END OF LOADING STAGE 1

APPROXIMATE CORE BOUNDARY AT END OF LOADING STAGE 2

CORE CONFIGURATION FOR CRITICAL EXPERIMENTS (Loading Stages 1 and 2)

Fig. 13.4
After being suitably normalized to account for different sensitivities, these rates are averaged and the reciprocal count rate plotted versus the number of loaded channels. Since the count rate becomes infinite at criticality, this plot approaches zero as criticality is approached. This property is used to define the critical loading. In addition to plotting the reciprocal average count rate, the reciprocal count rate is plotted for each detector individually. Operations involving anticipation of criticality are based on the extrapolation of the most conservative counter. A pulsed neutron generator may be used as a supplementary method for anticipating criticality.

After the minimum critical loading is reached, a number of reactivity effects are measured. When the core is subcritical, measurements are made utilizing the pulsed neutron generator. When the core is supercritical, measurements are based on reactor period. When the core is critical, the measurements are made using calibrated control rods. At least two such measurements are currently planned:

a. The pulsed neutron generator is used to measure the subcritical multiplication factor when various rods are inserted. Of particular interest are the combinations N, GOTM, and HS.

b. The reactivity effect of unloading the fuel from a single channel is examined for channels near the boundary and near the center of the loading.

At this stage of loading, the first operational test of the permanently installed flux scanning system is made. The flux level is increased to a power of two to three watts per channel, and wires are inserted to obtain measurements in scanner positions F-4, F-7, F-9, F-10, F-12, and F-15 which are either within the core or near the boundary (Figure 13.4). Operation at this power level does not produce activation of fuel elements or other materials that is high enough to cause handling problems.

Foils, such as gold, are standardized in a known flux and then irradiated in the core to verify the approximate power level in terms of the reading of the three in-core uncompensated ion chambers.

Foils for special studies are activated in specially designed fuel assemblies which are easily disassembled for retrieval of the foils. It is planned to irradiate foils of two enrichments of uranium to determine the ratio of uranium-238 absorption to uranium-235 fission. This ratio is simply related to the conversion ratio and resonance-escape probability.

13.5 Critical Experiments With Control Rods Inserted

After completion of all measurements with the minimum critical loading, the core buildup to a full loading proceeds in six major loading stages (stages 2 through 7, Table 13.5). Each step ends with a well-defined critical experiment for a selected pattern of fully-inserted control rods. In-core instrumentation location is altered, as required, prior to initiating fuel loading in each of these steps. During each of the loading steps, one or more control rods are fully inserted and remains inserted during the loading stage as indicated in Table 13.5. Fully inserted rods are electrically disconnected by use of the clutch supply key switches. With this provision, it is not possible to make the core critical during the loading process until the final loading step in the particular loading stage.

The first of this series of experiments (loading stage 2) is the critical loading around the fully-inserted central control rod. The preliminary estimate of the loading for this case is approximately 70 channels, and the resulting

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configuration on this basis is illustrated in Figure 13.4. As the assembly is
built up to this size from the minimum critical core, the central control rod
is left inserted. As with the initial loading operations, a number of control
rods are fully inserted while fuel is added and these rods are withdrawn after
specified additions of fuel to obtain a point on the plot of reciprocal count
rate.

In a similar fashion, critical experiments are assembled around other groups of
fully-inserted control rods during loading stages 3 through 6. Table 13.5 in-
dicates those rods that are to remain fully inserted during a particular loading
stage as well as the rods used for withdrawal during the experiments. These
loading stages build the pattern of control rods symmetrically out from the
center of the core. During loading stage 7 all control rods are fully inserted
and as fuel additions are made, partial withdrawal as a bank to several prede-
termined levels is made and the count rate is recorded. When the reciprocal
count rate is plotted versus number of loaded channels, the curve, which ex-
trapolates to zero when the core is fully loaded, defines the amount of rod
withdrawal to make the core critical. This is the first experiment to verify
the approximate shutdown margin of the fully-loaded core.

<table>
<thead>
<tr>
<th>Loading Stage</th>
<th>Rods Fully Inserted and Remaining Inserted Throughout The Loading Stage</th>
<th>Rods Fully Inserted During Loading, Then Withdrawn For Counting</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>None</td>
<td>NGOTM</td>
</tr>
<tr>
<td>2</td>
<td>N</td>
<td>GOTM</td>
</tr>
<tr>
<td>3</td>
<td>GOTM</td>
<td>FHUS</td>
</tr>
<tr>
<td>4</td>
<td>NGOTM</td>
<td>FHUS</td>
</tr>
<tr>
<td>5</td>
<td>FHUS NGOTM</td>
<td>CPXL</td>
</tr>
<tr>
<td>6</td>
<td>CPXL FHUS NGOTM</td>
<td>DVWE</td>
</tr>
<tr>
<td>7</td>
<td>None</td>
<td>All Control Rods</td>
</tr>
</tbody>
</table>

Rods not listed are fully withdrawn during loading and counting and
are capable of being inserted by the safety system.

When each of these critical cores is assembled, certain other measurements are
made. For example, when a critical assembly around rods GOTM is obtained, the
pulsed neutron generator is used to measure the multiplication factor when the
central rod, rod N, is inserted. Other observations, such as the amount of
uniform bank withdrawal of all rods for core criticality, are also noted during
these stages. Since fuel is loaded only at the rate of about two channels per
eight-hour shift, these control rod manipulations and observations of inter-
mediate critical cases add little, if any, to the total time of the program.
In the absence of unexpected difficulties, the estimated time to reach full
core loading is about 50 days. Of this time, the fuel loading operations alone
are expected to require 40 days.

Other measurements made in selected critical cores are flux scans with the
permanently installed system and the reactivity effect of discharging single
channels of fuel.

13.6 Zero Power Full Core Experiments

The program of zero power tests with the fully loaded core is a continuation of
the sequence of critical experiments, supplemented by measurements of neutron
distributions under conditions approaching those expected during normal
operation. During the full core tests, the uncompensated ion chambers remain in the positions shown in Figure 13.4. The other detectors are moved to the empty control rod channels directly below control rod nozzles A, K, Q, and Z (Figure 13.4).

Of particular value in terms of both safe operation and confirmation of design calculations is a series of observations to determine the number and location of control rods which may be withdrawn without reaching criticality. Of similar value are measurements of the shutdown margin by utilizing the pulsed neutron generator. The effect of control rod manipulation on the flux level in the thermal column is also measured.

A number of measurements of neutron flux fine structure distribution provide important information on the peaking factors to be encountered in operation. These measurements are made by irradiating foils in special fuel assemblies designed to be disassembled for foil removal. The following are examined:

a. Neutron flux variation at the ends of the $\text{UO}_2$ columns and the endcap, spider, and spacer regions

b. Radial and angular variations of neutron flux in the outer rods of the cluster

c. Variation of the neutron flux across a channel situated adjacent to a fully inserted control rod

d. Peaking factors in fuel channels adjacent to an empty fuel channel

e. Peaking factors in fuel channels adjacent to a graphite filled experimental channel

f. Variations in the neutron flux in the fuel near the tip of a partially inserted fuel assembly

g. Peaking factors in fuel channels near the reflector

h. Variations in the neutron flux in the fuel for channels adjacent to the tip of a partially inserted control rod.

Other flux measurements make use of the permanent flux scanning system and other foils in special fuel assemblies. The aim is to obtain data on the effect of control rods on the core power distribution and to calibrate the flux scanning system in terms of flux levels in the fuel channels. Other information related to use of the reactor as an experimental facility is obtained by measuring the reactivity effect and peaking factors obtained when four channels of fuel in the range of 3% to 4% enrichment are inserted into the core, one channel in each quadrant. The ratio of uranium-238 absorption to uranium-235 fission is re-evaluated to determine if the core size affects the results obtained in the initial critical experiment. The reactivity effect of loading or unloading a channel of fuel is repeated. The reactivity effect due to the service machine grapple is measured. Another effect to be measured is the change in the worth of a control rod caused by the position of other rods. The cold critical experiments are terminated by performing a full range rod calibration. Prior to this work the reactor coolant system is closed and tested. Blower operation is not required. It is presently planned to use fixed absorbers plus air or nitrogen under pressure to compensate for the withdrawal of rods. Reactivity accident analyses are performed to establish the maximum amount of reactivity which may be safely invested in air or nitrogen. A stepwise program permits full range calibration of the rods and determination of the free reactivity of the core. The absorbers are in the form of steel tubing in the fuel channels.
located equidistant from the control rod channels and extending the full length of the core. The principle of maintaining axial homogeneity in the fixed absorbers while extending the absorber range is accomplished by loading concentric tubing, thus effecting "grayness" control. The final loading of fixed absorbers permits the reactor to be depressurized with all control rods removed from the core. Pulsed neutron generator measurements are used to determine the subcritical multiplication factor. The control rods are then fully inserted and a bank of absorbers removed. The reactor is then made critical and as rods are inserted a subcritical calibration is performed using the pulsed neutron generator. This process is repeated until all fixed absorbers are removed from the core and the channels are reloaded with fuel elements.
14. TESTS AT POWER

14.1 General

The power test program includes:

a. The Phase IVb tests which are essentially zero power or low power (approximately 10 kw) nuclear tests combined with the plant mechanical tests and final checkout

b. All subsequent tests (Phase IVc and Phase V) including full power, full temperature operational testing of fuel assemblies and equipment.

Since detailed procedures for the tests at power are not completely developed, minor changes may be made to the program. Such changes, if required, are either in the direction of increasing reactor safety or are not directly related thereto.

A large number of additional tests and checks are made in the course of reactor start-up and approach to power. These are made at each step in the program as necessary to achieve safely the test objectives. For example, many of the mechanical tests described in Section 12 are repeated. Careful surveillance is maintained throughout the tests to insure continued satisfactory equipment and system performance. Detailed test and operating procedures are followed, and careful attention is given to verifying operating procedures including those procedures followed during emergency operation.

This test phase begins with verification of plant performance with the core fully loaded but operating at zero or near zero power. The power is then raised stepwise to the 50% power condition at constant 75% flow. A series of measurements and emergency operations are conducted during this relatively cold rise to power. During this period, selected fuel channels in the reactor are tested at near normal full power temperatures and are temperature cycled.

The reactor and associated plant are then tested at the design temperature levels, beginning with manual operation at 20% power and culminating in automatic operation at 100% power. The plant load following characteristics are then demonstrated.

After full power, full temperature is attained, further operations and tests are carried out to demonstrate the capability of special equipment and long-term performance of fuel assemblies. Long-term operation of such items as blower seals, control rod drives, and other equipment associated with the handling of high temperature helium are subsequently evaluated. Charge and service machine performance is evaluated. On-stream fuel handling is demonstrated during fuel shuffling operations. Fuel performance is verified by scheduled withdrawal and examination of assemblies. The resulting information is applicable to the design of improved clad fuel elements for gas-cooled reactors.
The testing program has the following more specific objectives:

a. Obtaining additional core physics data associated with temperature and power effects including:
   1. The combined temperature coefficient, i.e., the reactivity effects produced by temperature changes in the fuel, moderator, rod cables, and the reactor pressure vessel and internals
   2. The separate fuel and moderator temperature coefficients, including effects of temperature on rod worth
   3. Xenon poisoning measurements and their use to calibrate control rods.

b. Obtaining radiation exposures as determined by specimens placed at selected locations inside and outside the reactor pressure vessel, on certain internal components, and at points inside the biological shield

c. Verifying normal and emergency operating procedures

d. Measuring flux and temperature distribution at specific power levels, flows, and rod configurations

e. Observing changes in flux and temperature distribution as rods are moved through preplanned programs

f. Calibrating the nuclear instrumentation against gas and steam heat balances

g. Testing the burst slug detection system under actual flow conditions

h. Demonstrating the performance of the entire reactor plant including certain tests performed at full power, full temperature conditions. The following tests are included:
   1. Specially instrumented scram tests including several full power, full temperature scrams
   2. Tests on shutdown heat removal systems and methods, including natural convection tests and operation of the steam blowdown system
   3. Load transfer tests on the steam system
   4. Full-range calibration and testing of the automatic control system
   5. Load-following tests
   6. Continuing operational testing of the entire plant at full power.

i. Verifying the predicted fuel temperatures in the instrumented fuel assemblies and the predicted moderator temperatures under both steady-state and transient conditions. Verifying the predicted fuel channel outlet temperatures against specified channel orifice settings.

j. Performing radiation surveys to verify the adequacy of the biological shielding, to establish reference radiation levels within the reactor building, and to verify the performance of the radiation monitoring system

k. Obtaining early data on fuel performance by driving selected channels
to full power temperature levels in advance of full power operation of the entire core.

14.2 Organization

TVA has the responsibility for general planning and writing of the Phase IVb, IVc, and V test program. As in the case of the initial nuclear tests described in Section 13, a task force, comprising representatives from ORNL, TVA, and AEC, develops the initial outline plans for those physics tests included in the test program with the reactor at power.

Papers of principle are prepared for each test in the power test program. These papers set forth the objective, scope and theoretical basis for the test, and include a preliminary estimate of the time required to perform the test. The program is reviewed by ORNL, TVA, and AEC. As detailed procedures are developed, they are reviewed in a like manner.

Organizational responsibilities for the tests at power are as follows:

a. The Technical Program Group, under the direction of the technical program superintendent, has responsibility for planning and coordinating the performance of all tests at power.

b. The assistant to the operating superintendent coordinates the individual tests between the Technical Program and Operating Groups.

c. The Operating Group is responsible for execution of the tests.

d. Throughout the program the operation of the reactor as well as all other plant equipment is the direct responsibility of the assigned plant operations supervisor on each shift, under the general direction of the operating superintendent.

e. The shift operating crew is augmented by the addition of one or more representatives from the Technical Program Group to provide continuous technical assistance to the plant operations supervisor during all tests. The designated test engineer or duty physicist will be permitted to make minor changes in the test procedures which clearly do not alter the objective of the experiment or alter the hazards aspects of the work. Such changes are noted and considered in the final test evaluation. More significant changes require prior review and approval in accordance with the procedure described in Section 10.4.

f. TVA furnishes all required operators, craft personnel, data collectors, and technical personnel for performing the tests.

Analysis of test data is performed by the Technical Program Group with assistance from ORNL as required and mutually agreed upon.

14.3 Final Prepower Tests and Inspections

After the core is fully loaded and all Phase IVa physics tests are completed, the temporary start-up instrumentation is checked for operation at pressures, flows, and reactor coolant temperatures achieved during the Phase IVb tests. In lieu of using temporary start-up instrumentation in the core during Phase IVb tests, the four large neutron sources may be loaded to permit use of permanent nuclear instrumentation.

Activation samples are placed for measuring damage flux to the reactor pressure vessel and certain internal components, general flux levels inside the biological shield, and flux levels at the thermal column position. Several fission
product sources are attached to fuel elements for burst slug detection tests. The instrumented fuel assemblies are installed. Strict inventory control is maintained on all items brought into the reactor coolant system. The reactor coolant system is then closed and contains air at atmospheric pressure.

14.3.1 Measurement of Reactivity Effects

The reactor is brought to a stable critical condition and rod positions are recorded. Certain scrams and interlocks are temporarily bypassed. Scram protection is provided by temporary flux level trips set at low levels, as well as by other normal scram circuits. The reactor is then shut down and the reactor coolant system filled with helium by evacuation-purging-cleanup operations until approximate design purity is established at atmospheric pressure. The reactor is again brought to a stable critical condition, and rod positions are recorded to determine the increase in reactivity caused by removal of air.

The reactor coolant system is then charged with helium to a pressure of approximately 140 psia while maintaining the reactor just critical. At approximately 50-psi intervals, charging is interrupted to allow stabilization of counting rates. The helium pressure coefficient is thus obtained by the substitution method using previously calibrated rods. Alternatively, the pressure coefficient may be measured by the period technique.

Following this test, the reactor is shut down and the scram and interlock circuits previously bypassed are restored before proceeding with flow tests with helium.

14.3.2 Flow Tests With Helium

All plant systems are placed in operation culminating in 500 F isothermal conditions with the reactor shut down.

Parallel operation of the blowers is verified under the various normal operating conditions. The core pressure drop is measured. The system operating characteristics are measured for both reactor coolant blowers with and without a vessel cooling compressor operating, for each reactor coolant blower, and for each vessel cooling compressor. Flow changes during transfer of vessel cooling compressors are recorded along with other data.

Flow and temperature are carefully measured in the several flow-instrumented channels with the reactor coolant blowers running at various speeds. Changes in flow during charge and service machine operations are recorded. Control of flow with orifice settings by the charge machine is verified.

Flows to the top nozzle interiors and annuli are measured and adjusted. The operating characteristics of the nozzle and charge and service machine coolers are established and water flows are set.

A series of emergency operations are conducted to verify and establish flows and temperatures during transients and to verify general equipment operation under abnormal conditions. These include, among others:

a. Simulated scram and blower rampdown

b. Loss of 13.8 kv power, recovery of vessel cooling compressors by automatic start-up of the emergency power system. Operation of the shutdown heat removal systems including automatic closure of the reactor coolant loop inlet isolation valves, steam system blowdown to 135 psig, and operation of steam-driven shutdown feedwater pumps.
c. Natural convection cooling test including service machine nozzle cooling flow from high pressure storage system. Heatup of the nozzle to which the service machine is attached, after complete loss of flow except by natural convection. Decay of general system temperatures by natural convection.

d. Trip of one reactor coolant blower; verify ability of second blower to remain on-stream without surge damage. Verify associated emergency procedures including closure of isolation valves in loop with inoperative reactor coolant blower.

After completion of the flow tests with helium, an extended run of seven to ten days is performed prior to final inspection of the reactor coolant system, and removal of test apparatus, foils, and activation samples. During the extended run, the plant is in operation under conditions of full flow, gas temperatures at approximately 500 F isothermal, the steam plant operating occasionally at reduced conditions, and all plant auxiliary systems operating as near to normal conditions as possible without nuclear heat. Charge and service machine operations are simulated during the run.

14.3.3 Low Power Nuclear Experiments From Ambient to 500 F

Three nuclear experiments are included during the run; however, power levels and times at power are kept sufficiently low to permit subsequent re-entry of personnel into the reactor pressure vessel if it is determined to be necessary.

a. Measurement of the Combined Isothermal Temperature Coefficient up to 500 F

The reactor is brought to the critical condition at ambient or near ambient temperatures. The combined temperature coefficient at several temperatures is deduced by comparing relative rod positions at these temperatures to the positions at ambient temperature.

b. Damage Flux and Shielding Measurements

At an appropriate time during the extended proving run, the reactor is again brought critical. The flux level is raised to permit activation of the specimens previously installed inside the reactor pressure vessel and at various other locations inside the biological shield. Accurate times of exposure are recorded.

c. Burst Slug Detection (BSD) System Sensitivity Tests

During the experiment in item b. above, a simulated search test for a failed fuel element is made using the BSD system and a source of fission products. At least two channels are loaded. The ability of the BSD system to discriminate between channels with failed and normal fuel is demonstrated. The effects on signal level of flow change or of removing simulated failed elements with the charge and service machines is demonstrated.

Following these tests, all temporary instrumentation, foils and activation specimens are removed from the reactor pressure vessel and interior of biological shield. The access opening through the biological shield is permanently closed.

14.4 Approach to Power

Prior operations do not require that the containment system be in operation. As a prerequisite to the following tests, the containment system is operationally tested and placed in service.
14.4.1 Tests up to 10 MwT

The entire plant, except for the reactor, is placed in service using normal start-up procedures, and isothermal operation at approximately 500 °F is again stabilized. Protective circuitry is given a complete operational checkout; instrumented fuel assemblies are fully equipped with readout equipment and are under constant surveillance during the approach to power.

The synchronized reactor coolant blowers are manually adjusted to 75% flow and flow is held constant by manual control. Flows and temperatures are permitted to stabilize and a careful reactor coolant and steam heat balance is calculated for the zero power condition.

The reactor is made just critical and stable. Rod withdraw prohibits are checked. The flux instrumentation is set to calculated reactor power levels from previously obtained flux scanning and foil activation data.

Control rods are withdrawn on a preplanned program permitting the power level to stabilize after each withdrawal. When the reactor power reaches an indicated level of 5 MwT, power is stabilized, a general survey at operating conditions is made, and data is recorded of steam and reactor coolant conditions for a thermal heat balance. Instrumented fuel assembly and moderator temperatures are taken, and a reactivity balance is performed. The flux channels are reset to the actual reactor power, which is obtained by subtracting the measured thermal power for the zero power case from the measured thermal power as measured for this condition.

A radiation survey of the biological shield and auxiliary components is made using portable instruments; a complete survey of the shield is performed later at higher power levels using special instrumentation and film.

Using the newly calibrated flux channels as a guide, the reactor power is raised stepwise to 10 MwT. The reactor and plant are stabilized at an indicated 10 MwT and 75% flow. The turbine-generator is warmed, synchronized and placed in operation on initial pressure control. Operating conditions are checked and a second complete flux plot, heat, and reactivity balance is performed. The flux channels are recalibrated and power is adjusted as required.

A radiation survey is performed with dose rate measurements made at pre-established numbered grid points on the biological shield and reactor coolant ducts. Particular attention is given to areas where neutron and gamma streaming may occur.

14.4.2 Tests up to 20 MwT

The power level is raised on a programmed basis to the 20 MwT level as indicated by nuclear instrumentation, with approximate correlation between this reading and the temperature rise across the core. The mixed mean exit coolant temperature is approximately 700 °F.

When the power is stabilized, a general survey is made of operating conditions in supporting systems. A temperature survey, reactivity balance, and reactor coolant and steam system heat balance are calculated from measured parameters. The nuclear instruments are recalibrated to true thermal power as required. In addition, the following tests are then performed:

a. Main steam system relief valves are tested. Any effect on the reactor operation is recorded.

b. The electromagnetic valves are tested.

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c. A test of the turbine steam dump, and auxiliary apparatus is made.

d. The reactor is scrammed by remote manual actuation. Rod release and fall times, and flux decay are measured. Blower rampdown to approximately 23% speed is verified; gas flows and temperatures during the transient are recorded and the response of the entire steam system is checked. Blowers are returned to 75% flow.

e. Control rods are withdrawn, using the approved start-up program. A stepwise approach to 20 MWt is again performed at 5 MWt intervals. Reactor outlet temperature (TRO), reactor inlet temperature (TRI), and gas flows are noted and compared with flux indications. The ability of the nuclear channels to reproduce power level indications at the 10 and 20 MWt level is noted.

f. Regulated pressure operation is resumed at the 20 MWt, 75% flow condition and a general survey of operating conditions is performed.

14.4.3 Tests up to 40 MWt

The reactor is raised stepwise to 40 MWt as indicated on the nuclear channels with approximate correlations being maintained between these indications, TRO, TRI, and flow. The mixed mean exit coolant temperature is approximately 850 F. The plant is permitted to stabilize at approximately 5 MWt intervals.

A survey of operating conditions is made and an assessment made of the performance of each auxiliary system.

a. Simulation of Abnormal Conditions - Natural Circulation Test

A test is performed to demonstrate the ability of the turbine-generator to drop the outgoing line load and retain the plant load. With the turbine on pressure control, the 13.8 kv breaker 264 in the X-10 switchyard is momentarily opened and reclosed at a predetermined time. Breaker S-1340 in the EGCR substation opens and remains open. The turbine-generator retains the plant load at an acceptable frequency. The generator is synchronized and connected back into the outgoing line. Load is then removed from the turbine-generator until it is finally removed from the line with all steam going to the dump condenser. Breaker 264 is again momentarily tripped.

The reactor scrams and running equipment begins to decelerate except that which is battery supplied. The diesels proceed on an automatic start-up sequence, come to normal speed and connect with their respective 440 v emergency buses. The circuit breakers to both vessel cooling compressors and the 440 v shutdown feedwater pumps are opened at this time.

The system is then in the full loss-of-power emergency insofar as electrically driven blowers and feedwater pumps are concerned. The seal water system remains in normal service. Operation of the blowdown sequence is verified, including control of steam drum levels, and operation of reciprocating steam-driven shutdown feedwater pumps and other shutdown heat removal equipment. Coastdown of blowers, helium and steam flows, pressures and temperatures are recorded during the natural circulation shutdown.

Due to the 50% power, 75% flow condition, and comparatively small amount of afterheat stored in the core during short-term operation, the temperatures during this test are much lower than those which occur after the full power long-term operation at rated helium flows; however, it should
be possible to extrapolate these test results to the latter case.

When temperatures stabilize at a low level, the 13.8 kv system is re-energized, the blowers restarted, and a normal restart is made. Operations are restored at the 40-MwT 75% flow condition, again noting the relationship between the flux, TRO, and TRI at this flow.

After stable operation is achieved, a flux plot, temperature survey, reactor coolant and steam heat balance, and reactivity balance are made. The nuclear channels are reset to compensate for the discrepancy, if any, between actual thermal and indicated power.

b. **Verifications of Predicted Fuel, Moderator, and Reactor Coolant Temperatures**

The core physics and thermodynamics calculations are verified. Measured flux distribution is compared against calculated distribution. Individual channel orifice settings, TRO, and TRI are measured and compared with calculated values. The over-all core thermal power is compared to the sum of the calculated channel powers. The moderator temperature distribution is measured. Measured performance of the four instrumented fuel assemblies is evaluated against calculated performance. Channel orifice settings are trimmed as required, except as in item d. below, for the full power, full flow condition. Any discrepancies between calculated and measured parameters are evaluated and resolved.

c. **Auxiliary System Tests and Calibrations**

All auxiliary systems are checked for normal operation and are subjected to normal operational tests. Performance of systems under essentially full load, full temperature conditions is verified and any change in performance during this period is noted and resolved.

d. **Experiments with Selected Fuel Channels**

To provide advance information on the performance of fuel assemblies operating near normal temperatures and subjected to cyclic duty, a small number of channels are subjected to short-term, normal duty corresponding to full power operation.

Certain selected fuel elements used in these channels are specially prepared for subsequent analyses. Extensive records are assembled on these elements for future reference and analyses. These channels are operated for several weeks during a period when the plant is undergoing a system shakedown at 50% power.

The channels in one or two moderator columns are loaded as described above. At least one of the columns contains an instrumented fuel column to be used as a reference.

The selected channel orifices are adjusted to produce near full power coolant, fuel, and moderator temperatures. Nominal deviations from the 75% flow condition are used to adjust the channel temperatures. Protective scrams on these channel temperatures may be temporarily added for this condition.

When stable power and flow are achieved, a run of several days is performed noting any tendencies of the channel temperatures to drift. Close surveillance over the channels is maintained by exit thermocouples and by the pneumatic temperature monitoring (PTM) and burst slug detection (BSD)
systems. The individual channel exit gas temperatures reach 1050 F.

The charge machine is used to impose cyclic flow on the selected channels. Rates of change of temperature are limited to design levels. The elements are subjected to a predetermined number and range of cycles.

The reactor is shut down; the fuel assemblies are discharged from the selected channels and loaded into dry casks using the charge and service machines; the channels are reloaded and orifices reset for the full power, full flow condition. The irradiated elements are sent to ORNL for immediate hot cell examination.

14.5 Approach to Full Power - Normal System Temperatures

The following program is to verify experimentally the steady state values for TRI, TRO, and helium flow (Wg) with the system on manual control; to note disturbances in these parameters caused by changes in steam plant, reactor coolant system, and reactor conditions and to identify the general requirements for the automatic control system. The steam temperatures and flows associated with particular reactor power levels and coolant flows are established as dependent parameters.

The automatically programmed regime starts at 20% power (approximately 17 MwT); however, in the interest of proceeding on the convenient basis of 10 MwT per step, the level of 20 MwT is chosen for the first measurements. Tests at each subsequent 10-MwT step are similar to those performed at 20 MwT, except that the operating time at each step is extended as the power level is increased.

Before proceeding with the following tests, prior tests are evaluated in detail.

14.5.1 Tests up to 20 MwT

The entire plant except for the reactor, is placed in operation and the reactor coolant system brought to the 500 F isothermal state.

Using the approved procedures for operation in the start-up and intermediate ranges, the reactor power is raised to 20 MwT as indicated on the nuclear channels. The turbine is placed on initial pressure regulation at the 10-MwT level. Flow is held at 75% of full flow. TRO and TRI are compared to values previously obtained to verify that the reactor thermal power is 20 MwT. An approximate reactor coolant and steam system heat balance is performed if required.

Using the specified programs for TRO, TRI, and Wg versus power as a guide, the reactor coolant flow is reduced in steps of approximately 10% each, allowing the flows and temperatures to stabilize at each step. Close surveillance is maintained over all individual channel TRO's, as well as the performance of the BSD system, and the instrumented fuel columns.

As the reactor heats up, control rods are withdrawn to compensate for the fuel and moderator temperature rise; an additional compensation is necessary as xenon poison increases. Power is held constant at 20 MwT. Flow reduction is continued until the value prescribed for the 20-MwT level is reached. During each step a record is made of reactor coolant and steam flows and temperatures after TRI and TRO stabilize. Any discrepancies between the actual and predicted values are reviewed and corrective action taken if necessary. The reactor core and coolant system temperatures are higher than obtained at any previous time in the program.
Careful surveillance of reactor coolant purity is maintained as bake-out products evolve from the core and sleeves. General performance of the gas purification system is observed and a series of tests performed.

When stable operating conditions are obtained, a complete reactor coolant and steam system heat balance is made and a reactivity balance is calculated. Helium flow is now adjusted until the programmed TRO is obtained for the 20-MwT condition. Wg, TRI, and other reactor coolant temperatures, steam flows, and steam temperatures are recorded.

TRO is adjusted slightly above the design value; then slightly below the design value; at each condition, reactor coolant and steam temperatures and flows are recorded. The programmed TRO is then re-established.

The system is now perturbed using the steam plant and the reactor to impose transients. With blower speed constant, response of the system and new steady-state values of temperatures and steam flows are measured. The blower speed is then used as the perturbation, changing flows to balance points above and below the program line and recording time to obtain new steady-state temperatures and steam flows. Performance requirements for the automatic control system are deduced from this experiment and compared with values used in simulator studies.

The master power selector is calibrated for the 20 to 30 MwT range using the programmed TRO as the base parameter and compatible TRI and Wg values as measured at 20 Mwt and extrapolated to 30 MwT. A general checkout of the automatic control system is made.

The plant is placed on automatic control at 20 MwT. The perturbations previously performed on the system are again applied. The ability of the system to maintain desired conditions during operating transients at this power level is demonstrated.

14.5.2 Tests up to 30 MwT

The system is placed on manual control. The blower speed is advanced to the programmed value for the 30-MwT level and, using the pre-established rod program, reactor power is increased to this value, maintaining continuing surveillance over fuel, moderator, and reactor coolant temperatures. Essentially, the same operations are performed at this power level as were performed at 20 MwT. Calibration of the master power selector is extended to 40 MwT by extrapolation. At the end of the 30-MwT tests, the manual power selector is used to reduce power to 20 MwT and return power to 30 MwT at the maximum specified rates. Tendencies to overshoot and general performance of the system are evaluated.

14.5.3 Tests From 30 MwT to Full Power

The system is placed on manual operation at 40 MwT. Subsequent tests are similar to those performed at the previous 20 and 30-MwT levels. After these tests are finished, the reactor is stabilized on manual control at 40 MwT for the purpose of comparing in detail the performance at this condition to that obtained during the original 40-MwT run.

The combined temperature coefficient is again deduced by comparison of rod positions required to hold the reactor power constant at normal program temperatures and at reduced temperatures. These reduced temperatures are produced by increasing the blower flow to 75% at the maximum permissible rate and allowing temperatures to stabilize at this flow. Rod positions are compared with those recorded in the previous 40-MwT run.

By observing the equilibrium temperatures and different rates of change of
moderator temperatures and fuel temperatures (including sleeves) in the instrumented fuel assemblies, an attempt is made to deduce the separate fuel and moderator temperature coefficients.

After the stable run at 75% flow, 40 MwT, flow is reduced to the programmed value, again at maximum prescribed rates. The effect of the separate temperature coefficients is again checked and evaluated. The reactor is manually scrambled from the normal 40-MwT conditions. Previously installed equipment is used to record rod drop times, decay of flux, and other parameters of interest. Performance of the entire plant is evaluated with particular attention to rate-of-change of temperature in critical locations.

The reactor is again started and 40-MwT normal operation attained on manual control. Power is advanced to 50 MwT and calibrations, balances, temperature, and flow surveys and other operations carried out as described for the 20 and 30-MwT cases. Operations at the 60, 70, and nominal 80-MwT levels are carried out in like manner. At the 60-MwT and 80-MwT levels, the reactor is manually scrambled. As in the 40-MwT case, survey and analysis of performance is made of the transients following the scram. The final power level is determined by evaluation of temperatures as the nominal 80-MwT level is approached.

14.6 Tests After Attainment of Full Power

The previously described tests are based on the premise that time is saved by calibrating and partially verifying performance of the automatic control system during the approach to full power. An early appraisal of the system thus is possible. If difficulties arise, the program is continued on manual control alone, the intent being to attain nominal maximum power as early as safety and practicality permit. Wide-range load-following tests on the automatic control system are not performed, nor are any other tests done to show system response to unusual and abnormal situations prior to attainment of full power.

14.6.1 Control Rod Calibration Using Xenon Decay Technique

Reactor power is reduced to a critical condition near source level and the reactor is cooled to a nearly isothermal condition at approximately 500 F. During this experiment, the reactor temperature is held as near constant as possible by controlling blower speed and steam generator temperatures.

The reactor is held at steady-state near source level as xenon builds up and decays. The rod withdraw prohibit (RWP) on 75% blower flow is temporarily bypassed if necessary. Rod positions are observed and recorded as xenon poisoning changes. During xenon decay, the rods are inserted to hold the reactor critical. At stages the rods are withdrawn to a previously noted position and the positive period measured, thus measuring the rate of decay of xenon.

The xenon decay curve so produced is evaluated against the rod positions required to hold the reactor critical. Thus the rods are calibrated for this range. The values obtained are compared to the original rod calibrations and any differences resolved.

14.6.2 Load-Following Tests

These tests demonstrate the ability of the system to meet load swings which normally occur on an isolated electrical system or on a "peaking" plant in an interconnected system. The EGCR is, from the standpoint of load-following, a manually controlled plant in that the reactor coolant inlet and exit temperatures and reactor coolant flow are manually set on a master power selector. The master power selector is used to test the response of the system in a programmed series of tests ranging from 20 to 80 MwT. Tests are performed to
demonstrate ability of the system to hold TRI, TRO, and Wg to prescribed values in both power increase and decrease directions at maximum specified rates. Limitations on these rates are established by thermal stresses at specified points. To the extent possible, temperature changes at these points are verified for transient operation.

A preplanned program of experiments is carried out with the master power selector at several settings to demonstrate the response of the automatic control system in holding TRO, TRI, and Wg to specified levels against changes in steam demand including steam demands in excess of the master power selector setting. The program is accomplished in a series of small steps, utilizing the analogue studies as a guide; the objectives are to show that the plant operates safely and satisfactorily on both increasing and decreasing steam demands.

14.6.3 Other Tests

Operation of the EGCR is essentially a continuous test with principal emphasis on fuel performance. An important, but secondary, objective is to evaluate and, if necessary, improve operation of auxiliary equipment and systems in the design to provide a basis for future designs of high temperature gas-cooled systems. The tests may be conveniently grouped into fuel tests and equipment and systems tests. Only major tests are considered here; there are many additional tests not having particular significance from a hazards viewpoint.

a. Fuel Tests

1. Periodic Withdrawal and Examination of Fuel

   Fuel is discharged on a programmed basis from selected channels for examination and analyses in hot cells.

2. Special Tests

   The standard EGCR fuel element is uninstrumented. There are, however, four instrumented fuel assemblies. These assemblies perform the two principal functions of verifying calculational techniques for operation of the reactor and of monitoring the conditions in the channel. In the latter case, the assembly becomes an experimental unit and is operated to prescribed limits on cladding temperature based on direct measurement.

   Instrumented assembly channels become test channels for new fuel designs previously tested in out-of-pile rigs, capsule, and loop tests. Preliminary in-pile tests are performed under normal levels of flux and temperature but with single shortened capsules. Full-scale performance is determined in the EGCR.

3. Long-Term Burnup Tests

   Selected fuel elements are operated beyond normal exposures to develop information on fuel and cladding performance. This is done on a formally programmed basis. Among the important things to be learned in this operation is the fission gas release from the fuel and consequent pressure buildup.

4. Fuel Replacement

   As fuel elements are discharged from the core, they are replaced initially by elements of the same design. The initial charge in the reactor plus the 25% in spare elements are adequate to establish the
overall performance of the basic fuel design.

Subsequent channel loadings may include a variety of designs and cladding types, not necessarily restricted to metallic-clad elements. Prior to insertion of advanced fuel in the EGCR, the Technical Program Group evaluates the hazards implications of such a change, as described in Section 10.4 of this report.

Selected channels may be loaded with a given type of element. The reactor thus becomes a test bed for advanced fuel concepts with the testing program directed toward confirming and improving each design.

b. Equipment and Systems Tests

1. Plant Control and Safety Systems

In addition to the previously mentioned load-following tests, the system is tested to demonstrate performance for the following conditions:

(a) A single control rod drop while on automatic control

(b) Single rod runout

(c) Failure of a rod to fall on scram

(d) Response of system to fuel charging and discharging operations

(e) Response of system to flux scanning operations.

2. BSD System

(a) Tests to identify failed fuel elements

(b) Establishing criteria and program for removing failed fuel elements

(c) Observing effects of temperature and associated pressure changes on failed elements

(d) Mixed gas activity and deposition of fission products in reactor coolant system, blower seal water system, and other systems following fuel element failures

(e) Analyzing both recirculated coolant gas and coolant gas from the fuel channel containing the simulated failed fuel.

3. Charge and Service Machine Operations

(a) Charging and discharging operations under load

(b) Simulated failures of charge machine; supplementary operations by service machine

(c) Operations to verify ability to perform other design functions

(d) Wear surveillance (operating records and examination of parts) on machine components, particularly dry bearings.
4. **Fuel Handling**
   
   (a) Tests for handling of defective fuel assemblies
   
   (b) Dry handling of fuel assemblies
   
   (c) Testing of experimental storage holes.

5. **Shielding**
   
   (a) General shielding measurements at full power
   
   (b) Charge and service machine shielding tests during operations and while machines are fully charged with spent fuel.

6. **Reactor Coolant System**
   
   General surveillance program of coolant leakage to reduce leakage to economic minimum.

7. **Electrical System**
   
   Tests designed to simplify restart operations after momentary power loss.

8. **Purification System**
   
   Continuous testing including special tests following malfunctions which introduce contaminants into the reactor coolant or following deliberate injections of contaminants into the coolant.

9. **Flux Scanning System**
   
   **Bottom Dummy Orifice Settings**
   
   **Channel Exit Gas Thermocouple and PTM Systems**
   
   **Bulk Gas Flow-Measuring System**
   
   **Core Pressure Differential Instrumentation**
   
   **Instrumented Fuel Assemblies**
   
   **In-Core Thermocouples**

   A continuing series of tests is performed to verify and improve calculational techniques describing fuel, cladding, and graphite sleeve temperatures associated with plant performance as determined by the above items.

10. **Temperature Monitoring**
    
    (a) Vessel, temperature barrier, and nozzle temperature monitoring program to verify values used in design calculations of thermal stresses under steady-state and transient conditions.
    
    (b) Verify the adequacy of the shield cooling system by monitoring biological shield surface and internal temperatures.
15. NORMAL PLANT OPERATING PROCEDURES

15.1 General

Normal plant operating procedures discussed in this section include plant start-up, load increase, operation under load, load decrease, and plant shutdown. Operating procedures relating to special operations, such as fuel handling, waste disposal, decontamination, maintenance, and routine plant tests, are described in subsequent sections of this report.

During normal operation the power level is established by the reactor settings with the steam and electrical plant following the reactor. Electrical power delivery, up to the limit fixed by reactor thermal output, is adjusted by manipulation of the steam delivered to the turbine. Under these conditions, the reactor is subject only to those transients which originate from reactor control settings. The reactor is not subject to appreciable transients due to normal changes of electrical load. Operations during reactor transients caused by equipment malfunction or operator error are described in Section 16.

The structure and function of the EGCR operating organization and the training of the operating personnel are described in Sections 10 and 11 of this volume. The plant operations shift supervisor for each shift is responsible for the over-all operation of the plant, subject to the general supervision of the operating superintendent. He provides general supervision and technical assistance to the shift engineer who is directly responsible for the detailed operation of the plant. The shift engineer follows established operating procedures and practices to insure the safe and efficient operation of the plant. Directly responsible to the shift engineer are unit operators, aided by assistant unit operators, who perform the actual operating tasks. These employees are thoroughly trained in the theory and principles of plant operation. Training continues throughout the life of the plant to maintain a high degree of competency and to qualify new personnel. An instrument mechanic and a health physics technician on each shift provide specialized support to the Operations Group. Analytical laboratory services are available on the day and evening shifts, seven days per week.

Formal procedures are established for authorizing the operation of any plant system or equipment within specified limitations for either testing or regular use. These procedures require written authorization by the project manager or by his formally-delegated representative before such operations are begun. The authorization is issued to the operating superintendent only after all procedural requirements are met, and the project manager is satisfied that the operation can be conducted safely. The procedure requires certification by the appropriate staff members, including the technical program superintendent, the operating superintendent, and in many cases, the radiological health supervisor, that the equipment is ready for operation under the specified conditions and limitations. In all cases where specific approval by AEC (ORO) is required, this approval is obtained before the operation is authorized.

The authorization procedure includes assurance that the following prerequisites, where applicable, are satisfied:

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a. All equipment is inspected, prerequisite tests are completed, and the system is in safe and satisfactory condition for the specified operation.

b. All safety requirements are met and are thoroughly understood by the operators and supervisors.

c. Review and approval of the proposed operation by the Technical Program Group and Radiological Health Section includes, to the extent applicable:
   1. Over-all hazards control review and analysis.
   2. Inspection, check, and analysis of instrumentation, control and safety systems.
   3. Inspection and check of pressure-containing components.
   4. Check of step-by-step procedures and checklists with regard to sequences, timing, valve, control positions, etc.
   5. Radiological hazards surveillance and control.
   6. Reactor physics surveillance and control.
   7. Chemical laboratory control and materials surveillance.
   8. Cleanliness control.

d. All applicable operating procedures, checklists, instructions, and test procedures are completed, checked, approved, and formally issued.

e. Operating supervisors, shift engineers, and operators are thoroughly familiar with the system, equipment, and operating instructions, and meet all training requirements for the operation.

f. Operators and supervisors understand how to proceed in emergencies.

g. The effects of the new system or equipment operation or tests, in relation to existing operations, are anticipated and understood by all operating and test personnel.

h. All predictable indications of unsatisfactory or hazardous operation, their significance, and corrective actions are reviewed and understood by the operators and, where possible, are included in written instructions.

i. Clearance procedures for working on operating or de-energized equipment are established and thoroughly understood and enforced.

j. Definite lines of command are established and understood by all concerned.

k. Adequate systems of communication exist between the control room and all operating stations.

l. Adequate procedures are established for conveying information and instructions and transferring responsibilities at shift change.

m. All maintenance and test operations that can affect equipment operation are under the control of the operating superintendent.
n. All necessary operating supplies and auxiliary services are available.

o. Employees for supplementary or replacement duty are on call.

p. Adequate plant logs of operating events are kept by the control operators and operating supervisors, showing the time of each operation and significant occurrence. Test engineers keep detailed logs of test operations and results.

Most operating instructions are in a written form to furnish operating personnel with guidelines for plant operation. Step-by-step system and integrated plant operating procedures are given in the EGCR Operating Manual. Checklists are used for particularly important or complicated operations, e.g., plant start-up. Additional operating instructions are authorized by the operating superintendent and are issued by him in the form of Standard Operating Practices, Operating Instruction Letters, and Day Orders. The Standard Operating Practices include instructions for miscellaneous plant-wide measures, such as clearance procedures and plant security procedures. Operating Instruction Letters contain long-term instructions concerning the detailed operation of the plant, which may be subject to more frequent change than operating procedures. Day Orders are for temporary day-to-day instructions. Operating instructions for plant tests and experiments, other than those described in the Operating Manual and Hazards Summary Report, are issued by the operating superintendent as the need arises.

Operating records maintained include daily operating logs, recorder charts, graphs, equipment outages, and other records deemed necessary by the operating superintendent.

15.2 Plant Start-up

Plant start-up operations pertain to those actions necessary to raise the reactor power level from the source power level, where all control rods are fully inserted, to a power level corresponding to a reactor thermal output of 17 MwT. This is accomplished manually in two stages; start-up range and intermediate range. The start-up range is from a count rate equal to or greater than 5 counts/sec to 1 MwT, inclusive. The intermediate range is from 1 through 17 MwT.

Reactor start-up is initiated upon authorization from the plant operations shift supervisor on duty. A master checklist is used to outline the necessary steps to be taken and the sequence of action for the prestart-up and start-up phases. Upon completion of the prestart-up phase, the plant operations shift supervisor on duty must check and approve the master checklist before the reactor control rods are withdrawn.

Prior to withdrawing control rods, the reactor and associated coolant circuits are heated by using the heat of compression from the reactor coolant blowers. The time required to heat the plant to equilibrium conditions and to reach nominal reactor coolant inlet temperature depends upon the initial temperature and the limiting rates of temperature rise. During this period, the steam plant is placed in operation to remove heat at a rate commensurate with the operating limitations. When the reactor coolant inlet temperature is at the desired level, the control rods are withdrawn and reactor operation in the start-up range is initiated.

Limits on allowed operations are imposed by selection of the control range. When the start-up range has been chosen, up to 12 control rods may be selected for gang operation. Over-all safety system circuitry is described in Section 7.1 and Table 7.1.3 of Volume I. Selection of the start-up range engages the following special control and safety circuits:
a. More than or equal to 1 MwT rod withdrawal prohibit (RWP)
b. More than or equal to 3 MwT scram
c. Less than 75% flow RWP
d. More than 12 rods selected RWP
e. Count rate less than or equal to 5 counts/sec RWP

Coolant flow is maintained at greater than 75% full load flow. Reactor power is increased manually by incremental gang withdrawal.

When the reactor power approaches 1 MwT, operations are transferred to the intermediate range. Selection of the intermediate range engages the following special safety and control circuits:

a. Reactor coolant pressure less than or equal to 250 psig RWP
b. Less than 75% flow RWP
c. More than 1 rod selected RWP
d. Reactor coolant pressure less than or equal to 225 psig scram

Coolant flow is kept at 75% full load flow. Reactor power is increased by the incremental withdrawal of one control rod at a time. Steam is dumped to the shutdown condenser until approximately 5% thermal power is attained. The main condenser circulating water pumps are then started and steam is admitted to the turbine generator. Sufficient turbine generator speed is maintained to establish seals and maintain main condenser vacuum by means of the integral air ejector system. Excess steam is then dumped to the main condenser without overheating the turbine exhaust shell. Rod withdrawal is continued and excess steam is dumped to the main condenser until 17 MwT is reached. Shutdown equipment is stopped and placed on automatic stand-by.

At this power level, operations are transferred to the power-manual range. Selection of the range engages the following special safety and control circuits:

a. Flux less than 17 MwT RWP
b. More than 9 control rods selected RWP

Reactor coolant flow is reduced to 20% of full load while reactor power level is held constant. Approximately four hours are required for this operation due to the limitations on the rate of change of reactor coolant outlet temperature. Reactor coolant and steam conditions are established at normal operating levels corresponding to 20% reactor power. A maximum of nine control rods are withdrawn in this power-manual range.

The flux-minus-flow scram is the safety limit most likely to be approached during this transition. The steam temperature increases during this period to its normal value and is held essentially constant during further load increases. The turbine generator is brought up to operating frequency under speed control, synchronized, and loaded to a point approximating station demand. The initial pressure controller is placed in service and the turbine generator loaded by ramping up the turbine load limiter until no steam is being dumped. Ramping up of the turbine load limiter is continued past the point where no further load increase is produced. At this point, the turbine is under initial pressure control and ready for normal reactor operation under load.
15.3 Load Increase

Normal reactor operation to increase the load applies to increasing the power level from an initial steady-state value to a higher steady-state value with the power levels kept within the range of 17 to 85 MwT. The new power level is established with the turbine governor on either initial pressure control or on speed control. Load increases are made in either the power-auto or the power-manual mode.

Above the 17 MwT flux level in the power-auto range, the power level is increased simply by setting the master power level controller to the desired power position. A maximum of five rods are selected for power-auto operation. This controller automatically adjusts the set points for reactor coolant flow and temperature, and controls coolant blower speed and control rod withdrawal until the desired power level is reached.

As control rods are withdrawn, the reactor power level increases. The coolant temperature difference across the reactor increases slightly as reactor power level increases, but for a given reactor power output, it remains nearly constant. Reactor coolant inlet and reactor coolant outlet temperatures increase about 60 and 100 F, respectively, as reactor power goes from 17 to 85 MwT. Coolant flow is almost directly proportional to reactor power. Superheated steam temperature remains nearly constant for normal reactor operation at loads greater than 17 MwT.

Rapid repositioning of the master power level controller cannot result in a hazardous condition due to a built-in rate limiter. In the event of a failure of the rate-limiting feature of the master power level controller, the reactor safety system through a high reactor coolant outlet temperature of flux-minus-flow scram protects the reactor if a rapid change in power level is attempted.

Load increases are also made under manual control. The reactor operator positions the control rods and controls blower speed to establish the new reactor power level. The sequence of operations is first to increase blower speed, incrementally, followed by incremental adjustment of control rods. This sequence keeps the reactor gas outlet temperature and the rate of change in reactor gas outlet temperature within allowable limits during the load increase. Manual increase in reactor power is effected in small increments so that the desired total load increase is attained within the design limitations. During a load increase under manual reactor operation, close coordination of reactor operation with that of the steam plant and with the reactor plant auxiliaries, is maintained.

15.4 Operation Under Load

Normal reactor operation under load within the power range of 17 to 85 MwT is under automatic control. Operation is primarily at fixed load with the automatic controls acting to compensate for deviations from a preselected set point. Manual control is also possible within the power range. Whether reactor operation is under automatic or manual control, the safety system is adequate to prevent hazardous conditions resulting from either unsafe operating procedures or equipment malfunction.

Automatic operation of the reactor is initiated from any steady-state condition within the power range. Transfer from manual to automatic operation consists essentially of manually adjusting the controlling plant parameters to agree with the set points on the master power level controller and placing the controller in service. Close agreement between the actual parameters and the set points results in a smooth transition when the transfer is effected. The set points on the master power level controller are adjusted simultaneously from a single control.
The master power level controller directs the selected control rods to maintain the reactor coolant inlet temperature at a programmed value commensurate with power level. It also directs adjustments in blower speed to maintain the reactor coolant outlet temperature and reactor coolant flow at their programmed values corresponding to power level. These parameters are displayed and recorded in the main control room. Settings of the three master power level controller setpoints are also displayed in the main control room. Comparative readings between these two instrument groups are of major interest to the operator when adjusting the plant prior to placing the automatic control in service and while the reactor is under automatic control. Visible and audible annunciators warn of abnormal operating conditions.

For manual operation, the control room operator has at his option remote manual manipulation of control rods and remote manual control of blower speed and coolant flow. The control rods are adjusted to maintain a desired reactor coolant inlet temperature, and blower speed and coolant flow are adjusted to maintain a reactor coolant outlet temperature commensurate with the power level. Indicators and recorders provide the operator with the data on operating conditions necessary to maintain normal reactor operation. Visible and audible annunciators give the operator indications of abnormal conditions.

The steam plant controls permit operation of the turbine-generator either under initial steam pressure control or under speed control, depending upon plant requirements.

Under normal operating conditions, the turbine generator is connected to the electrical network and is operated under initial steam pressure regulation. This adjusts the position of the turbine control valves so that substantially constant steam pressure is maintained in the steam supply header. The turbine governor synchronizing device is set above the normal turbine operating range, and the turbine speed is determined by the network frequency. Therefore, when the turbine-generator is connected to the electrical network, it uses all the steam supplied and there is no steam dumping. Should the electrical load be lost when the turbine-generator is under initial steam pressure regulation, the regulation automatically shifts to speed control.

When the EGCR plant is isolated from the network and the turbine-generator is the source of site power, the turbine-generator is under speed governor control, the reactor is operated at constant power level, and the dump valve operates to prevent excessive steam pressure. The reactor power level is adjusted to just above the new load demand unless an experiment is in progress which would require constant reactor power. This operation is discussed under Section 16.

15.5 Load Decrease

A reactor load decrease involves essentially the same operations as a load increase. This applies to changing from a steady-state power level to a lower steady-state power level.

With the reactor power between 17 and 85 MwT and in the power-auto range, a load reduction is accomplished by resetting the master power level controller in increments to the desired power level. The control rods are inserted and the blower speed reduced at the predetermined rate until the power level is reached. The reactor is stabilized at this new power level before resetting of the master power level controller.

Reactor power level may also be changed manually. The operator inserts single control rods with individual rod switches or those selected for gang operation with the gang switch in desired increments. Blower speed is then reduced to a corresponding amount with special attention made to avoid a flux-minus-flow scram.
An electrical load change is accomplished with the turbine on either initial pressure regulation or speed governor control. If the turbine is on initial pressure regulation, the steam valves close and the electrical output is reduced by an amount necessary to maintain the desired steam pressure at the initial pressure regulator. If the turbine is on speed-governor control, the load is reduced by reducing the governor setting with either the load limit or the speed changer control by amounts necessary to maintain steam pressure at \(1250\) psig. If the turbine is on speed control and the steam dump valve is open, reactor power may be reduced without changing electrical load until the dump valve closes.

15.6 Plant Shutdown

During a normal plant shutdown, the turbine-generator is on initial steam pressure control until \(20\%\) load is reached; below this, it is on speed control. Following either a plant shutdown or a reactor scram, the plant is operated to remove reactor decay heat. After the control rods are fully inserted, blower speed is adjusted manually to provide cooling of the reactor core and associated components at a pre-determined cooling rate. This type of blower adjustment is continued until the reactor coolant outlet temperature reaches the final desired value. Thereafter, as long as practical, the blowers are operated to maintain a constant reactor coolant inlet temperature.

15.7 Control Rod Programming and Fuel Management

During normal operation of the reactor plant, one of the basic programs carried out by the operator is to establish control rod patterns, and fuel programming.

Control rod placement and fuel movements within the core are carried out by plant operators in accordance with instructions issued by the operating superintendent. Such instructions are based on recommendations issued by the Reactor Physics Staff and approved by the technical program superintendent. A fuel handling order (FHO) is issued and processed as described in Section 18.1. Fuel accountability procedures to be followed are outlined in Section 18.12.

Control rod programming and fuel loading operations are directed at limiting the maximum nominal fuel element clad temperatures. This objective is achieved during operation by limiting the radial peak-to-average power ratio to \(1.35\) when the reactor is operating at full power. At the same time, the maximum axial peak-to-average power ratio is maintained below \(2.0\) at full power. If the radial peak-to-average power ratio is limited to \(1.35\) when the reactor is operating at full power, with proper fuel channel orifice adjustment, the blowers are capable of providing sufficient coolant flow to accommodate the heat generation in each fuel channel. If the axial peak-to-average power ratio is limited to \(2.0\) in the maximum power channel when the reactor is operating at full power, the maximum nominal fuel element clad temperature is not exceeded.

Full insertion of control rods near the center of the core is used to maintain the desired radial power distribution. The axial power distribution is primarily controlled by partial insertion of rods into the core. The following method is used to provide control over the power distribution while achieving the wide range of reactivity control required throughout the life of the initial core. Desired ranges of reactivity control are obtained by full insertion of selected rods near the center of the core. Continuity in the range of control is obtained by slight movement of a bank of rods selected from those rods which are not fully inserted.

A possible sequence of control rod operation to obtain reactivity control throughout the range of \(k_{\text{excess}}\) of \(0.092\) to \(0.01\), which is encountered in the initial core while at operating temperature with equilibrium xenon and samarium,
is shown in Table 15.7. Control rod identification is as shown in Figure 4.7.5.2 of Volume I.

TABLE 15.7
A Possible Program of Control Rod Operation for the First Core

<table>
<thead>
<tr>
<th>Desired Range of Control (Δk)</th>
<th>Fully Inserted Rods</th>
<th>Δk Held by Fully Inserted Rods</th>
<th>Maximum Δk Required in Bank</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.092 - 0.073</td>
<td>G O T M L P</td>
<td>0.073</td>
<td>0.019</td>
</tr>
<tr>
<td>0.073 - 0.056</td>
<td>G O T M</td>
<td>0.056</td>
<td>0.017</td>
</tr>
<tr>
<td>0.056 - 0.030</td>
<td>M O</td>
<td>0.030</td>
<td>0.026</td>
</tr>
<tr>
<td>0.030 - 0.014</td>
<td>M</td>
<td>0.014</td>
<td>0.016</td>
</tr>
<tr>
<td>0.014 - 0.010</td>
<td>None</td>
<td>0.000</td>
<td>0.014</td>
</tr>
</tbody>
</table>

Table 15.7 indicates that in the initial core six control rods (G, O, T, M, L, and P) are fully inserted. These rods have a worth of 0.073Δk, while the remaining rods, acting as a bank, control a maximum Δk of 0.019. After the initial core burnup reduces kexcess from 0.092 to 0.073, the operator withdraws control rods L and P while compensating for this withdrawal by insertion of all partially inserted rods as a bank. Rods G, O, T, and M are the only fully inserted rods. This group of four rods is worth 0.056Δk, and the remaining rods, operating as a bank, control a maximum Δk of 0.017. As core burnup continues, the different patterns of fully inserted rods are changed so that the bank is never required to control a Δk of more than approximately 0.025. As long as the bank controls less than 0.025Δk, the required depth of insertion of the bank is less than approximately 62 in., which results in the axial peak-to-average power ratio being less than 2.0.

A number of different rod programs are possible. The one given in Table 15.7 is illustrative. It is one which minimizes the required depth of bank insertion. If excessive orifice adjustment is required by this program, then fewer configurations may be used at the expense of deeper bank insertion by incorporating other pairs and triplets of rods fully inserted.

Charging, discharging, or relocating fuel within the reactor is accomplished after a fuel handling order is initiated by the Reactor Physics Staff. The basis for fuel movement into, out of, and within the core depends upon the fuel cycle and the previous burnup history of the fuel. For the initial EGCR core, with an enrichment of 2.46%, the fuel lifetime for the case in which no fuel is shifted, is 6600 Mwd/MT. For the case of continuous fuel repositioning such that all fuel assemblies achieve the same exposure, the computed lifetime is 8200 Mwd/MT. In each of these cases k is more than or equal to 1.01. The exposure which is achieved during operation of the initial core depends upon the details of fuel management and control rod programming. The feed enrichment which is required to obtain 10,000 Mwd/MT in the equilibrium cycle is 2.25%.

Typical refueling procedures for the equilibrium core consist of loading fresh fuel into a channel in an outer region of the core, shifting the fuel assemblies from this channel into a central channel, and discharging the fuel from the central channel. The replacement of fuel in the outer channel is made when the fuel reaches an exposure of 5000 Mwd/MT. As fuel is shifted into a central channel, the six fuel assemblies will be reversed in order in the channel from position (1, 2, 3, 4, 5, and 6) to (4, 5, 6, 1, 2, and 3). The average exposure of fuel located in the central region of the reactor is thus 7500 Mwd/MT and in the outer region 2500 Mwd/MT, and a degree of radial power flattening is achieved. The inversion of fuel assemblies in the channel flattens the axial exposure variation, which is beneficial to the axial power distribution with the control rods partially inserted.
The fuel management program for the initial core is based on requirements for good fuel cycle economics and efficient transition into the equilibrium cycle. In the ideal case, the range of exposures and spatial variation of fuel composition at end of core life are identical to that for the equilibrium core. The wide range of exposures required for this condition is difficult to achieve, and the desired spatial variation cannot be obtained without some loss of core life, since the placement of all of the highly irradiated fuel near the center of the core represents a condition of lower reactivity than that with a spatially uniform composition. The ideal condition cannot be approached without resorting to a program of fuel repositioning during the operation of the first core.

During operation of the first core and during the transition into and operation of the equilibrium core, the Reactor Physics Staff directs a program of flux measurements using the flux scanning system for the various control rod configurations and conditions of core burnup. These measurements are used to verify that the actual peak-to-average power ratios are within the prescribed limits and to ascertain the exposure of the fuel assemblies. A continuous record of the burnup status of the fuel assemblies is maintained. Recommendations to the operating superintendent for fuel relocation and replenishment and changes in control rod configuration are based on calculations performed by the Reactor Physics Staff.
16. EMERGENCY PLANT OPERATING PROCEDURES

16.1 General

Emergency plant operating procedures are instituted upon malfunction or faulty operation of plant components or systems. The general categories of accidents, for which emergency plant operations are established, are uncontrolled addition of reactivity, loss of flow, electrical system failure, loss of coolant, loss of utilities, accidents involving the steam system and plant auxiliaries, miscellaneous system failures, fire, and the maximum credible accident. Accident analyses for these accidents are described in Section 8, Volume I.

Various means of issuing operating instructions for normal operations are described in Section 15. In addition to these instructions, emergency procedures are provided as a guide for corrective actions during emergency plant operation. These actions are taken to protect the public and plant personnel and to minimize damage to plant equipment. The emergency procedures summarized in this section present essential principles of the recommended course of action rather than detailed procedures.

In the absence of established written procedures, all operator control actions and corrective measures (at local stations or in the main control room) are made after reporting the conditions and intended actions to the plant operations supervisor and receiving permission to proceed.

The final responsibility, on shift, for the safe operation of the plant rests with the plant operations supervisor. Close cooperation between the plant operations supervisor and the shift engineer is maintained so that each of these responsible persons is kept fully informed as to the current status of the plant.

16.2 Reactivity Accidents

Uncontrolled addition of reactivity can occur as a result of uncontrolled or inadvertent rod withdrawal, by improper handling of fuel assemblies or control rods, or by addition of steam to the core from a leak in a steam generator tube. The consequences of such accidents are described in Section 8.6, Volume I.

All of these possible accidents are characterized by an increase in reactor power level. In some cases, the reactor coolant outlet temperature increases, and if blowers are on automatic control, blower speed also increases in an attempt to maintain the reactor coolant outlet temperature at the design level.

The reactor is adequately protected for these accidents by either the negative temperature coefficient of reactivity or by safety system action. The time to initiate scram by safety system action varies, depending upon the rate of reactivity insertion which is influenced by the method of control being used at the time of the postulated accident.

Operator action is to insert rods to compensate for reactivity addition or to scram the reactor manually. Following this initial action, the cause of the uncontrolled reactivity addition is determined, and necessary repairs or
corrective actions are taken to restore the reactor to its original state prior to resuming operation.

16.3 Loss of Flow Accidents

16.3.1 General

Accidents involving loss or reduction of reactor coolant flow to a fuel channel could result from a temperature barrier failure, plugged fuel channel, closure of reactor coolant loop isolation valves, single blower failure, or power failure. Such accidents may occur either as a rapid or nearly instantaneous flow reduction or as a gradual flow reduction. Analyses of the loss of flow accidents are presented in Section 8.9, Volume I.

16.3.2 Temperature Barrier Failure

Failure of the temperature barrier may occur at the bellows seal located at its lower end and at the bellows seals located at each of the nozzle penetrations. As a result of such a failure, some of the coolant bypasses the core and flows directly into the top plenum of the vessel. Failures of the temperature barrier are discussed in Section 8.9.2, Volume I.

When the reactor is on automatic control, the decrease in reactor coolant outlet temperature resulting from the coolant bypass of the core is detected by the control system which automatically reduces coolant flow. The negative temperature coefficient causes a slight decrease in reactor power due to increased core temperatures even though the rod control system, which detects a small decrease in the reactor coolant inlet temperature, is causing the rods to be withdrawn. Whether the reactor is on manual or automatic control, increase in reactor core temperature counterbalances the other effects and reactor operation is stabilized at a negative flux-minus-flow condition. Indication that a temperature barrier failure has occurred is obtained by comparing the reactor coolant outlet temperature with individual channel coolant outlet temperature as recorded by the pneumatic temperature monitoring system and channel thermocouples.

Operator corrective actions vary with the amount of imbalance in the flux-minus-flow indication, and the change in deviation between the reactor coolant outlet temperature and the average fuel channel coolant outlet temperature. At full power operation, the normal deviation between the reactor coolant outlet temperature and the average fuel channel coolant outlet temperature is about 12 °F. As power level decreases, this deviation increases to about 55 °F at 20% power. When the reactor coolant outlet temperature, the average fuel channel coolant outlet temperature, or the normal deviation between these temperatures changes, a systematic check is made of all circuits, instruments, and equipment to determine the cause of the change. If the cause appears to be leakage of coolant around the core, the reactor power level is reduced under manual control so that the fuel channel coolant outlet temperatures at the reduced load are safely below design values. If necessary, the operator initiates a controlled shutdown of the reactor to check and determine the causes of changes and to establish corrective measures.

16.3.3 Plugged Fuel Channel

The partial or complete blockage of the coolant passages in a fuel channel causes that fuel channel temperature to increase and, depending on the extent of blockage, may result in the failure of one or more of the fuel elements in that channel. This accident is described in Section 8.9.3, Volume I. The fuel channel coolant outlet temperature rise is detected by either the pneumatic temperature monitoring (PTM) system or by the temperature monitoring system. Off-normal conditions are indicated by the PTM system by a prin.out in red or
by alarm. If one or more fuel elements fail, fission product release is detected by the burst slug detection (BSD) system or by the radiation monitors adjacent to the reactor coolant piping. The initial action by the operator is to reduce temperature in the affected channel. This is done by reducing power or scramming depending on the channel coolant outlet temperature. Following this action, the operator takes steps to determine the cause of the plugged channel and corrects the situation using the charge or service machine as required.

16.3.4 Inadvertent Closure of Reactor Coolant Loop Isolation Valves

The isolation valves in the reactor coolant loops may be inadvertently closed by the operator or by a failure in the control system. The consequences of valve closure are discussed in Sections 8.9.4 and 8.9.7, Volume I.

If the accident occurs due to a valve or control system failure such that a reactor coolant loop isolation valve closes, the blower in that loop probably surges in approximately 50 sec or less. Closure of the valve and subsequent surging of the blower results in a sudden loss of coolant flow in the affected loop, thus causing the reactor to scram on high flux-minus-flow or on high reactor coolant outlet temperature.

If the operator inadvertently isolates one or both loops, a reactor scram occurs immediately from the blower trip.

Corrective action by the operator is to attempt to stop the surging blower to minimize equipment damage and to ensure that at least one coolant loop is in service to remove the decay heat following the shutdown.

As soon as it is determined that the isolation is due to a failure or improper action, the isolation can be reversed by either returning the manual isolation switch to the "off" position or by turning the "override automatic isolation" switch to the "override" position, depending on the cause. Opening of either or both of the reactor coolant outlet isolation valves makes the vessel cooling system available for heat removal.

The cause of the isolation is determined and the necessary inspections and repairs completed before the reactor is restarted. If a blower surge is detected or suspected, necessary tests and inspections are performed to determine that the mechanical integrity of the blower is unimpaired.

16.3.5 Single Blower Failure

The complete failure of one of the two reactor coolant blowers sharply reduces the coolant flow through the reactor. A high flux-minus-flow and a high reactor coolant outlet temperature occurs almost immediately in the power range, causing the reactor to scram due to one of these conditions, whether or not the reactor is under automatic control. A blower breaker trip from any cause also initiates an automatic scram. Corrective action is to isolate the defective loop and then follow normal shutdown procedures for removal of decay heat with a single blower. The blower in the normal loop ramps down to 23% speed following the scram.

If the single blower failure accident occurs during reactor start-up or operation in the intermediate power range when the coolant flow through each of the blowers is approximately 75% of the full load flow, a rod withdrawal prohibit occurs and power level stabilizes. The operator manually shuts the reactor down.

Partial loss of flow through either of the blowers, due to faulty operation of the blower or some component part, also causes an increase in flux-minus-flow and reactor coolant outlet temperature. If the speed, pressure rise, flow,
vibration, or motor load of one blower changes relative to the other beyond normal operating or control differences, the operator takes corrective action. Since relative speeds of the two blowers cannot be manually controlled, the only course of action is to reduce blower speed. This may also require a reduction of power. Speed reduction continues until blowers stabilize or until speed reaches 23% and the reactor is shut down. If conditions do not improve, the reactor is scrambled. The scram action automatically ramps the blower speeds down at the maximum rate. The blower speed is not increased again until the cause of the disturbance is determined. It is not necessary to determine which blower is in trouble. It is sufficient to know that the two are not operating within normal limits.

16.3.6 Power Failure

Loss of normal plant power results in a coolant flow reduction through the core since both of the reactor coolant blowers are tripped. The loss of normal plant power results in a reactor scram.

Adequate core cooling is provided during a normal power failure by operating the vessel cooling compressors on emergency power or by natural convection. When the service machine is connected to the reactor pressure vessel, emergency cooling is required to cool the reactor nozzle to which the service machine is attached until flow is established by the vessel cooling compressors.

Upon indication of the loss of normal power accident, operator action is to check the automatic operation of the control and safety system to assure that the following actions take place and to take corrective action in the event automatic action does not occur:

a. Diesel generators start and supply emergency power
b. Vessel cooling compressor is operating
c. The inlet isolation valves in the reactor coolant loops close
d. The electromatic relief valves open to blow down the steam system
e. The shutdown heat removal system goes into operation.

Corrective actions initiated by the operators include all normal plant shutdown procedures, supplemented by frequent check of such items as steam drum level, water supply to the condensate and the demineralized water storage tanks, diesel fuel storage and flow to engines, temperatures and pressures of reactor coolant systems to assure adequate removal of the decay heat, and close inspection of emergency shutdown equipment to assure that it continues to function properly.

When normal power supply is restored and check-out of the plant is completed, the plant may be manually restarted following normal startup procedures.

16.4 Electrical System Failures

16.4.1 General

An accident involving loss of electrical power can result from a failure external to the EGCR, from a failure within the plant with the plant connected to the external network through the X-10 substation or isolated from the system. Sequential failures or operating conditions at the time of or arising from the failure may scram the reactor by safety system action. The plant-wide power failure accident is described in Section 16.3.6. Other general electrical system failures are discussed below.

16.4.2 Failure External to Plant

Failure of the electrical transmission system external to the EGCR with the
plant generating power results in a partial loss of electric load on the turbine-generator unit. The effects of this accident are: an increase in turbine-generator speed, and hence frequency, causing the automatic transfer of turbine control from the initial pressure regulator to the speed governor; the main steam pressure rises until the steam dump valve opens and all excess steam is discharged to the turbine condenser; the deaerator steam supply control system locks in to maintain the deaerator steam pressure existing at the time the turbine-generator is transferred to governor speed control so that the temperature of the feedwater to the steam generators remains essentially constant regardless of decrease in the turbine load. The change in reactor thermal output resulting from this loss of power load is thus minimized and no scram or shutdown action is required. If warranted by plant operating conditions, the reactor power output is decreased under manual control at the discretion of the shift engineer.

16.4.3 Failure Within Plant

The design of the plant electrical distribution system is such that a 13.8 kv bus fault or loss of station generating capability with the plant isolated are the only failures within the plant that can result in a plant-wide power failure. This accident is described in Section 16.3.6. Most failures cause a partial outage resulting in a part of the plant equipment becoming inoperative. All critical equipment is duplicated and, upon loss of its normal power supply, is supplied from emergency diesel-generators or from the failure-free system. Thus, in general, corrective action by the operator is to insure that the automatic operations occur properly, to locate the cause of the failure, and to initiate repair measures. Separate emergency procedures are written as operating instructions for each failure of important plant equipment that can be credibly postulated.

16.4.4 Loss of Failure-Free Power Supply

Complete loss of failure-free power is not considered credible because it requires the simultaneous failure of two independent electrical circuits. The system consists of a 120-cell, 250 v battery, two separate rectifier chargers, two identical motor-generator sets with separate controllers, separate circuit breakers, separate feeders, and separate supply buses. If one-half of the failure-free power supply is interrupted, the load is manually transferred to the operating portion of the system. Corrective action is to institute normal plant shutdown and repair the fault.

16.4.5 Loss of Main Generator

The failure of the station turbine-generator unit results in a partial or complete power failure. The type of accident depends upon whether the plant is connected to or isolated from the external electrical network when the failure occurs.

Failure of the station generated power while the plant is connected to the external electrical network does not result in interruption of plant power. The generator failure trips the steam turbine stop valve. The main steam pressure then increases until the steam dump control valve opens and steam is dumped to the main condenser. If the cause of the trip-out is such that it can be corrected without discontinuing steam dump, then the reactor operation is continued without change until the turbine-generator unit is brought up to speed and loaded. Corrective action, if shutdown of the turbine-generator unit is required, is to reduce manually the reactor power until the turbine-generator can be returned to service after effecting the necessary repairs. If repairs are extensive, there is a normal shutdown of the plant.
Failure of the station generated power, when the plant is not connected to the external electrical network, results in interruption of plant power. This plant power failure scrams the reactor. The diesel generators automatically supply emergency power. The failure-free electrical systems continue to operate. Corrective action is to locate and repair the cause of station generator failure.

16.5 **Loss of Coolant Accidents**

16.5.1 **General**

The loss of coolant accident may occur as a result of an inadvertent opening or a rupture of the reactor coolant system. Depending on the size of the opening and its location, the depressurization of the reactor coolant system may be quite rapid or it may be relatively slow. The consequences of loss of coolant accidents are described in Section 8.10, Volume I.

The change in core cooling effectiveness due to loss of coolant through a small opening, and the consequent slow decrease in coolant system pressure is relatively insignificant. The system design is such that check valves or automatically operated valves close to limit or stop the coolant loss in the event of piping failures in auxiliary systems connected to the reactor coolant system. In addition, various remotely and locally operated hand valves are available to the operator for isolation purposes. In general, upon identification of an accident of this type, corrective action by the operator is to shut down the reactor, evacuate personnel from the reactor building if there is evidence of excessive airborne radioactivity, provide shutdown cooling, locate and isolate the failure.

Failure of thermocouple, BSD, flux scanning, or instrumentation penetrations may occur and may be located such that isolation cannot be accomplished. An accident of this nature results in a slow depressurization of the reactor coolant system.

During a loss of coolant accident, the reactor scrams on low pressure at 225 psig, and the reactor building is isolated.

16.5.2 **Reactor Coolant System Relief Valve Failure**

Three pairs of relief valves are provided to protect the reactor coolant system against overpressure, one pair for each steam generator and one pair for the reactor pressure vessel. The outlets from each pair of relief valves tie into independent discharge headers. Each header is fitted with an automatically actuated shutoff valve, interlocked so that either but not both of the headers may be isolated from the filter system. A defective relief valve is automatically isolated on low coolant system pressure at 270 psig.

Loss of coolant could result from excessive leakage from the valves, from a valve failing open, or as a result of failure of the valve to close following relief valve blowdown. Operator action for each of these cases varies as described below.

Relief valves remain closed during all normal operations of the plant. When a relief valve opens, the discharge from the valve passes through filters before venting to the atmosphere. Rupture discs upstream of the filters seal the system against leakage to the atmosphere through the relief valve seats. Any such leakage is discharged to the helium recovery system. If the leakage rate exceeds the capacity of the helium recovery system, the high pressure alarm for that system is actuated followed by operation of its pressure relief valve which discharges to the atmosphere through a separate filter system. Corrective action is to maintain coolant system pressure by admission of helium from storage, locate and isolate the leaking valve, and initiate controlled shutdown of the reactor so that valve repairs or replacement can be made.
If a relief valve fails in the open position, system pressure decreases to 270 psig. At this pressure, the failed relief valve is isolated and overpressure protection is provided by the alternate set of relief valves. Corrective actions by the operator include reactor shutdown, replacement of the rupture disc, and initiation of repairs to or replacement of the failed relief valve.

If a relief valve opens due to high system pressure caused by steam entering the reactor coolant system and then fails to close following blowdown, system pressure decreases at a rate controlled by the steam inleakage to the system. If the steam inleakage is not sufficient to compensate for the blowdown, system pressure decreases to 270 psig.

At this pressure, the failed relief valve is isolated and subsequent actions are as described in Section 16.7.6.

16.5.3 Depressurization During Refueling Operation

The charge machine and its component piping become a part of the reactor coolant system when the machine is connected to the vessel, the shield plug removed, and helium from the vessel cooling compressors is being circulated through the charge machine into the reactor. A loss of coolant during refueling operations, from either the charge machine or from the reactor coolant system is indicated by a decrease in reactor coolant system pressure.

Failures associated with the charge machine or in the vessel cooling system could cause a loss of flow to the charge machine. Excess flow valves in the suction line to the vessel cooling compressors or the valves in the helium supply line to the charge machine may close, depending upon the location and size of the failure. Reduction in flow to the nozzle annuli or interiors causes a reactor scram. If the rupture is located upstream of the check valves, the reactor coolant system is not depressurized but pressure decreases to a level determined by the heat removal in the system. Failures that cannot be isolated result in the complete depressurization of the reactor coolant system.

Operator action is directed at preventing the complete failure of fuel assemblies in the channel being refueled by the charge machine. If the depressurization results in a loss of flow to the charge machine, the operator immediately scrams the reactor. If the rate of depressurization is slow enough that isolation has not automatically taken place, the operator scrams the reactor and then attempts to isolate the failure by closing valves in the supply line to the charge machine. The charge machine operator may be permitted to complete refueling operations if conditions within the reactor building are satisfactory. The plant operations supervisor determines if refueling operations should be completed based on the rate of coolant loss, the environmental conditions within the reactor building, and the time required to complete a refueling operation. Core cooling is maintained by operation of the reactor coolant blowers.

16.5.4 Reactor Coolant System Rupture

If the rate of coolant loss and consequent depressurization is slow following a coolant system rupture, then a normal plant shutdown can be effected before a reactor scram occurs on low pressure. The corrective actions are to initiate reactor shutdown, locate and isolate the rupture, assure adequate core cooling, and repair or replace the failed component.

If there is a rapid depressurization of the reactor following a major rupture in the coolant system, the reactor scrams on low pressure. Analysis of accidents of this nature are described in Sections 8.10.2 and 8.10.5, Volume I, and are further discussed in Section 16.10.
16.6 Loss of Utilities to Reactor Plant

There are backup systems for each of the main systems (water, air, and electrical power) supplying utility services to the reactor. These systems are described in Sections 6.11, 6.10, and 5.7, respectively, of Volume I.

16.6.1 Service Water System

The normal operation of the service water system could be interrupted by pump failure, by electrical failure, by plugging of the service water strainers, or by a system rupture. Pump failure is indicated by the automatic start-up of one or both stand-by pumps. Electrical failure is indicated by the automatic switching to the emergency power system. An alarm annunciates on high differential pressure across a service water strainer, and operation is continued with the other strainer on the line while the clogged strainer is being cleaned. System rupture is indicated by pressure readings. Depending upon the nature of the rupture, stand-by pumps automatically start or backup fire protection water is supplied.

Extensive instrumentation permits a failure in the service water system to be located readily. The corrective procedure calls for location, isolation, and repair of the system failure. When the service water system fails completely and the fire protection system is the sole source of service water for the plant, the corrective action is to initiate a controlled shutdown of the plant.

16.6.2 Fire Protection Water System

Since fire protection water is delivered by gravity flow from a large elevated storage tank, its initial delivery can be interrupted only by a fire protection water system rupture. Corrective action for such a rupture is to institute a normal plant shutdown, isolate, and repair the rupture while using portable equipment for fire protection until the system integrity is restored.

Tank level is maintained by two pumps, one electrically driven and the other driven by a gasoline engine. The pumps are automatically started in sequence by a falling water level in the tank. Simultaneous failure of both pumps is not considered credible.

16.6.3 Condenser Circulating Water

The circulating water system is subject to the same type of failures as the service water system. Operation may continue on one circulating water pump at reduced plant efficiency without change of reactor power. Loss of one circulating water pump is indicated by a decrease in supply pressure to the condenser and an increase in condenser back pressure. Plugging of the intake screens also initiates a screen high differential pressure alarm. Corrective action by the operator is to place the failed pump back in service as soon as possible or, in the case of plugged screens, rotate and wash the screens.

A system rupture or loss of both pumps results in a rapid increase in main condenser back pressure and the turbine-generator unit trips on low vacuum. Corrective action by the operator is to scram the reactor, reduce the steam pressure, and place the shutdown steam system in service.

16.6.4 Instrument Air

The instrument air system is subject to compressor failures, electrical supply failure, and system rupture. Failure of an operating compressor results in the automatic start-up of one of the stand-by compressors. Instrument air is supplied from the air receivers during the interval between interruption of normal power and cut-in of emergency power. The operator determines the cause of
failure and initiates repairs. Subsequent operation continues at the discretion of the shift engineer.

A small leak places an additional load on the compressors without a significant change in air system pressure. Only a rather large rupture, such as the rupture of a complete air receiver, can seriously reduce instrument air pressure when the compressors are operating. A large rupture is indicated by reduced instrument air pressure with all compressors in operation. The plant air system is isolated automatically on low instrument air pressure.

Only a piping or receiver rupture can cause a complete failure of the instrument air system. To prevent hazardous conditions from arising from such accidents, pneumatic controls and instruments throughout the plant have been arranged to fail in a safe condition. The reactor scrams almost immediately from high flux-minus-flow due to the automatic reset to the 20% flow position by the blower scoop tube positioners. Corrective action by the operator is to insure that the reactor is shut down and that decay heat removal continues.

16.6.5 Plant Air System

Failures in the plant or instrument air systems which result in excessive reduction of pressure, automatically isolate the plant air system. Failures in the plant air system do not require reactor shutdown. The operator determines the cause and nature of the failure and makes necessary repairs.

16.6.6 Ventilation System

Failures in the biological shield cooling system due to either exhaust fan, electrical supply or by filter failure are automatically corrected by starting the stand-by fan which operates on normal or emergency power.

Loss of air flow within the reactor building could cause an increase in the concentration of radioactive contamination in working areas. Local monitors and alarms are provided to alert personnel in the event activity concentrations exceed the settings for permissible levels. Alarms are also received in the main control room. Upon receipt of such alarms, the operator notifies the Radiological Health representative on duty and evaluates the necessity for evacuating personnel from the reactor building. Entry into the affected area is prohibited until the cause is determined and appropriate action is taken so as to safely make the necessary repairs.

Rupture of an air filter could release radioactive materials to the stack. The stack radiation monitors alarm and initiate containment shell isolation on excessive levels as described in Section 7.5.2, Volume I. The stand-by filter unit is placed in service and the failed unit replaced.

16.7 Accidents Involving Steam System and Plant Auxiliaries

16.7.1 General

Various accidents involving the steam system and its related auxiliaries are described in Section 8.11, Volume I. Where practicable, critical equipment and services are duplicated by installed spares or alternate sources of supply. The duplication makes for safe plant operation and permits a normal plant shutdown rather than a scram following failure of such equipment.

16.7.2 Minor Accidents

Several minor failures, as described in Section 8.11.1, Volume I, can occur in the steam plant and auxiliary equipment. None of these failures result in a
radiological hazard. Certain failures require the operator to shut down the reactor on a programmed basis. However, even if the operator fails to take corrective action, the reactor and personnel are adequately protected by automatic shutdown actions. Operating instructions are established for all these accidents that can be credibly postulated.

16.7.3 Loss of Feedwater

A complete loss of normal feedwater to both steam generators is improbable but may occur due to a piping failure in the common feedwater line, loss of both main feedwater pumps, or loss of both condensate pumps. Shutdown feedwater is always available to at least one steam generator using the shutdown feedwater system.

The automatic actions of the plant following a feedwater failure are described in Section 8.11.4, Volume I. These actions place the plant in a safe shutdown condition. Corrective action by the operator is to insure the safe shutdown of the plant by monitoring the automatic actions and supplementing them as required. This accident is further discussed in Section 16.7.8.

16.7.4 Failure of Steam Dump System (Main and Shutdown)

The main steam dump system operates automatically when the main steam pressure increases to 1280 psig, which reflects inability of the turbine to accept the existing total steam flow. The steam dump system includes pressure reducing and desuperheating equipment. The effects of failures in this system are described in Section 8.11.5, Volume I, and indicate that no radiological hazard results.

Indication of the malfunction is apparent to the operator so that corrective action is taken.

If the shutdown condenser or drain cooler fails during a shutdown, which is a remote possibility, the operator corrective action is to allow the steam pressure to increase until the main steam dump system can be placed in service. If the main steam dump is unavailable, the operator opens the electromagnetic relief valves to vent steam to the atmosphere and supplies water to the steam generators from the condensate and demineralized water storage tanks. The domestic water backup supply is used when required.

16.7.5 Turbine Failure (Mechanical)

This accident is described in Section 8.11.6, Volume I. If it is determined that a prolonged shutdown of the turbine is necessary, the operator initiates a controlled shutdown of the reactor.

16.7.6 Steam Generator Tube Failures

With the exception of those failures which occur concurrent to a system depressurization accident (Section 16.10), steam generator tube failures are characterized by an increase in the moisture level in the reactor coolant system, an increase in the reactor coolant system pressure and finally by the generation of hydrogen as the steam or water reacts with the graphite in the core.

The consequences of a steam generator tube failure vary greatly, depending upon the size of the failure. A small leak is characterized by a gradual buildup of moisture over a long period of time while a large failure is followed almost immediately by reactor scram and relief valve operation due to overpressure.
a. Initial Corrective Action

In all steam generator tube failures, the operator shuts down the reactor. The method used, as well as subsequent actions, is determined by the rate of moisture in leakage and pressure buildup. In cases of small inleakage, immediate actions are directed toward locating and isolating the failure from the rest of the reactor coolant system.

Numerous automatic actions may accompany such an incident, especially the more severe type of failures. In all cases the operator monitors these automatic actions to obtain information upon which subsequent actions are based. Possible automatic actions or initial corrective actions by the operator are discussed below.

1. Reactor Shutdown

In all cases of steam generator failure, the operator shuts down the reactor. In severe cases, the reactor is scrammed automatically on high coolant pressure. For less severe leaks, this is necessary to permit isolation of the unsound loop.

2. Reactor Coolant Blowers

In the event the operator has time to systematically shut down the reactor, he also ramps the running blowers down to 23% speed. The blower in the failed loop is stopped to permit loop isolation. If the reactor is scrammed, or if it scrams on overpressure, the blowers automatically ramp down to 23% speed.

3. Reactor Coolant System Relief Valves

The relief valves will protect the reactor coolant system from overpressure. Discharge from the relief valves is normally to the stack but may be diverted into the reactor building in the case of high activity.

4. Containment Shell Isolation

If there is sufficient activity in the discharge from the relief valves, the containment shell is automatically isolated. Containment shell isolation also causes the containment shell spray cooling system to operate. If these actions are not automatically actuated, the operator evaluates the necessity of manually isolating the reactor building. If the reactor building is isolated, all employees within the building are evacuated and entry is prohibited.

5. Vessel Cooling Compressors

If moisture level is high, the vessel cooling compressors may automatically shut down due to an increased coolant density which appears as an apparent excess flow in the system. If such automatic operation occurs, and if the service machine is connected to the reactor pressure vessel, emergency cooling to the service machine nozzle is initiated.

6. Steam System

If the plant is operating disconnected from the external grid at the time the steam generator tube failure occurs and the reactor is scrammed, a loss of power results and the steam system blows down to 135 psig.
If the plant is operating connected to the external grid, the operator initiates blowdown of the steam system through the electromatic relief valves, lowering steam pressure to 350 psig. This reduces the carryover into the reactor coolant system, and at the same time reduces temperatures throughout the reactor system complex. The main feedwater pump is used to supply feedwater since the shut down feedwater pumps can't develop sufficient head at this pressure.

b. Emergency Cooling to the Core

After taking those initial corrective actions indicated above, the operator takes actions directed at providing adequate cooling to the core, isolating the failed steam generator and preventing the formation of a hydrogen mixture that could cause combustion. The actions the operator takes depend upon the information available during the assessment period as well as during the period that initial corrective actions are taken.

1. Reactor Coolant Blower Operating

With a reactor coolant blower operating at 23% speed, there is adequate flow through the core to cool the core and to maintain the hydrogen concentration within acceptable limits. If the vessel cooling compressor is not operating and the service machine is connected, the operator maintains the emergency cooling to the service machine until the temperature in the upper plenum is reduced to 650 F. When this temperature is reached, flow to the service machine nozzle may be halted to conserve helium. Isolation of the failed steam generator is performed if the failure location is known. Operator action may be required to prevent or rectify automatic isolation of the sound loop due to moisture condensation (via actuation of the flood-level isolation system).

2. Reactor Coolant Blower Not Operating

If a reactor coolant blower is not operating immediately following the initiation of the accident, core cooling is provided by natural convection. The operator attempts to isolate the failed steam generator and then restart the blower in the good loop. If the blowers can be started, core cooling proceeds as previously described. However, if the service machine is attached and the emergency cooling helium supply is used up before the top plenum gas temperature is reduced below 650 F then the operator must start depressurizing the reactor coolant system. If the blower is not restarted within one hour, the operator prepares to depressurize the reactor coolant system. The failed steam generator is isolated at this time. Before depressurizing the reactor coolant system, the operator initiates operation of the emergency cooling loop utilizing the bypass in the loop. After the emergency cooling system is in operation, the reactor coolant system is depressurized.

When the reactor coolant system is at a reduced pressure, the vessel cooling compressors may be used to provide cooling. Therefore, prior to using the emergency cooling loop, the operator attempts core cooling with the vessel cooling compressors. When a vessel cooling compressor is used the source of steam must be cut off and steam is purged from the coolant system using helium or nitrogen.

If adequate cooling cannot be provided, the operator proceeds to utilize the emergency cooling loop. The reactor coolant system is drained, both loops are isolated, and nitrogen purge is introduced.
to the core. The emergency cooling loop is then used to cool the core. Steps taken prior to operating the emergency cooling system are described in Section 16.10.3. As purge gas is added to the emergency cooling loop, the operator maintains pressure at or below 24 psia by discharging through the coolant system vent valves.

16.7.7 Failure of a Main Steam System Safety Valve

Steam generator overpressure is adequately relieved by either the superheater outlet relief valve or the drum relief valve; simultaneous failure of these two valves to open on overpressure is not considered credible. Additional protection against overpressure is provided by the use of electromatic relief valves which operate at a pressure below the setting of the main steam system pressure relief valves and which can also be operated by remote manual actuation.

If the steam system pressure relief valves develop small leaks during normal plant operation, or subsequent to opening on overpressure, plant operation can be continued until it is convenient to make repairs. Operation with a small steam leak through the relief valves requires additional feedwater makeup, and the plant operates at reduced efficiency.

Failure of an electromatic relief valve or of a steam superheater outlet relief valve to close constitutes a type of steam system failure described in Section 16.7.8.

Failure of a steam drum relief valve to close reduces flow to the superheater section of the steam generator, thus increasing steam outlet temperature and, depending upon the power level when the incident occurs, reduces the steam pressure in the steam generator drum. Corrective action is to institute a normal plant shutdown, initiate normal reactor decay heat removal procedures, isolate the affected steam generator, and repair or replace the defective valve.

16.7.8 Steam System Piping Failures

Small leaks in the steam, condensate, and feedwater systems usually are readily visible. Further indication of such a condition is increased make-up to the feedwater system. Many such leaks can be isolated and repaired without shutting down the plant.

A large rupture in the steam system, the sudden opening of the steam dump valve, or any of the steam system safety valves, is accompanied by a sharp reduction in main steam pressure. Mass flow of steam from each steam generator is limited by excess flow valves in the steam lines from each steam generator. These valves close automatically due to excess flow following rupture of a steam line. Steam flow is then restricted by means of a flow limiting venturi bypassing the valves. Operator corrective actions are to locate and isolate the leak or rupture; initiate a controlled shutdown of the reactor, if it has not scrammed by low level in the steam drum; and make repairs to or replace the defective equipment. Normal shutdown and reactor decay heat removal procedures are followed unless the location of the rupture is such as to require venting the steam generated by reactor decay heat to the atmosphere rather than to the shutdown condenser system.

A large rupture in the condensate system is indicated by a decreasing level in the deaerator storage tank. Corrective actions are to initiate a reactor shutdown, activate the shutdown cooling system, and make repairs to or replace defective equipment. Failure of the operator to take prompt action causes the main feedwater pumps to trip out because of low level in the deaerator storage tank, and ultimately scrams the reactor on low level in the steam drum.
If there is a feedwater pipe rupture downstream of the feedwater regulating valve, part or all of the feedwater flow to that steam generator is lost. Indications of the rupture are increased feedwater flow and a lowering of the water level in the affected steam generator. Corrective actions are to initiate reactor shutdown, institute reactor decay heat removal procedures using the unaffected steam generator; to locate, isolate, and repair the rupture. Failure of the operator to take prompt action results in a reactor scram on low level in the steam drum. If there is a feedwater pipe rupture upstream of the feedwater regulating valve, part or all of the normal feedwater flow to both steam generators is lost and the water level in both steam drums is steadily lowered. The corrective actions are to shut down or scram the reactor, to initiate reactor decay heat removal procedures using the shutdown feedwater pumps to supply the steam generators, to locate, isolate, and repair the rupture.

16.8 Other System Failures

16.8.1 General

Failure or maloperation may result in accidents in systems other than those previously discussed. Those of importance are described in this section. They include failures in the control system, vessel cooling system, and the blower seal water system.

Duplication of critical equipment in these systems is, in general, provided to minimize the hazards associated with failures. Established procedures and administrative control reduce the probability of maloperation.

16.8.2 Control System Failures

Wherever appropriate the control systems are designed to assure a fail-safe or a fail-as-is action. Some control system failures cause the reactor to scram by safety system action. Periodic testing, inspection, and maintenance together with duplication of components insure the maximum practical integrity of the control systems. Accidents involving control system failures are described in Section 8.12.2, Volume I.

a. Reactor Rod Control Failure

In general, control failures in the rod control system result in a fail-safe action by inserting rods. However, the rod control system does not directly fail safe in the event of a shorted resistance temperature sensing unit. The rods selected for automatic operation (as many as five) are withdrawn in this case. If the operator recognizes this condition in time, the rods are removed from automatic and placed on manual control. A controlled shutdown may be required at the discretion of the plant operations supervisor. If no corrective action is taken, the reactor scrams due to high flux-minus-flow.

b. Reactor Blower Control Failure

Loss of blower control could cause the blowers to lose synchronization which then may cause blower surge. Upon detecting a blower control failure, the operator shuts the reactor down, trips the blowers, and corrects the cause of the failure. If operation of the blowers is required for shutdown cooling, the operator uses the emergency manual blower control system if available to bring the blowers into operation. The emergency manual control is independent of the normal blower control system.
c. **Steam Pressure Control Failures**

Failure of the initial pressure regulator on the steam turbine-generator unit or failure of the steam dump system is indicated by an increase or decrease in the main steam line pressure, depending on the type of failure, without a corresponding change in the established reactor thermal power output.

Failures of the initial pressure regulator which result in increasing steam pressure cause the steam dump system to operate automatically, discharging excess steam to the main condenser, and electric power generation and feedwater inlet temperature are lower than normal for the reactor thermal power level. If the reactor is on automatic control, the decrease in reactor coolant inlet temperature resulting from the reduced feedwater temperature is detected by the control system which repositions the control rods to restore the reactor coolant inlet temperature to its set point value. The blower speed is increased due to the rise in reactor coolant outlet temperature until the deviations from the set point values for coolant flow and outlet temperature are in balance. There is a small change in steam temperature, pressure, and flow. The reactor power level increases slightly to compensate for lowered feedwater temperature. If the reactor is on manual control, the reactor coolant flow is unchanged, but its inlet and outlet temperatures are below the set point values, and there is a small increase in power due to the negative temperature coefficient of the reactor. Thus, reactor operation is stabilized whether or not the reactor is on manual or automatic control when this type of failure occurs. Rising main steam pressure, due to failure of the steam dump valves to operate when required to relieve excess pressure, causes the electromatic relief valves to discharge to the atmosphere.

Steam pressure control failures which result in lowered steam pressures, whether due to operator error or equipment failure, are reflected in reduced inlet and outlet temperatures of the reactor coolant and lower steam temperatures. If the reactor is on automatic control, the reactor control system functions essentially as described above, except that the deviations from the set points for the reactor coolant flow and outlet temperature tend to be greater, and the amount of the change in reactor power varies significantly with the reactor thermal power level when the failure occurs. When the reactor is on manual control, the effects of this pressure reduction on reactor operation are similar to, but of greater magnitude than, those described above for rising steam pressure. The corrective action includes determination of the cause of steam pressure control failure and making repairs and adjustments. Since there is no change in the basic requirements for reactor safety as a result of this condition, i.e., operation at reduced steam pressures or feedwater temperatures, the operation of the reactor at the same or the adjusted power level continues until shutdown and repairs are scheduled.

d. **Steam Generator Level Control Failures**

Failure of the steam generator level control system results in steam drum water levels above or below normal, depending on the type of failure. Changes in drum level are indicated and recorded in the main control room, and both high and low level alarms are provided. The reactor scrams on low level in the steam drum.

Malfunction of the feedwater regulating valve and its control system, which results in reduced feedwater flow, lowers the water level in the steam generator drum, and sounds a low level alarm. If corrective action is not taken, a reactor scram results. Corrective actions by the
operator, following low drum level indication or alarm, are to restore and maintain drum level by remote manual operation of the bypass valve around the defective feedwater control valve, to isolate the defective valve, and to initiate repairs or replacement as required. Operation of the reactor plant may continue at the discretion of the plant operations supervisor. Failure of the operator to take prompt action results in a reactor scram on low level in the steam drum.

Malfunction of the feedwater regulating valve and its control system, which results in excessive feedwater flow, raises the water level in the steam generator drum and sounds a high level alarm. Corrective action is to manually reduce feedwater flow immediately. If the failure is such that the operator cannot reduce feedwater flow promptly, the two feedwater isolation valves to the defective steam generator automatically close to prevent further increase in the level. Corrective action by the operator is to manually restore flow to the isolated steam generator. Operation of the plant may continue at the discretion of the plant operations supervisor.

If there is no subsequent operator action following isolation, the drum level decreases and scram occurs on low level followed by complete isolation on low-low level. The sound coolant loop is then used for decay heat removal.

16.8.3 Vessel Cooling System Failure

The automatic start-up of the stand-by vessel cooling compressor signals the operator that the operating compressor has failed or has been shut down by a control failure. Corrective action is to institute a normal plant shutdown and to initiate repairs.

Complete loss of flow to the reactor vessel nozzle annuli, the reactor vessel nozzle internals, or a high temperature of the nozzle internals results in a reactor scram. Corrective action is to institute normal reactor decay heat removal procedures and to locate, isolate, and repair or replace the defective equipment.

If the charge machine is refueling a channel when all flow in the vessel cooling system is lost, damage to fuel assemblies in the channel being refueled may occur. Operator action is described in Section 16.5.3.

These accidents are discussed in Section 8.12.3, Volume I.

16.8.4 Blower Seal Water System Failure

The loss of seal water to all the units is an extremely remote possibility. Essentially, separate water supplies are provided to the reactor coolant blowers and vessel cooling compressors. Emergency electrical power backup is also provided to the blower seal water pumps. In addition, high pressure head tanks provide approximately a 30-min supply to all units following a loss of both blower seal water pumps. Low flow indication and alarm warns the operator of a failure in the supply to the high pressure head tanks.

Corrective action by the operator, upon identification of a partial or complete loss of seal water, is to shut down the reactor, apply static seals, locate the cause of the failure, and initiate the necessary repairs.

If no corrective action is taken, the blower seals eventually fail resulting in a slow depressurization of the reactor coolant system and a low pressure scram at 225 psig. Ample time is available for the operator to shut down the reactor before the low pressure scram occurs.
If the reactor coolant blowers and the vessel cooling compressors are lost due to such a failure, core cooling is maintained by natural convection with the system at pressure. If the service machine is connected at the time that the accident occurs, and the upper plenum temperature exceeds 650 F when the emergency helium supply is depleted, the operator depressurizes the system. When the system is depressurized, core cooling is maintained using the emergency cooling loop.

16.8.5 Leakage of Activity from Equipment

Monitoring equipment is located throughout the plant where gaseous or airborne activity can credibly be released. Excessive activity is detected by air monitors or by the stack monitor. When airborne activity is detected, employees are evacuated from the immediate vicinity and the Radiological Health Section is notified. A contamination survey of the area is made by the Radiological Health Section to locate the source of the activity, and followup action is initiated. Plant shutdown or containment shell isolation may be initiated, depending upon the amount of activity discharged.

In addition to the release of gaseous or airborne activity, liquid wastes may be accidentally discharged. Upon detection of such accidental discharges, the immediate vicinity is isolated and the Radiological Health Section is notified. Contamination and radiation zones are established to protect plant personnel and to prevent the spread of contamination. The cause of the leak is determined and necessary corrective actions are initiated. Faulty equipment is repaired and the area is decontaminated.

16.9 Fire

Because greater than normal fire preventive measures are incorporated in the EGCGR design, the possibility of fire damage is less than that which would be expected in a conventional electric power generation plant. Carbon dioxide blanket, water spray, and deluge systems are available to combat fires in certain strategic locations. Critical items have been duplicated and physically separated. In addition, the 13.8 kv and 2.4 kv switchgear, the normal and emergency load centers, the failure-free power system, and motor control centers 1, 2, 3, and 5 are enclosed by fire-retarding walls.

An extensive fire alarm system is installed with alarms throughout the plant. The fire alarm circuits of the Oak Ridge National Laboratory can be activated through this system for additional assistance in fighting a plant fire. Upon receipt of a call for assistance, the ORNL fire captain dispatches a firefighting unit to the EGCGR site. This unit is met by a public safety officer who directs them to the scene of the fire. The ORNL Fire Department mobile unit assists the EGCGR fire brigade as directed by the plant operations supervisor on duty.

The duty shift is utilized in forming the local emergency fire organization. In the event of a fire, the shift engineer in charge of a designated fire brigade, proceeds to the scene of the fire and supervises the fire-fighting operation. If the fire is such that it may compromise the safety of continued operation, the plant operations supervisor requires that the plant be shut down.

All plant operating and maintenance employees are thoroughly familiar with and trained in the location and use of the installed on-site fire-fighting equipment. Frequent drills and training periods emphasize the production of informed, versatile emergency teams that can effectively handle any fire that might arise.
The operating procedures followed for the maximum credible accident are the same as those for a severe loss-of-coolant accident. Initially, there is no indication that the loss-of-coolant accident results in the consequences that are uniquely associated with the maximum credible accident.

As in the case of loss-of-coolant accidents, actions by the operator are directed at obtaining adequate cooling to the core to prevent excessive graphite oxidation. In cases where steam generator failures are associated with the accident, it is also necessary to minimize hydrogen generation and to assure adequate dilution to prevent a combustion hazard. The maximum credible accident differs from other loss-of-coolant accidents previously described in that the MCA results in the maximum number of fuel failures which, in turn, releases large amounts of radioactivity. Because of this release, operator action is directed at minimizing the release to the environment by taking steps to assure containment integrity.

The identification and consequences of the maximum credible accident are described in Sections 8.14 and 9.8, Volume I.

16.10.1 Detection and Identification of the MCA

Loss-of-coolant accidents are immediately indicated to the operator by a reduction in system pressure. Reactor coolant system pressure decrease to 250 psig causes an alarm, and a further reduction to 225 psig causes a reactor scram and automatic isolation of the containment shell. Although these indications are not sufficient to detect and to identify the MCA, initial operator action is on the basis that the depressurization accident is of the MCA level. The only basic difference between the MCA and a severe loss-of-coolant accident lies in the number of fuel element failures and the release of radioactivity from the failed fuel elements. Therefore, to determine if an MCA condition exists, information is obtained from the various activity monitoring systems, including the burst slug detection (BSD) system, the monitors adjacent to the reactor coolant system piping, and the radiation monitors within the reactor building. If these instrument systems record high levels of activity, there is sufficient indication that the depressurization accident resulted in a significant number of fuel element failures and that subsequent actions must be based on the assumption that the consequences of the accident are similar to those described in Section 9.8, Volume I.

16.10.2 Initial Corrective Actions

Initial corrective actions are directed at assuring that automatic actions by the safety system take place and include the following manual operations if such are not initiated by the safety system.

a. Scram reactor

b. Isolate containment shell, check position lights in control room of all isolation valves to assure that all valves are closed

c. Sound evacuation alarm in reactor building, assist personnel at air lock if required

d. Initiate operation of containment shell spray cooling system

e. Isolate control room.

In addition to these corrective actions, the operator initiates the Emergency Plan (Section 16.11).
Following these initial corrective actions, the operator obtains information upon which subsequent actions are based. Included are the following:

a. Determine if a relatively large steam generator failure exists. This is detected by comparing steam and feedwater flow, by noting change in steam drum level, by blower performance and other means. Extremely large failures cause isolation of a reactor coolant loop. Determine if such isolation was caused by low-low drum level or flood level. A small steam generator failure may not be detectable without proceeding to the subsequent operations.

After obtaining information relating to steam generator failures, the steam system is immediately blown down to 135 psia.

b. Determine if the reactor coolant blowers are operating and, if operating, at what speed. If not operating, determine if power is available or if blower is in an isolated loop.

c. Determine if a vessel cooling compressor is operating.

d. Determine if the gas inside the containment shell is being recirculated by the shield cooling system. This is done by checking the position lights of valves and determining if the fan is operating.

e. Determine if the emergency backup helium to the service machine is valved off. If automatic action does not take place, manually valve off the helium flow to the service machine by remote manual actuation.

f. Determine the pressure and temperature within the reactor building. Temperature indication is also used as a guide in determining failure location if blowers are operating.

g. Determine the activity level within the reactor building.

h. Check temperatures and flows in the reactor coolant loops if one or more blowers are operating.

i. Obtain information pertaining to the failure location from helium leak detectors.

16.10.3 Core Emergency Cooling

Following a severe loss-of-coolant accident or the MCA, core cooling is required to prevent runaway graphite oxidation and excessive hydrogen generation, since either could lead to the release of excessive radioactivity if not adequately controlled. Adequate control is achieved by providing sufficient flow to the core.

The following general principles are used by the operator to provide emergency core cooling during the maximum credible accident:

a. Reactor Coolant Blowers Not Operating

If neither of the reactor coolant blowers continues to operate, the operator attempts to restart one of the reactor coolant blowers. The operator attempts to select the blower in the sound loop based on indications outlined in Section 16.10.2. However, if there is insufficient or inconclusive information upon which to make a selection, either blower may be selected. If a failed steam generator is automatically isolated, the blower in the sound loop is selected.
If one of the blowers is operable, blower speed is increased to provide adequate flow to the core. To obtain maximum core flow for a given flow from a blower, the reactor coolant isolation valves in the loop opposite to the operating blower are closed. These valves are closed after it is established that the blower operates, but prior to increasing blower speed. If either of the reactor coolant system isolation valves closes, core bypass is prevented and blower operation continues. When a reactor coolant blower is operated, a helium purge is initiated within three hours after the start of the accident.

To obtain adequate core flow without exceeding the pressure and temperature limitations established for the reactor building, blower speed is limited. To insure that there is adequate flow through the core, core pressure differential is measured and used as a guide to proper operation of the blowers. Blower speed is increased until the core pressure differential is 0.54 psi. After the steam system blow down is complete, blower speed is increased until the core pressure differential becomes 0.72 psi. Due to the several combinations of helium, air, and steam, which may be present, pressure differentials cannot be closely associated with blower speeds.

Information provided by measuring the pressure differential across the core is used in conjunction with gas temperature leaving the steam generator, reactor coolant inlet and outlet temperature, reactor building pressure and temperature, blower speed and steam generator behavior in ascertaining the acceptability of continued operation with a given blower at a selected speed. Therefore, during these operations, the operator obtains indications of the following important parameters:

1. Reactor building temperature--This is obtained on panel K-1-1 by selecting a temperature measurement from the blower motor rooms, the steam generator rooms, the charge or service machine rooms, the BSD room, the vessel cooling equipment room, or by measuring the temperature at the inlet to the shield cooling system.

2. Reactor building pressure.

3. Temperature of coolant leaving the steam generator.

4. Temperature from core graphite thermocouples and the instrumented fuel columns.

5. Reactor coolant inlet temperature.

6. Reactor coolant outlet temperature.

7. Drum level indication in the steam generator being used to remove heat.

8. Activity level within reactor building.

Operation continues until the operating blower fails or becomes unavailable. Until there is sufficient information to determine that the operating blower is in the sound loop or that there is no steam generator failure, it is necessary to monitor the gas temperature leaving the steam generator. This is to prevent operation of a blower in the failed loop after steam generator level is lost. If there is a steam generator failure and if the selected blower is in the loop with the failed steam generator, automatic isolation occurs when the low-low level is reached in the steam drum or when flood level is reached in the steam generator shell. When
automatic isolation occurs, the blower is tripped. The operator, at this time, opens the isolation valves and attempts to restart the blower in the sound loop. If the blower operates, speed is increased to 100% and core cooling is continued until the blower fails or until core temperatures are reduced to an acceptable level.

The operator drains the water from the shell side of the steam generator in the failed loop after that loop is isolated.

It is desirable to increase reactor coolant blower speed to achieve the maximum allowable flow through the core to minimize the consequences of the accident. If such flows cannot be achieved, adequate cooling is provided if the flows corresponding to the core pressure differentials indicated in Table 16.10.3 are obtainable.

**TABLE 16.10.3**

<table>
<thead>
<tr>
<th>Time (min)</th>
<th>( \Delta p ) (psi)</th>
<th>( \Delta p ) (in. of ( \text{H}_2\text{O} ))</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>0.14</td>
<td>3.8</td>
</tr>
<tr>
<td>30</td>
<td>0.45</td>
<td>12.6</td>
</tr>
<tr>
<td>60</td>
<td>0.54</td>
<td>15.0</td>
</tr>
</tbody>
</table>

If a single blower is operable with speeds capable of producing a core pressure differential equal to or greater than those shown in Table 16.10.3, no further action is required until the blower becomes unavailable. If these core pressure differentials cannot be achieved, the operator must satisfy the core cooling by taking action as described in Section 16.10.3.c or 16.10.3.d.

b. **One Reactor Coolant Blower Operating**

If one reactor coolant blower is operating immediately following the accident, the operator proceeds as described in Section 16.10.3.a.

c. **Both Reactor Coolant Blowers Operating**

If both reactor coolant blowers are initially operating following the accident, have ramped down to 23% speed, and are providing adequate flow, no further action is immediately required. Helium purge is started within three hours after the start of the accident. However, if adequate flow is not available, one blower is shut down before increasing the speed of the other blower. This is done to prevent possible surging of one of the blowers.

After one blower is shut down, the operator proceeds as described in Section 16.10.3.a until further action is required.

If single blower operation is attempted and required core pressure differential cannot be achieved from either blower, the operator attempts to achieve adequate core pressure differential by operating both blowers.

d. **No Reactor Coolant Blowes Operable**

If reactor coolant blowers are not operable, or if a blower or blowers cannot achieve and maintain those core pressure differentials designated in Table 16.10.3 for a period of three days, core cooling is effected using the emergency cooling loop.
If a reactor coolant blower operates for three days and then is not capable of maintaining core pressure differential, that blower is shut down and a vessel cooling compressor is operated.

e. Emergency Cooling System

If core cooling is not available from the reactor coolant blowers or from a vessel cooling compressor in accordance with actions described in Sections 16.10.3.a through d, the emergency cooling loop (ECL) is placed into operation. The following steps are taken to put the emergency cooling loop into operation:

1. Close isolation valves in both reactor coolant loops and in vessel cooling system.

2. Open ECL inlet line drain valves.

3. Start flow of circulating water to the heat exchangers in the ECL system.

4. Start one ECL compressor in the heat removal system with the bypass open.

5. Close inlet drain valves (these valves to be in open position for at least five minutes).

6. Open the ECL isolation valves. Close compressor bypass valve.

7. If the ECL is used prior to three days after the accident, start the nitrogen purge: 1000 scfm for 1-1/2 hr, 200 scfm until the end of three days. Limit pressure in ECL to 24 psia or less by opening valves (HCV-14-56A, HCV-14-57) provided for depressurizing the reactor coolant system.

16.10.4 Containment Shell Integrity

Operator action is directed at assuring the integrity of the containment shell for the duration of the accident. Initial actions are to assure the proper operation of the containment shell spray cooling system. This system is operated until temperature inside the containment shell is reduced to ambient. At this time, operation of the system is halted.

If the emergency cooling loop is operated during the first three days following the start of the accident, nitrogen purge is required to minimize graphite oxidation and to dilute hydrogen to an acceptable level.

Addition of nitrogen results in a slow increase in pressure inside the containment shell. Within one day after the start of the accident, controlled venting is initiated through the fission product removal system to the stack. This prevents overpressurizing the containment shell due to continuing nitrogen purge.

During operation of a reactor coolant blower, if pressure or temperature inside containment shell approach the design values (9 psig and 200 F), the operating reactor coolant blower is stopped. The containment shell spray cooling system then reduces the pressure and temperature. Core cooling is accomplished using the reactor coolant blower in the opposite loop or by using the ECL.

16.10.5 Fission Product Release and Control

During the accident, fission product release from fuel elements is minimized by taking those actions described in Section 16.10.3. Such actions reduce
temperature in the core, which results in a decrease in the release of activity from the fuel. In addition, by preventing excessive graphite oxidation, additional fuel element failures are prevented.

Fission product control is maintained by assuring containment shell integrity and by taking any required action to assure the continuous operation of the shield cooling system.

Purging operations previously described minimize further release of fission products by oxidation of $\text{UO}_2$.

16.10.6 Accident Evaluation and Control

Operator actions described in Sections 16.10.3 through 16.10.5 are required to minimize the consequences of the maximum credible accident. All equipment that is operating to remove heat from the core is placed under surveillance by monitoring information in the control room. Corrective actions are taken in the event subsequent failures occur.

Further actions include obtaining and recording information relating to the accident, evaluating the consequences of the accident, and, finally, initiating decontamination and maintenance operations. Actions are also taken to minimize the consequences of the MCA to the general public and to other AEC facilities in the area.

16.11 Emergency Plan

On-site emergency situations require operator actions that are directed at restoring and maintaining the reactor facility to a safe level. In addition to these actions, the consequences of the accident may require off-site actions carried out with the cooperation of other organizations in the area.

The EGCR Emergency Plan assures rapid and orderly control of credible accidents, including fires, explosions, personnel injuries, and releases of radioactive material. The Emergency Plan defines the emergency organization and its responsibilities, and specifies actions taken both at the site and off site.

In the event of an accident that leads to the release of excessive amounts of radioactivity, off-site actions may require control of traffic on both the highways and on the lake adjacent to the site. In addition, evacuation of areas may be justified. To carry out such operations effectively, the cooperation of numerous organizations is required. The Emergency Plan describes the means for notifying and the extent of assistance provided by the following organizations that provide assistance during some accidents:

a. Union Carbide Nuclear Corporation (ORNL)
b. AEC-ORO
c. U. S. Weather Bureau
d. U. S. Coast Guard
e. TVA Division of Reservoir Properties (Public Safety Officers)
f. Civil authorities.
17. PERIODIC PLANT TESTS AND INSPECTIONS

Prior to initial power operation and during the approach to full power operation, extensive tests are conducted on all equipment and systems to assure conformance with the design intent. Such tests are described in Sections 12 through 14 of this report. In addition to these tests, periodic tests and inspections are necessary to assure that these components or systems continue to operate as required, or are capable of proper operation for those credible accidents described in Volume I of the Hazards Summary Report.

Periodic plant tests and inspections are conducted at specified frequencies. In general, those items of equipment which are required for the safe operation of the plant are tested and inspected at more frequent intervals. The frequency is, however, influenced by such factors as information available during normal operation; the backup protection provided to the equipment; and the causes, probability, and consequences of credible failures. In those cases where the design is based on extrapolation of existing technology or where reasonable uncertainty exists in the design analysis, more frequent tests and inspection are required.

There are three general areas where testing or inspection is required. The first deals with the testing of equipment that is normally in operation. Such tests are primarily of a preventive maintenance type that deal with improving the continuity of operation of the component and therefore are not performed primarily to improve or satisfy safety requirements. This type of test is described as a part of the preventive maintenance program in Section 19.3.1. Included in this program is an annual test of all safety relief valves and an inspection of all pressure vessels. The second general area of testing includes those tests or inspections on stand-by equipment. Stand-by equipment and associated instruments and alarms are rotated in their use. This is done primarily to extend the lifetime of the equipment, but also serves to check operability and proper functioning of all stand-by equipment. The third general area deals with the tests or inspections on those components or complete systems that are specifically included in the plant design to either prevent or reduce the probability of a serious accident or to minimize the consequences of credible accidents.

Periodic plant tests and inspections are carried out to assure safe operation and to protect plant equipment from damage. The tests and inspections included in this section are generally limited only to the third category of tests—those that are considered necessary for the safe operation of the reactor. These tests and inspections are tabulated in Table 17. Frequencies designated in the table represent minimum acceptable frequencies, corresponding to maximum time periods, to assure safe operation of the reactor.
### TABLE 17

**Periodic Tests and Inspections**

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>a. Safety Syst...</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

The scope of these tests is similar to that of the pre-operational tests. The reactor safety system has built-in test circuitry permitting each nuclear instrument channel (not including master key switch and manual scram switch) to be tested independently and prohibits testing more than one at a time. The design permits the insertion of test signals during a subchannel test (exclusive of detectors) which simulates an alarm, rod withdrawal prohibit (RWP), and scram signal. During the test, the other two active subchannels are half-tripped; therefore, any abnormal situation in either sub-channel causes a scram.

The tests performed on the safety system fall into three categories herein designated as Types I, II, and III. Type I is a "go-no-go" test where the test signals injected are of scram intensity to determine whether or not the stage trips. Type II is a calibration test performed at the amplifier input to determine the output readings with varying inputs. Type III tests check the calibration and operability of individual components (including detectors, if possible) and the entire system using simulated signals other than those supplied by the test circuits. Included in these tests is a check on the failure action of system and components upon loss of instrument power (electric or pneumatic) and input signals to components.

1. **Alarms**

<table>
<thead>
<tr>
<th>Visible and audible alarm in main control room on approach to or existence of an unsafe plant condition</th>
<th>1. The test is run to verify the operation of the alarms when the following conditions exist:</th>
<th>Type I, weekly</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>(a) The start-up rate is equal to or greater than 4 decades per minute in the start-up range.</td>
<td>Type II, monthly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Type III, yearly</td>
</tr>
</tbody>
</table>
### TABLE 17 (continued)

**Periodic Tests and Inspections**

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>(b)</td>
<td></td>
<td>The count rate is equal to or greater than $10^3$ counts/sec in the start-up range.</td>
<td></td>
</tr>
<tr>
<td>(c)</td>
<td></td>
<td>The start-up rate is equal to or greater than 4 decades per minute in the intermediate range.</td>
<td></td>
</tr>
<tr>
<td>(d)</td>
<td></td>
<td>The power (flux) is equal to or greater than 1 Mw in the start-up range.</td>
<td></td>
</tr>
<tr>
<td>(e)</td>
<td></td>
<td>The outlet gas temperature is equal to or greater than 1065 F.</td>
<td></td>
</tr>
<tr>
<td>(f)</td>
<td></td>
<td>The reactor coolant pressure is equal to or less than 250 psig.</td>
<td></td>
</tr>
<tr>
<td>(g)</td>
<td></td>
<td>The flux-minus-flow difference is equal to or greater than 5%.</td>
<td></td>
</tr>
<tr>
<td>(h)</td>
<td></td>
<td>The reactor coolant pressure is equal to or greater than 320 psig.</td>
<td></td>
</tr>
<tr>
<td>(i)</td>
<td></td>
<td>The water level in steam generator C-1 drum is equal to or greater than 7 in. below normal.</td>
<td></td>
</tr>
<tr>
<td>(j)</td>
<td></td>
<td>The water level in steam generator C-2 drum is equal to or greater than 7 in. below normal.</td>
<td></td>
</tr>
<tr>
<td>(k)</td>
<td></td>
<td>The flow of helium to the nozzle annuli is equal to or less than 7500 lb/hr.</td>
<td></td>
</tr>
<tr>
<td>(l)</td>
<td></td>
<td>The flow of helium to the nozzle interior is equal to or less than 4000 lb/hr.</td>
<td></td>
</tr>
<tr>
<td>System or Component</td>
<td>Safety Function</td>
<td>Objective and Scope</td>
<td>Frequency</td>
</tr>
<tr>
<td>---------------------</td>
<td>----------------------------------</td>
<td>---------------------------------------------------------------------------------------------------------------</td>
<td>------------</td>
</tr>
<tr>
<td>2. Rod Withdraw</td>
<td>Prevents withdrawal of control</td>
<td>2. The test is run to verify that the movement of the control rods is prohibited if the following conditions exist:</td>
<td></td>
</tr>
<tr>
<td>Prohibits (RWP)</td>
<td>rods (RWP)</td>
<td>(a) The count rate is equal to or less than 5 counts/sec in the start-up range only.</td>
<td>Type I,</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(b) The flow is less than 75% of full flow in the intermediate and start-up ranges.</td>
<td>weekly,</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(c) The number of rods selected in the start-up range exceeds 12, intermediate range 1, power range automatic 5, and power range manual 9.</td>
<td>Type II,</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(d) The power is equal to or less than 17 Mw in the power range only.</td>
<td>monthly,</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(e) The rod upper limit switch of each individual rod is reached. (RWP applies to individual rod for this case only.)</td>
<td>Type III,</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(f) The reactor outlet temperature is equal to or greater than 1065 F.</td>
<td>yearly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(g) The power is equal to or greater than 1 Mw in the start-up range.</td>
<td></td>
</tr>
</tbody>
</table>

(m) The temperature of the helium to the nozzle interior is equal to or greater than 125 F.

(n) The motor overload trip on any one of 25 motor starters and circuits opens (4 separate alarms).
### TABLE 17 (continued)

**Periodic Tests and Inspections**

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>(h) The reactor coolant pressure is equal to or less than 250 psig in the start-up range only.</td>
<td>Type I, weekly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(i) The flux-minus-flow is equal to or greater than 5%.</td>
<td>Type II, monthly</td>
</tr>
<tr>
<td>3. Scram</td>
<td>Drops all rods into the core</td>
<td>The tests are run to verify that the scram circuit breaker functions when any one of the following conditions exist:</td>
<td>Type III, yearly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(a) The reactor master key switch is opened.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(b) Any one of the manual scram switches is closed. (Main control room, service machine control panel, and charge machine control panel.)</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(c) The external power supply is interrupted.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(d) The loss of either reactor coolant blower.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(e) The power is equal to or greater than 3 Mw in the start-up range.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(f) The outlet gas temperature is equal to or greater than 1075 F.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(g) The reactor coolant pressure is equal to or less than 225 psig.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(h) The reactor coolant pressure is equal to or greater than 325 psig.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(i) The flux-minus-flow signal is equal to or greater than 10% (flux greater than flow).</td>
<td></td>
</tr>
</tbody>
</table>
TABLE 17 (continued)

Periodic Tests and Inspections

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nuclear Instrumentation</td>
<td>Provides flux indications and signals to safety system.</td>
<td>The nuclear instrumentation consists of seven channels, two fission chamber channels (start-up range), two compensated ion chamber channels (intermediate range), and three uncompensated ion chamber channels (power range).</td>
<td>Yearly</td>
</tr>
<tr>
<td>1. Channels 1 and 2</td>
<td></td>
<td>The positioning mechanism is checked to verify correct position indication and functioning of limit switches. The electronics, alarm, interlocks, and readout of each channel are calibrated using simulated signals. The simulated signals are supplied from safety system test chassis No. 1, and are $10^4$ counts/sec</td>
<td>Monthly</td>
</tr>
</tbody>
</table>

(j) The drum water level in steam generator C-1 is equal to or less than 9 in. below normal. (Opens relief valve PSV-1-145 and opens supply to steam driven shutdown feedwater pump.)

(k) The drum water level in steam generator C-2 is equal to or less than 9 in. below normal. (Opens relief valve PSV-1-146 and opens supply to steam driven shutdown feedwater pump.)

(l) The nozzle annuli coolant flow is equal to or less than 6000 lb/hr.

(m) The nozzle interior coolant flow is equal to or less than 3000 lb/hr.

(n) The nozzle interior coolant gas temperature is equal to or greater than 150°F.
TABLE 17 (continued)

**Periodic Tests and Inspections**

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>2. Channels 3 and 4</td>
<td></td>
<td>for calibration of count rate meters and recorders and ramp input to calibrate period alarm, period RWP, and period recorder.</td>
<td>Monthly</td>
</tr>
<tr>
<td>3. Channels 5, 6, and 7</td>
<td></td>
<td>2. The electronics, interlocks, and readout of each channel are calibrated using simulated signals. The simulated signals are supplied from safety system test chassis No. 1 and consist of ramp input to calibrate period alarm, period RWP, and period recorder and current input to calibrate log N amplifier, log N recorder, and signals to safety system.</td>
<td>Monthly</td>
</tr>
<tr>
<td>4. Channels 1 through 7</td>
<td></td>
<td>3. The electronics, interlocks, and readout of each channel is calibrated using a simulated signal. The simulated signal is a current input from safety system test chassis No. 1 to calibrate the flux amplifier, flux level recorder and signals to the safety system.</td>
<td>Monthly</td>
</tr>
<tr>
<td><strong>c. Radiation Monitoring System</strong></td>
<td>Personnel protection</td>
<td>4. Nuclear chambers are checked by the following tests:</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(a) The chambers of each channel are checked using a known neutron source. This check also includes the response of all electronics in each channel. The calibration of each individual component in each channel is checked.</td>
<td>Yearly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(b) Perform heat balance and correlate with flux level.</td>
<td>Monthly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>The radiation monitoring system is divided into the following eight subsystems:</td>
<td>Monthly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1. Stack monitoring system</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>2. Area monitoring system</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>3. Air particulate monitoring system</td>
<td></td>
</tr>
</tbody>
</table>
### TABLE 17 (continued)

**Periodic Tests and Inspections**

<table>
<thead>
<tr>
<th>System or Safety Component</th>
<th>Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>4. Coolant loop monitoring system</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>5. Liquid monitoring system</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>6. Local ratemeter</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>7. Hand and foot monitors</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>8. Spent fuel storage basin monitor</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

These systems are tested to verify reliability and component response. Tests include detector response, checks of high voltage power settings, and visible and audible alarm set points. Solenoid-operated check sources are provided for individual detector response, and are operable remotely from the main control room. Special source sets are provided for accurate calibration of area monitors, air particulate monitors, hand and foot counters, stack monitors, and some liquid waste monitors. Portable radiation monitoring instruments are maintained and calibrated at ORNL on a monthly cycle.

d. **Control Rod and Drive Mechanisms**

1. The control rod drive mechanism and rod are tested at the rehearsal shaft in the reactor building using a test panel installed on the service machine room floor. The test panel is equipped to test the functional operation of the drive motor, speed decreaser, brake and clutch, limit switches, scram velocity control loop, and position indicator. The manual drive also is checked while in the rehearsal shaft. The consequence of power failure to the motor and scram velocity control loop is investigated. At this time, a visual inspection is made of the control rod and cable to determine wear and damage that might lead to malfunctions.
### Periodic Tests and Inspections

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>e. Pressure Vessel, Nozzles and Core Internals</td>
<td>Confirm the integrity of the core and pressure vessel</td>
<td>2. Variation of control rod worth with operation is determined periodically using the xenon poisoning method as described in Section 14.</td>
<td>Yearly</td>
</tr>
<tr>
<td>1. Top Nozzles</td>
<td>1. Check alignment of a representative number of control rod nozzles and special nozzles with their respective core channels.</td>
<td>Yearly</td>
<td></td>
</tr>
<tr>
<td>2. Pressure Vessel</td>
<td>2. Check effects of irradiation on reactor vessel material. Information pertaining to these tests is included in Volume I, Section 4.8.5a.</td>
<td>See Volume I, Section 4.8.5a</td>
<td></td>
</tr>
<tr>
<td>3. Pressure Vessel</td>
<td>3. Check ability of individual rods to drop. Drop may be initiated from near bottom of travel.</td>
<td>Once per four months</td>
<td></td>
</tr>
<tr>
<td>4. Full time of flight tests of rods at approximately 500 °F isothermal core temperature and normal system pressures.</td>
<td>Yearly</td>
<td></td>
<td></td>
</tr>
<tr>
<td>System or Component</td>
<td>Safety Function</td>
<td>Objective and Scope</td>
<td>Frequency</td>
</tr>
<tr>
<td>---------------------</td>
<td>-----------------</td>
<td>-------------------------------------------------------------------------------------</td>
<td>-----------</td>
</tr>
<tr>
<td>3. Reactor Vessel</td>
<td></td>
<td>3. Inspect portions of the top grid plate, fuel channel extensions, welds, and</td>
<td>Yearly</td>
</tr>
<tr>
<td>Upper Plenum</td>
<td></td>
<td>temperature barrier insulation.</td>
<td></td>
</tr>
<tr>
<td>4. Graphite Columns</td>
<td></td>
<td>4. Visually inspect a representative number of fuel and of control rod channels</td>
<td>Yearly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>for cracks, pits, or channel bowing.</td>
<td></td>
</tr>
<tr>
<td>f. Fuel Assemblies</td>
<td></td>
<td>Surveillance of fuel assemblies and moderator material is maintained to back up</td>
<td></td>
</tr>
<tr>
<td>and Moderator</td>
<td></td>
<td>the basic design and to provide information to evaluate potential hazards.</td>
<td></td>
</tr>
<tr>
<td>1. Fuel Assemblies</td>
<td></td>
<td>1. Approximately 30 assemblies in the initial core have been measured in detail.</td>
<td>As required</td>
</tr>
<tr>
<td></td>
<td></td>
<td>These assemblies are removed one or two at a time for hot cell examination at ORNL.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Tests conducted at ORNL include:</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(a) Dimensional changes</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(b) Amount and identity of fission gas release</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(c) Metallographic examination of cladding (not all assemblies)</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(d) Isotopic composition of fuel (three assemblies only at 3000, 6000, and 10,000 Mwd/MT).</td>
<td></td>
</tr>
<tr>
<td>2. Moderator</td>
<td></td>
<td>2. One centrally located fuel channel is loaded with graphite specimens made from</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>ECCR graphite. Specimens are contained in an assembly consisting of six graphite</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>support sleeves. Periodically, the entire assembly is removed and sent to ORNL for</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>examination. The following tests are to be made:</td>
<td></td>
</tr>
</tbody>
</table>
TABLE 17 (continued)

**Periodic Tests and Inspections**

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>(a) Dimen3ion and weight change</td>
<td>At six months, and yearly thereafter</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(b) Coefficient of thermal expansion</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>(c) Irradiation-induced creep</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>(d) Oxidation rate in air</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>(e) Tensile strength</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>(f) Modulus of elasticity</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>(g) Stored energy</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>(h) Thermal conductivity</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

1. **Reactor Coolant System**
   - Prevent bypass of core during emergency cooling loop operation or during operation with single blower
   - Verify remote manual opening and closing of isolation valves.
   - Yearly

2. **Relief Valves**
   - Prevent overpressurization of reactor coolant system
   - Test relief valves using built-in testing system.
   - Yearly

3. **Reactor Coolant Blowers**
   - Utilized to remove heat from core
   - The following tests verify blower performance:
     - (a) Demonstrate reactor coolant blower operation using emergency remote manual control system.
     - Yearly
     - (b) Verify ability of blowers to ramp down on scram.
     - Yearly
     - (c) Confirm operability of static seal control system (valves are not actuated during test).
     - Yearly
TABLE 17 (continued)

Periodic Tests and Inspections

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>h. Reactor Containment</td>
<td>Limits release of radioactivity to the environment during all credible accidents</td>
<td>(d) Trip blowers, observe coastdown.</td>
<td>Yearly</td>
</tr>
<tr>
<td>1. Containment Shell</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>2. Penetra- Isolate contain-</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>tions - Isolate containment shell and Isolation Provisions</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(a) Inspection of containment shell structure for unequal settlement of foundations; corrosion; deterioration of sealing compounds, or other nonmetallic materials at connections, doors, or removable covers; mechanical impact damage; or cracking at points of stress concentrations.</td>
<td>Yearly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(b) The containment shell leakage rate is determined with all penetrations closed.</td>
<td>Once per five years</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2. All penetrations and isolation provisions are tested to verify remote manual operation and automatic closure of isolation valves. Valve leakage is tested as a part of the containment shell test. The following isolation provisions are tested:</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(a) Spent fuel discharge chute (one gate valve and one pneumatic lock).</td>
<td>Yearly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(b) Automatic closing valves in lines venting to atmosphere (28 valves).</td>
<td>Once per six months</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(c) Automatic closing valves in lines providing nonessential services (13 valves).</td>
<td>Once per six months</td>
</tr>
<tr>
<td>System or Component</td>
<td>Safety Function</td>
<td>Objective and Scope</td>
<td>Frequency</td>
</tr>
<tr>
<td>---------------------</td>
<td>----------------</td>
<td>-------------------------------------------------------------------------------------</td>
<td>-----------</td>
</tr>
<tr>
<td>3. Containment</td>
<td></td>
<td>(d) Remote manual isolation valves in lines providing essential services (16 valves).</td>
<td>Yearly</td>
</tr>
<tr>
<td>Shell</td>
<td></td>
<td>Normally closed, hand operated valves (13 valves) are checked to verify that the</td>
<td></td>
</tr>
<tr>
<td>Vacuum</td>
<td></td>
<td>valves are closed after each operation requiring their use.</td>
<td></td>
</tr>
<tr>
<td>Relief</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Valves</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>4. Containment</td>
<td></td>
<td>3. Verify the lifting and reseating of the valves with a pressure differential</td>
<td>Once per</td>
</tr>
<tr>
<td>Shell</td>
<td></td>
<td>across the valves. Leak check valve for proper reseating after operation (done as</td>
<td>two years</td>
</tr>
<tr>
<td>Air</td>
<td></td>
<td>part of containment shell test).</td>
<td></td>
</tr>
<tr>
<td>Locks</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>i. Containment</td>
<td></td>
<td>4. Inspection of air lock doors for mechanical damage or for deterioration of</td>
<td>Monthly</td>
</tr>
<tr>
<td>Shell</td>
<td></td>
<td>sealing compounds or other nonmetallic materials. (Same type of inspection as made</td>
<td></td>
</tr>
<tr>
<td>Spray</td>
<td></td>
<td>on the containment shell.)</td>
<td></td>
</tr>
<tr>
<td>Cooling</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>System</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>1. Verify equipment operation, flow distribution on the containment shell surface,</td>
<td>Once per</td>
</tr>
<tr>
<td></td>
<td></td>
<td>and automatic operation on signal due to containment isolation.</td>
<td>six months</td>
</tr>
<tr>
<td></td>
<td></td>
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</tr>
</tbody>
</table>

TABLE 17 (continued)

Periodic Tests and Inspections
## TABLE 17 (continued)

### Periodic Tests and Inspections

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>j. <strong>Heating and Ventilation System</strong></td>
<td>Provide recirculation of air inside the reactor building during containment isolation to remove airborne activity or to disperse hydrogen throughout the building</td>
<td>Test biological shield cooling system filter bank relief panels to verify that the panels are opened on pressure differential.</td>
<td>Yearly</td>
</tr>
<tr>
<td>k. <strong>Failure Free Power</strong></td>
<td>Power source for instrumentation, safety and control functions</td>
<td>1. Test each switchgear control power transfer scheme using the test switch.</td>
<td>Yearly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2. The failure-free feeder breaker trip alarm is tested by tripping each feeder breaker individually.</td>
<td>Once per three months</td>
</tr>
<tr>
<td></td>
<td></td>
<td>3. The condition of the batteries and the functional reliability of the chargers are checked by reading cell voltage and specific gravity of the electrolyte.</td>
<td>Monthly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>4. Load test on failure-free and power-lighting battery.</td>
<td>Yearly</td>
</tr>
<tr>
<td>1. <strong>Emergency Power System</strong></td>
<td>Provides emergency electrical power supply to critical equipment</td>
<td></td>
<td></td>
</tr>
<tr>
<td>System or Component</td>
<td>Safety Function</td>
<td>Objective and Scope</td>
<td>Frequency</td>
</tr>
<tr>
<td>---------------------</td>
<td>-----------------</td>
<td>---------------------</td>
<td>-----------</td>
</tr>
<tr>
<td>1. Motor Control Centers</td>
<td>1. The bus transfer schemes for MCC-1, MCC-2, and MCC-3 are tested by tripping each feeder breaker individually and checking that the bus tie breaker closes properly.</td>
<td>Yearly</td>
<td></td>
</tr>
<tr>
<td>2. Emergency Load Center and Diesel Generators</td>
<td>2. The following tests are performed to demonstrate the ability of the diesels to start up and connect to the emergency load center.</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>(a) Check that both diesel generators start and connect to their respective buses automatically by manually tripping both transformer feeder breakers. The standby vessel cooling compressor and one blower seal water pump should start after a time delay following restoration of bus voltage.</td>
<td>Once per six months</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(b) Repeat test, except lock out one diesel at a time and check that the other automatically starts and connects to both emergency load center buses. For each diesel apply the maximum anticipated load.</td>
<td>Once per six months</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(c) Start diesel, synchronize, and temporarily parallel with its respective transformer.</td>
<td>Once per two weeks</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(d) Test selective tripping of protective devices by individual tests and calibration of each of these devices.</td>
<td>Yearly</td>
<td></td>
</tr>
<tr>
<td>3. Diesel Engine</td>
<td>3. All support systems for the diesel generators are periodically tested to verify start-up and proper functioning of equipment. The following tests are included:</td>
<td>Once per two weeks</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(a) Diesel Starting Air--Compressor operation is verified by bleeding air receiver pressure down until the compressor</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
**TABLE 17 (continued)**

**Periodic Tests and Inspections**

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>4. Turbine Room Louvers</td>
<td>Protection of emergency power system from steam in turbine building</td>
<td>4. Inspect and check freedom of movement of louvers.</td>
<td>Once per six months</td>
</tr>
<tr>
<td>m. Shutdown Heat Removal System</td>
<td>Removal of reactor decay heat following reactor shutdown</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1. Verify automatic operation of the entire shutdown system by initiating a low drum level signal from each drum or by a simulated under-voltage condition of the normal load center. Checks are made to verify that the shutdown steam and feedwater system automatically come into service and operate properly.</td>
<td></td>
<td>Yearly</td>
<td></td>
</tr>
<tr>
<td>2. Verify proper operation of the following valves by remote manual operation:</td>
<td></td>
<td>Once per six months</td>
<td></td>
</tr>
</tbody>
</table>
### TABLE 17 (continued)

**Periodic Tests and Inspections**

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>(a) Feedwater regulating valves bypass</td>
<td>Once per six months</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(b) Shutdown feedwater control valves</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(c) Electromatic relief valves</td>
<td></td>
</tr>
</tbody>
</table>

3. Verify proper operation of the following equipment including appropriate checks of temperature, pressure, flow, and automatic control:

   (a) Desuperheater, C-9
   (b) Shutdown condenser, C-14
   (c) Drain cooler, C-29

4. Verify pump start-up by remote manual operation and satisfactory operation of the electrically-driven shutdown feedwater pumps and steam-driven shutdown feedwater pumps.

   Desuperheater, C-12, is operationally tested concurrently with the steam-driven shutdown feedwater pumps to verify correct steam temperature and pressure at the pump inlet.

5. Verify that the domestic water backup to the shutdown feedwater pump operates at the correct pressure by closing off the normal supply.

n. **Emergency Cooling System**

   Emergency backup protection for removal of heat from the core after reactor shutdown and reactor coolant system depressurization.
TABLE 17 (continued)

**Periodic Tests and Inspections**

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Heat Removal System</td>
<td>Remove heat from reactor core</td>
<td>1. All components and auxiliaries comprising the heat removal system are periodically tested to verify remote manual operation of the equipment and proper functioning of individual components. The main control room instrumentation is checked and an inspection is made of equipment in the cell including the following components:</td>
<td>Monthly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(a) Compressors--Operated individually using the bypass to verify proper operation and ability to isolate each compressor.</td>
<td>Monthly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(b) Aftercooler and heat exchanger--Checked for abnormal fouling of surfaces and for leaks.</td>
<td>Monthly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(c) Cooling water systems--Flow and temperature of cooling water to the compressors, aftercooler, and heat exchanger is checked by opening the supply and drain valves.</td>
<td>Monthly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(d) Compressor lubricating oil system--Auxiliary lube oil pump and oil cooler are checked during the compressor test.</td>
<td>Monthly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(e) Loop isolation valves--Valves are manually operated from the control room when the reactor coolant system is depressurized.</td>
<td>Yearly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(f) Hot leg-piping helium purge--The low flow alarm is tested by reducing flow until the set point is reached.</td>
<td>Monthly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(g) Cold leg-piping drain valves--Valves are opened and closed individually from the main control room when the reactor coolant system is depressurized.</td>
<td>Yearly</td>
</tr>
</tbody>
</table>
### TABLE 17 (continued)

**Periodic Tests and Inspections**

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>2. Purge Gas System</td>
<td>Provides an inert gas blanket to the reactor core to minimize oxidation of graphite and UO$_2$ during depressurization accidents. Also used to dilute and disperse hydrogen.</td>
<td>(h) Check operability of chase cooling fans.</td>
<td>Monthly</td>
</tr>
<tr>
<td></td>
<td>2. The purge supply is tested by opening the shutoff valve from the main control room. Satisfactory operation of the pressure reducing valves is checked. The low supply pressure alarm is tested by introducing a simulated low pressure signal. The emergency cooling loop remains isolated from the reactor for this test. The purge gas system is vented through the heat exchanger drains to test the system. Any nitrogen leaking into the cell from the compressor seals is vented to the stack through the cell exhaust fan and filter system. The nitrogen vaporizer is tested by operating the remote manual controls located in the main control room.</td>
<td></td>
<td>Monthly</td>
</tr>
<tr>
<td>3. Fission Product Removal System</td>
<td>Prevent overpressurization of the containment shell due to nitrogen purge while using the emergency cooling system heat removal system</td>
<td>3. Tests on this system include the following:</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(a) Remote manual operation of isolation valves</td>
<td>Monthly</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(b) Check efficiency of filters using polydisperse aerosol dioctyl phthalate (DOP) and radioactive iodine.</td>
<td>Once per six months</td>
<td></td>
</tr>
<tr>
<td>4. Cell Containment</td>
<td>Minimize the release of activity to the environment for all credible accidents</td>
<td>4. Tests and inspections directed at verifying cell integrity include:</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(a) General inspection as described for the containment shell, h.1(a)</td>
<td>Yearly</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(b) Cell leakage rate—Tested with all penetrations closed to determine the leakage rate.</td>
<td>Once per five years</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(c) Cell air lock described as for air locks on the containment shell, h.4.</td>
<td>Monthly or yearly as indicated in h.4</td>
<td></td>
</tr>
</tbody>
</table>
TABLE 17 (continued)

**Periodic Tests and Inspections**

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>(d) Cell drain control--Drain valves are opened and closed from the main control room to check proper operation. High water level alarm is tested at this time.</td>
<td>Monthly</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(e) Cell ventilation--Ventilation system valves are tested for proper operation. The two cell isolation valves are tested for closure by applying a small radioactive check source to the radiation detector element.</td>
<td>Monthly</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

**Plant Alarm System**

1. Evacuation Alarm System
   - Notifies personnel to evacuate the reactor building, the entire plant area, or to return to area
   - Verify that the evacuation alarm system functions properly by operating the evacuation alarm master control switch in the main control room. The reactor building evacuation alarm is checked automatically each time containment isolation is initiated.
   - Monthly

2. Fire Alarm System
   - To provide an alarm in case of fire
   - Verify that the fire alarm system is operative by initiating an alarm signal from an area fire alarm box.
   - Once per three months

**Fire Protection System**

1. Gasoline Engine-Driven Fire Pump
   - To act as a backup for the electric-driven fire protection pump
   - The following tests are performed:
     a. Check that this equipment starts as required by lowering the fire protection water tank level.
     - Once per six months
### TABLE 17 (continued)

**Periodic Tests and Inspections**

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>(b) Manually start engine</td>
<td>Protection of critical equipment from damage due to fire</td>
<td>2. Verify that the system functions automatically when a heat source is applied to the area thermostats.</td>
<td>Weekly</td>
</tr>
<tr>
<td>2. Automatic CO₂ Fire Protection System</td>
<td>Protection of critical equipment from damage due to fire</td>
<td>2. Verify that the system functions automatically when a heat source is applied to the area thermostats.</td>
<td>Yearly</td>
</tr>
<tr>
<td>q. Air System</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1. Plant Air Isolation Valves</td>
<td>Conserves compressed air for plant controls and instrumentation</td>
<td>1. Verify that the valves close at the correct pressure and isolates the plant air system.</td>
<td>Once per six months</td>
</tr>
<tr>
<td>2. Check Valves in Main Headers to Instrument Air Receiver F-22-3</td>
<td>Isolates instrument air receiver for use in the reactor building</td>
<td>2. Verify that the check valves in the main air headers effectively isolate instrument air receiver lowering pressure at inlet of check valves to below the outlet pressure.</td>
<td>Yearly</td>
</tr>
<tr>
<td>r. Blower Seal Water System</td>
<td>Provides seal water to both sets of blowers on failure of one blower seal water pump</td>
<td>1. Verify that the high pressure head tank and low pressure head tank equalization valves function properly by manually stopping one blower seal water pump.</td>
<td>Once per three months</td>
</tr>
<tr>
<td>System or Component</td>
<td>Safety Function</td>
<td>Objective and Scope</td>
<td>Frequency</td>
</tr>
<tr>
<td>---------------------</td>
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<td>---------------------</td>
<td>-----------</td>
</tr>
<tr>
<td>1. Steam Generator</td>
<td>Primary heat sink</td>
<td>1. A representative number of steam generator tubes are inspected to determine the degree of erosion or pitting and their general condition. A hydrostatic test of the steam-water side at 150% design pressure is performed.</td>
<td>Yearly</td>
</tr>
<tr>
<td>2. Excess Flow Valves</td>
<td>Protects the steam generator from failures of the main steam piping</td>
<td>2. Verify the proper operation of the steam excess flow valves and controls by using the hand switch in the main control room and by insertion of a simulated high flow signal.</td>
<td>Yearly</td>
</tr>
<tr>
<td>3. Steam Generator Isolation Valves</td>
<td>Provides protection for the reactor coolant loops, steam generators, and containment shell</td>
<td>3. Proper operation of the feedwater isolation valves and controls is tested by raising the drum level to actuate the high level switches and by operating the hand switches in the main control room. Proper operation of the steam, feedwater, helium valves and controls that trip the reactor coolant blower, close the reactor coolant system valves, and isolate the steam generator are tested by operation of the hand switch in the main control room, manually decreasing the drum level to actuate the low-low</td>
<td></td>
</tr>
<tr>
<td>2. High Level Controls</td>
<td>Prevents discharge of water into reactor coolant system from seal water tanks</td>
<td>2. Verify that bypass valve opens and tank level is maintained</td>
<td>Once per six months</td>
</tr>
<tr>
<td>3. Excess Flow Valves</td>
<td>Protects against failure of the pressure supply to high pressure head tanks</td>
<td>3. Check operation of valves by insertion of simulated signal.</td>
<td>Yearly</td>
</tr>
</tbody>
</table>
### TABLE 17 (continued)

**Periodic Tests and Inspections**

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>t. Service Water System</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fire Protection Water Backup Supply to Service Water Header</td>
<td>Provide equipment cooling on loss of service water pressure</td>
<td>Verify that fire protection water backup supply valves to service water headers open and operate properly by manually reducing service water pressure. Also verify that fire protection water backup supply valves to individual equipment and associated check valves operate properly to supply emergency cooling by manually reducing service water supply pressure to the individual equipment.</td>
<td>Yearly</td>
</tr>
<tr>
<td><strong>u. Vessel Cooling System</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1. Excess Flow Valves</td>
<td>Prevents depressurization of reactor coolant system due to piping failure in the vessel cooling system</td>
<td>1. Valves are closed automatically by inserting a simulated excess flow condition.</td>
<td>Yearly</td>
</tr>
<tr>
<td>2. Static Seal Remote Control System</td>
<td>Emergency backup for running seals</td>
<td>2. Confirm operability of static seal control system (valves are not actuated during test).</td>
<td>Yearly</td>
</tr>
</tbody>
</table>
### TABLE 17 (continued)

**Periodic Tests and Inspections**

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>3. Nozzle Cooling Piping Check Valves</td>
<td>Prevents depressurization of reactor coolant system due to piping failure in the vessel cooling system</td>
<td>The functioning of these valves is checked by depressurizing the vessel cooling system with the reactor coolant system pressurized.</td>
<td>Yearly</td>
</tr>
<tr>
<td>4. Reactor Coolant System Inlet Isolation Valves</td>
<td>Permits coolant flow through core with reactor coolant blowers shut down and VCC operating</td>
<td>Automatic closure of the inlet isolation valves in the reactor coolant system is tested by shutting down the reactor coolant blowers. The interlock is also tested for the case where the VCC's are not operated.</td>
<td>Yearly</td>
</tr>
<tr>
<td>v. Gas Vent System</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1. Reactor Coolant Loop Emergency Depressurization Valves</td>
<td>Allows reactor coolant loop to be depressurized remotely</td>
<td>Verify that HCV-14-56A and HCV-14-57 are operable by operating them one at a time from the main control room.</td>
<td>Once per six months</td>
</tr>
<tr>
<td>2. Main and Emergency Relief Transfer Valves</td>
<td>Prevents reactor coolant loop depressurization in the event of a safety valve failure</td>
<td>Verify that these valves function properly by operating them from the main control room, (PCV-14-45 A, B, C, and D).</td>
<td>Monthly</td>
</tr>
</tbody>
</table>
### TABLE 17 (continued)

#### Periodic Tests and Inspections

<table>
<thead>
<tr>
<th>System or Component</th>
<th>Safety Function</th>
<th>Objective and Scope</th>
<th>Frequency</th>
</tr>
</thead>
<tbody>
<tr>
<td>w. Service Machine Emergency Cooling Supply</td>
<td>Provide cooling supply on loss of vessel cooling system</td>
<td>Verify that the service machine emergency cooling supply controls and valves operate properly to supply emergency cooling by throttling the manual valve in the line. Also verify that the emergency cooling supply is shut off on low reactor coolant pressure by inserting a simulated signal.</td>
<td>Once per six months</td>
</tr>
<tr>
<td>x. Control Room Isolation and Emergency Air Supply</td>
<td>Minimizes the exposure dose to operators inside the control room during the maximum credible accident</td>
<td>1. Verify that the automatic isolation of the containment shell also closes the heating and ventilation lines to the control room.</td>
<td>Yearly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2. Verify that the coolers for removing the heat load inside the control room operate satisfactorily.</td>
<td>Yearly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>3. Check manifold pressure on air bottles provided for emergency breathing.</td>
<td>Monthly</td>
</tr>
</tbody>
</table>
1. Receiving fuel at EGCR from storage at Oak Ridge Gaseous Diffusion Plant

2. Unpacking and cleaning assemblies; visual inspection for shipping damage

3. Transferring assemblies to inspection table in reactor building

4. Inspecting assemblies and placing six acceptable fuel assemblies and one bottom dummy assembly in fuel loading tube

5. Transferring assemblies and bottom dummy from fuel loading tube to loading and rehearsal mechanism

6. Transferring assemblies and bottom dummy from loading and rehearsal mechanism to charge machine

7. Charging fuel assemblies and bottom dummy into reactor

8. Relocating fuel in reactor with charge machine

9. Removing fuel from reactor with charge machine

10. Loading into spent fuel transfer mechanism

11. Discharging from transfer mechanism into discharge chute

12. Monitoring assemblies in discharge chute for ruptured elements

13. Releasing assemblies from discharge chute to entry trough in spent fuel storage basin

14. Transferring sound assemblies, one at a time, to underwater work bench

15. Placing assembly in cluster ejection position on work bench and breaking shear pins with pneumatic plunger

16. Removing seven-element cluster from graphite sleeve

17. Visually inspecting cluster with periscope and reading number imprinted on top spider

18. Placing cluster in fuel storage rack

19. Storing spent fuel in basin for minimum of 100 days

20. Transferring fuel from storage rack to shipping cask rack

21. Removing loaded and sealed cask from basin, monitoring, decontaminating, and placing cask on truck trailer

22. Moving trailer to remote location and conducting temperature rise test

23. Moving cask on truck trailer to Oak Ridge Gaseous Diffusion Plant and transferring cask to railroad car

24. Shipping cask to Savannah River Plant by railroad

Fig. 18.1

EGCR FUEL HANDLING
18. FUEL HANDLING AND TRANSPORTATION

18.1 General
All fuel handling operations at the EGCR plant are initiated by specific instructions issued by a responsible member of the EGCR staff. All operations involving the handling of fuel are carried out so as to minimize or prevent the spread of radioactive contamination, to minimize direct radiation to operating personnel, and to preclude the possibility of criticality. In addition, all fuel handling operations are carried out so as to minimize the possibility of damaging the fuel cladding. The Radiological Health Section recommends procedures for contamination control and radiation safety. The Reactor Physics Staff recommends procedures governing the handling and storage of fresh and spent fuel to prevent accidental criticality. The fuel handling cycle at the EGCR is described diagrammatically in Figure 18.1.

To maintain complete control of the fuel element inventory, fuel handling operations are carried out only after a fuel handling order (FHO) has been issued by the accountability records technician and approved by the operating superintendent. The FHO completely identifies the fuel to be handled, its present location, the operation to be performed, and its final location.

An FHO for charging, discharging, or relocating fuel within the reactor is normally initiated by the Reactor Physics Staff. The FHO for all other fuel handling operations is originated by the Chemical Engineering Section. The accountability records technician prepares the FHO based upon a request from the responsible group and presents it to the operating superintendent for authorization and scheduling.

Fuel handling is carried out by the Plant Operations Section. Upon completion of the operation, duplicate copies of the executed FHO are returned to the accountability records technician and to the staff group that initiated the request.

A specially prepared FHO is incorporated within a standard operating procedure and is used in the event the plant operations supervisor determines that an unscheduled fuel handling operation is required. This is used, for example, when a sizable fuel element rupture is detected.

18.2 Receiving Fuel At Plant
Prior to the time initial fuel loading commences, fuel is stored in a warehouse at the Oak Ridge Gaseous Diffusion Plant, in accordance with instructions provided by Oak Ridge National Laboratory. The fuel remains under the jurisdiction of ORNL until it is received at the EGCR site by the operator. The initial procurement order of fuel consists of 1739 assemblies of the type described in Volume I of the Hazards Summary Report, Section 4.2. Fuel assemblies are shipped to the site in the same containers as received from the fabricator.

Shipsments of fuel to the site are normally scheduled so that no more than 112
fuel assemblies are on the site outside the confines of either the fuel storage vault or the reactor core. This measure prevents the possibility of accidental criticality during this phase of the fuel handling cycle. Calculations indicate that 16 steel drums of fuel assemblies, arranged in an optimum geometry and submerged in water, are subcritical.

At the time of shipment by the vendor, fuel is substantially free of radioactive contaminants. Fuel is checked by the Radiological Health Section to verify that surface contamination is not excessive for handling operations. Fuel assemblies are unpacked in a temporary enclosure located in the north end of the turbine building. Assemblies are removed from the mailing tubes, and the shredded cork, with which the mailing tubes are filled, is removed by means of compressed air and vacuum cleaning. The assemblies are moved on the fuel transfer cart to the fuel storage vault where visual inspection of the assemblies is made and the assemblies are sealed in polyethylene bags.

Fuel assemblies are retained in the fuel storage vault until they are to be loaded into the reactor. Prior to initial loading, approximately 500 assemblies are cleaned and placed in storage. Thereafter, unpacking and cleaning operations are suspended until nearly all of these assemblies have been loaded into the reactor.

18.3 Fuel Storage

The storage racks in the vault are designed to give a safe configuration from a criticality standpoint. The location of the fuel vault requires that the reactor site be flooded to elevation 845 ft 6 in. before any water enters the vault. Such an occurrence is not considered credible. The vault has a CO₂ fire protection system. No lines carrying water under pressure pass through the vault. A line from the chemical laboratory sink and floor drains passes through the vault. It is inconceivable that this line could carry enough water to flood the vault and create a critical configuration if the line were to rupture.

Access to the fuel storage vault is limited to those members of the Operations and Technical Program Groups that are involved in fuel handling operations. The door to the vault is locked to prevent access by unauthorized persons. Access to the fuel storage vault is limited to assist in maintaining clean conditions in the vault, to prevent damage to the fuel, and to assure the required control over the location of fissionable material.

18.4 Inspection

Fuel assemblies are moved on a cart from the fuel storage vault to the fuel loading room, area E102, where each fuel assembly is inspected before it is loaded into the reactor. Area E102 is designated a clean area. Employees handling fuel in this room are dressed in clean coveralls, caps, shoe covers, and gloves. All of the inspection gages that contact the fuel are maintained in a clean condition to prevent introducing contaminants (radioactive and non-radioactive materials) into the core. Inspection is accomplished using fuel assembly gages. The following inspections are performed by the Plant Operations Section and are of a go-no-go type:

a. Visual examination for cleanliness and for nicks, scratches, or other surface defects.

b. Check OD, ID, and length of the sleeves.

d. Check angle of bottom cone, radius of top spherical surface, and squareness of the ends of the sleeve.

e. Check seating of end surfaces using the column joint test mounts.

Fuel assemblies failing any of the above inspections are rejected and returned to the fuel storage vault where they are reinspected by the Chemical Engineering Section. Assemblies which, on reinspection, fail to meet specifications are retained in the storage vault.

18.5 Fuel Loading

Inspected fuel assemblies are first loaded into the loading and rehearsal mechanism and then into the charge or service machine for insertion into the reactor. Precautions are taken in the handling of fuel assemblies to prevent accidental damage and to prevent the entrance of foreign materials into the reactor, on the fuel assemblies, or charge and service machines fuel loading mechanisms.

18.5.1 Fuel Loading Into Loading and Rehearsal Mechanism

Fuel assemblies that pass inspection are carried to the loading tube of the fuel loading and rehearsal mechanism by electric hoist.

The fuel loading and rehearsal mechanism is located in the reactor building in the area extending from the charge machine vault to the floor of the service machine room. This mechanism consists of pneumatically operated control rod and fuel loading tubes mounted on a carriage fitted with casters riding on parallel tracks, a top grid structure, a bottom grid structure, and nozzles duplicating typical reactor vessel nozzles for charging control rods, special plugs, and fuel. Two nozzles, one a control rod nozzle and the other a special plug nozzle, penetrate the service machine room floor. A charge nozzle fitted with a shield plug penetrates the ground floor into the charge machine vault. The nozzles are equipped with couplings, seals, and air-operated ball valves similar to those on the reactor vessel. Helium supply, vent, and evacuation connections are provided for each nozzle. The top and bottom grid structures and tube channels provide a mock-up of a typical 16-channel section of the reactor core.

The loading and rehearsal mechanism is used in all fuel loading operations. The apparatus may be used to rehearse fuel loading and unloading operations, shield plug removal and replacement in a reactor nozzle, replacement of O-rings in nozzle shield plugs, and replacing top nozzle closure gaskets. These operations are performed by the charge machine or the service machine. The rehearsal device is also used to check out the control rod operations and the operation of the top and bottom dummy fuel assembly latches.

18.5.2 Transferring New Fuel to the Charge Machine

Utility services required before these operations can begin include 100 psig plant air at the pneumatic cylinder on the fuel loading mount, at the ball valves of the rehearsal device, and at the charge machine; 310 psig helium at the rehearsal nozzles and at the charge machine; power at the gas handling panel and the charge machine operating console.

Six new fuel assemblies and a bottom dummy assembly are placed in the loading tube of the fuel loading rehearsal device, with the dummy assembly located so that it will support six fuel assemblies when the tube is in a vertical position. The bottom grid shield plug of the selected nozzle extension is manually removed, and the fuel loading tube mount is positioned over the
opening in the floor and raised to a vertical position. The fuel loading tube is then carefully positioned over the preselected bottom grid nozzle extension and, when in the correct position, the carriage and tube positioners are locked in place. At this point, all personnel vacate the fuel loading area; subsequent operations are remotely controlled.

The charge machine is positioned under the rehearsal charge nozzle and the charge machine nozzle bellows extension is connected to the rehearsal charge nozzle adapter. The leak tests and pressure checks of the charge machine and charge nozzle, that are performed at the reactor prior to loading, may be rehearsed at the rehearsal mechanism if such is desired. The loading of the bottom dummy and fuel assemblies into the charge machine then proceeds. When all fuel assemblies are in the machine, the bottom dummy assembly is returned to the fuel loading tube and latched in place. The shield plug is replaced in the rehearsal charge nozzle, using the charge machine, and the ball valve is closed.

In recharging operations, the clean bottom dummy is usually returned to the fuel loading tube and latched in place before the shield plug is replaced. In some instances, it is placed in the charge machine to be used to replace any faulty dummy assemblies discovered during fuel-handling operations in the reactor.

18.5.3 Transferring New Fuel to the Service Machine

The operations involved in the loading of new fuel into the service machine from the rehearsal device assembly are essentially the same as those for the charge machine, except that the upper rehearsal grid structure is used and the fuel is transferred upward through the rehearsal control rod nozzle.

18.6 Loading and Removing Reactor Fuel

The charge machine is used for normal fuel loading and removal operations. These operations are performed while the reactor is on stream. The rate of fuel consumption, with the reactor operating at full thermal power, is approximately the equivalent of one fuel assembly per day, based on attaining an average fuel burnup of 10,000 Mwd/MT. Fuel loading and reshuffling operations are scheduled to permit charge machine operation about once every three days.

Most charge machine operations are remotely controlled, since employees are not permitted in the charge machine room when a shield plug is removed and the reactor is operating. The internal mechanisms are controlled from a console in the control room on the second floor of the reactor building.

The charge machine is positioned under the preselected reactor nozzle, and operations for charging new fuel and discharging spent fuel into and out of the reactor are begun. Subsequent procedures are: check reactor charge nozzle shield plug and blind flange seals for leaks; remove blind flange from reactor nozzle; position the charge machine accurately beneath the nozzle and attach it to the reactor nozzle; connect the required utilities, helium and air; perform the seal leak test on the ball valve; purge the charge machine by evacuating; and then pressurize it with helium. The foregoing operations are all performed from the operator’s platform on the charge machine. All other fuel loading operations are remotely controlled.

When the charge machine has been pressurized with helium from storage, the pressure across the ball valve is equalized, and the pressure across the shield plug is checked from the main control console. When equilibrium between the reactor vessel lower plenum and the charge machine vessel has been established, the ball valve may be opened. The shield plug is then removed and deposited in
the charge machine. The fuel transfer tube is extended and the bottom dummy assembly and the spent fuel assemblies of the selected channel are deposited in the charge machine. New fuel assemblies are then installed in the fuel channel and the bottom dummy assembly reset in position. The new, or refitted, shield plug is then installed, the ball valve closed, and local control at the charge machine is resumed. The space between the shield plug and ball valve is depressurized to 20 psia. The shield plug seals are checked and, if they are satisfactory, the charge machine is depressurized to approximately 20 psia. The space between the ball valve and the shield plug is then evacuated to remove helium before the charge machine is detached from the reactor. The helium vent lines are shut off and evacuated to preclude exposure of personnel to possible radioactive gases trapped therein.

While a fuel assembly is being replaced or cycled, the spent fuel assemblies, and the new fuel assemblies as they are installed in the core, must be cooled. Helium is, therefore, introduced through the charge machine into the reactor vessel at approximately 315 psig and 350 °F. The helium flows upward from the charge machine through the extensible charge tube into the preselected reactor core fuel channel. The cooling requirements during refueling are discussed in detail in Section 4.6.7 of Volume I of the EGCR Hazards Summary Report.

18.7 Discharge From Charge (or Service) Machine

From the standpoint of personnel hazards, a critical portion of the fuel handling cycle is the transfer of spent fuel from the charge or service machine to the spent fuel basin. Special precautions are necessary to provide adequate cooling to the fuel cladding, control the spread of contamination, and insure that operating employees are adequately shielded from radiation associated with the fuel assemblies during the transfer operation.

18.7.1 Transferring Spent Fuel From the Charge Machine

Except for checking out all required services and basic operating prerequisites, all operations involved in transferring spent fuel from the charge machine into the spent fuel transfer mechanism are conducted remotely from the spent fuel transfer panel, or from the charge machine operating console.

The utility services required and operating conditions which must exist before operation can begin include; 100 psig plant air at all pneumatically operated devices, electric power at the control panels, water in the discharge chute at 19-ft level, spent fuel storage basin monitoring system operable, a negative gage pressure in the spent fuel transfer shaft enclosure, the wall periscope functioning, and the 5-1/4-in. ID liner secured in place in the transfer tube.

Under normal operating procedures, six spent fuel assemblies are transferred by the charge machine in each operating cycle. The charge machine is depressurized and carefully positioned under the spent fuel transfer mechanism. It is then manually coupled to the bottom nozzle of the spent fuel transfer mechanism. These operations, which are executed from the operator's platform on the charge machine, are carried out under health physics surveillance. At this point all persons are required to vacate the charge machine vault and subsequent operations are remotely controlled. The ball valve in the charge machine is opened and the shield plug is removed from the spent fuel nozzle and stored in the machine. The spent fuel then is moved from the charge machine to the transfer tube of the spent fuel transfer mechanism. Air from the plant air system is circulated through the machine inlet nozzle and into the spent fuel transfer shaft to cool the fuel during transfer. The air is vented to the biological shield cooling system filters and fans. A backup water spray cooling system is provided in the transfer shaft for use in the event the plant air system fails. The circulation of plant air and operation of the charge machine
blower are continued until the fuel has been discharged from the transfer tube into the discharge chute. The shield plug then is replaced and operators and a health physics technician enter the charge machine room to disconnect the machine from the nozzle.

18.7.2 Transferring Spent Fuel From the Service Machine

The procedure for the transfer of spent fuel from the service machine to the spent fuel transfer mechanism is similar to that outlined above for the charge machine, except that the fuel element assemblies are individually handled and are fed into the transfer tube from the upper end. The time between removal from the reactor and insertion in the transfer tube by the service machine is long enough that the fuel does not require cooling beyond that which is provided by 50 cfm of air induced through the spent fuel transfer shaft by the biological shield cooling system fans.

It is desirable in certain instances, as when reinsertion into the reactor is desired, to store irradiated fuel assemblies of various types in a dry condition. For this reason, 5 of the 13 storage holes extending from the service machine room floor down into the biological shield are provided with a cooling water jacket and controlled venting to the gas vent system. Regular or experimental fuel assemblies and instrumented fuel columns can be stored in these storage holes. The duration of irradiated fuel storage in these holes depends upon the reason for storage and may vary from a few days to several months.

Fuel assemblies or instrumented fuel columns are inserted in and removed from the storage holes by the service machine, using a procedure similar to the one used for inserting and removing fuel and instrumented fuel columns from the reactor, except that there is no coolant flow through the service machine, and the service machine and storage holes are at atmospheric pressure.

18.8 Transfer to Spent Fuel Storage Basin

The transfer of spent fuel to the storage basin is part of a continuous operation in which fuel is discharged from the charge (or service) machine, tested for defective fuel elements, and transferred to the fuel storage basin for decay prior to shipment to the reprocessing facility.

All devices in the spent fuel transfer mechanism and the charge (or service) machine are fully instrumented and equipped with automatic supervisory controls and interlocks to assure that the fuel is fully supported during all transfer steps. For example, the transfer mechanism tube can be raised to its uppermost position only after the retaining latches have been closed and the load sensor indicates that the fuel load is fully supported.

After spent fuel is transferred from the charge or service machines to the spent fuel transfer tube, the retaining latches are closed and the load sensor is checked to see that the load is fully supported. The gate valve at the upper end of the discharge chute is opened, the transfer tube is hoisted to its uppermost position, and then is lowered to its inclined position in line with the discharge chute. This releases the retaining latches and the fuel is released from the tube into the chute. A visual check is made with the periscope and mirror to ascertain that all assemblies are in the chute. The transfer tube then is raised to its uppermost position and the gate valve is closed. The spent fuel assemblies are monitored for fission product release while in the discharge chute. If no activity is indicated, the quick-opening door at the bottom of the chute is opened, fuel is discharged into the storage basin, and the door is then closed.

18.9 Test for Defective Elements
Spent fuel elements are checked for defects in the discharge chute with the spent fuel storage basin monitoring system. This system, which monitors the water in the discharge chute, is placed in operation before spent fuel is admitted to the chute to assure that the system is functioning properly and to establish the level of background activity.

In the monitor, water is passed successively through a filter, a cation exchange resin bed, and an anion exchange resin bed. A gamma scintillation counter measures the activity that accumulates on the anion bed. If no significant activity increase is indicated after an hour, the water circulation is stopped and the fuel is discharged into the storage basin. If activity buildup is sensed by the monitoring system, the spent fuel assemblies are retained in the chute until the activity decreases to an acceptable level. Thus, the release of fission product gases from a ruptured fuel element into the storage basin water is minimized.

If monitoring in the chute indicates the presence of a ruptured element, the fuel assemblies are moved from the entry trough to the tanks where they are monitored in succession until the faulty assembly is identified. The handling of defective assemblies after discharge into the storage basin is described in Section 18.10.2.

18.10 Handling Spent Fuel in Storage Basin

Fuel handling operations at the storage basin are carried out by plant operators with assistance from the Radiological Health Section to measure radiation levels and advise on radiological safety. Employees wear protective clothing and respirators are available for use, if required.

Five containers holding 36 clusters each are used to store fuel in the basin. Each container has built into it a sheet of cadmium as a neutron poison. The containers, holding 36 clusters each, are critically safe in any array. Calculations have indicated that 36 unsleeved, bare fuel assemblies (clusters) may constitute a critical mass when submerged in water. These calculations are based upon a cylindrical array one fuel assembly-length high with the assemblies spaced so that the peripheries of their top spiders are in contact. Care is taken, when moving containers, to avoid spilling clusters into the basin, in particular adjacent to other storage containers. Fuel movements in the basin are restricted such that a container is never moved over or adjacent to another filled container in the basin. The storage containers are positioned in the basin such that the above requirement is met.

The water depth in the storage basin is maintained at 19 ft to provide adequate shielding. The basin may contain as many as 180 fuel assemblies which have been cooled for periods ranging from 1 to 180 days. Single fuel assemblies upon discharge from the reactor can be raised to within 13 ft of the water surface without exceeding a radiation dose rate of 0.75 mrem/hr. A radiation monitor with local alarm is located on the service bridge to indicate radiation levels in the range of 0.1-100 mrem/hr. Radiation surveys by a health physics technician are made when operators are moving fuel in the basin.

To minimize radiological hazards to personnel during fuel handling operations, a check is made to ascertain that the ventilation, basin cleaning, and basin demineralizing systems are in operation. Operations need not necessarily be halted upon failure of the latter two systems, but if the ventilation system fails, operations are halted.

A record of the exact location of each fuel assembly in the basin is maintained by the accountability technician.
18.10.1 Handling Sound Fuel Assemblies

Movement of spent fuel in the basin is carried out by operators working from the service bridge using long-handled tools. Sound assemblies are transferred one at a time to the underwater work bench, either directly from the entry trough or from the monitoring tanks. For removal of the sleeve, the assembly is placed horizontally on the bench with the top spider end against an anvil. A hydraulically actuated plunger is positioned against the bottom spider to break the two shear pins at the top spider. This requires a force of 900 lb; the hydraulic cylinder is designed to exert a maximum force of 1100 lb. The cluster is pushed a few inches out of the sleeve by the plunger. The assembly then is shifted to a vertical position under the cluster-removal anvil at the side of the work bench, and the cluster is pulled out of the sleeve. Finally, the cluster is placed in a fuel storage rack where it is held in the basin for at least 100 days. The sleeve is stored in the radioactive waste can, where it is held until its activity level is low enough for disposal in the ORNL burial ground (Section 21.4).

18.10.2 Handling Defective Fuel Assemblies

Ruptured fuel assemblies are stored in sealed cans. A can to be loaded is placed next to the monitoring can rack and the cover is removed. To assure that the can contains an air space for expansion of water, the cover is lifted to the surface to entrap air. It is then lowered in an upright position. The monitoring tank containing the defective element then is opened, and the assembly is removed from the monitoring tank and placed in the can without delay. The can then is closed and moved to the side of the basin where it is retained for future disposition.

18.10.3 Handling Instrumented Fuel Assemblies

After the instrumented fuel assemblies are monitored in the discharge chute, they are released to the entry trough by opening the lever-lock door. A long-handled, pneumatically-operated cable cutter is used to cut thermocouple leads, tubing, and supporting cables between the assemblies. The cutter design is such that it cannot cut into the fuel cladding. After dismantling, the assemblies are handled in the same manner as standard assemblies.

18.10.4 Waste Disposal

Small particles of graphite and metal resulting from spent fuel handling operations are removed with the basin cleaning tool. Water from the tool is filtered and recycled to the basin. Contaminated filter elements are handled as solid wastes. Graphite sleeves are disposed of by burial after a period of decay.

The basin water is circulated through the basin demineralizing system after fuel handling operations and at intervals between such operations to reduce the concentration of radioactive and nonradioactive impurities.

18.11 Shipment of Spent Fuel to Reprocessing Facility

After EGCR spent fuel has cooled a minimum of 100 days, it is shipped to the Savannah River Plant for storage. Fuel is shipped in a shipping cask, designed in accordance with the criteria set forth in Part 72 of the Atomic Energy Commission's "Regulations to Protect Against Accidental Criticality and Radiation Exposure in the Shipment of Irradiated Solid Fuel Elements" (10 CFR 72).

Conformance to these criteria affords reasonable assurance that the contents of the cask will not go critical, that excessive activity will not be released, and that radiation exposure to the public or personnel handling the cask will not be
18.11.1 Loading Spent Fuel into Shipping Cask

At the start of loading operations, the shipping cask is in the pit of the spent fuel basin. The first step in loading operations is removal of the cask lid. This is unbolted from the cask and placed on the basin floor using remote manual methods. Using the fuel element grapple, fuel clusters are transferred from the fuel basin storage rack to one of the cluster positions in the cask rack. After the cask rack is loaded, the cask lid is lowered into place on the cask and bolted using a long-handled wrench.

18.11.2 Preparing Shipping Cask for Shipment

All operations carried out with the loaded cask are done so as to minimize radiation exposure to personnel. The area around the shipping cask washing pad is roped off and designated a radiation or contamination zone.

As the loaded shipping cask emerges from the storage basin, it is monitored to assure that the gamma radiation level does not exceed 200 mrem/hr at the surface or 10 mrem/hr at a distance of one meter. Should either of these levels be exceeded, the cask is returned to the basin. To reduce the level to that permitted, some of the fuel is removed or the shipment is delayed for additional decay. If the radiation level is below these limits, the cask is placed on the cask washing pad and is washed down to remove loosely adhering contamination.

The cask is full of water as it leaves the basin. To assure that there is adequate space for expansion of water, the sampling return valve and the vent valve are opened to lower the water level to just below the bottom of the cask lid. Next, a check is made to ascertain that the sample valves are closed and the pressure relief valve (set at 50 psig) is installed. The cask surface is smeared to determine external contamination, and is scrubbed, if necessary, to reduce beta-gamma and alpha contamination below 4000 d/min/100 cm² and 500 d/min/100 cm², respectively.

The cleaned cask is lifted by the overhead crane and is positioned on the base, previously secured to the bed of a low-boy trailer. The cask is bolted to the base and the tie-down cables are secured to the trailer. The trailer is then hauled to an exclusion area several hundred feet away from the main buildings, where tests are made to determine the rate of temperature rise of the water in the cask. If it is determined that the temperature of the water in the cask when pressurized would exceed 267 F enroute to the processing plant, the cask is returned to the basin, and enough fuel elements are removed so that this temperature is not exceeded. Alternatively, the cask is held in the basin for additional cooling time. After temperature rise tests have been completed, the cask water is sampled, and the activity of the sample is measured to determine whether or not any of the fuel elements are leaking fission products. If the water sample has an activity level as high as 5 µc/cm³ of beta-gamma or 0.1 µc/cm³ of alpha, the cask is returned to the basin, and the fuel clusters are removed and monitored in the monitoring tanks. The cluster found to contain the ruptured element is canned. Immediately prior to shipment, the cask is pressure-tested with air to 40 psig.

18.11.3 Shipping to Reprocessing Plant

Fuel is shipped to the reprocessing plant by rail. The cask is trucked to the Oak Ridge Gaseous Diffusion Plant, where it is transferred to a railway flat car by a crane.

Shipments are made without a courier. The cask and trailer or flat car are
labeled in accordance with applicable regulations.

Rupture of the cask in an accident is improbable. Coolant may be lost if the cask is overturned; however, the fuel is not melted if the coolant is lost. EGCR fuel elements that have been irradiated to 10,000 Mw.d/MT contain a maximum of about 100 curies of fission products that have diffused from the UO\textsubscript{2} during normal operation. After 100 days' cooling, this activity is reduced by decay to about five curies. Cesium-137 is the principal activity present. Almost none of the activity is contributed by fission products which are gases at ordinary temperatures. Therefore, in the unlikely event that a fuel element ruptures during shipment, only a minor fraction of the activity escapes from the cask.

Should an accident occur during shipment, health physics and operating personnel are dispatched to the scene as required, to take necessary emergency measures in cooperation with the AEC radiological assistance team.

Upon receipt of the cask at the Savannah River Plant, handling is done by personnel of that plant. The cask is smeared, and the water inside is sampled. The cask is decontaminated, if necessary, and is unloaded in accordance with Savannah River Plant procedures. Fuel clusters are transferred from the cask racks to racks which are stored in the plant's storage basin.

After decontamination of the outside surface to comply with shipping regulations, the empty cask is returned to EGCR. Upon receipt at EGCR, the interior of the cask is decontaminated if necessary.

18.11.4 Shipping of Ruptured Fuel Elements

After cooling at least seven days, a ruptured assembly, in its can, is loaded into the fuel shipping cask for hauling to hot cell facilities at Oak Ridge National Laboratory. In the event of the loss of coolant from the cask, a pressure of 30 psig could be reached in the can at steady-state conditions. The can is designed to withstand a pressure of 50 psig.

Prior to shipment, a crew of operators is designated to take emergency measures in the event of an accident. Such measures include supplying water to the cask. Mobile cranes, capable of lifting the cask, are brought from nearby plants to any accident scene along the road between the EGCR and ORNL.

18.12 Fuel Control and Accountability

Each fuel assembly has a serial number imprinted on the top spider. All records concerning materials and manufacturing operations are recorded by the fuel manufacturer. These records include the weights and isotopic analyses of the UO\textsubscript{2} contained in each of the seven elements in each assembly.

The records maintained for fuel control and accountability at the EGCR contain the following information for each numbered assembly:

a. Date of receipt
b. Location in storage room or vault
c. Time and date of insertion and position in reactor
d. Information as in c. for subsequent reactor relocation
e. Time and date of removal from reactor
f. Location in spent fuel storage basin or in storage holes

g. Date and destination of shipment and position in shipping cask.

Irradiation level and plutonium and uranium-235 contents are calculated periodically for each assembly. When the fuel is reprocessed, calculated values are checked against values determined by isotopic analysis.
19. PLANT MAINTENANCE PROGRAM

19.1 General

The EGCR maintenance program ensures continued safe operation of the plant with a minimum of downtime consistent with economic considerations.

The routine inspection and maintenance program is similar to that for a high-pressure inert gas system and for a conventional TVA steam power plant, except it was modified to reflect more stringent requirements for the nuclear aspects of the plant. The inspection and maintenance programs for the conventional parts of the plant are standard. The maintenance program is based upon utilizing conventional methods and procedures for performing contact maintenance work. Special controls are incorporated to cover work within contamination and radiation zones. Work on contaminated equipment or systems is done under the surveillance of the Radiological Health Section which recommends required control measures. Careful planning, prewritten job procedures, and close coordination with operation and technical sections assure safe and efficient plant operation. Normal inspection contemplates an annual shutdown to permit inspection and maintenance of those portions of the plant not readily accessible during normal operation.

Where duplicate or stand-by equipment is provided for any component, which in itself is not vital to the safe operation of the plant, operations may be continued while repairs are made to stand-by equipment. If the defective component is vital to the safe operation of the plant, operations are continued using the spare or stand-by unit only until an orderly shutdown can be effected. Operating procedures preclude continued operation when serviceable stand-by units for critical equipment are not available.

Critical equipment, vital to the safe operation of the plant, is defined as items required to scram the reactor, remove afterheat, monitor the condition of the core, and maintain the integrity of the reactor coolant system and secondary containment. In general, any malfunctioning item of equipment which requires shutdown of the plant is classed as critical equipment.

A list, referred to as the Critical Equipment List, is compiled and issued to the Operations Group. This list states the requirements to be met before the equipment is taken out of service, and what tests are accomplished before the equipment is returned to service. All equipment not specifically designated on the Critical Equipment List is considered as noncritical and may be taken out of service, repaired, and returned to service according to normal steam plant practice.

19.2 Organization

Maintenance work on EGCR equipment and systems is performed by two groups. The Plant Maintenance Section is responsible for all mechanical and electrical maintenance work. The Controls Engineering Section is responsible for the maintenance of the instrumentation and controls system. Close cooperation between these two groups is maintained to facilitate scheduling, conserve manpower, and minimize plant downtime.
Under normal conditions, mechanical and electrical maintenance is accomplished on a day schedule, five days per week. Much of the routine instrument maintenance is carried out on a similar schedule; however, instrument mechanics are normally on shift with operations personnel.

19.2.1 Plant Maintenance Section

a. Personnel

The Plant Maintenance Section is composed of the section supervisor, mechanical engineers, an electrical engineer, and engineering aide, foremen, and craft journeymen. The crafts represented are machinists, steamfitters, boilermakers, electricians, carpenters, painters, and laborers.

The supervisor and engineers are responsible for:

1. Planning, scheduling, and controlling personnel, materials, equipment, and tools
2. Initiating training and educational programs for maintenance personnel
3. Establishing and supervising the maintenance of a readily accessible file of design and vendor information, parts data, preventive maintenance records, and historical records
4. Supervising foremen on all maintenance assignments, including instructions to cover safe working practices, radiation protection measures, and approved maintenance repair procedures
5. Making technical studies on maintenance of mechanical and electrical equipment, and making recommendations on design changes
6. Preparing labor and material costs estimates for nonroutine work.

The Plant Maintenance Section is directed by personnel with engineering backgrounds. Technical support is available from the Technical Program Group and the Radiological Health Section, which provides specialists as required.

b. Facilities

The Plant Maintenance Section and shop facilities are organized to perform field maintenance work primarily.

On-site shop work consists basically of minor repairs, replacement of defective components, and checkout of equipment. The bulk of the work is of short duration and minor complexity, and the shops are equipped accordingly. Machine, electric, pipe, carpentry, and welding shops are provided. In cases where maintenance operations require facilities not provided at the site, shops in the Oak Ridge complex of Union Carbide Nuclear Company and TVA Central Shops are utilized as required. Craft personnel from nearby TVA installations are available to assist EGCR crafts in cases of emergency.

19.2.2 Controls Engineering Section

a. Personnel
The maintenance of the instrumentation and controls system is the responsibility of the Controls Engineering Section. The supervisor and instrument engineers are responsible for:

1. Adequacy of the maintenance facilities and the training of personnel to meet all requirements, both routine and emergency
2. Planning and scheduling of all instrument maintenance in cooperation with the Plant Maintenance Section
3. Establishment of a preventive maintenance program for all control systems and components, especially those involving the safety of the plant
4. Planning and maintenance of a file system that contains the information necessary to analyze design, order spare parts and components, apply preventive maintenance procedures, and provide history of repairs on all equipment.
5. Supervision of the instrument foreman on all maintenance assignments.

b. Facilities

The instrument maintenance shop is located on the second floor of the reactor service building. The main shop area has 1350 ft$^2$ of floor space. The instrument storage area, located on the same floor, has 100 ft$^2$ of floor space. The shop is equipped with services (water, air, electricity), tools, and test equipment necessary for the calibration and maintenance of either pneumatic or electronic instruments. In addition, special instruments are provided for test and experimental work, and training purposes.

A small area in the reactor building, equipped with a work bench and hand tools, is provided for maintenance of slightly contaminated instruments. This eliminates the need for complete decontamination before repair. Instruments, tools, and equipment used in this area are checked by the Radiological Health Section before transfer to another area. Strict handling, cleanliness, and radiation control procedures are maintained in this area.

19.3 Maintenance Categories

The Plant Maintenance Section or the Controls Engineering Section perform three categories of work, preventive maintenance and inspection, routine maintenance, and nonroutine maintenance. Any of these categories of work may involve hazardous conditions due to radiation or contamination. The procedures used in performing this work depend upon both the category of work and the degree of hazard involved due to direct radiation or contamination.

19.3.1 Preventive Maintenance and Inspection

The preventive maintenance program minimizes shutdowns and breakdowns by systematically inspecting equipment, making calibrations or adjustments, and scheduling repairs and overhauls before failures occur.

Each piece of equipment is studied thoroughly, and a schedule of routine inspections is determined and established under the following classifications:

a. A-Class: Major inspection (complete check of equipment). Usually made
semiannually to annually.

b. B-Class: A "middle-of-the-road" inspection. Usually made quarterly to semiannually and, on occasions, monthly.

c. C-Class: A minor inspection (ordinarily visual and frequent). Usually made monthly to quarterly and, on occasions, weekly.

As each piece of equipment is studied, a complete list of items to be checked on each inspection is made. A central control system serves to indicate which inspection is due and when. If inspections do not interfere with normal plant operation, the inspections are scheduled and carried out in accordance with work loads in the section. Inspections that require shutdown of equipment or which interfere with normal plant operations are coordinated with the Plant Operations Section.

After inspection is completed, information is transferred from the inspection sheet to a card for the record. If any repairs are necessary, such repairs fall into the category of routine maintenance and are scheduled according to the urgency required.

19.3.2 Routine Maintenance

Routine maintenance includes all maintenance work on equipment or systems which is directed toward restoring the equipment or system to its normal functioning capability, but does not alter its basic design function. Routine maintenance is conducted during normal plant operation, as well as during scheduled shutdowns.

Normal routine maintenance work is either requested by the Operations Section or results from the preventive maintenance program.

There generally is a backlog of work. Thus, all work is given a level of priority to facilitate effective scheduling. Priority is based on safeguards considerations, production loss resulting from the equipment being shut down, or the probability of a breakdown if a repair is not made, and consequent damage to equipment.

19.3.3 Nonroutine Maintenance

Nonroutine maintenance includes modifications to systems or processes, as differentiated from repair or replacement of faulty equipment. Depending upon the nature and extent of the work, maintenance or construction forces are used. In the latter case, the Plant Maintenance Section or the Controls Engineering Section is responsible for estimating the job, maintaining close contact with the work to see that it is performed in accordance with specifications, within the cost estimate, and reporting on the progress of the job during the construction period.

19.3.4 Hazardous Maintenance

Before a maintenance operation is initiated on equipment, a complete radiation-contamination survey is made by the Radiological Health Section. If the direct radiation dose rate were to result in integrated doses exceeding those in Table 22.2.1, or if contamination levels exceed those values in Table 22.3.1, maintenance procedures are amended in accordance with the requirements indicated in Section 19.4.3.

19.4 Administrative Procedures for Carrying Out Program

19-4
All work performed in the various categories of the maintenance program, including those of the Plant Maintenance Section and the Controls Engineering Section, both during normal plant operation and during plant shutdown, are in accordance with established administrative procedures described below. These administrative procedures deal with the conditions or requirements that must be satisfied to initiate and complete a maintenance operation rather than to exercise control over the actual repair work.

19.4.1 Noncritical Components

Noncritical components are not required for the safe operation of the facility. Therefore, preventive maintenance or routine maintenance operations on noncritical components is carried out by one of the maintenance sections in accordance with normal steam plant practice, except as noted in Section 19.4.3.

The maintenance work on noncritical components is coordinated with the Operations Section to minimize downtime. Detailed maintenance procedures for most pieces of equipment are provided by the vendor or are written by the maintenance sections. If radiation contamination levels are associated with the maintenance operation, the operation must be altered and is administratively controlled as described in Section 19.4.3. Nonroutine maintenance of a noncritical component is discussed in Section 19.4.4.

19.4.2 Critical Components

Critical components are required for the safe operation of the facility. Therefore, prior to performing preventive maintenance or routine maintenance, it is necessary to evaluate the effect of performing the maintenance work. Such an evaluation is made on all items listed as critical equipment. The Critical Equipment List is prepared by the Operations Group and the Technical Program Group and approved by the project manager. If the maintenance work does not involve a radiation or contamination hazard, the work is initiated after approval by the operating superintendent. If a radiation or contamination hazard is associated with the maintenance job, it is necessary to alter the operation as described in Section 19.4.3. Nonroutine maintenance of critical components is discussed in Section 19.4.4.

19.4.3 Hazardous Maintenance

When hazardous conditions exist, it is necessary to alter normal maintenance procedures before maintenance is initiated.

In all cases, a Radiation Work Permit is required. The use of this permit provides maximum assurance that both the worker and management take adequate steps to minimize the consequences of radiation or contamination associated with the job. A complete description of the Radiation Work Permit is given in the Radiological Health Manual for the EGCR. Section 23.6.1 describes the use of the Radiation Work Permit and the control exercised to minimize radiation or contamination hazards.

In all cases involving hazardous maintenance, it is necessary to fulfill the requirements set forth in the Radiation Work Permit. After this is done, the maintenance operation is performed in accordance with Sections 19.4.1 and 19.4.2.

19.4.4 Nonroutine Maintenance

Nonroutine maintenance involves changes in the basic design. When it is necessary to perform this type of maintenance, on either critical or noncritical components, such maintenance is not carried out until a complete evaluation of
such a change is conducted. The evaluation and approval steps are described in Section 10.4. After the procedure is approved, the maintenance operation is performed in accordance with Sections 19.4.1 and 19.4.2.

19.4.5 Clearance Procedures

To protect the service to the public and the safety of the employees, TVA clearance procedures are adapted for use at the EGCR. Clearances are issued by number only to properly authorized persons whose names appear on the official clearance list which is approved and issued by the operating superintendent. All persons on this list must be familiar with the methods and purpose of each type of clearance. The duty shift engineer is responsible for the issuance of all EGCR clearances.

No work is performed on or near lines or equipment until the shift engineer is informed by a maintenance representative of the work to be performed. Before a clearance is issued by the shift engineer to the individual in charge of the maintenance work, necessary operations are carried out and protective tags applied by the operator. The maintenance representative receiving the clearance must be satisfied that the clearance is properly executed according to established procedures. Upon completion of any work for which clearances are held, the person holding this clearance reports to the shift engineer giving exact details of the work completed and then releases his clearance by number to the shift engineer. No further maintenance work is performed after clearances are released.

Strict adherence to the clearance procedures ensures that the reactor operators are fully informed of any proposed nonavailability of equipment or services, the length of the outage, and the work being performed.

19.4.6 Work Completion

Representatives of the Plant Maintenance Section, Controls Engineering Section, Plant Operations Section, and Technical Program Group (if involved) observe the testing and return to operation of the component or system involved in maintenance.
20. **DECONTAMINATION**

20.1 **General**

Decontamination of equipment and plant structures is necessary to:

- a. Prevent the spread of radioactive contamination.

- b. Reduce existing contamination levels to a point that will permit the continued use of an area, component, or tool.

- c. Reduce radiation intensities and contamination levels to provide adequate working time for direct maintenance of plant or equipment.

- d. Reduce the radiation intensity or contamination levels to allow sufficient working time for disposal operations.

Planning for the decontamination of major pieces of equipment is the responsibility of the Plant Maintenance Section. Technical assistance in the choice of decontamination reagents and methods is given by the Chemical Engineering Section. Radiological Health Section guidance is utilized from the earliest stages of planning and throughout the decontamination operation. An evaluation of a proposed decontamination operation is made to identify and to minimize any hazards to personnel or damage to the systems being decontaminated.

Major decontamination operations are carried out by the Plant Maintenance Section with surveillance by the Radiological Health Section. Decontamination solutions are, in general, prepared under the direction of the Chemical Engineering Section.

Various reagents are employed in decontamination operations. The selection of reagents depends upon the nature of the contaminant and the nature of the contaminated surface. Detergent solutions are used on many types of surfaces where the contamination is loosely bound. Contaminated dust is removed by vacuum cleaning, followed by flushing, or scrubbing with detergent solution. Dilute solutions of ammonium oxalate, citric acid, and hydrogen peroxide are used for dissolving UO$_2$ particles and associated fission products. Other reagents, both acidic and basic, are employed as the need arises. No reagent is used unless it has first been determined that its use is not detrimental to the system.

Liquid wastes resulting from decontamination operations are collected in the hot waste sump tank or the warm waste sump tanks, from which they are routed to the hot waste storage tank or to the warm waste retention basin. Solid wastes from decontamination operations such as absorbent pads, blotting paper, scrubbing brushes, mops, and the like are disposed of at the ORNL burial ground.

Major hazards in any decontamination program are those associated with airborne contamination, direct radiation exposure, and the possible spreading of contamination through the leakage of highly contaminated solutions. Prior to initiating a decontamination operation, the Radiological Health Section conducts a radiation and contamination survey. Following these surveys, a Radiation Work
Permit is issued prior to the start of work. (Refer to Section 22.2 for definitions of radiation zones and contamination zones.) Issuance of a Radiation Work Permit requires that the safety procedures to be used, including protective clothing, monitoring devices, etc., are reviewed by the employees, the employees' supervisor, and the attendant radiological health representative before a job is started.

Most decontamination operations are carried out under conditions where gaseous or airborne activity is discharged directly to the gas vent system or to an exhaust hood. When decontamination is done in the reactor or service buildings, the regular heating and ventilating systems carry off airborne contamination. Portable air samplers are used to monitor airborne particulate activity in areas in which decontamination is done in the open. Employees wear protective clothing and respiratory equipment as required.

20.2 Decontamination of Tools and Small Parts

Many contaminated tools and small parts are not decontaminated, but are disposed of by burial. If they are sufficiently valuable, however, or can be returned to service cheaply, decontamination is undertaken. Certain items such as instruments, which are contaminated to a relatively low level, are decontaminated prior to maintenance to prevent the spread of contamination in otherwise clean areas.

Decontamination of small parts and tools is accomplished in a small tank located in the decontamination area. Tank fittings include an exhaust hood with integral fan, an agitator, and a heating coil. Tools and small parts are segregated according to materials of construction, level, and type of contamination. When a suitable number are accumulated, they are placed in a wire basket and immersed in the tank. Items unable to withstand this total immersion are decontaminated by hand in a manner similar to that used to remove local skin contamination from personnel. During the decontamination operation, gaseous waste is discharged to the plant stack after passing through roughing and absolute filters.

20.3 Charge and Service Machines Decontamination

The charge and service machines are subject to contamination during normal operation or as a result of abnormal conditions either in the reactor coolant system or in the machines. While this contamination does not, under normal circumstances, interfere with the normal operation of the machines, such contamination precludes or severely limits direct access to the interior of the machines for maintenance.

Contamination of the service machine and the charge machine results from fission products or particulate UO₂.

The equipment provided for make-up and supply of decontaminating solutions for the machines is described in Sections 5.9.10, 5.10.6, and 6.7.2 of Volume I. This equipment is located in the reactor service building on the basement floor. Piping headers from this equipment to both the charge machine and the service machine areas terminate in blind flanged closures.

20.3.1 Charge Machine Decontamination

Decontamination of the charge machine for maintenance or replacement of machine internals is not expected to be required more often than once a year. Decontamination is carried out in three phases. The first phase consists of spraying decontaminants into the charge machine vessel through built-in sprays. This work is done by maintenance and operating personnel in the charge machine room. The second phase consists of direct contact decontamination of hot spots. This
is done after the charge machine internals are removed from the vessel and moved to the decontamination area. The third phase consists of decontamination of individual parts as the machine internals are dismantled. The second and third phases of decontamination are performed by maintenance personnel. The third phase of the decontamination is done on a nonpriority basis since a complete set of spare internals is available. This duplicate set of internals is exchanged with the operating set whenever required for maintenance, decontamination, or alterations.

Section 5.9.10 of Volume I describes the decontamination facilities provided for decontaminating the charge machine. Before a decontamination operation is started, the work area is completely surveyed by the Radiological Health Section, and radiation or contamination zones are established. Temporary shielding is erected where needed, and entrance and exit requirements for the area are set up. Decontamination of the charge machine is done following a step-by-step procedure. The charge machine is moved to the decontamination area of the charge machine room. The machine is purged to reduce the gaseous activity level to a minimum. Purging or venting of gaseous activity is in accordance with the requirements discussed in Section 21.2, Gaseous Waste Disposal. Flexible hoses are connected to the decontamination solution supply and drain lines which are normally stubbed off in this area. Alternate cycles of hot water, decontamination agent, and hot water are sprayed throughout the machine to decontaminate the internal mechanisms to acceptable levels. Plant air is used to dry the machine internals after decontamination.

Drainage from the machine to the hot waste tank is monitored to determine the effectiveness of the various reagents used. Floor drains in the area are opened to discharge solution to the warm waste sump tank in the event of a spill. The warm waste sump tank is set to discharge to the hot waste storage tank during these operations. All gases evolved in the decontamination operation and air used for drying are automatically directed to the gas vent system for cleanup and then discharged to the stack. If replacement or maintenance of the charge machine internals is desired, the head is removed from the machine and a radiation survey is made of the charge machine internals. If a hot spot is identified, an appropriate decontamination agent is directed on the area through a hose until the activity level is reduced to that required for limited duty one meter from the contaminated surface.

After the internals are removed from the machine, they are sealed in a large plastic bag to prevent spread of contamination. The internals are then transferred to the decontamination area. If the surface contamination level is acceptable, the bag is removed and direct maintenance is undertaken. If the surface contamination is too high to permit maintenance operations, further local decontamination is required. If radiation or contamination levels are such that this is impossible, the internals are stored in the yard and temporary shielding and personnel barricades are erected as required until the radiation or contamination decay to acceptable levels. Acceptable radiation and contamination levels are specified by the Radiological Health Section in accordance with the standards given in Tables 22.2.1 and 22.3.1.

20.3.2 Service Machine Decontamination

Decontamination of the service machine is done in a manner similar to that used for the charge machine. Prior to removal of any component from the service machine, the machine is depressurized and purged in accordance with the requirements discussed in Section 21.2. A survey is made to determine the level of radioactivity prevailing within the machine. This survey is conducted by inserting a probe through the 2-in. connection on the top of the machine. From the results, an indication of the source of the radioactivity and the decontamination requirements are established. The service machine is then moved to the
decontamination area of the service machine room. Decontamination of the service machine is performed in accordance with established procedures.

If the tip of the fuel handling chute becomes highly activated during service machine operations, it is necessary to postpone decontamination for several days until radiation level decays. If this is impractical from an operating standpoint, the chute tip is removed in the rehearsal area. The service machine is fitted with permanently installed internal headers and spray nozzles through which decontaminating fluids are sprayed throughout the machine. The machine is thoroughly rinsed with hot water and rechecked for high radiation levels. If high levels persist, the decontamination cycle is repeated until acceptable radiation levels are achieved.

20.4 Area Decontamination

The prime objective in area decontamination is to return the contaminated area to normal use quickly and economically without destroying its usefulness and without subjecting personnel involved in the decontamination operation to an excessive radiation or contamination hazard. Area decontamination is required following accidental spills of radioactive materials or as a result of any operation that results in contaminating surfaces in excess of the values designated in Section 22.3.

When excessive contamination is detected in an area, radiological health personnel immediately establish a contamination zone and post signs indicating the requirements for entry, occupancy, and departure. Area decontamination is then carried out by the Plant Maintenance Section with guidance by a radiological health representative. Both mechanical and chemical methods are employed in decontaminating areas. Mechanical methods include vacuum cleaning, abrasive cleaning, scraping, and surface removal. Chemical methods include the use of water and detergents, steam and detergents, acids, caustics, and solvents. In some cases, if repeated attempts to remove contamination are unsuccessful in reducing the contamination to an acceptable level, contamination is fixed to surfaces by painting or covering with other suitable bonding materials.

In all area decontamination operations, special care is taken to minimize the spread of radioactive contaminants. When applying mechanical methods, care is taken to maintain control over airborne contamination. Likewise, in applying chemical methods, special care is taken to prevent the spread of any liquid wastes.

Most areas that contain potential sources of contamination are equipped with floor drains which lead to one of the three waste sump tanks. These drains, which are normally closed, are opened for draining decontaminating solutions to the waste handling system. In areas that are not equipped with floor drains, decontaminating wastes are mopped up. In areas such as the waste tank sampling points, where the probability of small spills is relatively high, absorbent pads and tissue are supplied. The operator routinely wipes the sampling equipment and deposits the pads and any released contamination in a contaminated waste can.

20.5 Decontamination of Major Components and Systems

A major component is decontaminated to perform maintenance or to reduce the radiation level in the vicinity of the component. Radiation surveys of equipment are made on a routine basis, and special surveys are made following any abnormal operations that lead to excessive contamination buildup.

Some decontamination operations performed on major components or systems influence the safety of the reactor. Therefore, it is necessary to meet specific
requirements prior to initiating such an operation. Approval to commence de-
contamination is granted only after it is established that the operation can be
carried out with adequate heat removal from the core and with adequate contain-
ment of activity. Decontamination operations include provisions to assure that
the operation does not unfavorably alter the condition of the reactor coolant
system from the standpoint of increased leakage, degradation of physical
strength, or increased contamination by nonradioactive material.

Decontamination of a steam generator or a reactor coolant blower requires that
the helium pressure be reduced to slightly above atmospheric pressure. The
steam generator or blower being decontaminated is isolated from the rest of the
system by means of the reactor coolant isolation valves. Decay heat is removed
by operating the blower in the opposite loop or by using one of the vessel
cooling compressors. The emergency cooling system heat removal loop is used
as a backup for removing decay heat.

Specific procedures have not been developed for decontaminating such major
components as the steam generators and blowers. These components have not been
provided with built-in decontamination devices, as have the charge and service
machines, because they are not expected to require frequent decontamination.
Specific decontamination procedures for the steam generators, reactor coolant
blowers, and other major components will be devised on the basis of conditions
existing at the time.
21. WASTE DISPOSAL

21.1 General

Disposal of radioactive wastes produced during operation is controlled so that the associated radiation does not constitute a hazard to plant personnel or the general public. All equipment comprising the waste disposal system is operated to minimize the release of radioactivity to the environment and in all cases to maintain such releases below the levels designated in Section 22.

Equipment and facilities provided for the disposal and monitoring of radioactive waste are described in Volume I.

The following sections describe the methods of disposing of gaseous, liquid, and solid waste. In all operations, the basic objective is to control the location or mobility of waste products such that the disposal operation is accomplished safely. Disposal operations take advantage of the various methods for reducing the activity concentration in effluents, including deposition, absorption, filtration, dilution, and radioactive decay.

21.2 Gaseous Waste Disposal

Gaseous wastes, including airborne particulates, originate from the following sources:

a. Reactor coolant system leakage
b. Controlled venting of the reactor coolant system
c. Fuel chute vent
d. Warm waste sump tank vent
e. Spent fuel storage basin
f. Hot waste sump tank and storage tank vents
g. Decontamination of equipment
h. Laboratory hood exhaust
i. Activation of shield cooling air
j. Helium sampling system
k. Controlled venting of the charge and service machines
l. Miscellaneous equipment leakages

Those parts of the reactor facility which either normally, or at the time of an emergency, contain large amounts of gaseous radioactivity are located inside the reactor building. Spread of airborne contamination is minimized by directing flow from areas of least contamination to those areas that are potential sources of high airborne contamination. Gaseous waste products originating inside the containment shell are discharged at a controlled rate from the stack. Prior to discharge, all of this waste is passed through particulate filters. The discharge from the biological shield cooling system and the gas vent system also flows through a silver-plated copper mesh filter for iodine removal.

The hot waste storage tank, the experimenters and auxiliary warm waste sump tank, the warm waste retention basin, and the spent fuel storage basin are the only potential locations of gaseous activity exterior to the containment shell.
The reactor service building ventilation system is used for the disposal of gaseous waste originating in the hot waste storage tank, spent fuel storage basin, and warm waste sump tank.

21.2.1 Normal Discharge of Gaseous Waste

During normal operation of the reactor the amounts of radioactive gaseous wastes are not great. The primary source of radioactive waste is approximately 367 curies/day of argon-41 which is produced by activation of argon-40 in shield cooling air. The continuous discharge of 367 curies/day of argon-41 with the shield cooling air does not result in excessive exposure dose for the meteorological conditions expected at the EGCR site. Flows of other gaseous waste are either maintained at a low level of activity or are not discharged continuously. The primary methods of controlling the release of fission product activity during normal operation are based upon minimizing the activity available for release, and minimizing the leakage rates from equipment.

To minimize the activity available for release, the burst slug detection equipment is utilized to detect fuel element failures. When fuel failures occur that result in excessive levels of activity in the reactor coolant system, the failed fuel is removed from the core. Gas samples are taken from the reactor coolant system as required to determine the fission product concentration. Likewise, radiation detectors adjacent to the reactor coolant system piping are utilized to indicate when high concentrations of activity are contained in the reactor coolant system as mobile or plated-out sources. Activity buildup in the purification system or on filters in the biological shield cooling discharge lines is used to obtain indication of levels of activity in the coolant available for release.

The operator obtains information from the above sources. This information is analyzed to determine if conditions exist or are building up that might require excessive gaseous waste disposal. To minimize the leakage of contaminated helium from equipment, it is necessary to locate the area of leakage and perform the necessary maintenance. It is possible to determine leakage rates for such equipment as the charge and service machines before the machines are connected to the reactor coolant system. Leakage rates from the reactor coolant system and those systems connected to the reactor coolant system are obtained from the record of helium makeup. Gross leaks also are detected and located by the helium leak detection system. This system samples air at 16 locations within the reactor building and is capable of detecting up to 5% ±0.1% helium in air by volume. In addition, six continuous air monitors located throughout the reactor building serve to indicate areas where fission product leakage is significant.

Gaseous waste discharged to the stack from the reactor and service buildings is monitored. Continuous readout and alarms are provided in the main control room to indicate the activity of particulate, iodine, and total gas. For normal discharges of gaseous waste, the total gas monitor indicates the amount of argon-41 release. Indication of fission product release is obtained from the particulate and iodine monitors. By maintaining proper surveillance of the amounts of activity in the reactor coolant system as well as the systems connected to the reactor coolant system and by assuring that leakage rates are not excessive, the discharge of gaseous wastes to the stack does not result in levels which alarm in the control room, nor does the discharge of activity result in concentrations of activity at the point of maximum ground concentration in excess of the values specified in Section 22.

For normal discharge of gaseous waste, when there is sufficient information to conclude that the release of fission products is not significant, analyses of the filters in the particulate and iodine monitors are not required but may be
performed to obtain some indication of the isotopic content of the activity being released. When the instruments in the control room indicate an abnormal increase in stack activity level, the operator locates the cause of the increase and takes proper corrective actions.

21.2.2 Abnormal Release of Gaseous Waste

Any release of gaseous waste in quantities or concentrations that exceeds the stack activity concentrations for alarm is considered an abnormal release.

Releases of gaseous activity are of two types, controlled and uncontrolled. The consequences of the abnormal release depend upon the amount and isotopic content of activity being released, the duration of the release, and the meteorological conditions at the time of the release. Those actions taken by the operator to minimize the activity available for release during normal operation in many cases reduce the consequences of abnormal releases to acceptable levels with few further corrective actions required. However, in some cases, additional action is required to provide adequate safety.

a. Controlled Releases

Controlled releases include any of the venting operations that result in discharge of activity from equipment or systems. Although the quantity and isotopic content of gases vented from various locations differ, the procedure for carrying out the operation is the same in all cases. All controlled releases of activity are carried out in accordance with approved standard operating procedures. These releases fall into three general categories: discharges that result in activity concentrations in the stack that are less than the levels that cause a stack monitor alarm; discharges that result in activity concentrations in the stack that are greater than the levels that cause a stack monitor alarm, but less than the levels that cause automatic isolation of the containment shell; and discharges that result in activity concentrations that exceed the levels that cause automatic isolation of the containment shell. The concentrations that cause alarm and automatic isolation are indicated in Table 21.2.2.

<table>
<thead>
<tr>
<th>TABLE 21.2.2</th>
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<tbody>
<tr>
<td>Stack Activity Concentrations for Alarm and Automatic Isolation</td>
</tr>
<tr>
<td>Monitor</td>
</tr>
<tr>
<td>Total gas</td>
</tr>
<tr>
<td>Iodine</td>
</tr>
<tr>
<td>Particulate</td>
</tr>
</tbody>
</table>

The concentrations to cause an alarm are based on:

1. A continuous release of unidentified activity

2. A maximum concentration of 0.1 occupational maximum permissible concentration in air ($MPC_a$)

3. The most adverse meteorological conditions representing the EGCR site.

The concentration to cause automatic isolation is based on:
1. A release of unidentified activity for a period of 30 min.

2. An integrated exposure dose equal to 0.1 of the allowable occupational yearly exposure dose to the thyroid.

3. The most adverse meteorological conditions representing the EGCR site.

Before venting, the concentration of activity and the total amount of gas to be vented is determined. The concentration of activity is obtained by analyzing a gas sample or by monitoring a sample discharge. If the activity to be discharged is small enough such that the alarm setting in the stack monitor is not reached, the operation proceeds with no further action being required. During the venting period, the operators maintain close watch of the stack monitor readout in the control room. Unless the alarm setting is reached, the operation proceeds as planned. If the alarm setting is reached, the discharge rate is decreased to below the alarm setting. If this cannot be done, the venting operation is halted and additional action is taken in accordance with the procedures described below.

If gas sampling indicates that the discharge of the gas is sufficient to cause an alarm of the stack monitor or if venting causes an alarm, further venting is halted. An evaluation is made of the consequences of the abnormal discharge, based upon knowledge of the amount of activity to be discharged, the meteorological conditions at the time, and records of previous discharges. If it is concluded that the discharge can be made without additional information, the operation continues. It appears that the consequences will be excessive, further information such as the isotopic breakdown of the activity is required. In most cases where it appears that the activity level is excessive, venting is postponed until activity decays or until meteorological conditions are favorable. Gas sample analysis is utilized, as required, to guide the assessment of the hazard.

For those venting operations that would not only exceed the alarm setting but could also exceed the isolation setting if preventive action is not taken, venting is carried out only after evaluating the consequences of the release, based upon a knowledge of the amount and identity of the activity to be released, the meteorological conditions, and records of previous releases. This case differs from the previous case in two important respects; the amount of activity or the potential rate of its release is sufficiently high that, unless adequate steps are taken by the operator to control the release rate, the concentration in the stack would cause automatic isolation, and the total amount of activity that can be discharged is limited. It is necessary to prevent automatic isolation during the venting operation. This is accomplished by either discharging over longer periods of time or by holding up discharges until the activity concentration decreases by natural decay. The amount of activity that can be vented is limited, even for those cases where the discharge does not cause a concentration in the stack that results in automatic isolation. Any release due to venting must not only assure that the concentration is below the value that causes isolation, but also must assure that the total amount of release of specific isotopes does not exceed a given amount. The specific amount that is released varies according to the conditions that exist at the time the venting operation is scheduled, and is also influenced by previous venting operations or abnormal releases.

b. Uncontrolled Releases

21-4
Uncontrolled gaseous waste releases result from increased leakage rates from equipment or systems containing radioactive gaseous waste or from an increased source of gaseous waste. Uncontrolled releases are limited by automatic isolation of the containment shell or by operator action following a stack monitor alarm.

Immediately following an alarm, the operator determines the cause of the high activity concentrations in the stack. At the same time, the operator observes the monitor readings to determine if the concentrations are changing in a safe direction. Since the alarm settings are established at a level that does not result in excessive exposure to the general public, it is not immediately necessary for the operator to isolate the containment shell or shut down the reactor. These actions are taken by the operator if concentrations increase or if the source of the high activity discharge cannot be located within 30 min.

In all cases, if activity concentrations increase to a level of $1.2 \times 10^{-3} \mu\text{c/cm}^3$, the containment shell is automatically isolated.

21.3 Liquid Waste Disposal

The EGCR system for the collection, temporary storage, and disposal of warm and high level liquid waste is described in Sections 5.1.11 and 6.8 of Volume I.

21.3.1 Liquid Waste Collection

Liquid wastes from various plant areas flow by gravity through separate stainless steel piping systems to three sump tanks, two in the reactor building and one in the reactor service building. The contents of the sump tanks are discharged to the warm waste retention basin or to the hot waste storage tank depending upon the activity level. Material in the warm waste retention basin is periodically released at a controlled rate to the Melton Hill Lake. Liquid in the hot waste storage tank is transferred to ORNL by tank truck for final disposition.

Each of the sump tanks is emptied automatically by a pump or a pair of pumps controlled by level switches. Manual override of these controls is carried out only with the approval of the plant operations supervisor. Liquid levels are recorded at least once each shift, and a sample is taken at this time if the tank level has increased by 50 gal or more. Samples are collected in 50 ml bottles and after a preliminary gross beta-gamma survey are taken to the analytical laboratory.

The contents of any sump tank can be discharged to the hot waste storage tank or to the warm waste retention basin. The normal position of valving is such that the warm waste sump tanks discharge to a warm waste retention basin and the hot waste sump tank discharges to the hot waste storage tank. During most operations of the liquid waste collection system, it is possible to obtain samples prior to discharging waste from the sump tanks. If the gross beta-gamma activity concentration is less than $10^{-4} \mu\text{c/cm}^3$, the waste is treated as warm waste and is discharged to the warm waste retention basin. Wastes of higher concentration are discharged to the hot waste storage tank.

When it is necessary to decontaminate the charge or service machines, it is estimated that approximately 20,000 gal of decontamination solutions are required. These solutions flow into the hot waste sump tank at rates up to 150 gpm. Under such conditions, the 500-gal hot waste sump tank acts as a surge tank with both pumps operating almost continuously. Most of the decontaminating solutions are pumped to the hot waste storage tank. During this operation, samples are not taken from the sump tank, but the activity concentration of the
solution is monitored by the continuous monitors on the discharge lines from the sump tanks. Samples are taken from the hot waste storage tanks after each addition of a batch of solution to indicate the progress of the decontamination operation. During the latter stages of decontamination, rinse water may be pumped to the warm waste retention basin. Decontamination solutions may also be discharged to the warm waste retention basin when the level of activity concentration, as measured by the monitor on the discharge from the sump tank, decreases to a level of $10^{-4}$ $\mu$C/cm$^3$. The hot waste storage tank has a capacity of 13,000 gal. If the level of activity concentration of the decontamination solutions remains high for a large fraction of the 20,000 gal of solutions required, a number of transfers of hot waste to ORNL are necessary.

21.3.2 Normal Discharge of Liquid Wastes

Under normal operating conditions, all warm waste is discharged at a controlled rate into Melton Hill Lake, and all hot waste is transported to ORNL for disposal. The program established by the Operator to control liquid waste disposal operations is based upon minimizing the amounts of warm waste that are discharged to Melton Hill Lake. The requirement for the EGCR is that the liquid effluent, discharged into Melton Hill Lake, assuming complete mixing and minimum flow, shall not increase the activity concentration in the lake by more than $10^{-8}$ $\mu$C/cm$^3$ (Section 9.6.2 of Volume I). In order to meet this requirement, operations are generally directed at maintaining the concentration in the discharge to the lake below $10^{-7}$ $\mu$C/cm$^3$.

a. Warm Waste Disposal

The two 45,000 gal sections of the warm waste retention basin are filled and emptied alternately. Each section is provided with a level switch which annunciates in the control room on high level. Prior to discharge, a sample is taken from the basin and checked for gross beta-gamma activity. If the activity of the basin contents is at its maximum value of $10^{-4}$ $\mu$C/cm$^3$, approximately 28 hr are required to discharge the basin to the service water return sump. At this flow, the concentration of activity is less than $10^{-7}$ $\mu$C/cm$^3$ when mixed with the expected flows of 2300 gpm of service water and 25,000 gpm of condenser circulating water. Flow from the basin and diluting flows are varied to maintain a concentration activity in the plant effluent below $10^{-7}$ $\mu$C/cm$^3$. Periodic samples are taken from the combined warm waste and return water stream at the point in the lake where it leaves the plant discharge line to verify the adequacy of the in-plant controls. Off-site monitoring includes a water sampler at Melton Hill Dam in addition to mud, liquid, and biota sampling of the EGCR environment as described in Section 23.5.

b. Hot Waste Disposal

Liquid waste collected in the hot waste storage tank is transferred by tank truck to the ORNL waste disposal system. Transfers to ORNL are infrequent except when the charge or service machine is being decontaminated. Decontamination of the charge or service machine requires a number of transfers to ORNL. Prior to each transfer, the liquid in the hot waste storage tank is sampled. A determination of pH and major constituents of the waste is made by the analytical laboratory. The waste is neutralized at this time as required by ORNL for addition to their waste treatment system.

Hot wastes are pumped from the hot waste storage tank to a 4000-gal tank truck. The transfer from the storage tank to the truck is made by plant operators under the surveillance of the Radiological Health
Section. Prior to the transfer, the connections between the storage tank pump and the truck and between the truck and the storage tank vent are checked for leaks. A continual survey is made of the transfer operation to prevent excessive exposure to personnel or the release of contamination from the equipment. It requires approximately three hours to fill the transfer truck from the hot waste storage tank using the 25-gpm pump. During this period, excessive exposure to operators is minimized by isolating the immediate area of the transfer, by using temporary shielding and by changing operators. With the expected activity levels, only the first of these measures is needed under normal conditions. When the transfer operation is completed, hoses are rinsed and disconnected and the ends are capped. Decontamination of equipment and the area is conducted as required.

During the two mile trip to ORNL, radiation exposure to a driver is minimized by using temporary shadow shielding on the tank, partially filling the tank, or changing drivers. For certain conditions, it may become necessary to provide an escort for the shipment and schedule the transfer for periods when public usage of the connecting road to ORNL is low.

Upon receipt of wastes at ORNL, disposal is accomplished in accordance with the established ORNL procedures.

21.3.3 Abnormal Release of Liquid Waste

Any release of liquid waste or the discharge of liquid waste from the waste disposal system, with concentrations or discharge rates that results in increasing the concentration of activity in the lake by more than $10^{-8}$ $\mu$Ci/cm$^3$ is considered an abnormal release. Abnormal releases of liquid waste could result from either equipment failure or operator error.

a. Abnormal Release of Warm Waste

Abnormal releases of warm waste could result from the failure of heat exchange equipment allowing activity to contaminate the service water. The service water system supplies cooling water to heat exchangers and other equipment in which reactor coolant and blower seal water are handled. In the event of a leak, the service water is contaminated with radioactive impurities contained in the helium. An in-line radiation detector on the service water discharge line warns of activity concentrations. When the alarm associated with this instrument is actuated, steps are taken to locate, isolate, and repair the leak. If the activity concentration is less than $5 \times 10^{-6}$ $\mu$Ci/cm$^3$, the leak is not detected by the instrument. The service water return line is sampled and checked for activity content once a week. With the normal service water flow of 2300 gpm, only 0.5 curies of undetected activity could be released to Melton Hill Lake in one week.

Abnormal release could result from maloperation of the warm waste retention basin. If an operator makes an error by discharging at the maximum rate, the basin is emptied in approximately 40 min. If the warm waste retention basin is emptied at the maximum rate, the initial discharge rate is approximately 1350 gpm and decreases to approximately 950 gpm when the basin is emptied. With the maximum concentration allowed in the basin ($10^{-4}$ $\mu$Ci/cm$^3$), the concentration of activity entering Melton Hill Lake varies from approximately $4.7 \times 10^{-6}$ $\mu$Ci/cm$^3$ to $3.3 \times 10^{-6}$ $\mu$Ci/cm$^3$ during the emptying of the basin if maximum dilution is available. The concentrations would, therefore, exceed the $10^{-7}$ $\mu$Ci/cm$^3$ criterion for only 40 min. The total release of activity to
Melton Hill Lake would be approximately 0.017 curies. Although such a disposal operation is not a planned method of operation, the consequences of such an abnormal release would not severely restrict subsequent operations.

b. Abnormal Release of Hot Waste

Abnormal release of hot waste could result from diversion of hot waste in the hot waste sump tank to the warm waste retention basin or as the result of spills during the transfer to the tank truck or during transit to ORNL.

If hot waste is accidentally diverted to the warm waste retention basin, the monitor in the discharge line alarms, and operator action is required to stop the flow. If the activity discharge to the warm waste retention basin is sufficiently great to cause the concentration in the basin to exceed $10^{-4} \mu\text{c/cm}^3$, special precautions are taken by the operator to avoid discharging the waste to the lake at an excessive concentration. In addition to normal sampling, an isotopic analysis of the activity in the warm waste retention basin may be made for this special case. Discharge then proceeds on the basis of a known rather than an unknown mixture. Prior to the disposal operation, waste is retained as long as possible to allow for radioactive decay.

Any spills of hot waste either during transfer operations from the hot waste tanks to the tank truck or during transit to ORNL, in general, result only in localized hazards problems. If such spills occur, the operator treats the area around the spill as a contaminated area and immediately proceeds to prevent the spread of contamination. Those measures taken during area decontamination serve to minimize the consequences of such an abnormal release of hot liquid waste.

21.4 Solid Waste Disposal

The principal solid wastes resulting from EGCR operations are graphite sleeves removed from irradiated fuel assemblies, spent filter elements from gas and liquid handling systems, spent demineralizer beds from the blower seal water, and spent fuel basin cleanup systems. Other solid wastes include activated or contaminated components, hand tools, blotter paper, rags, brushes, and clothing. These wastes are collected and are transported periodically to the ORNL waste burial ground for disposal. Sections 22.4 and 22.5 discuss the radiological health aspects of these operations.

21.4.1 Graphite Sleeves

During spent fuel handling operations, a significant amount of solid waste results from the separation of the fuel elements from the graphite sleeves. This operation is carried out in the spent fuel storage basin. After the graphite sleeves are removed from the spent fuel assemblies, the sleeves are temporarily stored in a waste can located in the spent fuel storage basin. When the waste can is full (30 sleeves), and after a sufficient delay time to allow for radioactive decay, the waste can is hoisted from the basin by the overhead crane. Water is drained from the waste can prior to transfer to the transfer truck. The truck bed is lined with blotter paper to reduce the spread of any contaminated water. The waste can is then placed on the truck after which operators transfer the sleeves to a wooden box lined with a plastic bag. After all the sleeves have been transferred, the plastic bag is tied and the cover is nailed on the box. The waste can is returned to the basin, and the truck is driven to the ORNL burial ground. During all transfer operations, employees involved wear appropriate protective clothing and are under
radiological health surveillance.

Radiation dose rates associated with sleeve handling are not expected to be a serious problem. The graphite is of high purity, and a cooling time of at least three days is allowed between reactor discharge and sleeve disposal. Transfer of sleeves from the waste can should not require more than 30 min. If the radiation level is excessive, the transfer is carried out successively by several operators. Other operations associated with sleeve disposal do not require close contact for more than a few minutes.

21.4.2 Filter Elements and Demineralizer Beds

Contaminated filter elements and demineralizer beds are placed in plastic bags for transfer to the ORNL burial ground. Employees handling these items wear protective clothing, including masks when necessary, and the work is done under radiological health surveillance.

Filter elements in the ventilation systems are disposed of by entering the ventilation duct downstream of the filters, removing the filters, placing them in plastic bags, and hauling them to the ORNL burial ground.

Filter elements in helium service are mounted in housings which are, in effect, removable sections of pipe. To change a filter element, valves on either side of the housing are closed, the housing is disconnected, and the filter element is removed from the housing, bagged, and hauled to burial. A new element is inserted and the housing is reinstalled in the line.

To dispose of filter elements in the blower seal water and spent fuel basin systems, the filter inlet and outlet valves are closed, the blind flange on the filter housing is removed, and the filter element is removed, bagged, and transported to the burial ground. A new filter element is installed, the blind flange replaced, and the unit returned to service.

Demineralizer beds in the blower seal water and the spent fuel storage basin cleanup systems are of the type in which the container and resin are discarded as a unit. These units are disposed of by closing valves in the inlet and outlet lines, disconnecting two "snap-joint" couplings, placing "snap-joint" caps on the inlet and outlet of the unit, placing it in a plastic bag, and transporting it to the burial ground. A new unit is then installed.

Filter elements are replaced when the pressure drop due to particulate loading becomes excessive or before the filters collect enough radioactive material to become a radiation hazard. Similar criteria govern the replacement of demineralizer units.

21.4.3 Other Solid Waste

Contaminated solid waste such as blotting paper, absorbent pads, rags, scrubbing brushes, and shoe covers is collected in yellow-painted cans lined with plastic bags. Periodically, these bags are picked up, tied, and deposited in yellow Dempster Dumpster containers. These are trucked periodically to the ORNL burial ground and emptied.

21.5 Waste Disposal Records

All discharges of radioactive waste products from the EGCR to the uncontrolled environment are monitored, and records of such discharges are maintained by the Radiological Health Section. These records provide a technical basis for evaluating subsequent waste disposal operations, form the basis for evaluating the adequacy of the systems provided to dispose of radioactive waste, and
provide information required by the AEC for all incidents or abnormal operations.

The following records are maintained:

a. Gaseous waste activity release from the stack (rate of release and integrated release for a specific time period)
   1. Total gas
   2. Iodine
   3. Particulates

   These activities are, in general, in terms of curies of unidentified activity. Specific analyses to determine the isotopic content of the discharge are made as required. Following abnormal releases, the amount of specific isotopes released is estimated from an analysis of the filters in the stack monitor equipment.

b. Liquid waste activity release to Melton Hill Lake (rate of release and integrated release for a specific time period). As in the case of the gaseous waste release, the liquid waste release is in terms of curies of unidentified radionuclides. Records of sample analyses are maintained and correlated to the gross beta-gamma activity. Monthly composite samples of liquid discharged from the warm waste basin are analyzed to determine such long-lived isotopes as strontium-90 and cesium-137.

c. Activity concentration in Melton Hill Lake,

d. Airborne activity concentration at the perimeter air monitoring stations.

At the time of a controlled release or immediately following an uncontrolled release, such information as is required to evaluate the consequences of the release is obtained.
22. RADIATION STANDARDS

22.1 General

The function of the Radiological Health Section at the EGCR is to provide an effective program of radiation protection for the plant complement, visitors, and the general public not only during routine phases of plant operation, but also in cases of emergency.

Radiation standards, appearing in subsequent sections, comply with all Federal, state, and local regulatory agencies. They are based on current recommendations of the National Committee on Radiation Protection and Measurements, the International Commission on Radiological Protection, and the Federal Radiation Council. Personnel exposure limits, concentrations of radioactivity in air and water, and surface contamination limits are established to be consistent with these guides and still allow optimum operating flexibility. In all cases, safety of the individual is the primary consideration. Radiation protection criteria included in this section are explained in detail in the Radiological Health Manual for the EGCR. Future changes in philosophy, maximum permissible values, or plant operating limits will be reflected in revisions to the Radiological Health Manual.

The entire plant complement and visitors abide by rules and regulations set forth in the Radiological Health Manual. All personnel, including new employees, temporary employees, and trainees are instructed in the basic fundamentals of radiation protection. Periodic Radiological Health Bulletins are prepared and disseminated to all personnel. Prior to reactor startup and thereafter, as needed for new employees or research representatives, instruction in radiation protection disciplines is given to all persons whose duties involve a radiological environment. More detailed instructions, including health physics philosophy, radiation monitoring techniques, portable and fixed instrumentation, use of respiratory equipment and protective clothing, zoning and zone entry restrictions, safe handling and storage of radioactive materials, waste disposal, and criticality, are given to all plant operating supervisors, shift engineers, unit operators, and maintenance foremen.

22.2 Personnel Exposure Criteria

This section states current radiation protection guides used to prevent unnecessary exposure to ionizing radiation of individuals employed at, visiting, or residing near the EGCR.

22.2.1 External Exposure

Personnel exposure to ionizing radiation is maintained within acceptable limits by diligent monitoring and designation of all known radiation hazards, effective personnel exposure control, and safe work practices by plant personnel. Radiation Protection Guides for normal operations are given in Table 22.2.1. Whenever practical, operating procedures are controlled to limit exposure of personnel to 1/10 of the values for maximum annual and accumulation totals.
TABLE 22.2.1

**Recommended Limiting Dose to Body Organs of Occupational Workers**

<table>
<thead>
<tr>
<th>Body Organ</th>
<th>Average Weekly Dose in Rems</th>
<th>Maximum Dose in Rems&lt;sup&gt;a&lt;/sup&gt;</th>
<th>Quarterly Accumulation (13 weeks)</th>
<th>Annual Accumulation Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>Blood forming organs, total body, head and trunk, gonads, lenses of eyes</td>
<td>0.1</td>
<td>3</td>
<td>12</td>
<td>5(N-18)&lt;sup&gt;b&lt;/sup&gt;</td>
</tr>
<tr>
<td>Skin&lt;sup&gt;c&lt;/sup&gt;, thyroid</td>
<td>0.6</td>
<td>10</td>
<td>30</td>
<td></td>
</tr>
<tr>
<td>Hands, forearms, feet and ankles</td>
<td>1.5</td>
<td>25</td>
<td>75</td>
<td></td>
</tr>
<tr>
<td>Bone</td>
<td></td>
<td>30&lt;sup&gt;d&lt;/sup&gt;</td>
<td>30</td>
<td></td>
</tr>
<tr>
<td>Other body organs</td>
<td></td>
<td>4&lt;sup&gt;n&lt;/sup&gt;</td>
<td>n</td>
<td></td>
</tr>
</tbody>
</table>

|                             |                             |                                  |                                   |
|                             |                             |                                  |                                   |
|                             |                             |                                  |                                   |

a. The unit of human biological dose is the rem. It is equal to the absorbed dose in rads multiplied by the Relative Biological Effectiveness factor for the type of radiation being absorbed.

b. N is attained age in years.

c. This applies to very soft radiation. Any radiation applied to the total skin surface and having a half-value layer greater than 1 mm of soft tissue will constitute a total body exposure.

d. n is the "relative damage factor" and applies to internal exposure. Its value is between 1 and 5 depending upon the proportion of dose delivered by the various types of radiation present. See Radiological Health Manual for further information.

22.2.2 Radiation Zones

Areas where external exposure to personnel is encountered are designated as radiation zones. Such a zone where the dose rate exceeds 3 mrem/hr or the accumulated daily dose to personnel may equal or exceed 20 mrem is posted. Tags located at the boundary include information as to the dose rate at the boundary and the highest radiation level within the zone. Table 22.2.2 gives criteria for Radiation Zones.

TABLE 22.2.2

**Criteria for Radiation Zones**

<table>
<thead>
<tr>
<th>Dose Rate Range</th>
<th>Immediate Action</th>
<th>Follow-up Action</th>
</tr>
</thead>
<tbody>
<tr>
<td>3 mrem/hr to</td>
<td>Post low level tags.</td>
<td>Periodic review</td>
</tr>
<tr>
<td>6 mrem/hr</td>
<td></td>
<td></td>
</tr>
<tr>
<td>6 mrem/hr to</td>
<td>Post warning signs or tags.</td>
<td>Periodic review</td>
</tr>
<tr>
<td>1 rem/hr</td>
<td>Rope off if weekly accumulation may equal or exceed 1 rem.</td>
<td></td>
</tr>
</tbody>
</table>

22-2
TABLE 22.2.2 (Continued)

Criteria for Radiation Zones

<table>
<thead>
<tr>
<th>Dose Rate Range</th>
<th>Immediate Action</th>
<th>Follow-up Action</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 rem/hr to 3 rem/hr</td>
<td>Post warning signs or tags. Rope off.</td>
<td>Erect a barricade which provides absolute exclusion of unauthorized personnel if the weekly accumulated dose in the area can equal or exceed 12 rem. See Section 22.2.6</td>
</tr>
<tr>
<td>Over 3/rem/hr</td>
<td>Post warning signs or tags. Erect temporary barricade. Lock and/or block all entrances.</td>
<td></td>
</tr>
</tbody>
</table>

22.2.3 Internal Exposure

The values summarized in Table 22.2.1 apply equally well to internal and external exposure or combinations of the two; however, due to the uncertainties involved in determining internal dosages, every reasonable effort is made to maintain individual body burdens of radioactive material at the lowest possible level. A table of maximum body burdens, critical organs, and maximum permissible concentrations (MPC) of radioactive materials in air and water for 50-year occupational exposure and interim values for nonoccupational exposure is given in National Bureau of Standards Handbook 69, "Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and Water for Occupational Exposure." This table is used as a guide in controlling the concentrations of radionuclides in air and water at the EGCR.

22.2.4 Nonoccupational Exposure

Recommended maximum limiting yearly doses (excluding medical and background dosages) and maximum limiting concentration of radionuclides in the environment of nonoccupational groups are shown in Table 22.2.4.

TABLE 22.2.4

Recommended Limiting Dose of Ionizing Radiation to Nonoccupational Groups

<table>
<thead>
<tr>
<th>Nonoccupational Group</th>
<th>Total Body, Lenses of Eyes, Gonads</th>
<th>Air and Water Concentration MPC Factors</th>
</tr>
</thead>
<tbody>
<tr>
<td>Adults who work in vicinity of the controlled area or who enter controlled area occasionally</td>
<td>1.5 rem/year</td>
<td>30% of 40-hr/week occupational MPCa</td>
</tr>
<tr>
<td>Persons living in neighborhood of controlled area</td>
<td>0.5 rem/year</td>
<td>10% of 168-hr/week occupational MPCa</td>
</tr>
<tr>
<td>General population: individual</td>
<td>0.5 rem/year or 5 rem in 30 years</td>
<td>1% of 168-hr/week occupational MPCa</td>
</tr>
<tr>
<td>average</td>
<td>0.17 rem/year</td>
<td></td>
</tr>
</tbody>
</table>

a. Based on average over one year.

22.2.5 Accidental Exposure Values
It is recognized that an accidental dose in excess of the limiting continuous exposure values previously specified for both occupational and nonoccupational groups may sometimes be encountered. The following values are suggested as limiting dose values for short-term exposures resulting from accidents of less severity than the maximum credible accident.

TABLE 22.2.5

Accidental Exposure Limits
(Yearly maximum)

<table>
<thead>
<tr>
<th>Body Portion</th>
<th>Exposure to Public Near Exclusion Area (Rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Bone</td>
<td>3</td>
</tr>
<tr>
<td>Skin and thyroid</td>
<td>3</td>
</tr>
<tr>
<td>Total body and gonads</td>
<td>0.5</td>
</tr>
<tr>
<td>Other body organs</td>
<td>1.5</td>
</tr>
</tbody>
</table>

22.2.6 Emergency Exposure Values

In cases where grave danger exists or is imminent, or where human life is at stake, it is necessary to have pre-established maximum exposure limits. When any emergency exposure is planned, authorized approval must be obtained from the supervisor, Radiological Health Section, and the project manager or his duly authorized representative. Table 22.2.6 gives the recommended limiting values for remedial action and rescue operations.

TABLE 22.2.6

Maximum Limiting External Exposurea
During Extreme Emergency

<table>
<thead>
<tr>
<th>Dose</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>12 rem</td>
<td>For planned exposure</td>
</tr>
<tr>
<td>25 rem in 1 dayb</td>
<td>Taken only if necessary to prevent serious damage to plant or personnel</td>
</tr>
<tr>
<td>100 rem in 1 dayc</td>
<td>To be taken to save a life</td>
</tr>
</tbody>
</table>

a. Adequate protective equipment must be provided so that the internal exposure can be considered negligible (Section 23).

b. Considered "once-in-a-lifetime" dose. The individual receiving such an exposure is relocated in an area where he is not exposed to another emergency dose.

c. Personnel are made aware of the possible consequences of such exposure such as vomiting and nausea in some cases, but no serious disability.

22.3 Contamination Criteria

To minimize radioactive contamination within the plant area and to prevent its release to the environs, all areas involving the use, processing, storage, or disposal of radioactive materials are zoned. Contamination zones are established wherever personnel, equipment, or the environs are significantly contaminated and wherever the deposition of radionuclides in the body is possible. Criteria
for establishing contamination zones are given in Table 22.3.1.

### TABLE 22.3.1

<table>
<thead>
<tr>
<th>Type of Radiation</th>
<th>Airborne Contamination ($\mu$C/cm³)</th>
<th>Surface Contamination</th>
<th>Transferable Direct Reading (d/min/100 cm²)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Alpha</td>
<td>$5 \times 10^{-11}$</td>
<td>300 d/min/100 cm²</td>
<td>30</td>
</tr>
<tr>
<td>Beta-gamma</td>
<td>$3 \times 10^{-10}$</td>
<td>0.25 mrad/hr</td>
<td>1000</td>
</tr>
</tbody>
</table>

Nonzoned surfaces either conform to these limits, or are decontaminated to these levels, or in the case of alpha contamination where decontamination to these levels is not feasible, the contamination is permanently fixed to the surface by an approved bonding material, providing the contamination before bonding does not exceed ten times the values listed for alpha in Table 22.3.1.

Tables 22.3.2 through 22.3.4 show the maximum contamination levels for skin surfaces, clothing, and items given radiation and contamination clearance.

### TABLE 22.3.2

<table>
<thead>
<tr>
<th>Surface</th>
<th>Direct Survey Alpha (d/min/100 cm²)</th>
<th>Beta-Gamma (mrad/hr)</th>
<th>Transferable (Smear) Alpha (d/min/100 cm²)</th>
<th>Beta-Gamma (mrad/hr)</th>
</tr>
</thead>
<tbody>
<tr>
<td>General body</td>
<td>150</td>
<td>0.06</td>
<td>Nothing detectable</td>
<td></td>
</tr>
<tr>
<td>Hands</td>
<td>150</td>
<td>0.3</td>
<td>Nothing detectable</td>
<td></td>
</tr>
</tbody>
</table>

### TABLE 22.3.3

<table>
<thead>
<tr>
<th>Item</th>
<th>Direct Survey Alpha (d/min/100 cm²)</th>
<th>Beta-Gamma (mrad/hr)</th>
<th>Transferable (Smear) Alpha (d/min/100 cm²)</th>
<th>Beta-Gamma (mrad/hr)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Shoes, contamination zone inside</td>
<td>150</td>
<td>1.0</td>
<td>30</td>
<td>1000</td>
</tr>
<tr>
<td>outside</td>
<td>300</td>
<td>2.5</td>
<td>30</td>
<td>1000</td>
</tr>
<tr>
<td>Shoes, personal inside</td>
<td>150</td>
<td>0.3</td>
<td>30</td>
<td>200</td>
</tr>
<tr>
<td>outside</td>
<td>300</td>
<td>0.6</td>
<td>30</td>
<td>200</td>
</tr>
<tr>
<td>Clothing, contamination zone</td>
<td>150</td>
<td>0.75</td>
<td>30</td>
<td>1000</td>
</tr>
</tbody>
</table>

Clothing, personal | 150 | 0.25 |

a. The average reading shall not exceed 0.75 mrad/hr in any 644 cm² (100 in.²) area.
TABLE 22.3.4

Criteria for Contamination Clearance

<table>
<thead>
<tr>
<th>Direct Survey</th>
<th>Transferable (Smear)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Alpha Beta-Gamma</td>
<td>Alpha Beta-Gamma</td>
</tr>
<tr>
<td>(d/min/100 cm²) (mrad/hr)</td>
<td>(d/min/100 cm²)</td>
</tr>
<tr>
<td>&lt; 300</td>
<td>&lt; 0.05</td>
</tr>
<tr>
<td>&lt; 300</td>
<td>&lt; 0.05</td>
</tr>
<tr>
<td>&lt; 30</td>
<td>&lt; 200</td>
</tr>
</tbody>
</table>

a. "Clearance" as used here applies to transfer of items to clean shops, storerooms, offices, etc., within the plant area, transfer of such items within the controlled vicinity, or to items being released to the general public.

22.4 Waste Disposal Limits

All radioactive wastes, either solids, liquids, or gases, are disposed of or released in quantities and at rates consistent with existing Federal, state, and local regulations. All proposed discharges of liquid and gaseous wastes and proposed disposal of solid wastes are reviewed by the radiological health representative in cooperation with the chemical engineer. Wastes which are removed to ORNL for off-site disposal are placed in controlled disposal areas inaccessible by the public. The National Bureau of Standards Handbook 69 is used as a guide in radioactive waste disposal. Excerpts from it are given in the following sections.

22.4.1 Solid Waste

Solid radioactive wastes result from the accumulation of contaminated equipment and trash from various phases of plant operation. Activity levels associated with these wastes range from slightly above background to those necessitating shielding and remote handling devices, with radiation intensities of several rem/hr. In the latter case, safe handling in removal and transfer of solid wastes is necessary to prevent overexposure of personnel and extensive contamination of the plant or environs.

Radiation intensities and levels of contamination encountered in handling solid waste vary widely; consequently, no rigid limits are applied. Radioactive waste disposal is limited by accumulated doses to personnel engaged in disposal operations or the possibility of a contamination incident. Criteria are established (Sections 22.2 and 22.3) for both of these and are applied to all aspects of solid waste disposal.

22.4.2 Liquid Waste

Methods of collection, storage, sampling, and disposal of liquid wastes are described in Section 21.3.

The subcommittee on Permissible Internal Dose, National Committee on Radiation Protection, in National Bureau of Standards Handbook 52 states, "Because of the many uncertainties involved, this committee recommends that every effort be made to keep the concentrations of radioisotopes in air and water and in the body to a minimum. The goal should be no radioactive contamination of air and water and of the body if it can be accomplished with reasonable effort and expense. If such a goal cannot be attained, the average operating levels should be kept as far below these recommended values as possible, and not above them for any extended period of time."
Table 22.4.2 lists acceptable concentrations for interim releases of liquid wastes to the environment.

**TABLE 22.4.2**

**Provisional Maximum Permissible Concentration of Unidentified Radionuclides in Water (MPCU)\(_w\)**

Values that are applicable for occupational exposure (168-hr/week) to any radionuclide or mixture of radionuclides

<table>
<thead>
<tr>
<th>Limitations</th>
<th>(\mu)c/cm(^3) of water(b)</th>
</tr>
</thead>
<tbody>
<tr>
<td>If Sr(^{90}), I(^{129}), Pb(^{210}), At(^{211}), Ra(^{223}), Ra(^{224}), Ra(^{226}), Ac(^{227}), Ra(^{228}), Th(^{230}), Pa(^{231}), Th(^{232}), and Th-nat are not present(^a) the continuous exposure level (MPC)(_w), is not less than</td>
<td>(3 \times 10^{-5})</td>
</tr>
<tr>
<td>If Sr(^{90}), I(^{129}), Pb(^{210}), Po(^{210}), Ra(^{223}), Ra(^{226}), Ra(^{228}), Pa(^{231}), and Th-nat are not present(^a) the continuous exposure level (MPC)(_w), is not less than</td>
<td>(2 \times 10^{-5})</td>
</tr>
<tr>
<td>If Sr(^{90}), Pb(^{210}), Ra(^{226}), and Ra(^{228}) are not present(^a) the continuous exposure level (MPC)(_w), is not less than</td>
<td>(6 \times 10^{-6})</td>
</tr>
<tr>
<td>If Ra(^{226}), and Ra(^{228}) are not present(^a) the continuous exposure level (MPC)(_w), is not less than</td>
<td>(10^{-6})</td>
</tr>
<tr>
<td>In all cases the continuous occupational level (MPC)(_w), is not less than</td>
<td>(10^{-7})</td>
</tr>
</tbody>
</table>

\(^a\) In this case "not present" implies the concentration of the radionuclide in water is small compared with the MPC value in Table 1. of NBS Handbook 69.

\(^b\) Use 1/10 of these values for interim application in the neighborhood of an atomic energy plant.

**22.4.3 Gaseous Disposal**

All reasonable efforts are taken to filter, monitor, and dilute adequately radioactive gases discharged to the environment, and control the continuous and short-term releases of such gases so that both occupational and nonoccupational groups are not subjected to exposures exceeding those values listed in Tables 22.2.1, 22.2.4, and 22.2.5. The following table, taken from National Bureau of Standards Handbook 69, is used as a guide in all gaseous waste discharge.

**TABLE 22.4.3**

**Provisional Maximum Permissible Concentration of Unidentified Radionuclides in Air (MPCU)\(_a\)**

Values that are applicable for occupational exposure (168-hr/week) to any radionuclide or mixture of radionuclides

<table>
<thead>
<tr>
<th>Limitations</th>
<th>(\mu)c/cm(^3) of air(b)</th>
</tr>
</thead>
<tbody>
<tr>
<td>If there are no alpha-emitters, Sr(^{90}), I(^{129}), Pb(^{210}), Ac(^{227}), Ra(^{228}), Pa(^{230}), Pu(^{241}), and Bk(^{249}) are not present(^a) the</td>
<td></td>
</tr>
</tbody>
</table>
TABLE 22.4.3 (Continued)

Provisional Maximum Permissible Concentration
of Unidentified Radionuclides in Air (MPCU)\textsubscript{a}

Values that are applicable for occupational exposure (168-hr/week) to any radionuclide or mixture of radionuclides

<table>
<thead>
<tr>
<th>Limitations</th>
<th>(\mu\text{c/cm}^3) of air\textsuperscript{b}</th>
</tr>
</thead>
<tbody>
<tr>
<td>continuous exposure level, (MPC)\textsubscript{a}, is not less than</td>
<td>10\textsuperscript{-9}</td>
</tr>
<tr>
<td>If there are no alpha-emitters and if beta-emitters Pb\textsuperscript{210}, Ac\textsuperscript{227}, Ra\textsuperscript{228}, and Pu\textsuperscript{241} are not present\textsuperscript{a} the continuous exposure level, (MPC)\textsubscript{a}, is not less than</td>
<td>10\textsuperscript{-10}</td>
</tr>
<tr>
<td>If there are no alpha-emitters and if beta-emitter Ac\textsuperscript{227} is not present\textsuperscript{a} the continuous exposure level, (MPC)\textsubscript{a}, is not less than</td>
<td>10\textsuperscript{-11}</td>
</tr>
<tr>
<td>If Ac\textsuperscript{227}, Th\textsuperscript{230}, Pa\textsuperscript{231}, Th\textsuperscript{232}, Th-nat, Pu\textsuperscript{238}, Pu\textsuperscript{239}, Pu\textsuperscript{240}, Pu\textsuperscript{242}, and Cf\textsuperscript{249} are not present\textsuperscript{a} the continuous exposure level, (MPC)\textsubscript{a}, is not less than</td>
<td>10\textsuperscript{-12}</td>
</tr>
<tr>
<td>If Pa\textsuperscript{231}, Th-nat, Pu\textsuperscript{239}, Pu\textsuperscript{240}, Pu\textsuperscript{242}, and Cf\textsuperscript{249} are not present\textsuperscript{a} the continuous exposure level, (MPC)\textsubscript{a}, is not less than</td>
<td>7 \times 10\textsuperscript{-13}</td>
</tr>
<tr>
<td>In all cases the continuous occupational level, (MPC)\textsubscript{a}, is not less than</td>
<td>4 \times 10\textsuperscript{-13}</td>
</tr>
</tbody>
</table>

\textsuperscript{a} In this case "not present" implies the concentration of the radionuclide in air is small compared with the MPC value in Table 1. of NBS Handbook 69.

\textsuperscript{b} Use 1/10 of these values for interim application in the neighborhood of an atomic energy plant.

22.5 Transportation and Transfers

Shipment of radioactive materials to and from the EGCR is under the cognizance of the supervisor, Radiological Health Section, or his authorized representative. All such shipments comply with AEC, state, and local regulations, and are consistent with those regulations prescribed by the Interstate Commerce Commission (ICC). The Handbook of Federal Regulations entitled, "Transportation of Radioactive Materials," is used as a general guide in shipping radioactive materials. Methods of shipping irradiated fuel are given in detail in Section 18.11.

The following principal limitations, taken from the handbook, are in compliance with Interstate Commerce Commission regulations and are applied to all shipments of radioactive materials moved by highway carrier, rail freight, or rail express:

a. Group I radioactive materials, defined as those materials which emit gamma rays only or both gamma rays and electrically charged corpuscular rays, are packed in suitable inside containers (specifications are given in Article 78.34 of ICC Regulations) and shielded by a second container so that the gamma radiation during transportation does not exceed 10 mrem/hr at one meter from any point on the radioactive source or 200 mrem/hr at any accessible point on the surface of the shipping
container.

b. Group II radioactive materials, defined as those materials which emit neutrons and either or both the types of radiations characteristic of Group I, are packed so as to limit beta and gamma radiation to 10 mrem/hr at one meter, and neutron radiation which is the physical equivalent of 2 mrem/hr at one meter from the source.

c. Group III radioactive materials, defined as those materials which emit electrically charged corpuscular rays only; i.e., alpha or beta, etc., are packed in suitable containers which prevent the escape of primary radiation and which do not permit the level of secondary radiation to exceed 10 mrem/day at any surface of the container.

d. Outside shipping containers are of such design that gamma radiation does not exceed 200 mrem/hr or equivalent at any point of readily accessible surface.

e. The design and preparation of the package is such that there is no significant radioactive surface contamination of any part of the container. Container designs are in accordance with specifications given in Part 78 of ICC Regulations. The degree of fogging of undeveloped film at a distance of 15 ft will not exceed the fogging of 11.5 mrem of gamma rays that have been filtered by 1/2 in. of lead.

These limitations apply only to shipments of radioactive materials to locations outside the Oak Ridge Operations area and do not apply to transfers of material to and from the various AEC installations at Oak Ridge, Tennessee.
23. RADIATION EXPOSURE CONTROL PROCEDURES

23.1 General

Standards and criteria for radiation protection given in Section 22 are in accordance with present knowledge of the physical and biological effects of radiation on man. Application of these standards in actual practice is one of the most essential parts of any radiation protection program.

In maintaining personnel exposure to ionizing radiations at minimal levels, it is necessary to know: the number, type, location, and radiation levels from all sources of radiation, means of obviating hazards arising from these sources, and accurate methods of measuring the accumulated dose to personnel. The following sections present methods used to achieve these requirements.

23.2 Facilities and Space

The radiological health laboratory is located in the turbine building near the personnel air lock. It is used as a sample analysis and counting room and as a field office for the shift monitoring group. Figure 23.2 shows an arrangement plan of the laboratory.

23.3 Instruments

A basic requirement for any radiological health program is the detection and measurement of radiation. Before instrument selection is undertaken, it is necessary to have knowledge of the types and energy spectrums of those radiations likely to be encountered at a given facility. For the EGCR these are given in Table 23.3.1.

<table>
<thead>
<tr>
<th>Radiation</th>
<th>Energy Spectrum (Mev)</th>
<th>Average Range of Ionizing Particle (cm of air at STP)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Alpha particles</td>
<td>5-10</td>
<td>2-8</td>
</tr>
<tr>
<td>Beta particles</td>
<td>0.015-5</td>
<td>0.1-800</td>
</tr>
<tr>
<td>Gamma rays</td>
<td>0.05-2.9</td>
<td>0.4-450</td>
</tr>
<tr>
<td>Slow neutrons</td>
<td>10^{-7} - 10^{-1}</td>
<td>0.8</td>
</tr>
<tr>
<td>Fast neutrons</td>
<td>10^{-1} - 10</td>
<td>0.1 - 45</td>
</tr>
</tbody>
</table>

a Primary (alpha and beta) particles and secondary particles resulting from gamma and neutron interactions.

Instruments used consist of several types, some of which are designed to detect or measure only one aspect of radiation. Others are more versatile and are used to perform a number of tasks. Instrument types are categorized as to their function in the over-all radiation protection program. These functions
FIG. 23.2
RADIOLOGICAL HEALTH LABORATORY

5 DRAWER FILE CABINET

DESK

WORK TABLE

TABLE FOR COUNTERS

INSTRUMENTS & RELATED EQUIPMENT

FUME HOOD & CABINET

CABINET & DRAWERS

CABINET & DRAWERS

CABINET & DRAWERS

CABINET & DRAWERS

DRAWER CABINET

CORNER CABINET

SINK

METAL DOOR CABINETS
are: personnel monitoring, survey, area monitoring, and personnel and area contamination control. Tables 23.3.2 through 23.3.5 list the instruments utilized to achieve each of these functions. The program of calibration and maintenance of both fixed and portable instruments is directed by the supervisor, Radiological Health Section.

**TABLE 23.3.2**

### Personnel Monitoring Instruments (Portable)

<table>
<thead>
<tr>
<th>Instrument</th>
<th>Radiation Detected</th>
<th>Range</th>
<th>Application</th>
</tr>
</thead>
<tbody>
<tr>
<td>Film badge</td>
<td>Beta, gamma, fast and thermal neutrons</td>
<td>0.02 - $10^4$ rads</td>
<td>Permanent record of dose and type of radiation</td>
</tr>
<tr>
<td>Pocket chamber (indirect reading)</td>
<td>Gamma</td>
<td>0-200 mr</td>
<td>Provides day-to-day check of exposure</td>
</tr>
<tr>
<td>Pocket chamber (direct reading)</td>
<td>Gamma, thermal neutrons</td>
<td>0-200 mr</td>
<td>Permits wearer to check accumulated dose at any time</td>
</tr>
<tr>
<td>Personal radiation alarm</td>
<td>Gamma, high energy beta</td>
<td>0-10 r/hr</td>
<td>Maximum audible alarm at 0.5 r/hr; flashing light continuous at 10 r/hr</td>
</tr>
<tr>
<td>Chemical dosimeter$^b$</td>
<td>Gamma</td>
<td>5 to $2 \times 10^6$ rads</td>
<td>Measures gamma in a mixed radiation field</td>
</tr>
<tr>
<td>Glass dosimeter$^b$</td>
<td>Gamma</td>
<td>5 to several thousand rads</td>
<td>Dose measurement of gamma exposure over a wide range</td>
</tr>
</tbody>
</table>

$^a$ For discussion of the use of these instruments see Sections 23.6 and 23.7.

$^b$ Special components of the film badge used to evaluate abnormal exposures.

**TABLE 23.3.3**

### Portable Survey Instruments

<table>
<thead>
<tr>
<th>Instrument</th>
<th>Radiation Detected</th>
<th>Range</th>
<th>Application</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cutie pie</td>
<td>Gamma, and high energy alpha and beta</td>
<td>5-10,000 mr/hr</td>
<td>Dose rate instrument for X and gamma radiation detects high energy alpha radiation through end window</td>
</tr>
<tr>
<td>G-M survey meter</td>
<td>Gamma and beta ($\geq 0.2$ MeV)</td>
<td>0-20 mrad/hr</td>
<td>Used as a detection instrument, not as a dose rate meter</td>
</tr>
<tr>
<td>Alpha proportional counter (gas)</td>
<td>Alpha</td>
<td>5-100,000 counts/min</td>
<td>Alpha detector and counter</td>
</tr>
</tbody>
</table>
**TABLE 23.3.3 (continued)**

**Portable Survey Instruments\(^a\)**

<table>
<thead>
<tr>
<th>Instrument</th>
<th>Radiation Detected</th>
<th>Range</th>
<th>Application</th>
</tr>
</thead>
<tbody>
<tr>
<td>Alpha scintillation counter</td>
<td>Alpha</td>
<td>0-500 counts/ min</td>
<td>Used as a detector and counter when equipped with a register</td>
</tr>
<tr>
<td>Thermal neutron counter</td>
<td>Thermal neutrons</td>
<td>50-20,000(n_t/\text{cm}^2\cdot\text{s})</td>
<td>Used for thermal neutron measurement in presence of other radiations</td>
</tr>
<tr>
<td>Fast neutron counter</td>
<td>Fast neutrons</td>
<td>0-2500 mrad/hr</td>
<td>Measures dose rate resulting from first collision of neutrons with an energy range between 0.2 to 14 Mev</td>
</tr>
</tbody>
</table>

\(^a\) For discussion of the use of these instruments see Section 23.4.

**TABLE 23.3.4**

**Area Monitoring Instruments (On-Site and Off-Site)\(^a\)**

<table>
<thead>
<tr>
<th>Instrument</th>
<th>Radiation Detected</th>
<th>Range</th>
<th>Application</th>
</tr>
</thead>
<tbody>
<tr>
<td>Continuous beta-gamma particulate air monitor</td>
<td>Beta and gamma</td>
<td>50-50,000 counts/ min</td>
<td>Continuous recording of airborne particulate activity</td>
</tr>
<tr>
<td>Fast neutron dosimeter</td>
<td>Fast neutron</td>
<td>0.1 - 10^4 mrad</td>
<td>Integrates total fast neutron dose</td>
</tr>
<tr>
<td>Gamma detector</td>
<td>Gamma</td>
<td>0 - several r/hr</td>
<td>Gamma dose rate background monitor</td>
</tr>
<tr>
<td>Nuclear accident dosimeter</td>
<td>Thermal neutrons and fast neutrons</td>
<td>High intensity neutron flux</td>
<td>Provides data on the energy spectrum and dosage due to high intensity neutron bursts</td>
</tr>
<tr>
<td>Film and foil packets</td>
<td>Gamma and neutrons</td>
<td>0.02 - 10,000 rad gamma and relative intensity for neutrons</td>
<td>Used to supplement fixed instruments and nuclear accident dosimeters</td>
</tr>
<tr>
<td>Rain water collector</td>
<td>Beta and gamma</td>
<td></td>
<td>Collects radioactive particulates washed from the air</td>
</tr>
<tr>
<td>Gummed paper fall-out collector</td>
<td>Beta and gamma</td>
<td></td>
<td>Collects heavy particulates settling from air</td>
</tr>
</tbody>
</table>

\(^a\) For discussion of the use of these instruments see Sections 7.6, 23.4, and 23.5.
TABLE 23.3.5

Personnel and Area Contamination Monitoring Instruments

<table>
<thead>
<tr>
<th>Instrument</th>
<th>Radiation Detected</th>
<th>Range</th>
<th>Application</th>
</tr>
</thead>
<tbody>
<tr>
<td>Hand and foot counter</td>
<td>Beta, gamma</td>
<td>0.2 - 10 x maximum permissible limit</td>
<td>Personnel contamination detector and counter</td>
</tr>
<tr>
<td>Local ratemeter</td>
<td>Beta, gamma</td>
<td>200 - 20,000 counts/min</td>
<td>Beta, gamma monitors at each contamination zone portal or change room</td>
</tr>
<tr>
<td>Liquid waste monitor</td>
<td>Beta, gamma</td>
<td>$5 \times 10^{-6}$ $5 \times 10^{-1} \mu\text{c/cm}^3$</td>
<td>Used to detect abnormal levels of activity in the waste system</td>
</tr>
<tr>
<td>Stack monitors</td>
<td>Beta, gamma</td>
<td>Up to $10^{-4}$ $\mu\text{c/cm}^3$</td>
<td>Provides continuous information on activity concentrations in stack gases; can initiate containment shell isolation</td>
</tr>
<tr>
<td>Helium coolant monitor</td>
<td>Gamma</td>
<td>0.1 mr/hr to 100 r/hr</td>
<td>Detects and warns personnel of high activity levels in the coolant</td>
</tr>
<tr>
<td>Smear and air sample counter</td>
<td>Alpha, beta, and gamma</td>
<td>Background to several thousand counts/min</td>
<td>Used to obtain a quantitative result from surface contamination smears and air samples</td>
</tr>
<tr>
<td>Portable disc sampler</td>
<td>Alpha, beta, and gamma (later counted)</td>
<td>--</td>
<td>Used to take special air samples for specific purposes</td>
</tr>
</tbody>
</table>

a For discussion of the use of these instruments see Sections 7.6, 9.2, 21, 23.4, 23.5, and 23.6.

23.4 Radiation Survey and Monitoring

A radiation survey is defined as a critical examination of the radiation hazards incident to the production, usage, storage or disposal of radioactive materials, or other sources of ionizing radiation under specific circumstances. Such a survey takes the form of routine "probing and smearing" of an item that is to be released to an uncontrolled area or involve a meticulous survey of the reactor shield.

All of the portable detection and measuring instruments described in Section 23.3 are used in making radiation surveys. The smear technique is employed to determine qualitatively, and to some degree quantitatively, the extent of transferable contamination on surfaces and objects.

All controlled areas are probed and smeared on a regularly scheduled basis. Floor plan sketches are used to tabulate, by location, spots of contamination or areas of radiation that present an unsafe condition. Other areas normally uncontaminated or having low radiation levels are surveyed periodically at frequencies depending on their usage and potential hazard. Specific areas
such as the lunchroom, first-aid room, radiological health laboratory, and areas accessible to visitors are routinely surveyed for the detection and minimization of contamination and radiation levels therein. A log is maintained listing pertinent information relative to daily surveys. Continuous and special samples of air are obtained from contaminated or potentially contaminated plant areas. These samples are evaluated by standard counting techniques and the results are plotted on a daily basis for reference purposes. Data from the liquid waste monitors and from special liquid waste samples are also evaluated and plotted daily. Special surveys of radiation and contamination levels within the fenced area are made routinely and the data obtained are tabulated and incorporated in the plant monitoring record.

23.5 Environmental Monitoring

Each individual plant within the Oak Ridge Operations area has the responsibility for on-site monitoring. Union Carbide Nuclear Company has the responsibility for all environmental monitoring (off-site) for the Oak Ridge area. This is performed by the area monitoring group of the Health Physics Division, Oak Ridge National Laboratory.

23.5.1 Air Monitoring

Air monitoring is accomplished by utilization of the ORO area perimeter air monitoring and remote air monitoring systems.

a. Perimeter Air Monitoring (PAM)

The perimeter air monitoring system consists of eight air sampling stations surrounding the X-10, K-25, and Y-12 areas. The number of these stations will be increased to nine before EGCR startup. Locations of the existing stations are shown on Figure 23.5.1.1.

Each station consists of a gummed paper fallout collector, a rain water collector, and a continuous air sampler of the fixed filter paper type. Each station is provided with detectors and telemetering equipment with recorders at the ORNL area monitoring laboratory. An additional readout for the telemetered PAM system is located at the EGCR site.

b. Remote Air Monitors (RAM)

A remote air monitoring system is operated for the purpose of measuring air contamination in East Tennessee at locations remote from the Oak Ridge operations and for estimating the spread of airborne contamination in the event of a major accident. At distances of from 12 to 75 miles, air samplers are operated at six TVA dams and one Corps of Engineers dam. These locations are shown on Figure 23.5.1.2. All of the samplers are instrumented but none are telemetered.

Frequent operational checks (hourly by TVA) are made by the operating personnel at each of the locations and ORNL is notified by telephone of malfunctions which cannot be readily corrected. Fallout collectors, rain water collectors, and air filters are changed by operators at the hydroelectric plants and the samples are shipped to ORNL on a weekly basis. Processing and analysis of the samples are performed at ORNL, and the results are incorporated in routine reports issued by the ORNL Area Monitoring Section. Unusual or significant findings can be communicated to the EGCR Radiological Health Section immediately upon detection.

23.5.2 Water Monitoring
OAK RIDGE RESERVATION PERIMETER
AIR MONITORING SYSTEM
FIG. 23.5.1.1
STATION SITES FOR REMOTE AIR MONITORING SYSTEM

FIG. 23.5.1.2
The major source of radioactive wastes reaching the Clinch River is the Oak Ridge National Laboratory. In addition to the local sampling at the various sources and outfalls, the Laboratory operates a continuous monitoring and sampling station at White Oak Dam where the stream passes out of the controlled area. Releases here are regulated so that for the spectrum of radioisotopes identified as being present, the calculated concentration in the Clinch River after the intermixing of White Oak Creek will not exceed the permissible average concentration for people residing in the neighborhood of the controlled area.

The Clinch River is sampled at two downstream points, Mile 14.4 and Mile 12.0, by personnel at the K-25 plant. River sampling by ORNL includes collections at Clinch River Mile 37.5 (almost 17 miles above White Oak Creek) and at Clinch River Mile 4.5 (about 16 miles below White Oak Creek). Composite samples from these two locations are assayed for gross beta activity and specific radiochemical determinations are performed quarterly.

In connection with the Clinch River Ecological Study, additional water samples, composited in proportion to the flow for a weekly composite, are collected daily at the Oak Ridge water intake (Clinch River Mile 41.5), Clinch River Mile 5.5, Fort Loudoun Dam, Watts Bar Dam, and Chickamauga Dam. Radioactivity and radiochemical determinations are made by the Robert A. Taft Sanitary Engineering Center of the U. S. Public Health Service at Cincinnati and chemical analyses are made by the Stream Pollution Control Division of the Tennessee State Department of Public Health.

Annual mud surveys from ORNL to Guntersville, Alabama, are made utilizing a Flounder instrument. This instrument consists of a flat water-proof case containing 12 Geiger-Mueller tubes which are lowered to rest on the bottom sediment. Readings and mud samples for laboratory analysis are taken at 50-ft intervals in a traverse section of the Clinch River or at 10 equidistant points across a traverse section of the Tennessee River or TVA lakes. In general, these traverses correspond to silt ranges of TVA reservoir studies. The spacing of the traverses is close in the Clinch River and more distant in the Tennessee River. At appropriate intervals, the survey is extended to the mouth of the Tennessee River at Paducah, Kentucky.

Silt ranges for Melton Hill Reservoir are established at approximately one-mile intervals between the dam (Clinch River Mile 23.1) and EGCR (Clinch River Mile 23.5) and at two-mile intervals upstream from EGCR. After impoundment of the lake, the Flounder survey is extended up the lake at least to Mile 37.5.

Water sampling in Melton Hill Lake is designed to identify the pattern of dispersion of warm wastes released through the EGCR circulating water outlet. Continuous sampling at Melton Hill Dam proportions the sample to the flow.

There is a continuous monitor on the service water return line ahead of the seal well to alarm when radioactivity in this effluent is high. A sample tube is installed in the outfall pipe for the collection of samples of the final effluent. Releases of warm wastes are discussed under waste disposal (Section 21).

23.5.3 Special Environmental Monitoring

Aerial surveys, milk, vegetation, soil, and aquatic and terrestrial animal sampling are performed by ORNL. Some of these activities are expanded to include the environs and possible effects of EGCR operation.

a. Aerial Surveys

The pattern flown by UCNC's light airplane in its routine aerial survey
of Bethel Valley includes the EGCR site and its environs. The plane used for this survey is available for emergency surveys and it is anticipated that it could be used to trace ground deposition of radioactive materials resulting from an accident at EGCR.

b. **Milk Surveys**

Sampling of milk for the area is expanded to include milk produced on farms which might be affected by EGCR operation and to include grass and pasturage from area farms. In the event of an accidental release of large quantities of airborne radioactivity, an expanded milk survey is inaugurated by ORNL.

c. **Prestartup Background Surveys**

To provide a base line of data for poststartup plant operations, a radiation background survey was conducted at the EGCR site in the fall of 1962. Instrument readings, film exposures, and samples of soil, grass, forest litter, and tree leaves were obtained at specific locations within a pre-established grid (Figure 23.5.3). No air or water analyses were made during the first survey due to lack of equipment, but these will be added in future surveys (semiannually).

Background levels of gamma radiation were slightly higher than those normally detected for the Oak Ridge area and somewhat lower than in-plant levels at ORNL. Radioactive materials found in soil, grass, litter, and leaf samples consisted of the usual spectrum of atmospheric fallout.

23.6 **Access Control**

All spaces and areas within the plant where radiation dose rates and contamination levels necessitate special protective measures are considered controlled areas and personnel access thereto is restricted. Conversely, all spaces and areas not exhibiting hazardous conditions and requiring no special control measures are considered free access areas.

Zoning, protective clothing, and respiratory equipment are used in the radiation protection program. These methods may be used independently or collectively and may be temporary or continuous.

23.6.1 **Zoning**

To minimize personnel radiation exposures and maintain control over the spread of radioactive contamination, the reactor building and portions of the turbine and service buildings are zoned. Inadvertent, or unauthorized entry by individuals into areas having dangerous radiation and contamination levels, is controlled by a continuing familiarization of plant personnel with all hazardous or potentially hazardous situations; the use of informative warning lights, horns, signs, and tags; restrictive devices such as barriers, shields, interlocked doors and switches; and applicable fixed and portable instrumentation.

a. **Definitions**

1. **Radiation Zones** - Zones established where external radiation exposure to personnel is involved.

2. **Contamination Zones** - Zones established where the contamination of employees, equipment, and the environs is anticipated or encountered and where there is the possibility that radioactive material may become deposited inside the body resulting in internal radiation exposure.
3. Regulated Zones - Areas where operations are restricted for the purpose of radioactive contamination control. Such zones may contain Radiation Zones, Contamination Zones, or both, ranging in size from a small spot to a large area.

4. Contamination Zone Vehicles - Distinctly marked vehicles used to transport radioactive materials, contaminated equipment, and personnel wearing contamination zone clothing between contamination zones.

5. Contamination Zone Clothing and Equipment - Special wearing apparel and protective devices provided by plant management for use in a contamination zone.

6. Zone Portal - A designated point on the boundary of a zone where employees enter and exit.

7. Contamination Zone Change Facility - A facility located within a regulated zone and adjacent to a contamination zone portal equipped with adequate monitoring devices, protective clothing and equipment, decontaminating materials, and space for storage of personal effects.

Criteria for establishing radiation zones and contamination zones are given in Section 22, Radiation Standards. Figure 23.6.1 shows examples of zoned areas.

b. Authorization

Persons entering radiation zones are provided with special instrumentation and have radiological health surveillance. A Radiation Work Permit, initiated by the immediate supervisor and approved by the attendant health physicist, is required for entry into any area where one of the following limits is equaled or exceeded:

1. Single planned exposure 20 mrem
2. Dose rate 5 rem/hr
3. Air contamination (MPC) air (40-hr/week)

Zoned areas are identified by signs. The signs for radiation and contamination zones indicate the requirements for entry. Table 23.6.1 lists the requirements for entry into a Radiation Zone.

<table>
<thead>
<tr>
<th>Exposure Range (rem/hr)</th>
<th>Direct Reading Monitoring Instruments Required</th>
<th>Radiological Health Surveillance Required</th>
<th>Administrative Authority</th>
<th>Supervisory Authority</th>
<th>Radiological Health Project Manager</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.003 - 5</td>
<td>X</td>
<td>X</td>
<td></td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>5 - 20</td>
<td>X</td>
<td>X</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>20 - 50</td>
<td>X</td>
<td>X</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Over 50</td>
<td>X</td>
<td>X</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

* Trainees, personnel on loan to the plant, and visitors are required to have approval of the appropriate technical or operating staff supervisor sponsoring their EGCR activities.

* In the exposure range, 0.003 - 5 rem/hr, the requirements specified in
EGCR FLOOR PLAN SHOWING ZONED AREAS
FIG. 23.6.1
Columns II and III are ignored if the anticipated exposure time results in an accumulated daily dose of less than 20 mrem.

c. Regulations

The following general rules apply to all personnel:

1. Regulated zones are established in work areas surrounding, adjacent to, or connecting contamination zones. Regulated zones are accessible to all authorized plant personnel with restrictions only on contamination zone personnel and equipment as defined in item 2 below.

2. Entrance to, and exit from, contamination zones is made through specified portals. Contamination zone personnel and equipment are permitted entry into a regulated zone only when approved monitoring techniques indicate no transferable contamination.

3. Signs at portals give up-to-date information on requirements and conditions relative to entry, working conditions, and exit requirements.

4. If, in the absence of radiological health surveillance, it is deemed necessary by operating supervisors to proceed with entry into a radiation zone, the office of the health physicist is notified of the action taken.

5. Contamination zone clothing and equipment is not used outside of a contamination zone or a regulated zone except when the individual is an occupant of a contamination zone vehicle following a prescribed route.

6. Contamination zone vehicles, in transit between contamination zones, follow prescribed routes.

7. No lunchroom is established within a contamination zone. Eating, smoking, and drinking are prohibited in contamination zones except drinking from water fountains equipped with foot-treadle valves.

d. Management Responsibility

Management has the responsibility:

1. To provide suitable facilities for contamination zone personnel with provision of lockers for storage of personal effects.

2. To provide a supply of required contamination zone clothing and equipment for contamination zone use.

3. To see that all areas are surveyed by health physicists as required and properly zoned as specified in this procedure.

4. To assist health physicists in the posting of appropriate zone signs with up-to-date instructions pertaining to entry, occupancy, and departure.

5. To notify the Radiological Health Section in advance, where practicable, of changes in operations or maintenance schedules which should be considered in zoning requirements.

6. To support these regulations with administrative control.
e. **Radiological Health Section Responsibility**

The Radiological Health Section has the responsibility:

1. To provide personnel monitoring, perform building and area surveys, maintain exposure and survey records, and provide or obtain related services pursuant to these instructions.

2. To establish zoned areas where necessary and maintain up-to-date information relative to conditions therein by use of appropriate signs and tags.

3. To assist management in maintaining administrative control relative to these and any future regulations.

**23.6.2 Protective Clothing**

In addition to protection of personnel by all of the previously mentioned methods, the judicious use of protective apparel by employees is specifically encouraged. Those individuals whose duties bring them in contact routinely with radioactive material, or where work procedures are likely to cause or result in excessive levels of contamination, are required to wear protective equipment.

Coveralls and laboratory coats designated for use in a Contamination Zone are identified by the word "contamination" conspicuously stenciled in black letters across the back of the garment. Clothing so designated is not worn outside a Regulated Zone or Contamination Zone.

Protective clothing and related equipment is provided at each Contamination Zone change facility to be donned upon entry and removed upon exit.

Provisions are made for monitoring all items which are to be laundered, re-issued, or stored. Clothing is bagged in polyethylene containers, monitored for excessive radiation levels, and transported to the ORNL laundry for cleaning. Contamination limits for laundered coveralls and laboratory coats are given in Table 22.3.3.

**23.6.3 Respiratory Equipment**

Since the ECCR employs large quantities of pressurized helium as the reactor coolant, airborne radioactive contamination is an ever present possibility. Therefore, adequate respiratory equipment is necessary and is provided.

Respirator types available for use in any given situation include, but are not limited to:

- a. Portable air tank respirators (Scott Air-Pac type)
- b. Supplied air respirator (Air-line type)
- c. Oxygen generating respirators (Chemox type)
- d. Full face respirators (Army assault mask type)
- e. Dust and fume respirators.

Such equipment is used in areas where hazardous concentrations of airborne activity are known to exist or where there is the possibility of accidental dispersal of radionuclides into the plant environs.

Table 23.6.3 lists the concentration ranges for unidentified radionuclides in air and the required respiratory protection in each case.
<table>
<thead>
<tr>
<th>Alpha</th>
<th>Beta, Gamma</th>
<th>Immediate Action</th>
</tr>
</thead>
<tbody>
<tr>
<td>$2 \times 10^{-11}$</td>
<td>$&lt; 3 \times 10^{-9}$</td>
<td>Masks not required unless contamination determined to be above (MPC)$_a$ for 40-hr/week</td>
</tr>
<tr>
<td>$2 \times 10^{-11}$ to $2 \times 10^{-9}$</td>
<td>$3 \times 10^{-9}$ to $3 \times 10^{-7}$</td>
<td>Full face mask required; otherwise, evacuate personnel from area</td>
</tr>
<tr>
<td>$&gt; 2 \times 10^{-9}$</td>
<td>$&gt; 3 \times 10^{-7}$</td>
<td>Positive air supply required; otherwise, evacuate personnel from area; also consider external exposure to personnel</td>
</tr>
</tbody>
</table>

Upon termination of usage, all respirators are monitored for contamination, cleaned with a surgical soap solution, and if free of contamination, are packaged for future use.

### 23.7 Personnel Monitoring

Standard monitoring techniques are used in determining the accumulated radiation exposure to personnel. These include: individual film badges for all personnel, including temporary employees and visitors; special neutron film packets; pocket dosimeters, personal radiation alarms, etc., for specific personnel usage; regular and special body fluid analyses, including urine, feces, and blood analysis; whole body counting; and systematic records and reports.

#### 23.7.1 Film Badges and Dosimeters

Individual film and identification badges are worn by all employees while inside the fenced area of the plant. A quarterly schedule for changing and processing these badges is maintained. The films consist of two types: gamma packet containing a sensitive and a less sensitive emulsion, and a neutron sensitive packet containing a single emulsion.

Dose-density ranges of the sensitive and the less sensitive gamma film are 20 mrem to 50,000 mrem and 5 rem to 500 rem, respectively. The approximate upper limit for the neutron film is 5 rads.

All visitors to the EGCR are given a self-indicating pocket dosimeter and a visitor's badge containing a gamma sensitive film packet. These are issued at the security gate, collected at that point upon termination of the visit and, if indicated by the attendant, pocket meter readings are processed during the next work day.

Films are developed and read against controlled "batch" standards by the ORNL Applied Health Physics group. The processed films and the report are transferred to the EGCR where permanent storage is provided (Section 23.7.3).

In addition to the film emulsion for routine monitoring, the badge contains: a one-half gram pellet of elemental sulfur and a 0.005-in. thick gold foil assembly for the measurement of neutrons having energies greater than 2.5 Mev; a 0.015-in. cadmium, 0.005-in. gold, 0.015-in. cadmium wafer for the measurement of thermal neutron doses greater than 0.3 mrad; silver metaphosphate glass
rods for X-ray, gamma, and thermal neutron dosimetry in the range of 100 to 10,000 rads; a chemical dosimeter with a calibrated color change; and a 1-1/4 in. by 1/4 in. by 0.015 in. strip of indium for neutron activation in case of a criticality accident. The indium strip provides a "tattler" and, in cases of neutron doses of approximately 1 rad, will indicate on a survey meter a reading of 0.5 mr/hr one hour after an exposure. The indium strip affords a means of rapidly separating exposed from nonexposed personnel. Figure 23.7.1 is an exploded view of the film badge used at the EGCR.

Pocket dosimeters are issued to and carried by all employees whose duties involve potential radiation exposure. These instruments are either the ion chamber type, requiring specialized equipment for reading and charging, or the self-indicating electroscope type which can be read by the wearer at any time. Both types of dosimeters are read on a daily shift basis and the results are used to supplement standard radiation control procedures. If the full scale range (200 mr) is exceeded, the individual's badge is processed and the dosage evaluated. In certain instances, personal radiation alarms are issued to provide more convenient visual and audible indications of radiation hazards.

23.7.2 Medical and Biological Assays

Pre-employment, periodic, transfer, and termination physical examinations conform to the standard practice of the TVA Employee Health Office Branch with certain additional examinations and greater frequency being required for radiation workers. These variations from standard practice are discussed in the Radiological Health Manual.

Where significant deposits of radioactive materials are known or suspected inside an individual's body, specialized techniques are employed to determine the resultant dose. These take the form of radiochemical analyses of urine and feces as a regular check of environmental conditions, or in the event of a serious accidental exposure, include complete blood examination and several consecutive collections of body wastes for specific radioisotope identification and dose determination. Blood, sputum, nose swabs, and tissue samples are also obtained and analyzed for the presence of radioactive materials.

All employees working in areas of possible neutron exposure are given slit lamp eye lens examinations semiannually.

The body counting facility at ORNL may be used to discriminate selectively between types and energies of radiations emitted from materials deposited in the individual.

23.7.3 Records

A comprehensive records system is maintained which permits the Radiological Health Section to regulate and control personnel radiation exposures. Provision is made for permanent storage of all developed film from badges worn by employees and visitors. IBM cards are utilized for speed and accuracy in recording both external and internal exposure results and for other special sample analyses. Information contained thereon is readily available in compiling routine reports or in obtaining the cumulative exposure record for any individual.

Upon termination, transfer, or at any other time upon request with justification, any employee is furnished a record of his radiation exposure.

23.8 Personnel Decontamination

Prompt removal of radioactive contamination from the skin is necessary to
FIG. 23.7.1

EGCR FILM BADGE

BADGE BACK
- LEAD FILTER
LATCH
WINDOW
PLASTIC FILTER
CADMIUM, GOLD, CADMIUM FILTER
ALUMINUM FILTER
IDENTIFICATION INSERT (INDIUM FOIL)
CHEMICAL DOSIMETER
PLUG

FILM PACKS
LEAD
COPPER
PLASTIC
SULFUR
GOLD
METER NUMBER

LAMINATED IDENTIFICATION INSERT
FRONT FRAME

PHOSPHATE GLASS

FILM PACKED
W/000
COPPER
PLASTIC
SULFUR
GOLD
METER NUMBER

LAMINATED IDENTIFICATION INSERT
FRONT FRAME
avoid the possibility of ingestion, injection, or inhalation, and to avoid
direct radiation exposure of the skin. Ordinarily, methods used for personal
cleanliness also apply in removing radioactive contamination from the skin.
The following procedures are utilized for personnel decontamination:

a. Remove any clothing or equipment known to be contaminated before attempt-
ing to determine levels of skin contamination.

b. Decontaminate any significantly contaminated spots or areas of the body.

c. If the contamination is general over the body surfaces, take a thorough
shower, placing particular emphasis on the hair areas, hands, and finger-
nails.

d. If the contaminant is fixed on a part of the skin that is rough and
scaly, a special paste is available in decontamination kits in the change
rooms. Massage the contaminated area with this paste using a surgical
brush for fingernails and calloused areas. Rinse, dry, and monitor.

If localized areas or spots of contamination cannot be removed by repeated use
of the foregoing methods, the Radiological Health Section is consulted for
advice.

23.9 Waste Disposal

The disposal of solid, liquid, and gaseous radioactive wastes is described in
Section 21. These operations are monitored by the Radiological Health Section
which is responsible for advising on radiation control procedures during normal
and abnormal operation.

23.10 Transfer and Shipment of Radioactive Materials

All transfers and shipments of radioactive materials to locations outside the
controlled vicinity are categorized as off-site transfers. The controlled
vicinity is that area immediately surrounding the controlled area (site) in
which people enter occasionally or work. Areas into which persons enter during
the normal course of travel, recreation, etc., are not considered a part of the
controlled vicinity. Waterways and highways may pass through the controlled
area without changing the classification, provided there are adequate means of
evacuation in case of an emergency.

23.10.1 Off-Site Transfers

Off-site transfers comply with regulations of the Interstate Commerce Commission
and with such additional regulations as may be prescribed by the project manager.
Section 22.5 of this document and the Radiological Health Manual give criteria
for radioactive material shipments to off-site locations. Specific requirements
relative to shielding, packaging, and labeling such shipments are outlined in
ICC regulations. Methods of shipping irradiated fuel are prescribed in detail
in Section 18.11.

Upon receipt of fuel or other radioactive material and prior to placing it in
storage, a radiation survey and inspection of the container is made for surface
contamination, damage in transit, leaking container, etc. Prior to transfer or
shipment of spent fuel elements or other radioactive materials, a thorough in-
spection is made to ascertain integrity of the shipping container, adequacy of
gaskets, covers, fastenings, and tie downs and conformance with ICC regulations,
where applicable.

23.10.2 On-Site Transfers
On-site transfers comply with current radiological health and operational requirements. Each transfer is packaged and identified by appropriate markings, and has affixed to it a radiation hazard sign or tag with specific information relative to the type, nature, and intensity of radiation from the radioactive material therein. Dose rates and contamination levels are such that personnel who come in contact with or are involved in a given transfer, do not exceed the limits prescribed for normal occupational exposures.

23.11 Radiation Incidents

Official policy regarding accidents which occur at AEC facilities is established in AEC Manual, Chapter 0502, "Health and Safety Reporting." Type and severity of accidents which require investigation and special reporting are listed therein.

23.11.1 Definition

A radiation incident is any situation involving radioactive material which results in: significant exposure to personnel, appreciable damage to or loss of usefulness of plant property or equipment, or potentially adverse effects upon public relations.

23.11.2 Initial Action

Radiation incidents may be discovered and reported by individuals at the scene or may be detected by one or more of the fixed radiation monitoring instruments. When any radiation monitoring instrument which records or alarms in the main control room indicates an abnormal condition, the reactor operator immediately notifies the health physicist on duty. If the situation is of a routine nature (instrument malfunction, etc.), the health physicist performs the necessary corrective action or advises the reactor operator of the existence of hazardous conditions.

23.11.3 Emergency Action

If emergency conditions develop, the senior member of the Radiological Health Section is notified immediately, and emergency procedures are initiated.

Normally, the first consideration is that of personnel protection. However, situations may arise in which rescue operations or emergency actions are necessary to ensure the functioning of essential equipment and, as such, may require submitting employees to emergency exposures. In general, the sequence in emergency procedures is as follows:

a. Evacuate all personnel immediately and segregate exposed and contaminated personnel for further attention.

b. Isolate radiation and contaminated areas.

c. If airborne contamination is present, shut off forced ventilation systems, insofar as plant operation permits. Where contaminated liquids are involved, prevent their release to the environment, if possible.

d. Obtain promptly, samples of air, liquid, and surface contamination for analysis. The specific method of analysis depends upon the source of contamination.

e. Initiate decontamination procedures to return the plant to normal operation.
The EGCR Emergency Plan describes the emergency organization, responsibility of units within the emergency organization, and procedures that are to be followed during various emergency conditions.

23.11.4 Investigation and Report

All radiation incidents are promptly investigated to determine the probable cause and a report made of the findings.
24. **PLANT SECURITY**

The program of plant security at the EGCR is directed at minimizing both intentional and unintentional acts that endanger the safe operation of the plant. Protection against sabotage is provided primarily by access control to the plant. The EGCR is located within an AEC security area and the site is enclosed by a fence. Public safety officers (guards) are assigned to each shift. Actions necessary to prevent access of unauthorized personnel into the plant area are directed by the guards. In addition, each member of the duty shift is responsible to the plant operations supervisor to detect, report, or take actions to prevent acts of sabotage to the plant. Additional steps taken to provide protection against sabotage are discussed in Volume I, Section 8.13.3.

The unintentional act of a duly authorized employee could possibly be as harmful as the intentional act of the saboteur. Therefore, the protection provided to minimize the probability of the unintentional act is based on the following:

a. **Training**

The responsibility associated with each job is clearly established. All employees are adequately trained prior to assuming this responsibility. A continuing training program (Section 11.6) is provided to inform employees of changes in design, method of operation, plant security, etc. By maintaining a well-informed staff, unintentional acts are minimized.

b. **Operating Instructions**

The detailed procedures for all phases of operation are written and approved prior to execution.

c. **Access Control**

Areas within the reactor site are controlled by one or more of the following methods:

1. Administrative Control--Only those persons authorized by the project manager or his representative are allowed in certain areas. (Control room, reactor building, etc.)

2. Physical Control--Control of personnel in some areas is limited physically by one or more of the following: locked doors, barricades, roped areas, warning lights, signs, etc.

The operating staff is responsible for continuous checking of access control and for reporting any infractions of the control. As a part of this program, all employees are provided with adequate instruction concerning critical areas where entrance is restricted. When it is necessary to enter a restricted area (to perform maintenance for instance), the plant operations supervisor is required; to insure that adequate supervision is provided the employee while in the area, to determine if the operation in the limited access area presents a possible hazard, and
to inform the project manager or his representative of actions to be taken in the restricted area.

d. **Periodic Plant Tests**

Periodic and special plant tests are performed on all critical systems (Section 17). The primary purpose of conducting these tests is to assure the proper functioning of these systems. These tests are capable of determining if an unintentional act results in the loss of any critical equipment in the plant. At any time maintenance is performed on a component or system, such proof tests are required.

e. **Plant Design**

The plant design includes many safety features, including duplication of equipment or systems. A single unintentional act, in general, results in consequences no greater than the normal failure of any given component in the plant. In cases where an unintentional act either directly or indirectly leads to a serious situation, additional protection is provided by access control.